DOCUMENT COVER SHEET

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UK AP1000® Environment Report

UKP-GW-GL-790, Revision 7

Westinghouse Electric Company LLC
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REVISION HISTORY

Revision	Description of Change
0	Initial Submittal
1	See EDMS.
2	See EDMS.
3	See EDMS.
4	See EDMS.
5	See EDMS.
6	See EDMS.
7	The COMAH tier for the AP1000 plant changed from lower tier to upper tier as a result of the amount of hydrazine stored on the site. Therefore, incorporates UKP-GW-GL-037 Rev 3_ADL in the following sections and table: - Section 2.9.2.1 - Section 2.10 - Section 5.4 - Table 2.9-1 Incorporates UKP-GW-GL-790 Rev 7_ADL to revise: - Table 2.9-1, Table 2.9-2, and Table 2.9-3 to list the mass of solution and not the mass of chemical only. - Table 2.9-1 and Table 2.9-6 to remove use of glycol on the AP1000 plant site. - Section 3.3.1.1 such that the maximum input flow rate from the degasifier separator to the WGS is 0.99 m^3/h (0.58 scfm). - Section 3.3.1.2 such that the delay bed average holdup times are 41.6 days for xenon and 2.3 days for krypton based on an input flow rate to the WGS of 0.99 m^3/h (0.58 scfm). - Revises Table 3.4-2 to update instrumentation terminology. - Revises Table 3.4-3 to standardize parameter properties. Design Change Proposal APP-GW-GEE-5349, Rev. 0 was reviewed and no changes within the accuracy of the estimates were required. Design Change Proposal APP-GW-GEE-5215, Rev. 0 was reviewed and no changes were required. Design Change Proposal APP-GW-GEE-4392, Rev. 0 was reviewed and no changes were required. Editorial changes.

OPEN ITEMS

Open Item #	Description
	N/A

Trademark Notices

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LIST OF ACRONYMS AND TRADEMARKS

Acronym	**Definition**
ac	Alternating Current
AD	Annual Average Dissolved
AGR	Advanced Gas-cooled Reactor
ALARP	As Low As Reasonably Practicable
AOX	Adsorbable organically bound halogens
AP1000	referring to the **AP1000**® nuclear power plant
ASS	Auxiliary Steam Supply System
AT	Annual Average Total
BAT	Best Available Techniques
BDS	Steam Generator Blowdown System
BEGL	British Energy Generation Limited
BPEO	Best Practicable Environmental Option
BPM	Best Practicable Means
BWR	Boiling Water Reactor
C	(degree) Celsius
CA	Concrete Filled-in-Place Form Modules
CCS	Component Cooling Water System
CCTV	Close Circuit Television
CDM	Construction (Design & Management)
CDS	Condensate System
CEC	Cavity Enclosure Container
CFA	Conditions for Acceptance
CFR	Code of Federal Regulations
CFS	Turbine Island Chemical Feed System
CNS	Civil Nuclear Security Division of ONR
COD	Chemical Oxygen Demand
COMAH	Control of Major Accident Hazards
CPS	Condensate Polishing System
CR	Concentration Ratio
CVS	Chemical and Volume Control System
CWS	Circulating Water System

LIST OF ACRONYMS AND TRADEMARKS (cont.)

Acronym	**Definition**
DAC	Design Acceptance Certificate
DAW	Dry Active Waste
DBD	Different By Design
dc	Direct Current
DCD	AP1000 Design Control Document
DF	Decontamination Factor
DfT	Department for Transport
DOE	Department of Energy
DOP	Dioctyl phthalate
DOS	Standby Diesel and Auxiliary Boiler Fuel Oil System
DPUR	Dose Per Unit Release
DTS	Demineralised Water Treatment System
DWS	Demineralised Water Transfer and Storage System
EA	Environment Agency
EDS	Non Class 1E DC and UPS System
EHS	Environmental, Health and Safety
EOL	End-of-life (core life)
F	(degree) Fahrenheit
FHM	Fuel Handling Machine
FHS	Fuel Handling and Refueling System
FPS	Fire Protection System
ft	feet
FWS	Feedwater System
GALE	Gaseous and Liquid Effluents (Calculation of releases of radioactive materials in gaseous and liquid effluents from pressurised water reactors)
GDA	Generic Design Assessment
GDF	Geological Disposal Facility
gpm	U.S. gallons per minute
Gy	Gray
h or hr	hour
HEPA	High Efficiency Particulate Air
HHISO	Half Height ISO (container), (meeting low level waste repository WAC)

LIST OF ACRONYMS AND TRADEMARKS (cont.)

Acronym	**Definition**
HLW	High Level Waste
HPA	Health Protection Agency
HRGS	High Resolution Gamma Spectroscope (a waste package assay instrument)
HP	High Pressure
HVAC	Heating, Ventilation, and Air Conditioning
HV-VLLW	High Volume Very Low Level Radioactive Waste
IAEA	International Atomic Energy Agency
IDS	Class 1E DC and UPS System
lb	pound
ILW	Intermediate Level Waste
IRWST	In-containment Refueling Water Storage Tank
IWMS	Integrated Waste Management Strategy
LLW	Low Level Waste
LLWR	Low Level Waste Repository
LOCA	Loss of Coolant Accident
lop	Life of plant
LOS	Main Turbine and Generator Lube Oil System
LP	Low Pressure
LRGS	Low Resolution Gamma Spectroscope (a waste package assay instrument)
LV-VLLW	Low Volume Very Low Level Radioactive Waste
m	meter
MAC	Maximum Allowable Concentration
MATTE	Major Accident to the Environment
MCERTS	Monitoring Certification Scheme
MCR	Main Control Room
mil	thousandth of an inch
MP	Monitoring Point
MPC	Multi-Purpose Canister
MTU/MWD	Megawatt–days per metric ton of uranium metal
NDA	Nuclear Decommissioning Authority
NPP	Nuclear Power Plant
NRC	United States Nuclear Regulatory Commission

LIST OF ACRONYMS AND TRADEMARKS (cont.)

Acronym	Definition
ONR	Office for Nuclear Regulation
ORNL	Oak Ridge National Laboratory
P&I Document	Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs
Pa	Pascal (could be with a preceding letter such as k, M)
PCS	Passive Containment Cooling System
PCSR	Pre-Construction Safety Report
PGS	Plant Gas System
ppb	Parts per billion
ppm	Parts per million
PSA	Probabilistic Safety Assessment
psi(a or g)	pound per square inch (absolute or gauge)
PWR	Pressurised Water Reactor
PWS	Potable Water System
QA	Quality Assurance
QMS	Quality Management System
Radwaste	Radioactive Waste
RCA	Radiation Controlled Area
RCDT	Reactor Coolant Drain Tank
RCPB	Reactor Coolant Pressure Boundary
RC	Reactor Coolant
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RNS	Normal Residual Heat Removal System
RO	Regulatory Observation (raised by regulators)
RQ	Risk Quotient
RWMC	Radioactive Waste Management Case
RWMD	Radioactive Waste Management Directorate (predecessor to RWM)
RWM	Radioactive Waste Management
RWS	Raw Water System
RXS	Reactor System

LIST OF ACRONYMS AND TRADEMARKS (cont.)

Acronym	Definition
SAP	Safety Assessment Principle
scfm	Standard cubic feet per minute
SCV	Secondary Containment Vessel
SDS	Sanitary Drainage System
SEPA	Scottish Environmental Protection Agency
SFS	Spent Fuel Pool Cooling System
SG	Steam Generator
SGS	Steam Generator System
SMPP	Secure Military Power Plant
SQEP	Suitably Qualified and Experienced Persons
SUDS	Sustainable Urban Drainage System
SWS	Service Water System
TCS	Turbine Building Closed Cooling Water System
TEI	Technology Evolution Index
TQ	Technical Query (raised by regulators)
UF	Uncertainty Factor
UK	United Kingdom
U.S. NRC	United Status Nuclear Regulatory Commission
VAS	Radiologically controlled area ventilation system
VBS	Nuclear Island Nonradioactive Ventilation System
VCS	Containment Recirculation Cooling 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
VLLW	Very Low Level Radioactive Waste
VTS	Turbine Building Ventilation System
VVM	Vertical Ventilated Module
VXS	Annex/Auxiliary Building Nonradioactive Ventilation System
VWS	Central Chilled Water System
VZS	Diesel Generator Building Heating and Ventilation System

LIST OF ACRONYMS AND TRADEMARKS (cont.)

Acronym	**Definition**
WAC	Waste Acceptance Criteria
WCPD	Worst Case Annual Plant Discharges
WGS	Gaseous Radwaste System
WHO	World Health Organisation
WLS	Liquid Radwaste System
WRS	Radioactive Waste Drain System
WSS	Solid Radwaste System
WWRB	Waste Water Retention Basin
WWS	Waste Water System
y or yr	year
ZIRLO	referring to **ZIRLO**® fuel cladding

1.0 INTRODUCTION

The United Kingdom (UK) Nuclear Regulators have developed a Generic Design Assessment (GDA) process for evaluating alternative designs for the next generation of nuclear power plants (NPPs) to be built in the UK. Westinghouse Electric Company LLC (Westinghouse) has submitted an application for its **AP1000** NPP design to be considered in this process.

As part of this application, Westinghouse has submitted UKP-GW-GL-793, "**AP1000** Pre-Construction Safety Report" (PCSR) (Reference 1-10) as the primary source of technical information for the NPP design.

This UK **AP1000** Environment Report has been prepared to consolidate and summarise the UK **AP1000** NPP waste management and environmental information and to compliment the PCSR (Reference 1-10) to meet the environmental requirements of the GDA process. This revision incorporates UKP-GW-GL-060, Revision 10, "AP1000 Design Reference Point for UK GDA" (Reference 1-19). The report comprises six chapters:

Chapter 1 Introduction

This chapter contains an introduction to the need for new nuclear power and the nuclear regulators responsible for permitting the new NPPs. The chapter also includes a description of the relationship between this submittal and other documentation submitted as part of the GDA process. The chapter contains a section on the management systems applied to the GDA application.

Chapter 2 Generic Plant Description

This chapter provides an overview of the development, layout, and design features of the **AP1000** NPP that are particularly relevant to the generation of emissions, discharges, and wastes. The chapter includes information on the storage of radioactive water and process chemicals.

Chapter 3 Radioactive Waste Management Systems

This chapter contains quantitative information about the radioactive emissions, discharges, and wastes produced in the **AP1000** NPP. The chapter includes a description of the minimisation and abatement techniques employed.

Chapter 4 Non-Radioactive Waste Management Systems

This chapter contains quantitative information about the non-radioactive emissions, discharges, and wastes produced in the **AP1000** NPP. The chapter addresses non-radioactive discharges of waste water and cooling water, and associated discharges of residual chemicals from the required chemical dosing regimes.

Chapter 5 Environmental Impact

This chapter defines the bounding characteristics of a UK coastal generic site and provides assumed data that is input into dose assessments for human and non-human species. The dose assessments for the UK generic site are reported.

Chapter 6 Environmental Monitoring

This chapter identifies the proposed emission and discharge limits for the **AP1000** NPP and the arrangements for monitoring emissions and discharges from the generic site.

Chapter 7 Selected Considerations for Specific Sites

For the GDA, it is assumed that the generic site will be occupied by one **AP1000** NPP and the information in this Environment Report reflects this case. This chapter provides a commentary on the relevance of the Environment Report to sites which may have multiple **AP1000** NPP units. Detailed environmental impact of such sites will be evaluated at the site-specific design stage.

1.1 Need for Nuclear Power

In January 2008, the UK government published "A White Paper on Nuclear Power" (Reference 1-2) on the future of nuclear power in the UK. The White Paper sets out the decision the government has taken in response to a consultation on nuclear power. The consultation considered the following issues:

- nuclear power and carbon emissions
- security of supply impacts of nuclear power
- the economics of nuclear power
- the value of having a low-carbon electricity generation option: nuclear power and the alternatives
- the safety and security of nuclear power
- transport of nuclear materials
- waste and decommissioning
- nuclear power and the environment
- the supply of nuclear fuel
- supply chain and skills implications
- reprocessing of spent fuel

The White Paper (Reference 1-2) concluded that:

"The Government believes it is in the public interest that new nuclear power stations should have a role to play in this country's future energy mix alongside other low-carbon sources; that it would be in the public interest to allow energy companies the option of investing in new nuclear power stations; and that the Government should take active steps to open up the way to the construction of new nuclear power stations. It will be for energy companies to fund, develop and build new nuclear power stations in the UK, including meeting the full costs of decommissioning and their full share of waste management costs."

The majority of the UK's nuclear power stations are due to close over the next two decades. It is appropriate that, as part of a balanced approach to electricity generation, a second phase of nuclear power station construction is encouraged to help ensure a clean, secure, and sufficient supply of energy demanded by modern society.

1.2 Regulatory Bodies

1.2.1 Regulatory Approach

Regulatory bodies have important roles in the nuclear power program. Their remit is to ensure that any new nuclear power station built in the UK meets the highest standards of safety, security, environmental protection and waste management. The following organisations are involved in the authorisation of new nuclear power stations.

1.2.2 Environmental Regulators

1.2.2.1 Environment Agency

The Environment Agency (EA) regulates the environmental aspects of NPPs in England:

- radioactive waste (radwaste) disposal, including discharges;
- abstraction from and discharges to controlled waters, including rivers and estuaries;
- the sea and groundwater;
- operation of specific "non-nuclear" activities;
- assessment and, where necessary, clean-up of contaminated land;
- disposal of conventional waste; and
- certain flood risk management matters.

It also has wider responsibilities with regard to Euratom Article 37 requirements concerning the impact of nuclear sites on other European Union Member States. Operators have to satisfy the EA that discharges and disposals made into the environment are minimised and their effects are acceptable, such that people and the environment will be properly protected throughout the whole lifecycle of the plant, from construction to decommissioning.

1.2.2.2 Natural Resources Wales

Natural Resources Wales provides the same regulatory function for NPPs in Wales as does the EA in England.

1.2.3 Office for Nuclear Regulation

The Office for Nuclear Regulation (ONR) grants site licences to the operators of nuclear power stations. Applicants must satisfy ONR about the safety aspects of the design, manufacture, construction, commissioning, operation, maintenance, and decommissioning of the installation, and the management of radwaste on the site, before a licence is granted.

1.2.4 Civil Nuclear Security Division of ONR

ONR's Civil Nuclear Security Division (CNS) regulates security at all civil nuclear sites. It is concerned with physical security of nuclear material, IT security, security of nuclear material in transit, and vets people who access nuclear sites. CNS requires the holder of a nuclear site licence to submit a site security plan, which must be approved before nuclear material arrives on site.

1.2.5 ONR Transport

ONR Transport is the UK Competent Authority for the safe transport of all radioactive material by all modes. It issues Design and Shipment Approvals for certain package designs.

It directly regulates road transport and some aspects of rail transport, and advises/supports the Civil Aviation Authority and the Maritime and Coastguard Agency in air and maritime transport matters.

1.2.6 Local Planning Authorities

Local planning authorities have a role in approving the planning applications required for the nuclear power stations.

1.3 The Generic Design Assessment Process

As part of this regulatory role, the EA and ONR (referred to as Nuclear Installations Inspectorate at the time) proposed that the new nuclear power stations be subject to a methodical, well-defined, multi-stage assessment and licensing/permitting process. This process would implement a "pre-authorisation" system for reactor designs to allow generic designs to be assessed in advance of any application to build a nuclear power station at a particular location. This process is called the generic design assessment or GDA. It involves looking at all the design issues separately from the other important factors such as whether the siting of a new nuclear power station is suitable, or whether the potential operator is competent.

The GDA process was introduced jointly by the UK nuclear regulators – the ONR, the CNS (referred to as Office of Civil Nuclear Security at the time), and the EA. The Scottish Environmental Protection Agency (SEPA) is not taking part in the GDA.

The ONR produced guidance to reactor vendors or vendor/operator partnerships in preparing a GDA application (Reference 1-3). As part of the GDA process, the ONR and the CNS carry out a detailed assessment of the safety and security elements of a design, based on a submission made by the reactor vendor or vendor/operator partnership.

In addition, the EA has issued its "Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs" (P&I Document) (Reference 1-4), which describes the information on waste management and environmental issues that the EA needs to use to perform a generic assessment of new NPPs in England.

The regulators consider that it is important for potential site operators/licensees to be engaged in the GDA process, as they are required to demonstrate sufficient knowledge of the design before receiving permission to construct and operate a nuclear power station. The operator may also wish to be part of the design process to allow the design to be adapted to its particular needs.

The regulators intend that the GDA process operates in a transparent and open way so that the documentation provided for the GDA is made available to the public by the Requesting Party, with the exclusion of sensitive nuclear information and commercially confidential information.

The regulators make public statements on their progress and interim findings at key stages during the GDA process, and also publish their technical reports.

When the GDAs are completed, the regulators issue reports on their findings. If the design is judged to be satisfactory, the regulators issue the following:

- EA: Statement of Generic Design Acceptability
- ONR: Design Acceptance Confirmation

- CNS: Generic Conceptual Security Plan approval

The Requesting Party is expected to cooperate with the regulators by:

- Liaising with regulators
- Responding to regulatory issues
- Providing additional information as requested
- Responding to public comments

When applications are made for site-specific permissions (nuclear site licence, environmental authorisations and permits, and security plan approval), the regulators follow their existing procedures. Where these site-specific applications are based on a design that has undergone GDA, the regulators take full account of the work that they have already carried out and the advice that they have provided. It is expected that following a GDA, the regulatory bodies would typically be involved as follows:

- EA – for site-specific applications, the EA will take full account of the detailed design of the proposed station, including any changes since GDA, and generally focus on local impacts associated with the permissions sought and the suitability of the potential operator. Potential operators will need to address exclusions or caveats from the GDA process.

- ONR – will focus their licensing assessment on those site-specific issues that have consequences for the safety of the station and matters relating to the organisational structure and capabilities of the potential operator.

- CNS – will require the security plan to be taken forward and developed into a site security plan that could be considered for approval of material that is brought to the site. A construction security plan, that builds upon the conceptual security plan of the intended location and articulates how this will be developed into the site security plan, will be required to be approved by CNS before construction activities begin.

1.3.1 Westinghouse AP1000 NPP Generic Design Assessment – Overview of Documentation

Westinghouse is seeking approval to have an **AP1000** NPP simplified, passive advanced light water reactor plant built in the UK. The general plant description is included in Chapter 1 of the PCSR (Reference 1-10). The **AP1000** NPP design has been incorporated into the United States Nuclear Regulatory Commission's (U.S. NRC's) Design Certification Rule for the **AP1000** NPP design, Section II.A of Appendix D to 10 Code of Federal Regulations (CFR) Part 52. However, to show compliance with the UK regulations, additional information on the **AP1000** NPP is required.

The EA reviewed previous Westinghouse submittals, including the European DCD (Reference 1-1) against its "Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs" (Reference 1-4), and "Generic Design Assessment – Regulatory Issue RI-**AP1000**-0001" (Reference 1-5) was raised to provide guidance on the additional information needed.

In December 2008, Westinghouse responded by issuing Revision 1 of the "UK **AP1000** Environment Report" (Reference 1-6) to supplement the European DCD and to provide the additional information required. The Environment Report has been reviewed by the EA and ONR who issued regulatory queries (previously referred to as technical queries, TQs) and regulatory observations (ROs) which required further response or clarification from

Westinghouse. This report addresses the regulatory queries and ROs raised by the EA and ONR. The document was substantially restructured to present the information by subject rather than by the P&I Document (Reference 1-5). This is intended to make the information more accessible to the public.

Table 1.3-1 is attached to help readers understand where the information requested by the EA in their P&I Document is addressed.

This report forms part of the 2017 design reference point for the GDA of the **AP1000** plant (Reference 1-19).

1.3.2 Relationship to Integrated Waste Management Strategy

The EA has requested the production of an integrated waste management strategy (IWMS) for the wastes generated by the **AP1000** NPP. UKP-GW-GL-054, "UK **AP1000** Integrated Waste Strategy" (Reference 1-7), has been produced and will be referenced throughout this Environment Report to provide details of the recycling, treatment, and disposal of radioactive and non-radioactive solid wastes, liquid waste, and gaseous waste.

1.3.3 Relationship to Radioactive Waste Management Case

The UK Regulators have also requested that a radioactive waste management case (RWMC) is prepared for the **AP1000** NPP waste treatment systems. An RWMC has been prepared for the **AP1000** NPP which demonstrates the long-term safety and environmental performance of the management of specific Intermediate Level Waste (ILW) (Reference 1-8) and High Level Waste (HLW) (Reference 1-9) from their generation, conditioning, storage, and disposal. These documents hold information that is relevant to the Environment Report and vice versa.

1.4 Management System

1.4.1 Westinghouse Management System

The management system used in preparation of the Environment Report is also used in preparing other documentation supporting the UK GDA of the **AP1000** NPP. These other documents include:

- **AP1000** PCSR (Reference 1-10). The PCSR includes a similar description of the Management System.
- UK **AP1000** Integrated Waste Strategy (Reference 1-7)
- UK **AP1000** Radioactive Waste Management Case Evidence Report for Intermediate Level Waste (Reference 1-8)
- UK **AP1000** Radioactive Waste Management Case Evidence Report for High Level Waste (Reference 1-9)

It is the Westinghouse Policy to design, produce, market, and distribute products and services and to conduct operations in an environmentally sound, socially responsible manner. We consider the impact our actions may have on the environment and the health and safety of our employees, subcontractors, customers, and public (Reference 1-11). Westinghouse is committed to the integration of environmental, health and safety (EHS) into the design process, as well as during construction and commissioning. Implementation of the Policy is through the "Westinghouse Environmental, Health and Safety Management System" (Reference 1-12).

The Westinghouse Electric Company "Quality Management System" (Reference 1-13) has been developed to comply with regulatory, industry, and customer quality requirements imposed by customers or regulatory agencies for items and services provided by Westinghouse world-wide operations. The Quality Management System (QMS) describes the Westinghouse commitments to the quality assurance (QA) requirements of ISO 9001; 10CFR50, Appendix B; and ASME NQA-1.

Westinghouse, headquartered in Cranberry Township, Pennsylvania, U.S.A., has operations located throughout the world that are responsive to energy industry, utilities, and government needs. Westinghouse operations are made up of organisations that are responsible for specific business areas. These operational organisations are responsible for marketing, design, procurement, manufacture, installation, inspection, testing, servicing, project management, and operation of certain NPP items, radioactive material packaging and transportation, and non-nuclear items. Westinghouse also offers engineering services such as life-extension studies, diagnostics, service analyses, and item and service testing. The New Plants & Major Projects Business Unit is responsible for designing and licensing the **AP1000** NPP.

The QMS applies to activities that affect the quality of items and services supplied by Westinghouse. It defines the basic requirements applicable to customer contracts and is a commitment to our customers. It serves as a directive for all functions in establishing necessary policies and procedures that comply with the requirements of ISO 9001; 10CFR50, Appendix B as applicable for safety-related activities; and ASME NQA-1.

Westinghouse implements all applicable requirements of the QMS for all safety-related items and services. Westinghouse implements those requirements of the QMS consistent with ISO 9001 for items and services that are not safety-related, as a minimum.

An inspection conducted by UK regulators from 31 March to 3 April 2009 found that Westinghouse uses a well-developed set of quality processes that include sub-tier procedures that are periodically reviewed and audited. Joint MSQA Inspections were conducted with the ONR during GDA Step 4 in July and December 2010. In the Step 4 report, ONR-GDA-AR-11-013, the UK Regulators stated, "From a MSQA view point, the Westinghouse **AP1000** NPP design is suitable for construction in the UK". The work for GDA is guided by UKP-GW-GAH-001, "Project Quality Plan for UK Generic Design Assessment (GDA) Issue Resolution" (Reference 1-14) and procedures.

Safety-related items, services, and activities are those that may impact those NPP structures, systems, and components that are relied upon to remain functional during and following design basis events to assure: 1) the integrity of the reactor coolant pressure boundary (RCPB), 2) the capability to shut down the reactor and maintain it in a safe shutdown condition; or 3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set by the governing regulatory agency, if applicable. In addition, safety-related items, services, and activities may be those defined by a governing regulatory agency or contract.

Project Quality Plans, for example, the Project Quality Plan for the UK GDA, may be developed to supplement the requirements of the QMS and provide for specific contractual requirements and alternate QA standards when necessary.

Westinghouse complies with the regulatory requirements applicable to the items and services it provides for use in NPPs, as imposed by the governing regulatory agency.

The Project Quality Plan for the UK GDA (Reference 1-14) establishes the Project QA Plan and defines the QA objectives for the conduct of activities to be performed by Westinghouse related to the GDA of the **AP1000** NPP and supporting licensing activities in the U.K. Work performed by Westinghouse shall be performed in accordance with the QMS described above. It is the policy of Westinghouse to provide accurate and reliable information to fully satisfy the EA and ONR and regulatory requirements. This Project QA Plan for the UK delineates the QA requirements that Westinghouse uses to meet its stated objectives for providing the necessary technical information and documentation for the EA and ONR to conduct a complete GDA of the **AP1000** NPP design. Activities affecting quality are documented in accordance with Westinghouse manuals, procedures, instructions, specifications, and drawings that contain appropriate information to accomplish prescribed activities in a complete and satisfactory manner.

The Management of Safety starts at the senior level of an organisation, expressed in terms of a safety policy committing the organisation to objectives, actions, and behaviours that will deliver effective safety through all phases of plant life. The policy provides organisational commitment to continuous safety improvement. Safety management permeates the whole of the organisation as evidenced by an effective safety culture.

1.4.2 Licensee's Management System

1.4.2.1 Intelligent Customer

Westinghouse has an established organisational structure and arrangements to deliver effective safety management up to the end of its input to the GDA, and during subsequent plant construction and commissioning prior to handover to the Operating Organisation. Thereafter, the Operating Organisation's Licensee will assume the responsibility for safety and environmental management of the operating power station through the operating life and eventual decommissioning of the plant. The Westinghouse organisational structure, processes, and competences to control the design have the attributes of a Design Authority in the context of the GDA.

Licensees of the Operating Organisation will possess the characteristics of an Intelligent Customer. The ONR has published a Technical Assessment Guide (Reference 1-15) that provides guidance on the required attributes of an Intelligent Customer. This guidance signposts the specific parts of the Safety Assessment Principles (SAPs), the Licence Conditions, and International Atomic Energy Agency (IAEA) documentation that are relevant to the Intelligent Customer role.

The Intelligent Customer should:

- Understand the safety requirements of all their activities relevant to safety including those of its contractors and to take responsibility for managing their safe operation.

- Understand their duties under the law with respect to safety.

- Set, interpret, and deliver safety standards relevant to their nuclear operations.

- Have sufficient breadth and depth of knowledge and experience to understand the safety envelope of their plant(s) and the nuclear safety hazards represented.

- Understand and support all aspects of the safety case/report and the facility operation over the full facility lifetime – including, where necessary, decommissioning and disposal.

- Know where and when to seek advice and, on receipt of this advice, understand the implications for safety.

- Maintain and develop the corporate memory with an ability to readily extract nuclear safety-related business intelligence.

- Ensure adequate numbers of Suitably Qualified and Experienced Persons (SQEP) are available to make safety judgments.

The Licensee must develop an organisational baseline which will ensure that the Licensee operates as an Intelligent Customer and the Licensee is capable of exerting proper controls of contractors' activities. The proper implementation of the above attributes will ensure that this is achieved.

1.4.2.2 Pre-Construction Design and Safety Case Consolidation

The GDA Safety Case, and the associated design which receives a Design Acceptance Certificate (DAC) from the regulator (including exclusions and conditions), will be the basis for the Site-Specific Safety Case.

The Licensee may wish to make alterations to the GDA design prior to (and possibly even during) construction and commissioning. This will require a defined design modification process as well as alterations to the safety case. The management arrangements (both Westinghouse and Licensee) will clearly state that the Licensee is responsible for these processes to ensure the safety of any such changes. The Licensee is expected to have an acceptance process before any changes that could affect safety or the environment are put into effect. The relationship between Westinghouse and the Licensee will require Westinghouse to provide any information and support to the Licensee to enable them to make informed decisions and to be able to present the case for change knowledgeably to the regulators. In this case, clear and effective lines of communication are to be defined.

The Licensee will take full responsibility for ensuring that Westinghouse is fully informed as to all aspects of the EHS significance of any work which Westinghouse might be asked to do in this context, and this should be fully incorporated in the Licensee's management arrangements.

1.4.2.3 Construction

During the construction phase, responsibility for safety and the environment will rest with the Licensee (and Westinghouse). The Licensee will also have to implement arrangements to ensure full compliance with the legal regulations such as the Construction (Design and Management) (CDM) Regulations (Reference 1-18).

The management arrangements for any design modifications will be similar to these described in subsection 1.4.2.2 above and will be documented in the management arrangements of the Licensee and Westinghouse. The Licensee will require adequate quality arrangements to be in place to manage testing and inspection requirements during construction. During manufacturing, appropriate contractor quality arrangements need to be in place so that the Licensee can ensure that safety, reliability, and environmental targets are achieved.

1.4.2.4 Operation

Westinghouse will make clear in the GDA safety documentation its initial identification of those operational constraints and requirements which will be needed to operate the plant safely and to protect the environment. The GDA safety documentation will also define the limiting conditions of operation, the expectations from the structures, systems, and components, and the necessary testing and maintenance. The GDA safety documentation will reflect the role of the operator insofar as this affects safety and the environment.

Westinghouse expects the prospective Licensee to utilise this information during the Licensee's determination of technical specifications and operating rules. Westinghouse also expects the Licensee to utilise this information in developing their Station Operating Instructions.

During operation, Westinghouse will provide services that the Licensee may need to operate their plant safely. In addition, Westinghouse will encourage and assist the Licensee to ensure that they have management arrangements in place to maintain in-house core competences which will permit them to demonstrate competence as a Licensee. It will be the Licensee's responsibility to ensure that the essential services they require to maintain safety and the environment continue to be available either from Westinghouse or an alternate supplier if Westinghouse is unable or unwilling to provide such support.

1.4.2.5 Decommissioning

Only high-level reference will be made to management arrangements for decommissioning in this report. More details can be found elsewhere in the GDA documentation (Reference 1-16). It is anticipated that knowledge transfer and the management arrangements developed to support operation will provide the foundations of the arrangements required for the decommissioning process. However, a review of this will be carried out during the creation of the operational management arrangements to ensure that this is true. It is expected that the decommissioning arrangements will be reviewed every five years, consistent with the review of site specific Funded Decommissioning and Waste Management Plans.

1.4.3 Westinghouse Support to a Licensee's Management System

Any contract between Westinghouse and a prospective Licensee is expected to define the tasks and the interrelation between the organisations. The management arrangements and the related communication processes between the Licensee and Westinghouse have to be agreed within this framework.

1.4.3.1 Design Authority

The GDA process requires that the Licensee of the Operating Organisation establishes a Design Authority, and that arrangements are put in place that ensure that sufficient information is transferred from the Design Organisation to the Licensee such that it can function as an effective Design Authority. The Licensee Design Authority is a key component of the Intelligent Customer organisation.

In the context of the **AP1000** NPP introduction to the UK, Westinghouse has processes in place that ensure that design and operational knowledge is transferred to the Licensee of the Operating Organisation to permit it to perform as an Intelligent Customer. These processes include the provision of design information and comprehensive training and education programmes such that the Licensee can establish a credible Design Authority.

The Design Authority may not be bounded by the Operating Organisation and may include inputs from external bodies, particularly the Design Organisation, Westinghouse. In the Intelligent Customer role, the Licensee will show that it can maintain continuity of the necessary engineering skills and knowledge, access to appropriate research, and control intellectual property issues such that it can demonstrate full control of the plant independent of any changes in the external contracting environment.

The future Licensee of an **AP1000** NPP is likely to join and to contribute to the AP1000 Owners Group and the Pressurised Water Reactor Owners Group (formerly the Westinghouse Owners Group) which provides a focus for information, services, and development programmes from which Owners and Licensees of **AP1000** plants can benefit. The group is coordinated centrally by Westinghouse. The services provided by the group include the optimisation of Technical-Specifications, performance improvements, and access to a common knowledge base of plant and licensing issues.

1.4.3.2 Life Cycle Support

Generally, throughout all phases of the plant life cycle, the key areas for interface with the site Licensee are expected to include the following:

- Westinghouse's intention is to work closely with the Licensee, providing all necessary technical information to enable the development of the safety cases and all other essential operating documentation, including emergency arrangements and maintenance schedules.

- Westinghouse is expected to demonstrate to the Licensee the adequacy of its management arrangements to support safety, the environment, and the quality of the plant.

- Westinghouse will work with the Licensee to support the production of a comprehensive Licensee quality management system insofar as the safety and environmental aspects of operation of the Westinghouse **AP1000** NPP design are concerned.

- Westinghouse expects the Licensee to have a document management system that ensures appropriate records are retained. Westinghouse will support the Licensee in transferring **AP1000** NPP information into their document management system.

- Westinghouse will identify the aspects of the design that need special consideration from a security point of view and will relay them to the Licensee. Whilst these will not explicitly be identified in the management arrangements, a requirement to transfer such information will be included in the Westinghouse management arrangements as will an undertaking to support the Licensee in any discussions with CNS.

- SQEP from within both the Licensee's organisation and Westinghouse will work to their own respective management systems to support safety and the environment. However, where the Licensee has a high dependency on individual contractors (e.g., Westinghouse staff) to support safety or the environment, the Licensee is expected to have processes for ensuring the competency of contractor staff. All contracted SQEP personnel (Westinghouse, other) will work according to the Licensee's management system. Where this is done, it will be clearly communicated and agreed upon between both parties. Westinghouse will cooperate with the Licensee in demonstrating the competency of its staff for their assigned roles. Westinghouse expects that the Licensee will secure an independent review of proposals prior to submission to the regulators (e.g., through appropriate Independent Nuclear Safety Assessments and Nuclear Safety Committee arrangements).

There is an overview of lifetime management arrangements in the Life Cycle Safety Report (Reference 1-11).

1.4.3.3 Knowledge Transfer and Competence Retention

On a contractual basis, Westinghouse will support the Licensee to ensure that their knowledge of the aspects of the design which affect each of these topics is transmitted in an effective and appropriate way.

Arrangements for knowledge transfer will be defined and discussed in detail with prospective Licensees. The process will be applied throughout the stages leading up to and beyond the start of operation. These will include, but are not be limited to:

- Programme and processes (GDA, for construction and operation)
- Technical knowledge of the plant systems, staffing, and related competences
- Arrangements for training
- SQEP arrangements and competence retention
- Arrangements for experience feedback (construction, commissioning, operation/project management, and communication)

The ability of the Licensee to satisfy the safety and environmental requirements of the site licence shall be demonstrated via the safety management prospectus. This may be one document which embraces safety, environment and security, or it may be split according to the Licensee's preference. As required, Westinghouse will assist to ensure that such a safety management prospectus is created and maintained. In addition, Westinghouse will advise and assist in the creation of a Licensee's nuclear baseline, whereby it can demonstrate the adequacy of its organisational structure, staffing, and competences to maintain safety. This will be based on requirements from the operator to support safety and the environment.

Knowledge transfer will be systematically carried out starting from the arrangements in place during the GDA process. One such method already in place during the GDA is the involvement of the utilities in the safety and environmental document specification and review process.

Once a site has been selected, Westinghouse will provide the utility with all necessary technical, safety, and environmental input to prepare a site-specific PCSR and Environment Report.

During the GDA, pre-construction, construction, commissioning, and operational stages, Westinghouse will engage with the Licensee in respect of the Learning from Experience processes which Westinghouse has in place. The Learning from Experience processes will benefit from the plants being constructed in China and the USA. The requirements to alert and involve Licensees in the discussion and resolution of learning events, which have relevance to safety or the environment, will be built into the Westinghouse management arrangements.

1.5 References

1-1 EPS-GW-GL-700, Rev. 1, "**AP1000** European Design Control Document," Westinghouse Electric Company LLC, January 2010.

1-2 "A White Paper on Nuclear Power," Department for Business, Enterprise & Regulatory Reform, January 2008.

1-3 ONR-GDA-GD-001, Revision 2, "New Nuclear Reactors: Generic Design Assessment Guidance to Requesting Parties, Office for Nuclear Regulation," June 2016.

1-4 "Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs," Version 1, Environment Agency, January 2007[1].

1-5 Grundy, C., Environment Agency Letter No. WEC70020R, Rev. 0, "Generic Design Assessment – Regulatory Issue RI-**AP1000**-0001," February 2008.

1-6 UKP-GW-GL-790, Rev. 1, "UK **AP1000** Environment Report," Westinghouse Electric Company LLC, December 2008.

1-7 UKP-GW-GL-054, Rev. 1, "UK **AP1000** Integrated Waste Strategy," Westinghouse Electric Company LLC, March 2011.

1-8 UKP-GW-GL-055, Rev. 2, "UK **AP1000** Radioactive Waste Management Case Evidence Report for Intermediate Level Waste," Westinghouse Electric Company LLC, March 2011.

1-9 UKP-GW-GL-056, Rev. 2, "UK **AP1000** Radioactive Waste Management Case Evidence Report for High-Level Waste," Westinghouse Electric Company LLC, March 2011.

1-10 UKP-GW-GL-793, Rev. 1, "**AP1000** Pre-Construction Safety Report," Westinghouse Electric Company LLC, January 2017.

1-11 UKP-GW-GL-737, Rev. 2, "Plant Life Cycle Safety Report," Westinghouse Electric Company LLC, March 2011.

1-12 "Environmental, Health and Safety Management System,", EHS MS Rev. 1, Westinghouse Electric Company LLC, January 2013.

1-13 QMS-A, "Rev. 7, "Westinghouse Electric Company Quality Management System," Westinghouse Electric Company LLC, October 2013.

1 Version 3 (October 2016) of "Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs," is now available, but when the "UK AP1000 Environment Report" was reviewed against this doucment, only version 1 was available.

1-14 UKP-GW-GAH-001, Rev. 6, "Project Quality Plan for the UK Generic Design Assessment (GDA) Issue Resolution," Westinghouse Electric Company LLC, October 2016.

1-15 NS-TAST-GD-049, Rev. 5, Licensee Core and Intelligent Customer Capabilities", April 2016.

1-16 UKP-GW-GL-795, Rev. 0, "UK **AP1000** NPP Decommissioning Plan" Westinghouse Electric Company LLC, March 2011.

1-17 APP-GW-GL-700, Rev 19, "**AP1000** Design Control Document", Westinghouse Electric Company LLC, June 2011.

1-18 Health and Safety Executive, "Construction (Design and Management) Regulations", 2015".

1-19 UKP-GW-GL-060, Rev 10, "AP1000 Design Reference Point for UK GDA", Westinghouse Electric Company, LLC, January 2017.

Table 1.3-1

RELATIONSHIP BETWEEN ENVIRONMENT REPORT AND PROCESS AND INFORMATION DOCUMENT REFERENCE ISSUES

Environment Report Section	Environment Report Topic	P&I Document Reference Number
1	INTRODUCTION	
1.1	Need for Nuclear Power	
1.2	Regulatory Bodies	
1.3	The Generic Design Assessment Process	
1.4	Management System	1.1
2	GENERIC PLANT DESCRIPTION	
2.1	General Facility Information	1.2
2.2	Development of the **AP1000**	1.5 part, 2.9 part
2.3	External Appearance and Layout	1.2
2.4	Reactor Power Conversion System	1.2
2.5	Engineered Safety Features	1.5 part
2.6	Best Available Techniques Applicable to **AP1000** Reactor Design	1.5 part
2.7	Plant Water Use	3.1
2.8	Transportation of Radioactive Fuel	1.4, 2.4, A-1
2.9	Radioactive and Non-Radioactive Materials	3.2 ,3.4
3	RADIOACTIVE WASTE MANAGEMENT SYSTEMS	
3.1	Introduction	
3.2	Minimisation of Waste at Source	2.1 part
3.3	Gaseous Radioactive Waste	1.5 part, 2.1 part, 2.2 part, 2.9 part
3.4	Liquid Radioactive Waste	1.5 part, 2.1 part, 2.2 part, 2.9 part
3.5	Solid Radioactive Waste	1.4, 1.4.3, 1.5 part, 2.1 part, 2.4, A-1, 2.5.1, 2.5.2, 2.5.3, 2.5.4, 2.5.5

Table 1.3-1 (cont.)

RELATIONSHIP BETWEEN ENVIRONMENT REPORT AND PROCESS AND INFORMATION DOCUMENT REFERENCE ISSUES

Environment Report Section	Environment Report Topic	P&I Document Reference Number
4	NON-RADIOACTIVE WASTE MANAGEMENT SYSTEMS	
4.1	Gaseous Non-Radioactive Waste	3.3
4.2	Liquid Non-Radioactive Waste	3.3
4.3	Solid Non-Radioactive Waste	2.4
5	ENVIRONMENTAL IMPACT	
5.1	Characteristics of the Generic Site	1.3, 2.9 part
5.2	Radiological (Human Dose Assessment)	2.7, 2.8
5.3	Radiological (Non-Human Dose Assessment)	2.10
6	ENVIRONMENTAL MONITORING	
6.1	Proposed Regulatory Limits	2.3
6.2	Monitoring Programmes	2.6

2.0 GENERIC PLANT DESCRIPTION

2.1 General Facility Information

The GDA site consists of one **AP1000** NPP on a coastal site. During its operating and decommissioning life, an **AP1000** NPP will generate solid waste, liquid discharges, and gaseous emissions. To assist in understanding the sources the **AP1000** NPP is described below.

The **AP1000** NPP is designed to provide net electrical power to the grid of at least 1000 MWe. The overall goal is plant availability of greater than 90 percent considering all forced and planned outages. The plant design objective is 60 years without the need for replacement of the reactor vessel, although the design provides for the replacement of other major components, including the steam generators.

Westinghouse has received standard design certification from the U.S. NRC for the **AP1000** NPP design.

2.2 Development of the AP1000 NPP

The history of the development of the **AP1000** NPP design has been previously documented (Reference 2-1). Throughout the design process, consideration has been given to safety, environmental protection, and waste minimisation through concepts comparable to the UK regulatory principles of As Low as Reasonably Practicable (ALARP), Best Available Techniques (BAT), and the waste management hierarchy.

2.2.1 Design Principles – Safety & Simplicity

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

1. No alternating current (ac) power is required to perform any safety function. This includes the three key safety functions of:

   - 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 (MCR) habitability
   - beyond design basis security-related mitigation features.

2. The fission product barriers of the fuel clad, the reactor vessel and coolant system, and the containment vessel are maintained. The containment vessel is an ideal barrier against radioactive releases to the environment. To transfer decay heat out of the core, natural, non-pumped mechanisms like natural circulation, evaporation, conduction, convection, and condensation are used.

3. Core damage frequency and large release frequency, as calculated by a robust probabilistic safety assessment (PSA), are minimised, 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 Class 1 responses to excessive leakage must be provided for these plants. In the **AP1000** NPP, the shaft seals are eliminated altogether through the use of wet winding 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 number of potential reactor coolant release scenarios, each requiring a mitigation strategy. In **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 tube leak.

In addition to the design objectives of safety first and no ac power for Class 1 functions, the **AP1000** NPP design process included making constructability, reliability, operability, and maintainability part of the design.

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.

Detailed discussion of the **AP1000** NPP design safety case can be found in UKP-GW-GL-793, "**AP1000** Pre-Construction Safety Report" (Reference 2-2).

2.2.2 Development of the AP1000 NPP Design

The design of the **AP1000** NPP is a development of the AP600 design (References 1-17 and 2-1). The AP600 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 of shutting down the reactor, keeping it cool, and containing its coolant.

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

The development of the AP600 was a large design and licensing effort to produce the safest, simplest, least expensive NPP on the world market. However, other nuclear plants were not AP600's competition, other non-nuclear power stations were. In particular, natural gas plants were the economic plants of choice in the United States. In order to compete against natural gas plants 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 its safety margins and increase its 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 the cost of electricity generated by 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:

1. Safety first – maintain large margins to safety limits
2. Maintain passive nature of all safety functions
3. Maintain no operator actions for safety functions

4. Maintain use of proven components and technology
5. Do not change the plant footprint and lose layout and analysis already completed
6. No design impacts unrelated to power
7. Minimise design impacts. 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.

2.3 External Appearance and Layout

A schematic of the **AP1000** NPP is shown in Figure 2.3-1. The site and plant layout are illustrated in Figure 2.3-2. The location of the functional components of the **AP1000** NPP power generation complex is shown in Figure 2.3-3.

Each **AP1000** NPP unit is composed of the following principal building structures, each constructed on their own individual foundation slabs.

2.3.1 Nuclear Island

The nuclear island comprises the containment building, shield building, and auxiliary building:

- The containment building is a freestanding cylindrical steel containment vessel with elliptical upper and lower heads. The containment building provides shielding for the reactor core and the reactor coolant system (RCS) during normal operations. The containment building houses the RCS and other related systems and provides a high degree of leak tightness. It provides containment of the releases of airborne radioactivity following postulated design basis accidents.

- The shield building is the reinforced concrete structure that surrounds the containment vessel. During normal operations, a primary function of the shield building is to provide shielding for the containment vessel and the radioactive systems and components located in the containment building. Another function of the shield building is to protect the containment building from external events. The shield building protects the containment vessel and the RCS from the effects of tornadoes and tornado produced missiles.

- The auxiliary building houses and protects Class 1 mechanical and electrical equipment located outside the containment building, including the MCR. The auxiliary building also provides an area for handling and storage of new and spent fuel, ion exchange columns and liquid radwaste system (WLS) components. The auxiliary building rail car bay is used to accommodate the ILW stabilisation equipment when ILW is being treated (see subsection 3.5.7.2).

The nuclear island structures are designed to withstand the effects of natural phenomena such as hurricanes, floods, tornados, tsunamis, and earthquakes without the loss of capability to perform safety functions. Further details are found in Chapter 12 of the PCSR (Reference 1-10).

2.3.2 Turbine Building

The turbine building is a steel column and beam structure. The turbine building houses the main turbine, generator, and associated fluid and electrical systems. It provides weather protection for the laydown and maintenance of major turbine/generator components. The turbine building also houses the makeup water purification system. No Class 1 equipment is located in the turbine building.

2.3.3 Annex Building

The annex building is a combination of reinforced concrete and steel-framed structure with insulated metal siding. The annex building provides the main personnel entrance to the power generation complex. It includes access ways for personnel and equipment to the clean areas of the nuclear island in the auxiliary building and to the radiological control area. The building includes the health physics facilities for the control of entry to and exit from the radiological control area as well as personnel support facilities such as locker rooms.

The annex building also contains the non-Class 1 electric power systems, the ancillary diesel generators and their fuel supply, other electrical equipment, the control support area, and various heating, ventilating, and air conditioning systems.

2.3.4 Diesel Generator Building

The diesel generator building is a single-story, steel-framed structure with insulated metal siding. It houses two identical diesel generators separated by a three hour fire wall. These generators provide backup power for plant operation in the event of disruption of normal power sources. No Class 1 equipment is located in the diesel generator building.

2.3.5 Radwaste Building

The radwaste building includes facilities for dealing with low level waste (LLW) produced during the operation of the **AP1000** NPP. The building is used for the sorting and conditioning or treatment of various categories of LLW prior to processing, and transfers to shipping and disposal containers (see subsection 3.5.7.1).

Six liquid waste monitor tanks are located within the radwaste building. These tanks contain processed effluents which are ready for release to the environment. The liquid radwaste processing areas are designed to contain any liquid spills. These provisions include a raised perimeter and floor drains that lead to the WLS waste hold-up tanks.

The radwaste building is used to store the mobile ILW waste stabilisation equipment when not in use (see Figure 3.5-9).

2.3.6 Radioactive Waste Stores

2.3.6.1 LLW Store

The LLW store is a non-seismic building with sufficient space to accommodate two years of LLW production. It is located within the boundary of the licensed site to the rear of the radwaste building.

2.3.6.2 ILW Store

The ILW store for the generic site is a reinforced concrete structure that can be extended at suitable intervals to suit new ILW arisings. Initially, the ILW store will be 33 m (110 feet (ft)) long, 13.5 m (44.3 ft) wide, and 14 m (46 ft) high (externally) and has walls 1 m thick. The ILW store facility incorporates a package receipt area and assay equipment and shielded vault serviced by a certified nuclear crane. Additionally, office and administration space is provided for real time record keeping, as well as an equipment room that houses heating, ventilation, and air conditioning (HVAC), and small electrical and mechanical equipment. Extension to the store will be sized to suit future waste arisings and are expected to be added in 20-year increments.

The ILW store is located within the confines of the licensed site in an area large enough to accommodate future extensions of the store. The location has been selected to minimise the transportation distances between the auxiliary building and the ILW store and to facilitate safe transfer of waste.

2.3.6.3 Dry Spent Fuel Store

The spent fuel store for the generic site is a seismically qualified facility and comprises spent fuel flasks, flask loading equipment within the **AP1000** NPP, a suitable transport vehicle, and below ground storage cells.

The spent fuel store is also located within the confines of the licensed site and retains the potential for future extension of the store. The location has been selected to minimise the transportation distances between the auxiliary building and the spent fuel store and to facilitate safe transfer of waste.

2.3.7 Other Buildings and Structures

Additional plant structures include warehouses, administration/office buildings, and the switchyard and transmission towers. At coastal sites, the circulating water systems (CWS) uses once through direct seawater cooling systems with appropriate seawater intake and discharge structures. The exact layout of these seawater cooling structures will be site specific.

The overall plant arrangement for an **AP1000** NPP is such that building configurations and structural designs minimise the building volumes and quantities of bulk materials (concrete, structural steel, rebar) consistent with safety, operational, maintenance, and structural needs to provide an aesthetically pleasing effect. Natural features of the site are preserved as much as possible and are utilised to reduce the station's impact on the environment. Landscaping for the site, areas adjacent to the structures and in the parking areas blend with the natural surroundings in order to reduce visual impacts.

2.4 Reactor Power Conversion System

The **AP1000** NPP reactor power conversion system comprises a single reactor pressure vessel, two steam generators, and four reactor coolant pumps for converting reactor thermal energy into steam. A single high-pressure turbine and three low pressure turbines drive a single electric generator.

A simplified diagram of the reactor power conversion system is shown in Figure 2.4-1.

2.4.1 Reactor

The **AP1000** NPP reactor contains 157 mechanically identical fuel assemblies. Each fuel assembly consists of 264 fuel rods in a 17 x 17 square array. There is substantial operating experience with this type of fuel assembly.

The fuel rods consist of **ZIRLO**® tubing containing cylindrical pellets of sintered uranium dioxide enriched in U-235. The **ZIRLO** tubing is plugged and seal-welded at the ends to encapsulate the fuel. An axial blanket comprised of fuel pellets with reduced enrichment may be placed at each end of the enriched fuel pellet stack to reduce the neutron leakage and to improve fuel utilisation.

The reactor core is cooled and moderated by light water at a pressure of 15.5 MPa (2250 psia). Soluble boron in the moderator/coolant serves as a neutron absorber. The concentration of boron is varied to control reactivity changes that occur relatively slowly, including the effects of fuel burn-up. Burnable absorbers are also employed in the initial cycle to limit the amount of soluble boron required, and thereby maintain the desired negative reactivity coefficients.

Some spaces of the 17x17 fuel rod array contain guide tubes in place of fuel. These guide tubes house instrumentation and accommodate either rod cluster control assemblies or gray rod cluster assemblies, both of which provide in-core reactivity control. Gray rods and control rods assist primarily in controlling core power distribution. Gray rods and control rods can also control reactivity to compensate for minor variations in moderator temperature and boron concentration during power operations. They can also assist in compensating for reactivity changes caused by power level and xenon changes during load following transients without the need for changing boron concentration.

Normally, the reactor will operate approximately 18 months between refuelling; accumulating a cycle burn-up of approximately 21,000 megawatt–days per metric ton of uranium metal (MWD/MTU). However, a maximum fuel rod average burn-up of 62,000 MWD/MTU has been established.

Refer to Chapter 6 of the PCSR for detailed information regarding the reactor design (Reference 1-10).

2.4.2 Steam and Power Conversion System

The design of the major components required for power generation such as the steam generators, reactor coolant pumps, fuel, internals, turbine, and generator is based on equipment that has successfully operated in power plants.

The steam and power conversion system is designed to remove heat energy from the RCS via the two steam generators and to convert it to electrical power in the turbine-generator.

The reactor is connected to two steam generators via two primary hot leg pipes and four primary cold leg pipes. A reactor coolant pump is located in each primary cold leg pipe to circulate pressurised reactor coolant through the reactor core. The coolant flows through the reactor core, making contact with the fuel rods containing the enriched uranium dioxide fuel. As the coolant passes through the core, heat from the nuclear fission process is transferred from the fuel rods to the coolant. The heat is transported to the steam generators by the circulating reactor coolant and passes through the steam generator tubes to heat the feedwater from the secondary system. Reactor coolant is pumped back to the reactor by the reactor coolant pumps, where it is reheated to start the heat transfer cycle over again. Inside the steam generators, the heat from the primary system is transferred through the tube walls to convert the incoming feedwater from the secondary system into steam. The steam is transported from the steam generators by the main steam piping to drive the high-pressure and low-pressure turbines connected to the electric generator. After passing through three low pressure turbines, the steam is condensed back to water by cooled water circulating inside the tubes of three main condensers. The heat rejected in the main condensers is removed by the CWS. The condensate is then preheated and pumped back to the steam generators as feedwater to repeat the steam cycle.

For detailed information regarding the steam and power conversion system refer to Chapter 6 of the PCSR (Reference 2-2).

2.4.3 Turbine Generator

The turbine generator system converts the thermal energy of the steam flowing through the turbine into rotational mechanical work, which rotates a generator to provide electrical power. The turbine-generator has an output of approximately 1,200 MW for the thermal output of the Westinghouse nuclear steam supply system of 3,415 MWt.

2.5 Engineered Safety Features

Engineered safety features protect the public in the event of an accidental release of radioactive fission products from the RCS. The engineered safety features function to localise, control, mitigate, and terminate such accidents and to maintain radiation exposure levels to the public below applicable limits and guidelines. A basic premise of the **AP1000** NPP design is to maintain safety and respond to accidents without reliance on ac power.

A detailed description of the safety features can be found in the PCSR (Reference 2-2).

2.6 Best Available Techniques (BAT) Applicable to AP1000 Reactor Design

Over the 15 years of design duration of the **AP1000** NPP and AP600, there were many design decisions that reinforced the concept of safety through simplicity, ALARP, and BAT (Reference 2-3). Examples of the decisions that relate to waste minimisation, waste generation, and waste disposal are identified below.

2.6.1 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, in previous designs, 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, and therefore, waste arisings that could result from these activities.

Penetrations have been minimised by implementation of a variety of innovative techniques:

- Service systems in containment, for example, 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 HVAC in containment share common penetrations since they won't be used at the same time. The fire protection water and containment spray supply systems also share a common penetration.

- 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.

The minimisation of containment penetrations reduces the risk for containment leakage and public or operator radiation exposure and minimises the potential of waste generation from this source.

2.6.2 Reactor Coolant Pump Selection

The function of the reactor coolant pump is to deliver adequate cooling water for power operations and for accident shutdown situations. The classic type of reactor coolant pump 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 and associated liquid effluent production. Alternatives considered included direct current (dc) powered safety pumps, canned motor pumps, wet winding pumps, no pumps (natural circulation), and others.

In the UK **AP1000** NPP, hermetically sealed wet winding pumps have been selected to eliminate the potential for reactor coolant leakage from shaft seals. This decision sacrifices the efficiency of shaft seal pumps for higher inherent reliability, safety, and simplicity of maintenance. The selection of wet winding pumps also eliminates the shaft seal pump support systems such as seal injection, seal leak off, lube oil, and fire protection systems (FPSs). The design and operation philosophy for the wet winding pumps is one of minimal maintenance. Unlike shaft seal pumps, wet winding pumps cannot be repaired in situ and require the entire pump (motor, hydraulics, flywheel, etc.) to be removed as a unit and replaced by a spare pump. This reduces the radioactive hazard and lowers the risk to the operators.

A basic premise of the **AP1000** NPP design is to maintain safety and respond to accidents without reliance on ac power. 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 acceptable 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 shutdown transient. The passive solution is the addition of rotating inertia to the wet winding 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 wet winding pump was chosen over the shaft seal pump for reactor coolant service because it meets the design requirements with the 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.

2.6.3 Load Follow with Rods

Most central station NPPs 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, many existing 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 use of boron and water handling systems are often associated with the production of radioactive effluent.

Load follow control in the **AP1000** NPP incorporates the proven, safe, and simple method of shim rods rather than the complex method of boron recycle. Shim control is the use of moveable control rods with a 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.

This solution 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.

2.6.4 Chemical and Volume Control System (CVS)

The functional requirements for the CVS are to fill, make up, let down, drain, and maintain the proper chemistry of reactor coolant water. This includes removal of impurities (both radioactive and non-radioactive) from the RCS. In many operating plants today, this function is performed by taking a portion of the reactor coolant into a variety of Class 1 subsystems that are outside containment. These systems reduce pressure and temperature of the reactor coolant, purify it, and pump it back into containment and the RCS with a high pressure pumping system. This process introduces potential reactor coolant leak sites outside containment with associated waste production, as well as imposing additional reactor coolant inventory control requirements.

The **AP1000** NPP design improvements (see Sections 2.6.2 and 2.6.3) have eliminated the requirement to continuously pump borated makeup water into the RCS or to include complicated water processing systems in the design. This has allowed simplifications to the CVS where coolant purification was developed to perform continuous purification of a portion of the reactor coolant at reactor coolant pressure (~200 bar(a)) using the reactor coolant pump head as a motive pressure and keeping all the purification equipment and reactor coolant within the containment vessel. The high pressure water purification uses ion exchange that is an industry proven process.

In the **AP1000** NPP, the basic design philosophy requires passive systems that eliminate the need for Class 1 coolant charging or letdown. The functions of reactor coolant makeup, boron injection, letdown, purification, and others are non-Class 1 making most of the system non-Class 1. Redundancies and potential Class 1 failure modes associated with these functions have been eliminated.

In summary, the CVS functional requirements were satisfied by simple designs using in-containment, high pressure coolant purification rather than out of containment, pumped, low pressure purification. This created a process that satisfies the design requirements with 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.

2.6.5 Use of Demineralisers for Treatment of Reactor Coolant System Let Down

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 the reactor or spent fuel pool coolant water. During plant heat up or cool down, boron concentration adjustments are made using a feed and bleed system where 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 has no planned leakage of reactor coolant from the pump shaft seal leak-off systems (see Section 2.6.2). In addition, the **AP1000** NPP has no plans to recycle dissolved boron in the reactor coolant for load follow changes (see Section 2.6.3). By these design decisions, the **AP1000** NPP's radioactive water sources are reduced with the main source coming from letdown during heat up.

Three options were considered for the treatment of the borated radioactive let down water:

1. Storage and Recycle

   The potential for storage and reuse during the next plant cool down was considered. However, it was dismissed for several reasons. 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. Small amounts of additional demineralised make up water are easily made between shutdowns to fulfil the cool down requirements. The approach is unnecessarily complicated and adds radiological hazard risks and additional containment and handling issues.

2. Evaporation

   Evaporators concentrate the radionuclides in liquid radwaste. Evaporators were dismissed because their operation is complicated, involve a number of fluid systems, use plant energy that could be used as net electrical output, and increase the potential for operator dose during maintenance.

3. Demineralisers

   Ion exchangers or demineralisers use disposable resin to capture radionuclides in a highly concentrated solid form.

Demineralisers were selected on the basis of 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. Section 3.4.5 addresses these issues further.

2.6.6 Zinc Addition

Zinc addition has been shown to have two benefits:

- It reduces the mobility of corrosion products, which in turn reduces occupational and environmental radiological doses, and

- It reduces the build up of boron and boron-lithium compounds in the core which has the potential to cause crud-induced power shift.

It may also reduce stress-corrosion cracking. To reduce these effects, the **AP1000** NPP incorporates a zinc addition subsystem as part of the CVS to produce and maintain a zinc oxide film on primary piping and components. This zinc addition has also been found to significantly reduce occupational radiation exposure when incorporated as early as hot functional testing.

Zinc concentrations ranging from 10 parts per billion (ppb) (+/- 5 ppb) are dosed into the RCS. Higher injection rates are typically used when zinc addition is first initiated in order to more quickly saturate RCS surfaces with zinc and achieve a residual zinc concentration in the coolant. Typical zinc uptake rates up to 95% occur in the first few months of zinc injection. On average these reduce to 20% - 90% of the injection rate over a given operational cycle. It is expected that up to around 5 kg (11 pounds) (lbs)) of zinc will be injected during the first zinc cycle to maintain a target concentration of 10 ppb, assuming a CVS purification flowrate of 380 l/min (100 gpm). This zinc usage will decrease over time, potentially to 2 - 3 kg (4.4 - 6.6 lbs) per cycle.

2.6.7 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 trade-off was to continue with this variety approach or to 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 diaphragm pumps. The use of similar pumps for common service has the advantage of requiring the minimum number of spares for storage and familiar maintenance requirements. It also satisfies its 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.

2.7 Plant Water Use

The **AP1000** NPP requires water for both plant cooling and operational uses. The plant water consumption and water treatment are determined from engineering evaluations based on the **AP1000** NPP design requirements and the water resources available at the specific site.

For the coastal generic site, it is assumed that the cooling water requirement will be met by seawater abstraction. Mains water will be supplied to meet other facility water demands during construction and operation.

A block diagram of the water supply and waste water system is shown in Figure 2.7-1. This diagram identifies the normal and maximum flow rates in the water system.

Refer to Chapter 6 of the PCSR for detailed information regarding the water systems (Reference 1-10).

2.7.1 Circulating Water System (CWS)

The CWS is a once through seawater cooling system that supplies seawater cooling water to remove heat from the main condensers, the turbine building closed cooling water system (TCS) heat exchangers, and the condenser vacuum pump seal water heat exchangers, under varying conditions of power plant loading and design weather conditions. Circulating water from the seawater intake basin is pumped by three 33 1/3 percent capacity vertical turbine pumps into the main condensers and heat exchangers and then returned to the seawater outfall basin. The **AP1000** NPP will not operate with a significantly reduced cooling water flow. If the water flow becomes too low, the plant will trip and enter the passive cooling mode. There is sufficient hold-up capacity for a plant trip.

The heat removed is rejected to the seawater cooling return basin (see subsection 4.2.3.3 and Reference 2-4). The potential impacts of the cooling water discharge are also discussed in Reference 2-4. The once through seawater cooling system will be dosed with sodium hypochlorite to control biofouling when seawater temperatures exceed 10°C (50°F) (Reference 2-4).

2.7.2 Service Water System (SWS)

The SWS supplies cooling water to remove heat from the non-Class 1 component cooling water system (CCS) heat exchangers in the turbine building. For the generic site, the SWS is also a once through seawater cooling system. However, the option to use a cooling tower is retained for particular site-specific requirements (see Section 7.2). Service water is pumped through strainers to the CCS heat exchangers for removal of heat.

The once through seawater cooling system will also be dosed with sodium hypochlorite to control biofouling when seawater temperatures exceed 10°C (50°F) (Reference 2-4).

2.7.3 Demineralised Water Treatment System (DTS)

The DTS receives water from the mains supply and processes this water to remove ionic impurities, and provides demineralised water to the demineralised water transfer and storage system (DWS).

The DTS consists of two 100 percent cartridge filters, two 100 percent reverse osmosis units normally operating in series for primary demineralisation, clean in place unit, sample panel unit, and two 100 percent electrodeionisation units for secondary demineralisation. The capacity of the DTS is sufficient to supply the plant makeup demand during startup, shutdown, and power operation.

Depending on the feedwater supply quality, a pH adjustment chemical may be added upstream of the cartridge filters to adjust the pH of the reverse osmosis influent. The pH is maintained within the operating range of the reverse osmosis membranes to inhibit scaling and corrosion. A dilute antiscalant (polyphosphate), which is chemically compatible with the pH adjustment chemical, may also be metered into the reverse osmosis influent water to increase the solubility of salts (that is, decrease scale formation on the membranes). The pH adjustment chemical, chlorine reducing agent, and antiscalant are injected into the demineralised water treatment process from the turbine island chemical feed system (CFS).

The DWS provides a reservoir of demineralised water to supply the condensate storage tank and for distribution throughout the plant. In addition to supplying water for makeup of systems that require pure water, the demineralised water is used to sluice spent radioactive resins to the solid radwaste system from the ion exchange vessels in the CVS, the spent fuel pool cooling system, and the WLS.

2.7.4 Potable Water System (PWS)

The PWS is designed to furnish water for domestic use and human consumption. Potable water is supplied from the mains supply.

2.7.5 Fire Protection System (FPS)

The FPS provides water to points throughout the plant where wet system-type fire suppression (e.g., sprinkler, deluge, etc.) may be required. The FPS is designed to supply fire suppression water at a flow rate and pressure sufficient to satisfy the demand of any automatic sprinkler system plus 114 m^3/h (500 U.S. gallons per minute (gpm)) for fire hoses for a minimum of 2 hours. Make-up water for the FPS is provided by the mains supply which feeds the fire protection storage tanks.

2.8 Transportation of Radioactive Fuel

2.8.1 New Fuel

Details of the fuel storage and handling systems can be found in Chapter 6.10 of the PCSR (Reference 1-10).

New fuel assemblies are transported to the site by truck, in accordance with ONR Transport regulations. The initial fuel loading consists of 157 fuel assemblies for one unit. Every 18 months, refuelling requires an average of 64 fuel assemblies for one unit. The fuel assemblies are fabricated at a fuel fabrication plant and shipped by truck to the site shortly before they are required. The details of the container designs, shipping procedures, and transportation routings depend on the requirements of the suppliers providing the fuel fabrication services. Truck shipments do not exceed the applicable gross vehicle weight restrictions.

The new fuel storage facility is located in the auxiliary building fuel handling area. Fuel is received in the rail car bay at the 100 m (328 ft) grade elevation. The fuel containers (travellers) are offloaded one container at a time by the rail car bay overhead crane. The traveller is positioned over an embedment at grade elevation, fastened, and manually rotated to the vertical position by the traveller upender. To assist with unloading, a man-lift is provided. The traveller is equipped with accelerometers that are installed to monitor any excessive acceleration during transport. The accelerometers are checked to ensure that they have not tripped prior to fuel assembly removal. Each traveller contains one fuel assembly which is removed from the traveller using new fuel handling tool and the fuel handling machine hoist. The fuel assembly is raised through the “Bay Door” at elevation 110.74 m (3911 ft) and transported to the new fuel storage rack. The new fuel is visually inspected while it is being inserted into the rack. No scaffolding is required to support the visual inspection process. Once the fuel is seated in the storage rack, the new fuel handling tool is disengaged and removed from the fuel assembly. The cell covers are reinstalled and the process is repeated for the remaining fuel assemblies.

The new fuel rack includes storage locations for 72 fuel assemblies with the maximum design basis enrichment. The rack layout provides a minimum separation between adjacent fuel assemblies which is sufficient to maintain a subcritical array even in the event the building is flooded with unborated water or fire extinguishant aerosols or during any design basis event. The racks include integral neutron-absorbing material to maintain the required degree of subcriticality. In the case of first-time fuelling, the spent fuel pool is also used for new fuel storage.

The rack rests on the floor of the reinforced concrete new fuel storage pit and is braced as required to the pit wall structures. The 5.2 m (17 ft) deep pit is dry and unlined. Materials used in rack construction are compatible with the storage pit environment, and surfaces that come into contact with the fuel assemblies are made of annealed austenitic stainless steel. Structural materials are corrosion resistant and will not contaminate the fuel assemblies or pit environment. Neutron absorbing “poison” material used in the rack design has been qualified for the storage environment. Venting of the neutron absorbing material is considered in the detailed design of the storage rack. The new fuel storage pit is drained by gravity drains that are part of the radwaste drain system, draining to the waste holdup tanks which are a part of the WLS. These drains preclude flooding of the pit by an accidental release of water. The new fuel pit is covered to prevent foreign objects from entering the new fuel storage rack.

The new fuel handling crane is used to load new fuel assemblies into the new fuel rack and transfer new fuel assemblies from the new fuel pit into the spent fuel pool. A gated opening connects the spent fuel pool and fuel transfer canal. The fuel transfer canal is connected to the in-containment refueling cavity by a fuel transfer tube.

A new fuel elevator in the spent fuel pool lowers the new fuel to an elevation accessible by the fuel handling machine (FHM). The FHM is part of the fuel transfer system. The fuel transfer system is used to move up to two fuel assemblies at a time between the auxiliary building fuel handling area and the refuelling cavity in the containment building. The FHM performs fuel handling operations in the fuel handling area. Fuel is placed into a basket of the underwater transfer car for passage through the fuel transfer tube and into the refuelling cavity. The refueling machine performs fuel handling operations in the containment building. Fuel is moved between the fuel transfer system and the reactor vessel by the refueling machine. It withdraws the fuel from the refueling cavity, moves over the core area, and inserts the fuel assembly into a vacant core location. During refuelling, the vacant core location is created by prior removal of a spent fuel assembly.

2.8.2 Spent Fuel

Spent fuel assemblies are discharged from the reactor every refuelling outage and are placed into the spent fuel pool. The spent fuel storage pool has the capacity to store approximately 617 fuel assemblies. Each refuelling offload is 68 fuel assemblies. Therefore, the spent fuel storage pool has the capacity for six refuelling offloads, which represents approximately 10 years, plus a full core offload.

The spent fuel is transferred from containment to the spent fuel pool by the fuel transfer system described in Section 2.8.1. The fuel handling equipment is designed to handle the spent fuel assemblies underwater from the time they leave the reactor vessel until they are placed in a container for shipment from the site.

The spent fuel pool provides storage space for spent fuel. The pool is approximately 42 feet (13 m) deep and constructed of reinforced concrete and concrete filled structural modules. The portion of the structural modules in contact with the water in the pool is stainless steel and the reinforced concrete portions are lined with a stainless steel plate. The normal water volume of the pool is about 191,500 U.S. gallons (725 m^3) of borated water with a nominal boron concentration of 2700 ppm. A spent fuel pool cooling system is provided to remove decay heat which is generated by stored fuel assemblies from the water in the spent fuel pool.

Spent fuel is stored in racks which include integral neutron absorbing material to maintain the required degree of subcriticality. The racks are designed to store fuel of the maximum design basis enrichment. The design of the racks is such that a fuel assembly cannot be inserted into a location other than a location designed to receive an assembly. An assembly cannot be inserted into a full location.

The spent fuel assemblies can normally be stored in the spent fuel pool for approximately 10 years, until fission product activity is low enough and cooling is sufficient to permit transfer to the HLW dry storage cask. Westinghouse has proposed the option of using the Holtec system as the dry storage casks of choice (see subsections 3.5.7.3 and 3.5.8.3).

The spent fuel assemblies are then transferred to the dry storage canister that is then placed in an underground storage cask which is designed to shield radiation. The process of loading spent fuel is carried out in the following steps:

- A clean, empty canister is brought into the cask washdown pit by the cask handling crane and washed with demineralised water. The cask lid is removed and stored while the remainder of the cask is washed.

- The clean, empty cask is then properly positioned in the flooded cask loading pit.

- The FHM is positioned over the specific fuel assembly to be shipped out of the spent fuel storage rack. The fuel assembly is picked up and transported into the cask loading pit. During the transfer process, the fuel assembly is always maintained with the top of the active fuel at least 2.9 m (9.5 ft) below the water surface. This provides confidence that the direct radiation from the fuel at the surface of the water is minimal.

- Once the fuel transfer process is complete, the lid is placed on top of the canister.

- The canister is then moved to the washdown pit and cleaned with demineralised water. Decontamination procedures can be started at this time.

- When the canister is satisfactorily decontaminated, it is lifted out of the cask washdown pit by the cask handling crane and prepared for transfer to the HLW store (see subsection 3.5.8.3). During the operations, sufficient water is maintained between plant personnel and fuel assemblies that are being moved to limit dose levels to those acceptable for continuous occupational exposure.

2.9 Radioactive and Non-Radioactive Materials

This section identifies the radioactive materials and non-radioactive chemicals that are stored on an **AP1000** NPP site. The primary storage and secondary containment systems that prevent releases to the environment are described.

2.9.1 Inventory of Radioactive Materials and Radioactively Contaminated Chemicals

2.9.1.1 Fuel Rods

The fuel rods consist of uranium dioxide ceramic pellets contained in cold-worked and stress-relieved **ZIRLO** tubing. The fuel rods include integral fuel burnable absorbers, for example boride-coated fuel pellets.

The reactor contains 41,448 fuel rods in 157 fuel assemblies. The total fuel weight is 95,975kg (211,588 lb) of uranium dioxide.

2.9.1.2 In-Core Control Components

In addition to the burnable absorbers, reactivity control is provided by neutron-absorbing rods and gray rods.

2.9.1.3 Reactor Coolant

The reactor coolant liquid volume at power conditions is 9600 ft^3 (272 m^3) which includes (1000 ft^3) 28 m^3 pressuriser liquid.

The CVS provides a means for adding chemicals to the RCS. The reactor coolant contains the following chemicals which become contaminated with radioisotopes by passage through the reactor:

- Boric Acid

  The RCS contains demineralised and borated water that is circulated at the flow rate and temperature consistent with achieving the reactor core thermal and hydraulic performance. The soluble boron in the form of boric acid is added to the reactor coolant to serve as a neutron absorber/moderator. The concentration of boron is varied to control reactivity changes that occur relatively slowly, including the effects of fuel burn-up (chemical shim control). Boron concentrations in the reactor coolant typically range between 612 ppm to 2700 ppm.

- Lithium 7 Hydroxide

  The pH control chemical is lithium hydroxide monohydrate, enriched in the lithium-7 isotope to 99.9 percent. 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. The lithium-7 hydroxide is introduced into the RCS via the charging flow. The concentration of lithium-7 hydroxide in the RCS is maintained in the range specified for pH control. The concentration of lithium-7 in the RCS is varied between 0 ppm and 3.5 ppm as a function of the boric acid concentration of the RCS.

- Hydrazine

  During reactor start up from the cold condition, hydrazine is used as an oxygen-scavenging agent. The hydrazine solution is introduced into the RCS at a stoichiometric amount based on the measured oxygen concentration, typically 1.5 times [O_2] to reduce oxygen to less than 0.1 ppm in the reactor coolant. Once above 200°F (93.3°C), the oxygen concentration is maintained below 0.005 ppm by the addition of hydrogen.

- Zinc Acetate

  Addition of soluble zinc acetate to the reactor coolant leads to the incorporation of zinc into oxide films on wetted reactor component surfaces, steam generator tubing and RCS piping. This reduces ongoing corrosion of austenitic stainless steel and nickel-based alloys. Zinc acetate is injected into the RCS to maintain a maximum zinc concentration of 10 ppb (+/- 5 ppb).

2.9.1.4 Borated Water

Boric acid is added to water in the following systems (in addition to the reactor coolant):

- Spent Fuel Pool/Fuel Transfer Canal

  The spent fuel pool provides storage space for spent fuel and the fuel transfer canal provides an underwater passage for the transfer of spent fuel. The normal water volume of the pool is about 725m^3 (191,500 U.S. gallons) and the fuel transfer canal is 243 m^3 (64,100 U.S. gallons). Both the spent fuel pool and transfer canal are kept full of borated water with a nominal boron concentration of 2700 ppm. Demineralised water can be added for makeup purposes, including replacement of evaporative losses, from the DWS. Boron may be added to the spent fuel pool from the CVS.

- In-Containment Refueling Water Storage Tank (IRWST)/Refueling Cavity

  The volume of the IRWST is 2141 m^3 (565,500 U.S. gallons) and it contains a nominal boron concentration of 2700 ppm.

  The borated water is transferred to the refuelling cavity prior to a refuelling and then back to the IRWST by the spent fuel pool cooling system. The volume of borated water required to flood the refuelling cavity is (1325 m^3) 350,000 U.S. gallons).

- Cask Washdown Pit

  The cask washdown pit is flooded with borated water to provide shielding before spent fuel assemblies are transferred to a shipping cask. The minimum volume of the cask washdown pit is 148 m^3 (39000 U.S. gallons). The nominal boron concentration is 2700 ppm.

2.9.1.5 Ion Exchange and Absorption Media

- Chemical Volume Control System (CVS) Demineralisers

  The CVS demineralisers maintain RCS fluid purity and activity level within acceptable limits. The purification loop operates at RCS pressure. The purification fluid flows through a mixed bed demineraliser, optionally through a cation bed demineraliser, and through a filter. It returns to the suction of a reactor coolant pump after being heated in the regenerative heat exchanger.

  Two stainless steel vessels contain mixed ion exchange resin beds in the Li(-7)OH form. Each bed contains approximately 50 ft^3 (1.4 m^3) of resin. A third bed of similar size contains a cationic resin in the H^+ form.

- Spent Fuel Pool Demineralisers

  Two mixed bed type demineralisers are provided to maintain spent fuel pool purity. The demineralisers are initially charged with a hydrogen type cation resin and hydroxyl type anion resin to remove fission and corrosion products. The demineralisers will be borated during initial operation with boric acid. Each demineraliser is sized to accept the maximum purification flow from its respective cooling train. The ion exchange resin are held in two stainless steel vessels that hold approximately 75 ft^3 (2.1 m^3) of resin each.

- Liquid Radwaste System

  The WLS provides treatment of radioactive effluent. The WLS has four stainless steel vessels containing ion exchange media in the treatment train. The first vessel contains 50 ft^3 (1.4 m^3) of layered activated charcoal above the zeolite resin. The next vessel contains 30 ft^3 (0.85 m^3) of cationic resin. Each of the final two vessels contain 30 ft^3 (0.85 m^3) of mixed resin.

- Gaseous Radwaste System

  The gaseous radwaste system (WGS) provides treatment of radioactive air emission. The WGS activated carbon guard bed removes residual moisture as well as iodine from the gas stream. The guard bed is a stainless steel vertical pipe with a nominal volume of 8 ft^3 (0.23 m^3). The main part of the WGS is two activated carbon delay beds. These are in series and each comprise a carbon steel vertical serpentine tube containing of 80 ft^3 (3 m^3) of activated carbon absorption media.

- Stabilised ILW

  When the ion exchange and the absorption media described above nears exhaustion, they are removed as ILW for treatment, stabilisation, and storage. Details can be found in Section 3.5.7.2.

2.9.2 Non-Radioactive Chemical Inventory

Chemicals are stored at the **AP1000** NPP on the turbine island, the nuclear island, and in the yard when a seawater cooling system is used. The inventory and chemical content of each tank on the turbine, nuclear island, and seawater cooling system are summarised in Tables 2.9-1, 2.9-2, and 2.9-3, respectively.

2.9.2.1 Control of Major Accident Hazards

An evaluation of the applicability of the Control of Major Accident Hazards (COMAH) regulations has been carried out on the storage quantities of chemicals and fuel oil in Tables 2.9-1, 2.9-2, and 2.9-3 (Reference 2-5).

Based on the current specification, the **AP1000** NPP site will be an upper tier COMAH site. This is because the proposed hydrazine inventory in Tank MT-01 is 3.1 tonnes (3.4 tons) (see Table 2.9-1) which exceeds the upper tier COMAH threshold of 2 tonnes (2.2 tons) (Reference 2-5).

Other chemicals listed in Tables 2.9-1, 2.9-2, and 2.9-3 do not fall under COMAH because they do not have the defined hazardous properties or they are not stored in volumes that exceed the COMAH threshold quantities.

2.9.2.2 Substances Under the Groundwater Directive

The existing Groundwater Directive (2006/118/EC, Reference 2-6) aims to protect groundwater from pollution by controlling discharges and disposals of certain dangerous substances to groundwater. In the UK, the directive is implemented through the Groundwater Regulations 2009. Groundwater is protected under these regulations by preventing or limiting the inputs of listed substances into groundwater. Substances controlled under these regulations fall into two lists:

- List I substances are the most toxic and must be prevented from entering groundwater. Substances in this list may be disposed of to the ground, under a permit, but must not reach groundwater. They include pesticides, sheep dip, solvents, hydrocarbons, mercury, cadmium, and cyanide.

- List II substances are less dangerous, and can be discharged to groundwater under a permit, but must not cause pollution. Examples include sewage, trade effluent, and most wastes. Substances in this list include some heavy metals and ammonia (which is present in sewage effluent), phosphorus, and its compounds.

Some of the chemicals used on the **AP1000** NPP fall within the List I and List II substances under the Groundwater Directive. These are identified in Table 2.9-4.

2.9.2.3 Substances Under the Dangerous Substances Directive

The Dangerous Substances Directive (76/464/EEC) and its "daughter" directives (Reference 2-7) control discharges that are liable to contain dangerous substances and that go to inland, coastal, and territorial surface waters. Dangerous substances are toxic substances that pose the greatest threat to the environment and human health. The directive specifies two lists of Dangerous Substances. List I covers those which are particularly toxic, persistent, and which may tend to accumulate in the environment. List II covers substances whose effects are still toxic, but less serious. The directive requires that pollution by List I substances is eliminated and pollution by List II substances is minimised. All discharges that are liable to contain dangerous substances must be authorised. The directive also specifies some requirements for environmental monitoring.

Some of the chemicals used on the **AP1000** NPP fall within the List I and List II substances under the Dangerous Substances Directive. These are identified in Table 2.9-5. The list includes halogenated by-products of seawater chlorination, which is discussed further in Section 4.2.5.

2.9.3 Chemical Storage

The inventory of chemicals stored on-site is presented in Table 2.9-6.

2.9.3.1 Turbine Island

The chemical inventory on the turbine island is stored in the northeast corner of the turbine building at ground level (reference elevation 100' 0") in an area reserved for chemical storage (see Figure 2.9-1). There are six tanks in the southern group (left side) and one spare location for a future addition, if needed. The five tanks on the north side (right side) of the chemical storage area are separated into groups of two and three.

The storage and handling equipment arrangement for each type of chemical is almost identical. Figure 2.9-2 presents an elevation view of an arrangement of the southernmost group of storage tanks. Each chemical is stored in a tank assembly consisting of an upper removable tote container and a permanent lower stainless steel covered rectangular tank. The movement of chemical tote containers to the storage area is via fork truck. The upper tote container and its associated lower tank are each nominally (1.514 m^3) 400 U.S. gallons) in capacity. The total capacity of each tank assembly is (3.028 m^3 (800 U.S. gallons).

A filled tote container is connected to the lower permanent tank using a quick connect on a flexible hose. An isolation valve is opened to permit the draining of the upper tote into the lower tank. Vents at the tops of each tank allow air to fill in behind the draining liquid and flow out when displaced by the filling liquid. Vented air is directed to a pipe vent leading to the outside of the turbine building at a high elevation. Each vent line can be individually isolated. The lower tank has a level transmitter with a local display to permit the operator to observe the level inside the tank. When the lower tank is full, the isolation valve between tanks may be closed, holding any excess in the upper tote container until needed later. However, it is not necessary to isolate the two tanks until it is time to remove an empty tote container for refill. The piping, valves, and hoses are suitable for more than 1 MPa (150 psig) internal pressure, and under normal operation only need to contain a few centimetres (inches) of solution head pressure (< 14 kPa (2 psig)).

The chemical handling procedures require closing the isolation valve prior to disconnecting the quick connects. The preferred quick connect type is self-closing. Upon disconnection, the ends automatically seal and retain the contents. Inadvertent mispositioning of valves does not result in a spill upon disconnection. The storage arrangement and scheme for liquid transfer between tanks eliminates the potential release of chemicals within the turbine building.

Figure 2.9-3 is a photograph of a typical rugged tote container certified for transportation of hazardous materials. Operating plants have typically chosen stainless steel tote containers similar to the one shown. The exact container used for transporting hydrazine to an **AP1000** NPP will be a site-specific choice by the owner/operator of the plant.

2.9.3.2 Standby Diesel Fuel Oil

The Standby Diesel and Auxiliary Boiler Fuel Oil System (DOS) fuel storage tanks are each 227 m^3 (60,000 U.S. gallons) and are located away from the main plant buildings (see Figure 2.3-2, Item No. 18).

There are two smaller diesel generator day tanks located in the Diesel Generator Building. These two tanks each have a volume of 7.6 m^3 (2000 U.S. gallons).

There is also an ancillary diesel generator fuel oil tank 2.46 m^3 (650 U.S. gallons) located in the southeast corner of the Annex Building.

2.9.3.3 Central Chilled Water System (VWS)

There is low capacity chemical storage associated with the VWS. The chemicals include sodium molybdate/tolytriazole.

The volume of the sodium molybdate/tolytriazole storage is 0.076 m^3 (20 U.S. gallons).

2.9.3.4 Chemical Volume Control System (CVS)

A zinc injection package is included in the CVS design to allow for continuous addition of zinc acetate solution into the RCS. The zinc acetate is kept in the stainless steel zinc addition tank which has a volume of 0.76 m^3 (200 U.S. gallons).

2.9.3.5 Reactor Coolant Supply

The stainless steel boric acid tank holds 303 m^3 (80,000 U.S. gallons) of boric acid at a maximum concentration of 4375 mg/l (0.0365 lb/gallon). The tank is located outside the Annex Building (see Item in 20 Figure 2.3-2).

The boric acid storage tank capacity is sized to permit one shutdown to cold shutdown, followed by a shutdown for refueling, at the most limiting time in core cycle with the most reactive control rod withdrawn. This assures that the tank size is large enough to not interfere with normal plant operations. The concentration of boric acid is selected to eliminate the need to provide heat tracing for the purpose of preventing boric acid precipitation. Only normal freeze protection is required to maintain solubility of the 2.5 weight percent boric acid. This freeze protection is provided by an immersion heater in the tank, which maintains a minimum temperature of 7.2°C (45°F) to ensure boric acid solubility.

A Boric Acid Batching Tank with a volume of 3.03 m^3 (800 U.S. gallons) is used to supply the Boric Acid Tank. This tank is located in the Annex Building.

Lithium-7 hydroxide is injected on a manual, as needed basis through the CVS chemical mixing tank. The amount of lithium-7 hydroxide stored on site in a warehouse will be decided by the utility.

2.9.3.6 Fire Protection System

The diesel fuel supply for the diesel-driven fire pump is in the diesel-driven fire pump enclosure which is located in the yard more than 15 m (50 ft) from Class 1 structures. The diesel is supplied from a 0.91 m^3 (240 U.S. gallon) diesel tank.

2.9.3.7 Seawater Cooling System

The chemical dosing system required for the seawater cooling supply is a site-specific design. Three (37.9 m^3) 10000 U.S. gallon) chemical tanks are located in the yard next to the CWS. These tanks contain sodium hypochlorite, ammonium hydroxide and polyacrylate/ polyphosphate/orthopolyphosphate.

2.9.4 Prevention of Contamination - Chemical Storage Systems

The secondary containment systems provided for the **AP1000** NPP chemical storage tanks are shown in Table 2.9-6.

2.9.4.1 Turbine Island

All of the CFS storage tanks are in the Turbine Building Chemical Storage Tank Dike Area (see Figure 2.9-1). The concrete curb around the chemical storage area is 11.2 m (36.7 ft) long x 6.9 m (23 ft) wide x 0.2 m (0.7 ft) high giving a retention volume of 15.5 m^3 (4090 gallons). This volume is more than the volume of the largest tank stored within the bund (3.03 m^3 800 gallons) and more than 25% of the total volume of the chemicals (~22 m^3 (5800 gallons)) stored within the bund (see Table 2.9-6). The bund complies with the UK guidance that it should have a minimum capacity of either 110 percent of the capacity of the largest tank, or 25 percent of the total capacity of all the tanks within the bund, whichever is greater (Reference 2-8).

The chemical storage area has a plugged floor drain. The Waste Water System (WWS) catch basin serving this area is covered so that spills can be contained. When washing down the area, the cover can be removed to allow water to enter the catch basin. The catch basin then feeds into the Turbine Building sumps. The Turbine Building sumps are concrete lined with a fibre-reinforced epoxy coating. The two Turbine Building sumps provide effective tertiary containment, each having a volume of (92 m^3) 25,675 U.S. gallons).

2.9.4.2 Standby Diesel Fuel Oil

Each DOS fuel storage tank is surrounded by a concrete dike which is sized to hold 110 percent of the tank capacity (i.e., 252 m^3 (66,000 U.S. gallons)). The secondary containment complies with the requirements of the Control of Pollution (Oil Storage) (England) Regulations 2001. There is a drain in each of the dikes which drains them to the Diesel Fuel Oil Area Sump (MT04) which has a capacity of (7.6 m^3 (2000 U.S. gallons). The manual ball valves from the dike drain sump both have to be normally “closed.” The operator has to check the liquid level inside the dike regularly or after the storm event. Then, the operator has to decide which valve to be opened. If there is no oil leak, the clean storm water can directly drain to the clean water sewer. If oil is presented, the valve discharging to the Diesel Fuel Oil Area Oil Sump has to be opened to allow controlled collection of the oil.

The Diesel Generator Building Sump (2.2 m^3 (580 U.S. gallons)) and the Annex Building Sump (9.5 m^3 (2500 U.S. gallons)) provide the secondary containment for the small diesel oil day tanks for the diesel generator and ancillary diesel generator, respectively. This will be reviewed during site-specific analysis and designed as necessary to ensure that the secondary containment will comply with UK guidance (Reference 2-8).

2.9.4.3 Central Chilled Water System

Secondary containment for the small volume chemical storage located in the Turbine Building is provided by the Turbine Building Sump. This is a concrete-lined, fibre reinforced, epoxy coated sump with a volume of 194.4 m^3 (51,350 U.S. gallons).

2.9.4.4 Chemical Volume Control System

The Turbine Building Sump also provides the secondary containment for the zinc acetate stored in the zinc addition tank.

2.9.4.5 Reactor Coolant Supply

There is currently no secondary containment designed for the Boric Acid Storage Tank. This will be reviewed during site-specific analysis and designed as necessary to ensure that the secondary containment will comply with the UK guidance (Reference 2-8).

The secondary containment for the Boric Acid Batching Tank and Chemical Mixing Tank is provided by the Auxiliary Building Sump. This is a concrete-lined, fibre reinforced, epoxy coated sump with a volume of 9.5 m^3 (2500 U.S. gallons).

2.9.4.6 Fire Protection System

The diesel tank feeding the diesel driven fire pump is located within a concrete dike in the yard. This will be reviewed during site-specific analysis and designed as necessary to ensure that the secondary containment will comply with UK guidance (Reference 2-8).

2.9.4.7 Seawater Cooling System

The three 37.9 m3 (10000 U.S. gallon) tanks used to contain sodium hypochlorite, ammonium hydroxide and polyacrylate/polyphosphate/orthopolyphosphate currently do not have a secondary containment design. This will be reviewed during site-specific analysis and designed as necessary to ensure that the secondary containment will comply with UK guidance (Reference 2-8).

2.9.5 Prevention of Contamination – Radioactive Systems

In the **AP1000** NPP, radioactive contamination from the facility is minimised by using structure, system, and component designs and operational procedures that limit leakage and/or control the spread of contamination. Westinghouse prepared a document (Reference 2-9) to demonstrate that these practices address the U.S. NRC Regulatory Guide 4.21 (Reference 2-10) which is considered good practice. Reference 2-12 provides evidence that the design of the **AP1000** plant fuel handling provides capabilities of detecting a long term leak from the pools, pits, and canals within the fuel handling area.

2.9.5.1 Construction Features

The nuclear island basemat, extending beneath the containment and the auxiliary building, is made using techniques which result in a monolithic basemat, without expansion joints. The walls are built as a single monolithic structure with the basemat, without expansion joints or other building joints. The concrete used for building construction is thick and extensively waterproofed. This precludes leakage from the radioactive equipment located in this building to the environment.

The modular construction technique used results in a large number of walls in the radiologically controlled areas which are comprised of concrete contained within permanent steel forms. This left-in-place steel will be coated with paints and sealants to minimise the potential for contamination to penetrate. This not only reduces waste and background radiation during operation, but also greatly reduces decommissioning wastes, since the need for concrete scarification will be minimised.

The number of passageways (doors) between the radiologically controlled area and the environment has been minimised. Where such doors are incorporated, systems of drains and floor and exterior concrete sloping are used to prevent (potentially radioactive) fluid from the interior of the buildings from exiting the buildings, and also to prevent surface water from entering the buildings.

Decontamination of potentially contaminated areas and equipment within the plant is facilitated by the application of epoxy paints and suitable smooth-surface coatings to the concrete floors and walls. Sloping floors with floor drains are provided in potentially contaminated areas of the plant. In addition, radioactive and potentially radioactive drains are separated from non-radioactive drains.

2.9.5.2 Radioactive Liquid Tanks

Radioactive liquids in an **AP1000** NPP are contained within the reactor vessel, steam generators, the RCS, the WLS, and connected auxiliary systems.

All radioactive tanks are located inside appropriately designed buildings within the radiologically controlled area. Therefore, any leakage from these tanks would accumulate in the radiogically controlled area of the plant, where adequate provisions have been made through floor drains and sealed surfaces to prevent the spread of contamination. In particular, incursion of radioactive fluid into the groundwater due to a long-term leak in one of these tanks is completely precluded.

No underground radioactive tanks, other than tanks located in buildings, are used in the design. No flat-bottomed radioactive tanks are used in the design.

Radioactive tanks are equipped with high level alarms to alert the operators before a tank overflows. Overflow lines are piped to optimum collection points (generally to another waste collection tank or the radioactive waste drain system).

2.9.5.3 Piping

The simplicity of the **AP1000** NPP design reduces lengths of contaminated piping and the number of nuclear systems pumps and valves, thereby reducing the potential for leaks.

To the extent possible, all radioactive piping is located inside the auxiliary building and the containment vessel. This minimises the potential for leakage to the groundwater from piping or fittings. The few exceptions are:

- Process piping to and from the radwaste building:

  This piping can be fully, visually inspected from the radwaste building pipe trench to the auxiliary building wall; only the short portion of the pipe embedded in the auxiliary building wall is not visible.

- Drain lines from the radwaste building and annex building routed back to the auxiliary building:

  This piping is not normally water-filled. These lines can be fully, visually inspected from the radwaste building pipe trench to the auxiliary building wall; only the short portion of the pipe embedded in the auxiliary building wall is not visible.

- The monitored radwaste discharge pipeline:

  The monitored radwaste discharge pipeline is engineered to preclude leakage to the environment. This pipe is routed from the auxiliary building to the radwaste building (the short section of pipe between the two buildings will be fully available for visual inspection as noted above) and then out of the radwaste building to the licensed release point for dilution and discharge. The discharge radiation monitor and isolation valve are located inside the radiologically controlled area. The exterior piping is designed to preclude inadvertent or unidentified releases to the environment; it is either enclosed within a guard pipe and monitored for leakage, or is accessible for visual inspection. No valves or vacuum breakers are incorporated outside of monitored structures.

The use of embedded pipes has been minimised consistent with maintaining radiation doses ALARP. To the extent possible, pipes have been routed in accessible areas such as dedicated pipe routing tunnels or pipe trenches, which will provide good conditions for decommissioning.

2.9.5.4 Floor Drains and Sumps

All floor drains in radioactive areas are grouted into the surrounding concrete to ensure that any leakage will be collected in the floor drain, and not bypass the drain.

Sumps which may contain radioactivity (the containment sump and the radioactive auxiliary building sump) are constructed from stainless steel and are fully surrounded by concrete to ensure that any leakage will be collected in the sump and that no bypass paths exist.

Sumps are covered to keep out debris. Covers are removable, or manholes are provided for access. The total capacity of each sump includes a 10 percent freeboard allowance to permit operation of high-high level alarms and associated controls before the overflow point is reached.

2.9.5.5 Spent Fuel Pool

The spent fuel pool and connected pools that contain borated water are designed with redundant means of protection and detection to eliminate unidentified leakage to the groundwater (Reference 2-9):

- The walls of these pools will be constructed using modular construction techniques, allowing higher quality than typical “in the hole” construction.
- The pools have liners to prevent borated water from corroding the concrete or structural steel behind the liner.
- Leaks in spent fuel pools are primarily associated with welds. In the **AP1000** NPP spent fuel pool design, the number of welds has been significantly reduced.
- The advanced welding techniques which will be employed will minimise the potential for weld failures during operation, and allow for inspection to verify weld quality.
- The pools walls are made of a 0.5-inch (1.3 cm) stainless steel plate joined to one another with full penetration welds; these welds will be fully inspected prior to being placed into service.
- The thickness of the wall plate and the use of full penetration welds ensure that the walls will not be damaged by fuel handling, including tool manipulation and storage.
- All welds in the liners are equipped with leak chases, which means that the tank is effectively double-walled in the area of plate joint. The leak chases provide for evidence of leakage and direct any contaminated leakage flow to the waste handling systems (see subsection 2.9.5.6). They also prevent leaching of active fluid into concrete, if a leak occurs.
- To the extent possible, these pools are located entirely inside the auxiliary and containment building, so that any theoretical leakage from the tanks would accumulate in the building rather than to the environment. Specifically, for pools other than a portion of the fuel transfer canal, the concrete support structure of the pools may be inspected from rooms adjacent to or below (i.e. outside) the pool.

2.9.5.6 Leak Chase Subsystem

Leak chases are provided for pools inside containment and in the auxiliary building, which are filled with borated water. The leak chases are provided to prevent borated water from getting behind the various pool liner plates and potentially corroding the structural elements behind the pool liners.

The leak chase subsystems consist of collection channels (which are part of the Concrete Filled-in-Place Form Modules (CA structural modules)) surrounding pool welds, collection piping, headers, collection pots, and the associated valves and instrumentation. The leak chase subsystems are zoned to allow identification of the source of leakage. Headers are provided for each of the zones with isolation valves to support leakage testing and localisation of the leak. Pots with level instruments are provided to collect, detect, and quantify any leakage.

This leak detection system will use piping which is adequately sized to allow testing and to minimise the potential for blockage by encrustation of precipitates (boric acid), and will facilitate removal of any such blockage.

The leak chase subsystems are part of the WLS and the Radioactive Waste Drain System (WRS). The leak chases for the pools inside containment are part of the WLS. These leak chases would capture potential leakage from the fuel transfer tube, refuelling cavity, and the IRWST. The leak chases for the pools outside containment are part of the WRS. These leak chases would capture potential leakage from the fuel transfer canal, spent fuel pool, cask loading pit, and the cask washdown pit.

2.10 References

2-1 APP-GW-GER-005, Rev. 1, "Safe and Simple: The Genesis and Process of the **AP1000** Design," Westinghouse Electric Company LLC, August 2008.

2-2 UKP-GW-GL-793, Rev. 1, "**AP1000** Pre-Construction Safety Report," Westinghouse Electric Company LLC, January 2017.

2-3 UKP-GW-GL-026, Rev. 2, "**AP1000** Nuclear Power Plant BAT Assessment," Westinghouse Electric Company LLC, March 2011.

2-4 UKP-GW-GL-034, Rev. 1, "Generic Assessment of the Impacts of Cooling Options for the Candidate Nuclear Power Plant **AP1000**," Westinghouse Electric Company LLC, February 2010.

2-5 UKP-GW-GL-037, Rev. 3, "Applicability of COMAH Regulations to **AP1000**," Westinghouse Electric Company LLC, March 2017.

2-6 "Directive 2006/118/EC of the European Parliament and of the Council of 12 December 2006 on the Protection of Groundwater Against Pollution and Deterioration (Daughter to 2000/60/EC)", European Commission, 2007.

2-7 "Directive 76/464/EEC – Water pollution by discharges of certain dangerous substances," European Commission, 2006/11/EC.

2-8 Health and Safety Executive, *The Control of Major Accident Hazards Regulations 2015*, L111, Third edition, 2015," HSE Books, 2015.

2-9 APP-GW-GLN-098, Rev. 0, "Compliance with 10CFR20.1406," Westinghouse Electric Company LLC, April 2007.

2-10 Regulatory Guide 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning," U.S. Nuclear Regulatory Commission, June 2008.

2-11 Not used.

2-12 UKP-GW-GL-799, Rev. 2, "**AP1000**® Plant ALARP Assessment of Structural Impact from Fuel Handling Area Pools Leakage", Westinghouse Electric Company LLC, August 2016.

Table 2.9-1

CHEMICAL INVENTORY ON AP1000 NPP TURBINE ISLAND

Tank	Chemical[1]	Volume		Concentration (%w/v)	Mass Chemical (tonnes)*	Affected System
		U.S. gallons	m^3			
Turbine Island Chemical Feed System (CFS)						
MT01	Hydrazine	800	3.028	35%	3.1	ASS, FWS, CDS, BDS
MT02	pH control chemical – monoethanolamine (actual pH control additive used may differ)	800	3.028	40%	3.08	FWS, CDS, BDS
MT03	Ammonium Hydroxide	800	3.028	30%	3.0	ASS, FWS, CDS, BDS
MT04	Ammonium Hydroxide	800	3.028	30%	3.0	ASS, FWS, CDS, BDS
MT05	Sodium Sulfite and Sodium Hydroxide	240	0.9	30%	1.0	ASS
MT07	Polyphosphate	800	3.028	<100%	<3.0	DTS
MT08	Ammonium Chloride	800	3.028	25%	3.2	SWS
MT11	Ammonium Hydroxide	800	3.028	30%	3.0	DTS
Standby Diesel and Auxiliary Boiler Fuel Oil System (DOS)						
MT01 A/B	No. 2 Diesel Fuel Oil	120000	454.2	100%	454.2	DOS
MT02 A/B	No. 2 Diesel Fuel Oil	2600	9.8	100%	15.2	DOS
MT03	No. 2 Diesel Fuel Oil	650	2.5	100%	2.5	DOS

*Mass of solution containing chemical.

Table 2.9-1 (cont.)

CHEMICAL INVENTORY ON AP1000 NPP TURBINE ISLAND

Tank	Chemical[1]	Volume		Concentration (%w/v)	Mass Chemical (tonnes)*	Affected System
		U.S. gallons	m^3			
Fire Protection System (FPS)						
MT02	No. 2 Diesel Fuel Oil	240	0.9	100%	0.9	FPS
Plant Gas System (PGS)						
Liquid	Nitrogen	1500	5.7	100%	4.6	PGS
Gas (3 bottle)	Nitrogen	27735 scf	785.4	100%	0.9	PGS
Liquid	Hydrogen	1500	5.7	100%	0.4	PGS
Gas (1 bottle)	Hydrogen	500 scf	14.2	100%	0.001	PGS
Liquid	Carbon Dioxide	104800 scf	2967.6	100%	6	PGS
Central Chilled Water System (VWS)						
MT02/MT05	Sodium Molybdate/Tolytriazole	45	0.2	<50%	<0.2	VWS
Zinc Addition Subsystem						
Zinc Addition Tank	Zinc Acetate	200	0.8	<40%	<0.8	CVS

Note:
1. The selection of chemicals may change based on site-specific requirements and site-operating experience.

*Mass of solution containing chemical.

Table 2.9-2

CHEMICAL INVENTORY ON AP1000 NPP NUCLEAR ISLAND

Tank	Chemical	Volume		Concentration (% w/v)	Mass of Chemical (tonnes)*	Affected System
		U.S. Gallons	m^3			
Boric Acid Tank	Boric Acid	80,000	302.8	0.4375%	302.6	CVS
Hydrogen (liquid)	Hydrogen	1500	5.7	100%	0.4	CVS
Lithium7 Hydroxide Tank	Lithium7 Hydroxide	5	0.019	<12%	<0.017	CVS
Hydrazine Tank	Hydrazine	5	0.019	35%	0.02	CVS

*Mass of solution containing chemical.

Note: Actual chemical selections may vary based upon Owner's site-specific chemistry program selections.

Table 2.9-3

CHEMICAL INVENTORY ON SEAWATER NUCLEAR ISLAND

Tank	Chemical	Volume		Concentration (% w/v)	Mass of Chemical (tonnes)*	Affected System
		U.S. Gallons	m^3			
MT06	Polyacrylate or polyphosphate or orthopolyphosphate	10,000	37.85	<100%	<37.85	SWS
MT09	Sodium Hydrochlorite	10,000	37.85	30%	38	CWS, SWS
MT10	Ammonium Hydroxide	10,000	37.85	30%	38	CWS, SWS

*Mass of solution containing chemical.

Table 2.9-4

AP1000 NPP CHEMICALS THAT ARE LIST I AND II SUBSTANCES UNDER THE GROUNDWATER DIRECTIVE

Chemical	List I or List II	Basis for Classification
Ammonium hydroxide	List II	Ammonia
Polyphosphate or orthopolyphosphate	List II	Inorganic compounds of phosphorus and elemental phosphorus
Sodium hypochlorite	List II	Nominated chemical
Boric acid	List II	Inorganic compounds of boron
Zinc compounds	List II	Nominated metal compounds
Diesel fuel oil	List I	Mineral oils and hydrocarbons
Halogenated By-Products of Chlorination in Seawater (see Table 4.2-3)	List I	Organohalogen compounds (and substances which may form such compounds in the aquatic environment) i.e., any organic compound which contains one or more covalently bonded halogen atoms

Table 2.9-5

AP1000 NPP NON-RADIOACTIVE EFFLUENT COMPONENTS THAT ARE LIST I AND II SUBSTANCES UNDER THE DANGEROUS SUBSTANCES DIRECTIVE

Chemical	List I/List II	Basis for Classification
Ammonium hydroxide	List II	Ammonia
Polyphosphate or orthopolyphosphate	List II	Inorganic compounds of phosphorus and elemental phosphorus
Hydrazine	List II	Substances which have an adverse effect on the oxygen balance, particularly: ammonia, nitrites
Sodium hypochlorite	List II	Biocides and their derivatives not appearing in List I
Boric acid	List II	Inorganic compounds of boron
Diesel fuel oil	List I	Mineral oils and hydrocarbons
Halogenated By-Products of Chlorination in Seawater (see Table 4.2-3)	List I	Organohalogen compounds (and substances which may form such compounds in the aquatic environment) i.e., any organic compound which contains one or more covalently bonded halogen atoms

Table 2.9-6

AP1000 NPP CHEMICAL SYSTEM TANKS AND SECONDARY CONTAINMENT

		Primary Containment					Secondary Containment				
			Volume		Dimensions			Volume		Dimensions	
Tank	**Chemical / Radioactive Material**	**Material**	**U.S. Gallons**	**m^3**	**ft**	**m**	**Material**	**U.S. Gallons**	**m^3**	**ft**	**m**
Turbine Island Chemical Feed System (CFS)											
MT01	Hydrazine	Stainless Steel Tote Container (see Figure 2.9-3)	800	3.03	~4.9 long ~4.9 wide ~4.9 high	~1.5 long ~1.5 wide ~1.5 high	Concrete	4095	15.5	33.75 long 22.75 wide 0.67 high	11.2 long 6.9 wide 0.2 high
MT02	pH control chemical – monoethanolamine (actual pH control additive used may differ)		800	3.03							
MT03	Ammonium Hydroxide		800	3.03							
MT04	Ammonium Hydroxide		800	3.03							
MT05	Sodium Sulfite and Sodium Hydroxide		240	0.91							
MT07	Polyphosphate		800	3.03							
MT08	Ammonium Chloride		800	3.03							
MT11	Ammonium Hydroxide		800	3.03							

Table 2.9-6 (cont.)

AP1000 NPP CHEMICAL SYSTEM TANKS AND SECONDARY CONTAINMENT

Tank	Chemical/ Radioactive Material	Primary Containment					Secondary Containment				
		Material	Volume		Dimensions		Material	Volume		Dimensions	
			U.S. Gallons	m^3	ft	m		U.S. Gallons	m^3	ft	m
Standby Diesel and Auxiliary Boiler Fuel Oil System (DOS)											
MT01A	No. 2 Diesel Fuel Oil	Storage Tank	60000	227	Note (2)		Concrete	66000	250	Note (2)	
MT01B	No. 2 Diesel Fuel Oil	Storage Tank	60000	227	Note (2)		Concrete	66000	250	Note (2)	
MT02A	No. 2 Diesel Fuel Oil	Day Tank	2000	7.6	Note (2)		Diesel Generator Building Sump. Concrete lined with fiber-reinforced epoxy coating	580(1)	2.2(1)	4 diam 6 high	1.22 diam 1.83 high
MT02B	No. 2 Diesel Fuel Oil	Day Tank	2000	7.6	Note (2)						
MT03	No. 2 Diesel Fuel Oil	Ancillary diesel generator fuel oil tank	650	2.45	Note (2)		Annex building Sump. Concrete, fibre-reinforced epoxy coating	2500	9.5	6 long 8 wide 7 high	1.83 long 2.44 wide 2.13 high

Table 2.9-6 (cont.)

AP1000 NPP CHEMICAL SYSTEM TANKS AND SECONDARY CONTAINMENT

		Primary Containment					Secondary Containment				
			Volume		Dimensions			Volume		Dimensions	
Tank	**Chemical/ Radioactive Material**	**Material**	**U.S. Gallons**	**m^3**	**ft**	**m**	**Material**	**U.S. Gallons**	**m^3**	**ft**	**m**
Standby Diesel and Auxiliary Boiler Fuel Oil System (DOS)											
MT11							Annex building Sump. Concrete, fibre-reinforced epoxy coating	5000	18.9		
Central Chilled Water System (VWS)											
MT02/MT 05	Sodium Molybdate/ Tolytriazole	SA-516 Grade 70 carbon steel, chemical addition tank	20	0.076	1.17 diam 3.83 high	0.36 diam 1.17 high	Turbine building sumps. Concrete lined with fibre-reinforced epoxy coating.	51350	194.4	2@ 13 long 24 wide 11 high	2@ 3.96 long 7.32 wide 3.35 high

Table 2.9-6 (cont)

AP1000 NPP CHEMICAL SYSTEM TANKS AND SECONDARY CONTAINMENT

		Primary Containment					Secondary Containment				
			Volume		Dimensions			Volume		Dimensions	
Tank	**Chemical/ Radioactive Material**	**Material**	**U.S. Gallons**	**m^3**	**ft**	**m**	**Material**	**U.S. Gallons**	**m^3**	**ft**	**m**
Chemical Volume Control System (CVS)											
Zn Addition Tank	Zinc Acetate	Stainless steel	200	0.76	Note (2)	Note (2)	Turbine building sumps. Concrete lined with fibre-reinforced epoxy coating.	51350	194.4	2@ 13 long 24 wide 11 high	2@ 3.96 long 7.32 wide 3.35 high
Reactor Coolant Supply											
Storage tank	Boric Acid	Stainless steel	80000	303	24 diam 23 high	7.32 diam 7.01 high	Note (1,2)				

Table 2.9-6 (cont)

AP1000 NPP CHEMICAL SYSTEM TANKS AND SECONDARY CONTAINMENT

Tank	Chemical/ Radioactive Material	Primary Containment					Secondary Containment				
		Material	Volume		Dimensions		Material	Volume		Dimensions	
			U.S. Gallons	m^3	ft	m		U.S. Gallons	m^3	ft	m
Batching tank	Boric Acid	austenitic stainless steel	800	3.03	~4.9 long ~4.9 wide ~4.9 high	~1.5 long ~1.5 wide ~1.5 high	Annex building Sump. Concrete, fibre-reinforced epoxy coating	2500(1)	9.5(1)	6 long 8 wide 7 high	1.83 long 2.44 wide 2.13 high
Chemical Mixing Tank	Normally empty	stainless steel	5	0.02	Note (2)		Auxiliary Building Sump. Concrete, fibre-reinforced epoxy coating	2500(1)	9.5	6 long 8 wide 7 high	1.83 long 2.44 wide 2.13 high

Table 2.9-6 (cont)

AP1000 NPP CHEMICAL SYSTEM TANKS AND SECONDARY CONTAINMENT

Tank	Chemical/ Radioactive Material	Primary Containment					Secondary Containment				
		Material	Volume		Dimensions		Material	Volume		Dimensions	
			U.S. Gallons	m^3	ft	m		U.S. Gallons	m^3	ft	m
Fire Protection System (FPS)											
MT02	No. 2 Diesel Fuel Oil	Ancillary diesel generator fuel oil tank	240	0.91	Note (2)		Concrete	Note (1)			
Circulating Water System (CWS) (Seawater)											
MT06	Polyacrylate or polyphosphate or orthopolyphosphate	Note (2)	10000	37.9	Note (2)		Concrete	Note (1)			
MT09	Sodium hypochlorite	Note (2)	10000	37.9	Note (2)		Concrete	Note (1)			
MT10	Ammonium hydroxide	Note (2)	10000	37.9	Note (2)		Concrete	Note (1)			

Notes:
1. Secondary containment design to be made compliant with UK requirements during site specific design.
2. To be determined.

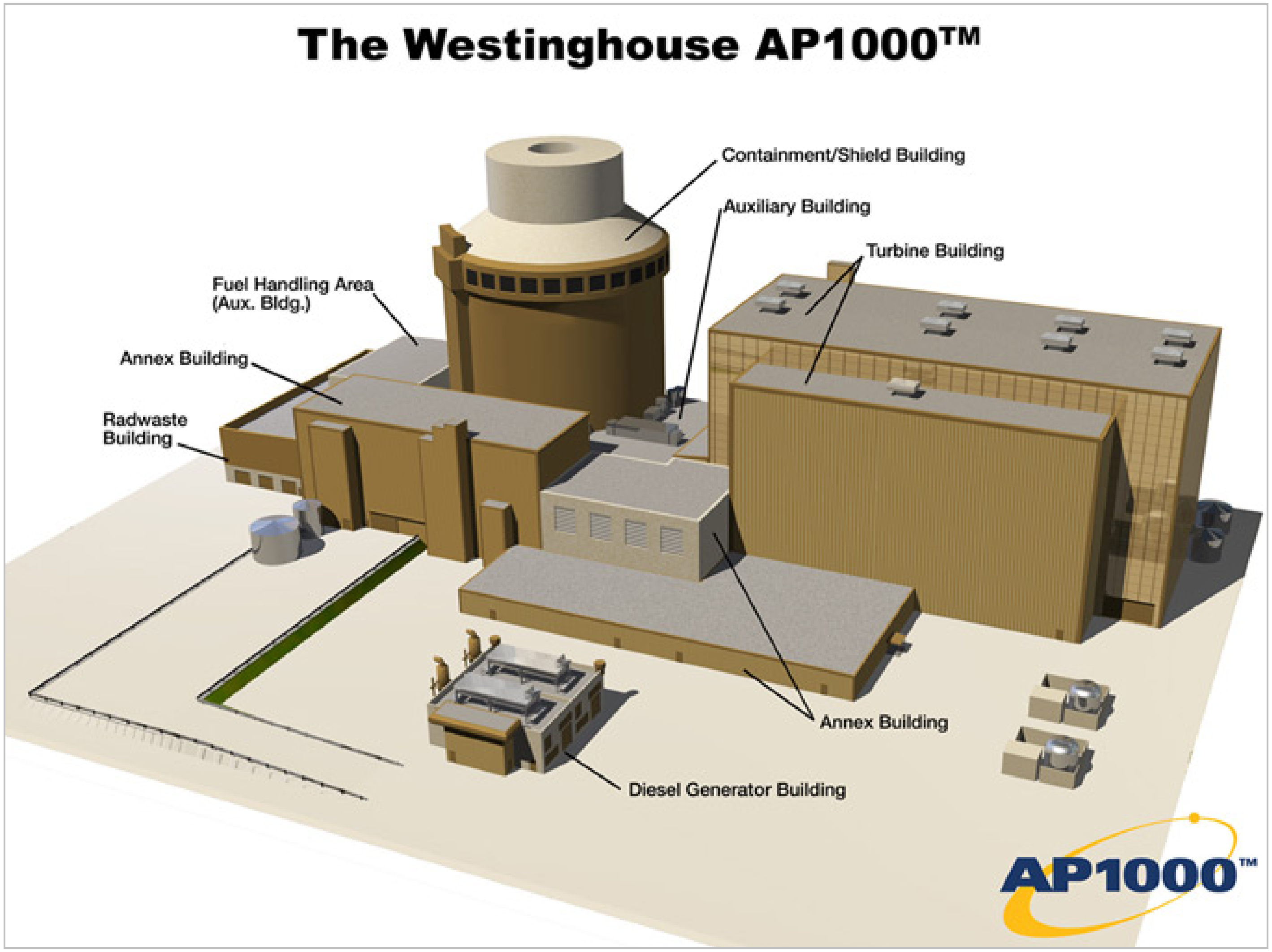


Figure 2.3-1. AP1000 NPP Schematic

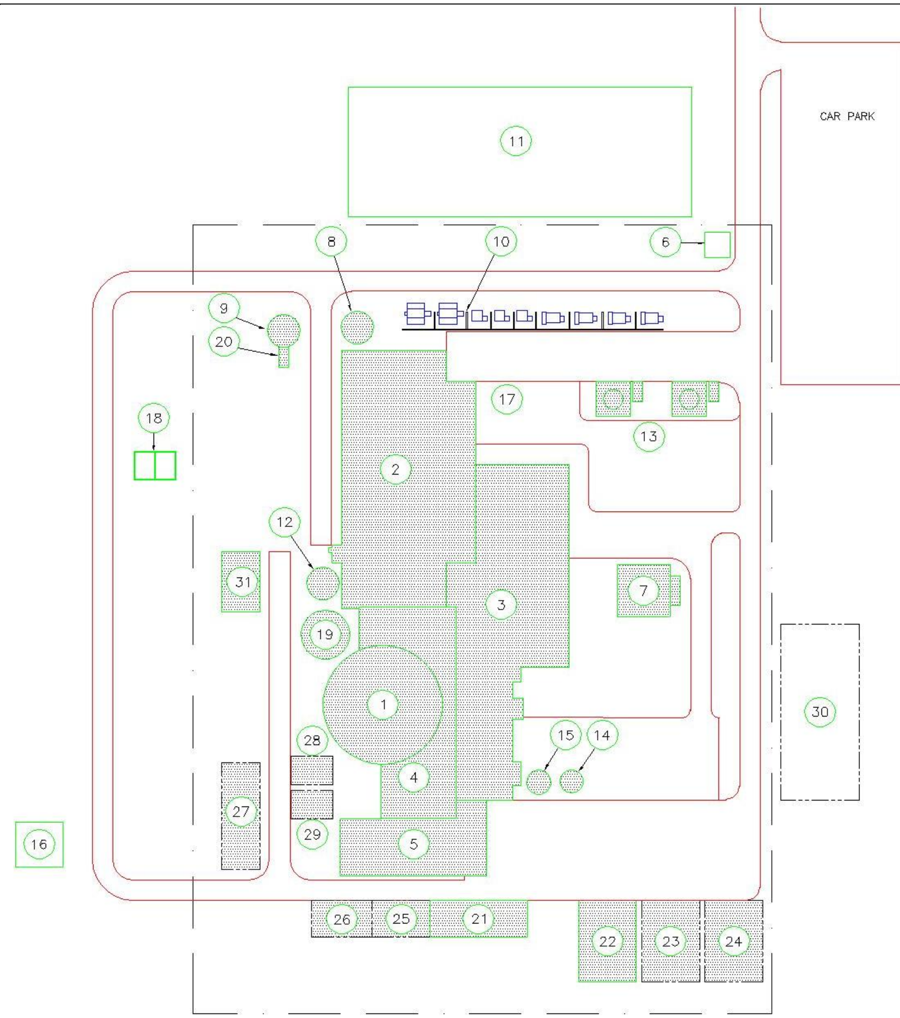


Figure 2.3-2. Typical AP1000 NPP Plot Plan

Figure 2.3-2 (cont)

KEY TO TYPICAL AP1000 NPP PLOT PLAN

Item	Description
1	Containment Shield Building
2	Turbine Building
3	Annex Building
4	Auxiliary Building
5	Radwaste Building
6	Plant Entrance
7	Diesel Generator Building
8	Fire Water / Clearwell Storage Tank
9	Fire Water Storage Tank
10	Transformer Area
11	Switch Yard
12	Condensate Storage Tank
13	Diesel Generator Fuel Oil Storage Tanks
14	Demineralised Water
15	Boric Acid Storage Tank
16	Hydrogen Storage Tank Area
17	Turbine Building Letdown Area
18	Waste Water Retention Basis
19	Passive Containment Cooling Ancillary Water Storage Tank
20	Diesel Driven Fire Pump/Enclosure
21	ILW Store (20 Years)
22	Spent Fuel Store (20 Years Storage)
23	Spent Fuel Store Extension 1 (40 Years)
24	Spent Fuel Store Extension 2 (60 years)
25	ILW Store Extension 1 (40 Years)
26	ILW Store Extension 2 (60 Years)
27	Decommissioning Facility (Future)
28	Storage Area for Non-Radioactive Waste
29	Storage Area Low Level Waste
30	Area for Contractors Compound
31	Area for Storage of Large Radioactive Components

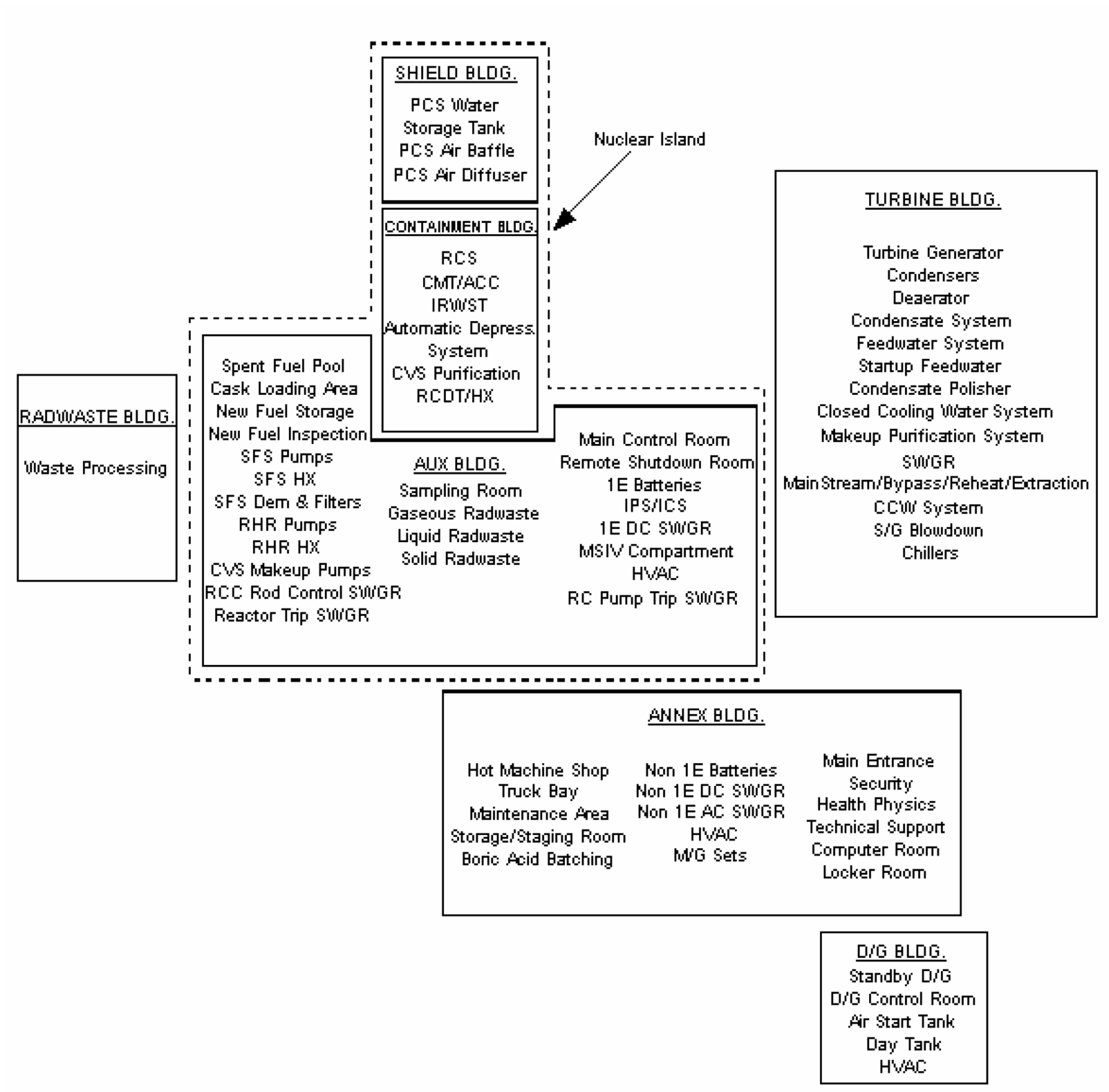


Figure 2.3-3. Location of System Functions within the AP1000 NPP Power Generation Complex

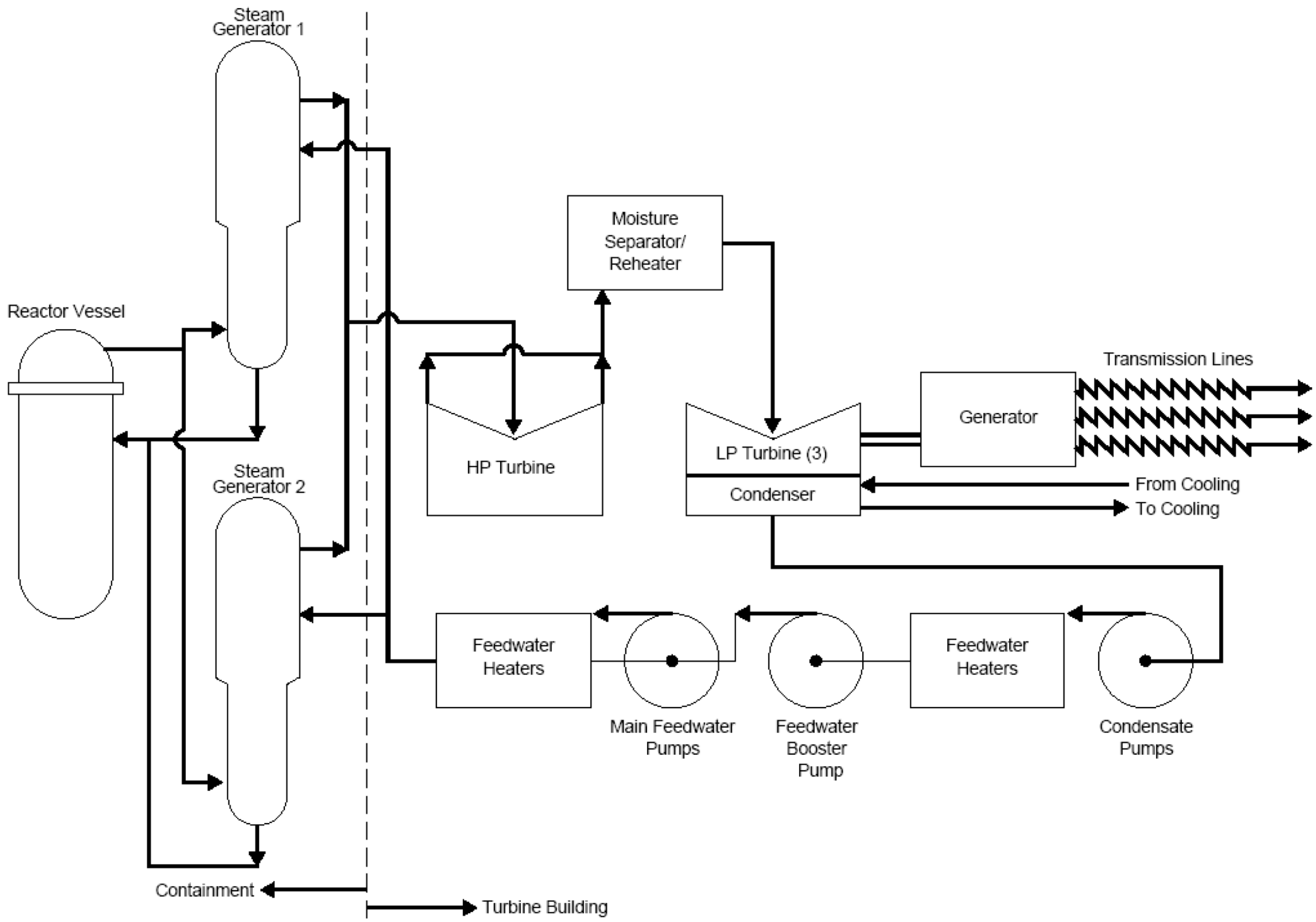
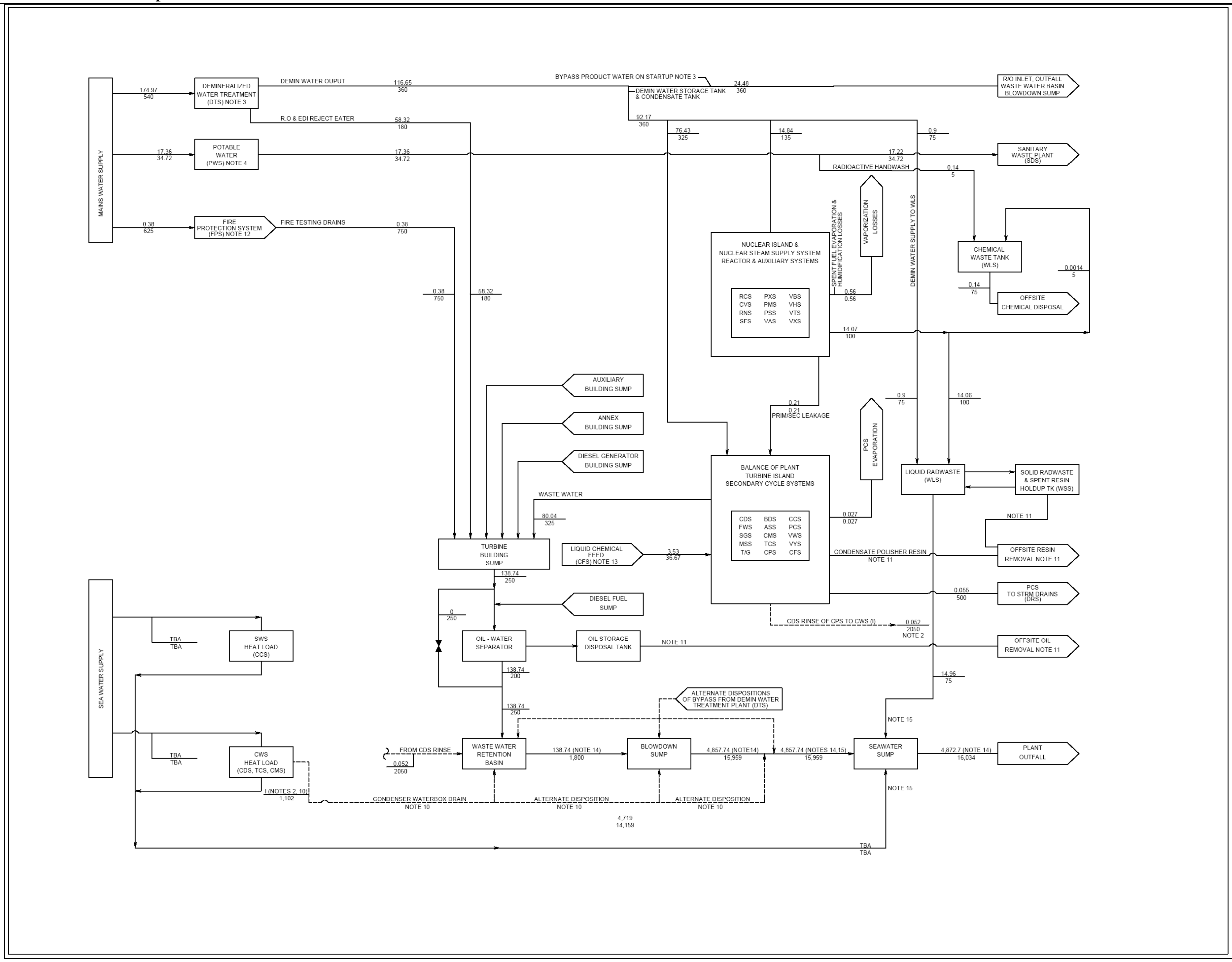


Figure 2.4-1. Reactor Power Conversion System – Simplified Flow Diagram



SYSTEM	KEY LOCATOR
ASS	AUXILIARY STEAM SUPPLY SYSTEM
BDS	STEAM GENERATOR BLOWDOWN SYSTEM
CCS	COMPONENT COOLING WATER SYSTEM
CDS	CONDENSATE SYSTEM
CFS	TURBINE ISLAND CHEMICAL FEED SYSTEM
CMS	CONDENSER AIR REMOVAL SYSTEM
CPS	CONDENSATE POLISHING SYSTEM
CVS	CHEMICAL AND VOLUME CONTROL SYSTEM
CWS	CIRCULATING WATER SYSTEM
DRS	STORM DRAIN SYSTEM
DTS	DEMINERALISED WATER TREATMENT SYSTEM
FPS	FIRE PROTECTION SYSTEM
FWS	MAIN FEEDWATER SYSTEM
MSS	MAIN STEAM SYSTEM
PCS	PASSIVE CONTAINMENT COOLING SYSTEM
PSS	PRIMARY SAMPLING SYSTEM
PWS	POTABLE WATER SYSTEM
PXS	PASSIVE CORE COOLING SYSTEM
RCS	REACTOR COOLANT SYSTEM
RMS	RADIATION MONITORING SYSTEM
RNS	NORMAL RESIDUAL HEAT REMOVAL SYSTEM
SDS	SANITARY DRAINAGE SYSTEM
SFS	SPENT FUEL POOL COOLING SYSTEM
SGS	STEAM GENERATOR SYSTEM
SWS	SERVICE WATER SYSTEM
TCS	TURBINE ISLAND CLOSED COOLING WATER
T/G VAS	SYSTEM TURBINE/GENERATOR RADIOLOGICALLY CONTROLLED AREA VENTILATION SYSTEM
VBS	NUCLEAR ISLAND NONRADIOACTIVE VENTILATION SYSTEM
VHS	HEALTH PHYSICS AND HOT MACHINE SHOP HVAC SYSTEM
VTS	TURBINE BUILDING VENTILATION SYSTEM
VWS	CENTRAL CHILLED WATER SYSTEM
VXS	ANNEX/AUXILIARY BUILDING NONRADIOACTIVE VENTILATION SYSTEM
VYS	HOT WATER HEATING SYSTEM
WLS	LIQUID RADWASTE SYSTEM
WSS	SOLID RADWASTE SYSTEM
WWS	WASTE WATER SYSTEM

Figure 2.7-1.
AP1000 NPP Standard Plant Water Balance

GENERAL NOTES

1. PATH FLOW RATES ARE DENOTED IN GALLONS PER MINUTE FOR A SINGLE UNIT. NORMAL VALUES OF FLOW RATES ARE SHOWN ABOVE THE LINE, AND MAXIMUM VALUES OF FLOW RATES ARE SHOWN BELOW THE LINE. THE BASES FOR NORMAL AND MAXIMUM FLOWS FOR THE DTS, PWS, FPS, SWS, AND CWS ARE PROVIDED IN THE SPECIFIC SYSTEM NOTES BELOW. SITE PRECIPITATION IS NOT INCLUDED IN THE WATER BALANCE DIAGRAM.

2. MANY OF THE FLOWS REPRESENTED IN THE WATER BALANCE ARE INTERMITTENT TO A CERTAIN DEGREE. THOSE FLOWS WHICH ARE VERY INFREQUENT OR WHICH PERSIST FOR VERY SHORT DURATIONS ARE SHOWN AS DOTTED LINE INTERMITTENT FLOWS. THESE INCLUDE THE CONDENSER WATER BOX DRAIN, THE STRAINER BACKWASH FLOWS FROM THE CWS AND SWS STRAINERS, AND THE CONDENSATE RINSE OF THE CONDENSATE POLISHER RESINS. DEMINERALISED WATER FLOWS ORIGINATING FROM THE DEMINERALISED WATER TREATMENT PLANT (DTS), AND DISTRIBUTED THROUGHOUT THE PLANT VIA THE DEMINERALISED WATER TRANSFER & STORAGE SYSTEM (DWS), ARE, WITH CERTAIN EXCEPTIONS, INDIVIDUALLY INTERMITTENT, BUT ARE SHOWN AS THE AVERAGE OF THESE INDIVIDUAL INTERMITTENT FLOWS OVER A COMPLETE FUEL CYCLE FOR PURPOSES OF DENOTING A NORMAL FLOW RATE. MAXIMUM FLOW RATES FOR DEMINERALISED WATER AND CONDENSATE ARE TYPICALLY BASED ON COMPONENT AND SYSTEM CAPACITIES (E.G., PUMP DESIGN FLOWS).

SPECIFIC SYSTEM NOTES

3. THE DEMINERALISED WATER TREATMENT PLANT (DTS) OPERATES IN A BATCH MODE TO REFILL THE DEMINERALISED WATER STORAGE TANK ON DEMAND. IN THE SINGLE PASS MODE, THE SYSTEM PRODUCES 360 GPM OF PRODUCT WATER FROM 540 GPM OF MAINS WATER, REJECTING 180 GPM OF WASTE WATER; THESE VALUES FORM THE BASIS FOR THE MAXIMUM FLOWS. AN ALLOWANCE HAS BEEN INCLUDED TO BYPASS THE PRODUCT WATER FROM THE DTS ON INITIATION OF BATCH PROCESSING TO BRING THE PRODUCT WATER INTO SPECIFICATION BEFORE DIRECTING THE PRODUCT INTO THE DEMINERALISED WATER STORAGE TANK. THE DISPOSITION OF THE BYPASS WATER IS SITE DEPENDENT. IT CAN BE DIRECTED BACK TO THE PLANT OUTFALL, TO THE WWRB, OR TO THE BLOWDOWN SUMP DIRECTLY WITHOUT PASSING THROUGH THE WASTE WATER SYSTEM PROCESSING.

4. NORMAL PWS FLOWS ARE BASED ON CONSUMPTION OF 50 GALLONS PER DAY PER PERSON FOR 500 PERSONS. MAXIMUM PWS FLOWS ARE BASED ON CONSUMPTION OF 50 GPD FOR 1000 PERSONS, REPRESENTATIVE OF SHUTDOWN PERIODS.

5. NORMAL SERVICE WATER FLOWS ARE BASED ON 100% POWER OPERATION AT 4 CYCLES OF CONCENTRATION. MAXIMUM SERVICE WATER FLOWS ARE BASED ON PLANT COOLDOWN WITH 4 CYCLES OF CONCENTRATION.

6. BACKWASH FROM SERVICE WATER STRAINERS IS ESTIMATED AT 3000 GPM FOR 1 MINUTE EVERY 12 HOURS THAT A STRAINER IS IN SERVICE. ONE STRAINER IS IN SERVICE FOR POWER OPERATION AND TWO STRAINERS ARE IN SERVICE DURING PLANT COOLDOWN. STRAINER BACKWASH CYCLES ARE STAGGERED TO AVOID SIMULTANEOUS DISCHARGES. THEREFORE, NORMAL STRAINER BACKWASH IS 3000 GPM FOR 1 MINUTE EVERY 12 HOURS (4.17 GPM AVERAGE) AND MAXIMUM STRAINER BACKWASH IS 3000 GPM FOR 1 MINUTE EVERY 6 HOURS (I.E., 8.33 GPM AVERAGE). STRAINER BACKWASH IS CONSIDERED TO BE INTERMITTENT AND IS NOT FIGURED INTO THE WATER BALANCE NUMERICALLY SINCE THE AVERAGE DISCHARGE OVER TIME IS <= 0.1% OF BLOWDOWN FLOW.

7. NORMAL CIRCULATING WATER FLOWS ARE BASED ON 100% POWER OPERATION WITH 4 CYCLES OF CONCENTRATION. MAXIMUM CIRCULATING WATER FLOWS ARE BASED ON 100% POWER OPERATION WITH 2 CYCLES OF CONCENTRATION.

8. SERVICE WATER BLOWDOWN IS DIRECTED TO THE CWS. CREDIT FOR THIS INFLOW HAS NOT BEEN CONSIDERED IN ESTABLISHING THE REQUIRED MAKEUP TO THE CWS. ALTERNATIVELY, SERVICE WATER BLOWDOWN CAN BE DIRECTED TO THE WWRB.

9. THERE ARE THREE STRAINERS IN THE COOLING WATER SUPPLY LINES FROM THE CIRCULATING WATER PUMPS TO THE PLATE HEAT EXCHANGERS IN THE TURBINE ISLAND CLOSED COOLING WATER SYSTEM. EACH STRAINER AUTOMATICALLY BACKWASHES AT 1820 GPM FOR A PERIOD OF 30 SECONDS, 5 TIMES PER 24 HOUR PERIOD DURING NORMAL OPERATION. THE BACKWASH CYCLE TIMERS ARE STAGGERED TO AVOID SIMULTANEOUS BACKWASHES FROM THE STRAINERS; HENCE, THERE ARE 15 CYCLES OF BACKWASH, EACH LASTING 30 SECONDS, PER 24 HOUR PERIOD. STRAINER BACKWASH IS CONSIDERED TO BE INTERMITTENT AND IS NOT FIGURED INTO THE WATER BALANCE NUMERICALLY SINCE THE AVERAGE DISCHARGE OVER TIME IS <= 0.2% OF BLOWDOWN FLOW.

10. THE DISPOSITION OF THE CONDENSER WATERBOX DRAIN IS SITE DEPENDENT. IT CAN BE DISCHARGED TO THE WWRB, THE BLOWDOWN SUMP, OR DIRECTLY TO THE PLANT OUTFALL. THE WATERBOX DRAIN OCCURS ONLY DURING SHUTDOWN PERIODS AND IS CONSIDERED TO BE AN INFREQUENT AND INTERMITTENT FLOW BASED ON THE UTILITY REQUIREMENTS DOCUMENT. THE QUANTITY OF THE WATERBOX DRAIN HAS NOT BEEN QUANTITATIVELY CONSIDERED IN DETERMINING THE CIRCULATING MAKEUP WATER REQUIREMENTS OR THE DISPOSITION TO WASTE WATER.

11. THE WATER BALANCE DOES NOT QUANTIFY OR REPRESENT THE VOLUME OF RESINS DISCHARGED OFFSITE FROM THE SPENT RESIN STORAGE TANK (WSS) OR FROM THE CPS. NOR DOES IT QUANTIFY THE VOLUME OF OIL WASTES REMOVED BY THE OIL-WATER SEPARATOR THROUGH THE OIL STORAGE DISPOSAL TANK.

12. NORMAL FIRE WATER CONSUMPTION FLOWS ARE BASED ON APPLICATION OF NFBA TESTING REQUIREMENTS. MAXIMUM FIRE WATER MAKEUP FLOW IS BASED ON REQUIREMENT TO REPLENISH 300,000 GALLONS FIRE RESERVE IN 8 HOURS.

13. LIQUID CHEMICALS ARE TREATED AS AN INFLUENT SOURCE OF LIQUID. THE CHEMICALS ARE PURCHASED AS LIQUIDS AND INJECTED INTO THE VARIOUS SYSTEMS AND ARE DISCHARGED VIA THE WASTE STREAMS.

14. TOTAL DISCHARGES EXCLUDE INTERMITTENT STRAINER BACKWASH AND INFREQUENT CONDENSER WATERBOX DRAIN AND SEA WATER RETURN FLOWS.

15. DISCHARGE MONITORING POINT (MP)

Figure 2.7-1 (cont.).
AP1000 NPP Standard Plant Water Balance (Notes)

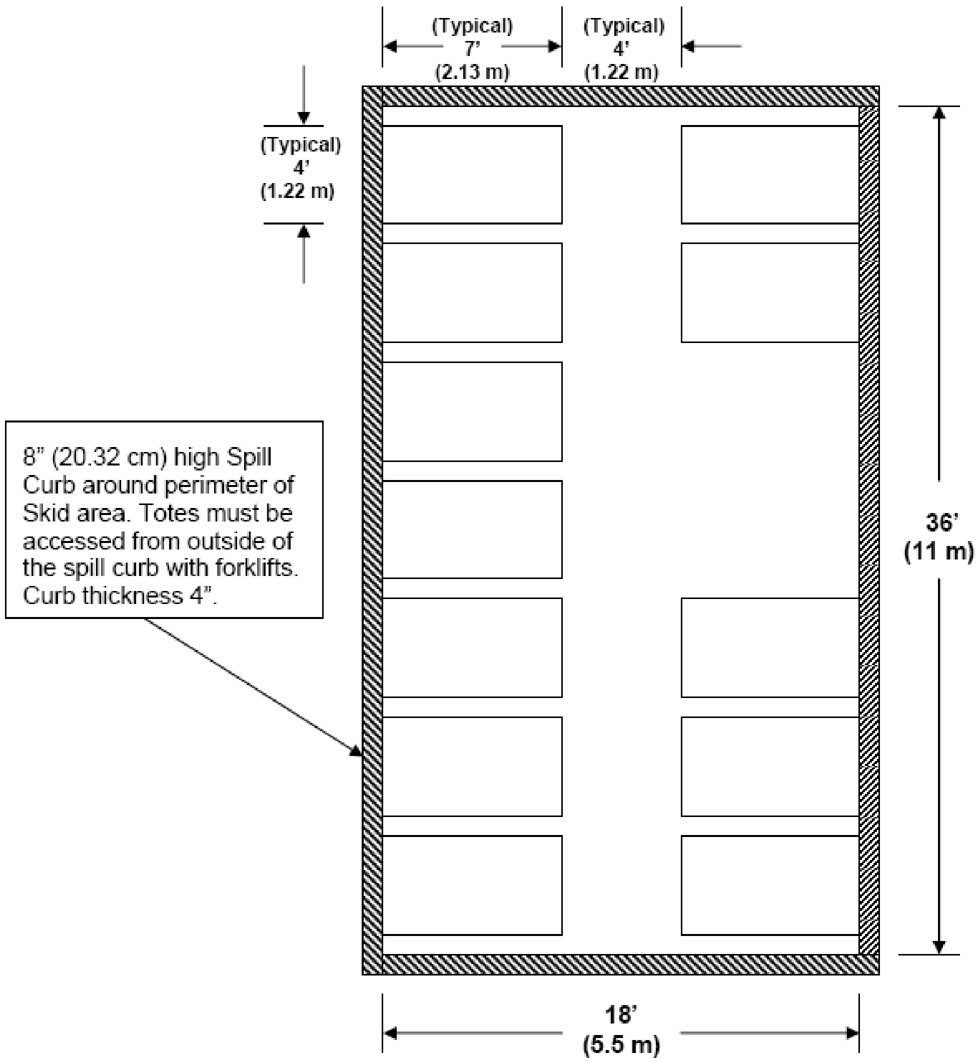


Figure 2.9-1. Plan View of AP1000 NPP Turbine Chemical Storage

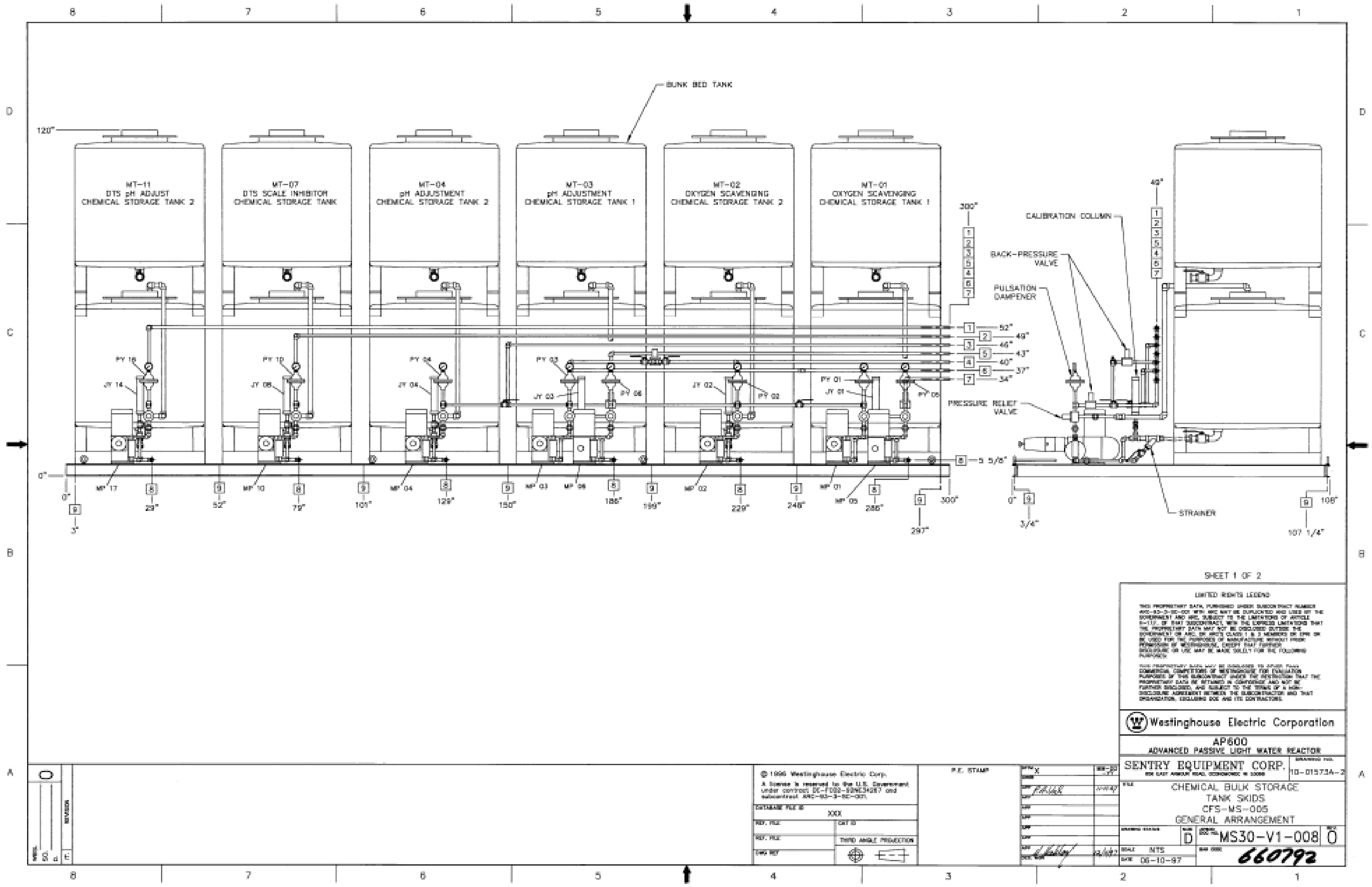


Figure 2.9-2. Turbine Island Chemical Storage Arrangement

Figure 2.9-3. Rugged Hazardous Liquid Tote Container

3.0 RADIOACTIVE WASTE MANAGEMENT SYSTEMS

3.1 Introduction

During the GDA process, an IWMS has been developed to ensure that radioactive material and radwastes generated by the **AP1000** NPP are managed in a manner which minimises the need for future processing, and that is compatible with anticipated facilities for ultimate disposal or end-use (Reference 3-1). During site-specific analysis, the operator of an **AP1000** NPP may develop its own strategy in line with its own corporate waste disposal procedures and policies.

The **AP1000** NPP IWMS (Reference 3-1) is consistent with the waste hierarchy shown in Figure 3.1-1 and the key BAT management factors for optimisation of releases from nuclear facilities shown in Figure 3.1-2 (Reference 3-2). These factors correlate closely as follows:

Waste Hierarchy	BAT Waste Management Factor
• Avoid	• Use of low waste technology
• Minimise	• Use of low waste technology • Efficient use of resources • Reduce emissions
• Reduce/Recycle	• Use of low waste technology • Use less hazardous substances
• Abatement	• Reduce emissions

It is not possible to avoid the production of radwaste entirely. Radioisotopes are produced during the normal operation of nuclear reactors, primarily through the processes of fission and activation. Fission products may enter the reactor coolant by diffusing from the fuel and then passing through the fuel cladding through leaks. The primary cooling water may contain dissolved or suspended corrosion products and non-radioactive materials leached from plant components which can be activated by the neutrons in the reactor core as the water passes through the core. These radioisotopes leave the reactor coolant either by plant systems designed to remove impurities, by small leaks that occur in the RCS and auxiliary systems, or during maintenance. Therefore, each plant generates radwaste that can be liquid, solid, or gaseous.

The other aspects of the waste hierarchy, minimisation, reduce/recycle and abatement of emissions, are all implemented within the **AP1000** NPP design in a manner that is consistent with the BAT waste management factors. Examples of these BAT techniques are shown in Table 3.1-1 and these and other techniques are discussed in more detail in the following sections.

3.2 Minimisation of Waste at Source

In the **AP1000** NPP, there are several ways in which the release of radioactive emissions is reduced at source. These ways are described below.

3.2.1 Fuel Rod Burn-up

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

The concept of high burn-up fuel is generally environmentally and economically beneficial. However, high fuel burn-up can result in extended cooling time for spent fuel. This will be a consideration for any future repository operator to address (see subsection 3.5.8.3).

3.2.2 Operational Cycle

Utilities can operate an **AP1000** NPP on different cycle lengths (e.g., 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 3 years, the cost of replacement power during the outage, the impact of outage length on average capacity factor, etc., 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.

3.2.3 Tramp Uranium

Uranium contamination on the exterior surface of Westinghouse fabricated fuel rods, sometimes called tramp uranium, is insignificant. Normally, Westinghouse does not smear their rods for contamination. However, when smears were taken, analysis showed that the levels of alpha contamination, and therefore any tramp uranium levels, were so low that statistically, smears no longer need to be taken.

3.2.4 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 uranium dioxide 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.

The design of the fuel that will be used in the **AP1000** NPP is an improvement over previous designs in that vibrations in the assembly are reduced. This design has already been used in existing plants. Since the implementation of the Westinghouse 17x17 RFA in 1998 the overall leakage rate of this design, incorporating all the Westinghouse debris protection features, is 0. The overall leakage rate, on a rod basis, of the basic RFA fuel product including designs that do not use all the debris protection features is less than 10^{-5} (Reference 3-42).

3.2.5 Materials Selection

In order to reduce Co-60 production by activation of Co-59, the latter is limited to below 0.05 weight percent in reactor internal structures and below 0.2 weight percent in primary and auxiliary materials. Low or zero cobalt alloys used for hard-facing 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 the reactor coolant. Cobalt free wear resistant alloys considered for this application include those developed and qualified in nuclear industry programmes.

The use of cobalt base alloy (e.g., **Stellite®**) is limited to applications where its hardness, low friction, and resistance to wear provide the best reliability for critical operations. Examples include use on guide surfaces of motor-operated gate valves and the seating surfaces of air-operated globe valves where tight shutoff and durability are required.

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.

Co-58 is produced from activation of Ni-58. For this reason, nickel-based alloys in the reactor coolant system are used only where component reliability may be compromised by the use of other materials. The major use of nickel-based alloys in the reactor coolant system is the INCONEL® steam generator tubes.

There are general prohibitions on copper, lead and antimony and other low-melting-point metals used in engineered safety features. In addition, the reactor coolant pump mechanical design criteria prohibit antimony completely from the reactor coolant pump and its bearings.

3.2.6 Minimisation of Leakage Pathways

The **AP1000** NPP is designed with fewer valves and components than predecessor plants which will result in fewer leakage pathways and lower overall input to the radwaste systems.

3.2.7 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 CVS. The following chemicals are added to the borated RCS system:

- Lithium hydroxide (Li7OH) is used to control the pH of the RCS and is chosen for its compatibility with borated water chemistry and the stainless steel and zirconium materials. The effective control of pH reduces the formation of radioactive corrosion products that may be released in liquid effluent. The use of Li7OH, where the Li-7 isotope has been enriched, also removes an important formation mechanism for tritium. The neutron absorption cross-section of lithium-7 is five orders of magnitude smaller than that of lithium-6, and the use of Li7OH substantially reduces the potential for tritium formation associated with neutron absorption by lithium-6 present in natural lithium hydroxide.

- Hydrazine is introduced as an oxygen scavenger during plant startup from cold shutdown to reduce corrosion product formation associated with dissolved oxygen.

- Dissolved hydrogen is added to the RCS during power operations to eliminate free oxygen produced by radiolysis in the core and to prevent ammonia formation.

- Zinc acetate is added initially during hot functional testing and during operations to minimise corrosion and to reduce radiocobalt and activated nickel concentrations.

The RCS water chemistry is routinely analysed to ensure that the chemistry is correct and that corrosion product particulates are below specified limits.

3.2.8 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, 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.

3.2.9 Reactor Coolant Pressure Boundary

Airborne releases can be limited by restricting reactor coolant leakage. The 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.

3.2.10 Reactor Coolant Purification

The CVS purifies the RCS to maintain low RCS activity levels. The CVS purification loop contains two mixed-bed demineralisers, a 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 coarse filters for infrequent large particulates. Fine filtration is provided by reactor coolant filters which are provided downstream of the demineralisers to collect particulates and resin fines. They are specified to retain 98% of 0.10 micron (0.004 mils) particles with sufficient filter capacity to maintain a differential pressure of less than 0.30 MPa (44 psi). It is expected that during plant startup, plant testing, and possibly during the first few cycles that filter cartridges with larger micron ratings may be required (to increase filter life and reduce cartridge changeout).

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. The mixed-bed and cation-bed demineralisers are expected to have considerable excess capacity to handle primary circuit purification during shutdown operations.

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 backup CVS mixed bed can be placed in service without the need to enter containment. At that point, it would be the left to the judgment of the operating utility whether there is a pressing need to replace the exhausted CVS change-out with the unit in power operation. Radiological conditions at any specific time must be carefully assessed.

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 “backup” CVS mixed-bed demineraliser (not yet in service) has the capability to perform deboration to roughly 70 ppm boron at the end of the fuel cycle. If the backup 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.

The mixed-bed demineralisers also remove zinc during periods of zinc addition (see Section 2.6.6). 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 37 kBq/cm^3 (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 WLS 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.

3.2.11 pH Control with Li7OH

Lithium hydroxide is used to control pH within the RCS. Naturally occurring lithium would produce large amounts of tritium when exposed to neutron irradiation. In order to reduce this source of tritium production, lithium hydroxide enriched in the Li-7 isotope (Li7OH) is used to largely avoid addition of the Li-6 isotope prone to such tritium production.

3.2.12 Recycling Steam Generator Blow Down

Fluid recycling is provided for the steam generator blowdown fluid which is normally returned to the Condensate System (CDS).

3.3 Gaseous Radioactive Waste

3.3.1 Gaseous Radwaste System

The WGS is described in detail in Chapter 26 of the PCSR (Reference 1-10).

3.3.1.1 Gaseous Radwaste System Sources

During reactor operation, tritium 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. The reactor coolant may also contain tritium and gaseous carbon-14 (mainly in the form of methane) produced by various activation reactions within the reactor coolant. Leakage of reactor coolant thus results in a release to the containment atmosphere of these gases.

Airborne releases can be limited both by restricting reactor coolant leakage and by limiting the concentrations of radioactive gases in the RCS.

WGS inputs are as follows:

- RCS degassing:

  Removal of the noble gases from the RCS is not normally necessary because the gases do not build up to unacceptable levels when fuel defects are within normally anticipated ranges. However, the WGS periodically receives influent when CVS letdown is processed through the WLS degasifier. This occurs when the CVS letdown flow is diverted to the WLS degasifier during dilutions, borations, and RCS degassing before shutdown.

  The WLS degasifier discharge is the largest input to the WGS. Since the degasifier is a vacuum type and requires no purge gas, the maximum gas influent rate to the WGS is essentially equal to the maximum dissolved hydrogen rate into the degasifier. The maximum input flow rate from degasifier separator is 0.99 m^3/h (0.58 standard cubic feet per minute (scfm)) based on RCS hydrogen concentration 45 cm^3/kg (1.2 in^3/lb).

- WLS reactor coolant drain tank (RCDT) degassing:

  The RCDT contents are also degassed by the WLS degasifier, and the resulting gas is then routed to the WGS. When enough gas has naturally come out of the RCDT contents, the tank is also vented to the WGS. The maximum input flow rate from the RCDT is $0.85m^3/h$ (0.5 scfm).

The estimated gaseous radwaste emissions arising from the system operations are shown in Table 3.3-1.

3.3.1.2 Gaseous Radwaste Treatment System

If noble gas removal is required because of high RCS concentration, the CVS is operated in conjunction with the WLS degasifier to remove the gases. Iodine is removed by ion exchange in the CVS.

Gases generated by RCS or RCDT degassing are passed to the WGS for further treatment. The WGS is used intermittently. Most of the time during normal operation of the **AP1000** NPP, the WGS is inactive. Based on the maximum input gas volume, the WGS is expected to operate approximately 100 hours per year.

The WGS is designed to perform on an intermittent basis the following major functions:

- Collect radioactive or hydrogen bearing gaseous wastes

- Process and discharge the waste gas while keeping offsite releases of radioactivity within acceptable limits

The **AP1000** NPP WGS is a once-through, ambient temperature, activated carbon delay system (see Figure 3.3-1). The system includes a gas cooler, a moisture separator, an activated carbon-filled guard bed, and two activated carbon-filled delay beds. The radioactive fission gases entering the system are carried by hydrogen and nitrogen gases that pass through the system tanks to their incoming pressure. These influents successively pass through:

- The gas cooler, where they are cooled to about 4ºC (40°F) by the chilled water system.

- The moisture separator, which is a stainless steel receiver, collects condensed water vapour (including condensed tritiated water vapour) from the cooled gas thus removing it from the gaseous radioactivity stream. The collected water is periodically discharged automatically to the WLS.

- An activated carbon-filled guard bed, which protects the delay beds from abnormal moisture carryover or chemical contaminants. It absorbs radioactive iodine with efficiencies of 99 percent for methyl iodine and 99.9 percent for elemental iodine. It also provides increased delay time for xenon and krypton and deep bed filtration of particulates entrained in the gas stream.

- 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 avearge calculated holdup times are 41.6 days for xenon and 2.3 days for krypton, based upon a continuous input flow rate to the WGS of 0.99 m^3/h (0.58 scfm).

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 gaseous radwaste system 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 gaseous radwaste system 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.

- A radiation monitor before discharge to the ventilation exhaust duct.

The use of a gas cooler, a moisture separator, a guard bed, and delay beds is a common gas treatment system throughout the nuclear power industry.

3.3.2 HVAC Systems for Radiologically Controlled Areas

The **AP1000** NPP design uses several HVAC systems, which are identified in Table 3.3-2. The classification of working areas being served by the HVAC systems and the systems that are fitted with abatement systems are also shown in Table 3.3-2.

Detailed descriptions of the HVAC systems can be found in Chapter 23 of the PCSR (Reference 1-10). The HVAC systems which extract air or process from radioactively controlled areas are described in the following subsections.

3.3.2.1 Containment Recirculation Cooling System (VCS)

The Containment Recirculation Cooling System (VCS) recirculates and cools air within the containment during power operations and shutdown. This results in an energy savings and a reduction in waste generated in the form of used filters when compared to that which would arise from the level of once-through HVAC ventilation system. The air recirculated by this system is expected to contain some activity, mostly noble gases and some iodine. The VCS does not penetrate the containment boundary and thus does not give rise to any discharges to atmosphere.

During shutdown operations in the containment, local, filtered extract systems are available for particular operations where airborne activity may be generated. This reduces the potential for airborne activity to be released into the general containment atmosphere which the recirculation system could spread to other parts of the containment building.

The extracted air is continuously monitored for airborne activity which would give an early warning to workers in the containment of the presence of airborne activity.

3.3.2.2 Containment Air Filtration System (VFS)

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 postfilters. 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 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 postfilter bank, and an exhaust fan. The efficiencies of the individual filtration elements are in Table 3.3-3. 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.

3.3.2.3 Radiologically Controlled Area Ventilation System (VAS)

The **AP1000** NPP 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.

The auxiliary/annex building ventilation subsystem is routed to minimise 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 buildup 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 flow rate of 16.42 m^3s^{-1} (34,900 cfm).

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 flow rate of 5.52 m^3s^{-1} (11,700 cfm) before discharge via the plant vent.

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 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:

- Primary Sampling Room
- Chemistry Laboratory
- Fuel Handling Area 1
- Auxiliary Building Rail Car Bay/Filter Storage Area
- 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.

3.3.2.4 Health Physics and Hot Machine Shop HVAC System (VHS)

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).

3.3.2.5 Turbine Building Ventilation System (VTS) – Bay 1 Area

The Turbine Building Ventilation System (VTS) serves all areas of the turbine building. The HVAC systems serving the switchgear rooms, rectifier room, security rooms, and plant control system cabinet rooms do not have a credible source of radioactive contamination and are considered non-radioactive ventilation systems (see subsection 4.1.1.3). The Bay 1 area of the turbine building contains the reactor coolant pump variable-speed drives, CCS equipment (a nonradioactive system), and the Steam Generator Blowdown System (BDS). The BDS may be contaminated in the very unlikely event of concurrent fuel defects, steam generator leak, radiation monitor or BDS isolation failure, and a BDS leak. The general area of the turbine building is ventilated using about 850 m^3s^{-1} (1,800,000 cfm) exhausted direct to atmosphere through roof ventilators without abatement.

3.3.2.6 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, as follows:

- 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 (Reference 3-3).

All HEPA filter housings will be of the safe change type and supplied in accordance with NF0153/1 (Reference 3-4). Filter inserts will comply with AESS30/95100 (Reference 3-5).

3.3.2.7 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.

3.3.3 Condenser Air Removal System

Air in-leakage and non-condensable gases contained in the turbine exhaust steam are collected in the condenser. The condenser air removal system removes the air and non-condensable gases from the condenser during plant start-up, cool down, and normal operation from the steam side of the three main condenser shells and exhausts them into the atmosphere via the condenser air removal stack.

During normal operation and shutdown, the main condenser has no significant inventory of radioactive contaminants; thus the non-condensable gases and vapour mixture are discharged to the atmosphere are not normally radioactive. However, it is possible for the mixture to become contaminated in the event of a steam generator tube leak. Radiation monitors associated with the BDS, steam generator system (main steam), and the condenser air removal system provides the means to determine if the secondary side is radioactively contaminated. Upon detection of unacceptable levels of radiation, operating procedures are implemented to precipitate corrective action.

3.3.4 Air Emission Release Points

The point source release points to the atmosphere are shown in Figure 3.3-2. The main plant vent and ILW ventilation stack provide the only potential sources of gaseous radioactive emissions under normal operating conditions. Both these sources are subject to abatement (see Sections 3.3.2, 3.3.2, and 3.3.3). The other emission points identified in Figure 3.3-2 only act as sources of radioactive air emissions in abnormal conditions, such as a loss of coolant accident (LOCA) or primary-secondary tube leak failure.

The main plant vent is located on the side of the containment building. Data relating to the main plant vent is shown in Table 3.3-4.

The details of the Condenser Air Removal Stack are shown in Table 3.3-5.

No design data is currently available for the ILW store ventilation stack.

3.3.5 BAT Assessment for Gaseous Radwaste Treatment

3.3.5.1 BAT – Optimisation of Delay Bed Sizing

The carbon delay beds in the WGS 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 maximise 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. No failure mechanisms have been identified that could increase discharge flow rates high enough to suspend activated carbon particulates from the delay beds.

Each serpentine has four legs. The number of legs, and hence the volume of carbon in the delay bed, has been optimised by evaluating the radioactive releases (using the GALE code) expected as a function of the number of legs. Figure 3.3-3 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 Category 1 seismic building; the cost of purchase, installation, and decommissioning of the additional serpentine legs and the additional cost of activated carbon.

3.3.5.2 BAT – HEPA Filter Selection

The VFS, VAS (fuel handling area), and VRS will have HEPA filter housings capable of holding a range of different specification filters. Higher specification HEPA filters than those shown in Table 3.3-3 are available. However, these filters may increase differential pressure and have shorter replacement intervals than the specified filters. This would result in increased energy use on the extraction fans and a 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 cost of filters, cost of filter disposal, and filter performance can be evaluated.

3.3.5.3 BAT – Radiologically Controlled Ventilation Areas

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 ventilation system is emitted to the atmosphere via the plant vent without treatment. The advantage of this system is that the exhaust air filtration units of the VFS are not being used to filter uncontaminated air. This prolongs the life of the filters and charcoal adsorber and minimises the generation of LLW.

3.3.6 Air Emissions

3.3.6.1 Annual Air Emissions

The radioactive air emissions from the **AP1000** NPP following abatement comprise radioiodines, noble gases, and particulates. The annual releases are presented in Tables 3.3-6, 3.3-7, and 3.3-8, respectively. The data are based on proprietary calculations determined from the revised GALE Code (Reference 3-6). The emissions data are for annual average air emission; no account is taken of short term variability of emissions.

3.3.6.2 Monthly Air Emissions

The monthly radioactive discharges over an 18-month fuel cycle were estimated for an **AP1000** NPP in Reference 3-7. The 18-month cycle includes startup, shutdown, and maintenance.

The radionuclides considered in evaluating gaseous emissions were those from the nuclear power reactors listed in EU Commission Recommendation 2004/2/Euratom for discharges to the atmosphere and liquid discharges. However, some of the radionuclides were omitted due to insufficient source data. Those radionuclides included in the analysis are listed in Table 3.3-9.

Table 3.3-10 lists the total monthly discharges of radionuclides in gas from an **AP1000** NPP during an 18-month operating cycle. Tables 3.3-11 through 3.3-22 list radioiodine, noble gases, H-3, C-14, Ar-41, Co-60, Kr-85, Sr-90, I-131, Xe-133, Cs-137, and other particulates discharged in the gas, respectively.

Sources of gas discharges include the following:

- RCS
- Containment building
- Auxiliary building
- Turbine building
- Condenser air removal system.

The variability in gaseous discharges associated with the RCS over the fuel cycle is the same as the variability in liquid discharges from which they are derived. The non-RCS gas discharges are expected to be constant during the fuel cycle. The variability of the gaseous discharge over the 18-month fuel cycle is shown in Figure 3.3-4.

3.3.7 Comparison of AP1000 NPP Discharges with Existing Plants

The gaseous discharges from the **AP1000** NPP were compared with those from the following operating plants (Table 3.3-23):

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

These plants were selected for comparison to the generic **AP1000** NPP because South Texas 1, Braidwood 1, Cook 1, and Vogtle 1 are more recently built Westinghouse pressurised water reactors (PWRs) in the United States; Sizewell B is a PWR in the UK. When the values are normalised to an annual basis and 1000 MW output, the **AP1000** NPP has conservative but comparable discharges to the operating plants.

3.4 Liquid Radioactive Waste

The management of liquid radwaste is described in detail in Chapter 26 of the PCSR (Reference 1-10).

The WLS is designed to control, collect, process, handle, store, and dispose of liquid radwaste generated as the result of normal operation, including anticipated operational occurrences.

3.4.1 Sources

The WLS receives liquid waste from the following sources at flow rates shown in Table 3.4-1:

3.4.1.1 Reactor Coolant System Effluents

The effluent subsystem receives borated and hydrogen-bearing liquid from two sources: the RCDT and the CVS. The RCDT collects leakage and drainage from various primary systems and components inside containment. Effluent from the CVS is produced mainly as a result of RCS heat-up, boron concentration changes, and RCS level reduction for refuelling. The RCS effluents contain dilute boric acid at concentrations up to 2700 ppm. This borated water is the principal input in terms of volume and activity.

3.4.1.2 Floor Drains and Other Wastes with High Suspended Solids

Floor drains and other wastes are collected by various building floor drains and sumps, and routed to one of two waste holdup tanks, with a volume of 45 m^3 (12,000 gal) each. They potentially have high suspended solid contents.

3.4.1.3 Detergent Wastes

Detergent wastes coming from the plant hot sinks and showers as well as some cleanup and decontamination processes are routed to the chemical waste tank with a volume of 19 m^3 (5,000 gallons).

3.4.1.4 Chemical Wastes

Chemical wastes collected from the laboratory and other relatively small volume sources are transferred to the chemical waste tank. It may be mixed non-hazardous, hazardous, and radwastes or other radwastes with high dissolved-solids content. These wastes are generated at a low rate.

3.4.1.5 Steam Generator Blowdown

Steam generator blowdown is normally non-radioactive and is accommodated within the BDS. However, if steam generator tube leakage results in significant levels of radioactivity in the steam generator blowdown stream, this stream is redirected to the WLS for treatment before release. In this event, one of the waste holdup tanks is drained to prepare it for blowdown processing.

3.4.2 Storage and Containment of Liquid Radwaste

The liquid radwaste is collected in five tank systems:

- RCDT
- Effluent hold up tanks
- Waste hold up tanks
- Chemical waste tanks
- Monitor tanks (treated liquid radwaste, see subsection 3.4.3.6)

Details of these tanks, their locations, and their secondary containment can be found in Tables 3.4-2 and 3.4-3.

Sludge is not expected to accumulate significantly on the bottoms of liquid waste treatment tanks. These tanks are designed to minimise sludge formation by including provisions to slope the effluent hold-up tank from one end to the other into a dirt pan section where particulate material can collect; the vertical tanks are sloped to the low point where particulates will be collected. The discharge pump suction is taken from the dirt pan or low point for drawing waste water out of the tank, thus performing a self-cleaning action. The tanks are also fitted with oversized manways that allow for access in case any additional cleaning is necessary for tank maintenance or during decommissioning.

3.4.3 Liquid Radwaste System

The WLS is located in the nuclear island auxiliary building. A schematic of the WLS is shown in Figure 3.4-1.

3.4.3.1 Degasification of Reactor Coolant System Effluent

The input to the RCDT is potentially at high temperature. Therefore, provisions are made for recirculation through a heat exchanger for cooling. The tank is inerted with nitrogen and is vented to the WGS.

The cooled RCS effluents then pass to the vacuum degasifier to remove hydrogen and dissolved radiogases before storage in the three effluent holdup tanks, with a volume of 106 m^3 (28,000 gallons) each. The stripped gases are vented to the WGS.

The degasifier column is designed to reduce hydrogen by a factor of 40, assuming inlet flow of 22.7 m^3/h (100 gpm) at 54°C (130°F).

The contents of the effluent holdup tanks may be:

- Recirculated and sampled.
- Recycled through the degasifier for further gas stripping.
- Returned to the RCS via the chemical and CVS makeup pumps.
- Passed through the filtration and ion exchange treatment units of the WLS before being sent to the monitor tanks for discharge.

3.4.3.2 Pre-Filtration

The contents of the effluent holdup tanks and waste holdup tanks are normally passed through a treatment system comprising an upstream filter followed by four ion exchange resin vessels in series and a downstream filter.

A pre-filter is provided to collect particulate matter in the effluent stream before ion exchange. The unit is constructed of stainless steel and has disposable filter bags. The pre-filter has a nominal particulate removal efficiency of 90 percent for 25 µm (0.98 mils) particles.

The pre-filter is particularly important for removal of solids present in effluent collected from the floor drains which are directed to the waste holdup tanks.

3.4.3.3 Deep Bed Filtration

The deep bed filter is a stainless steel vessel containing a layered bed of activated charcoal above a zeolite resin. The activated charcoal provides an adsorption media for removal of trace organics and provides protection for the ion exchange resins from contamination with small amounts of oil from the floor drain wastes. Moderate amounts of other chemical wastes can also be routed through this vessel.

The top layer of activated charcoal collects particulates and, being less dense than the zeolite, can be removed without disturbing the underlying zeolite bed which minimises solid-waste production.

The lower layer of the deep bed filter is clinoptilolite zeolite which possesses an affinity for caesium (see Table 3.4-4).

3.4.3.4 Ion Exchange

Three ion exchange beds are provided following the deep bed filter. The ion exchange vessels are stainless steel, vertical, cylindrical pressure vessels with inlet and outlet process nozzles plus connections for resin addition, sluicing, and draining. The process outlet and flush water outlet connections are equipped with resin retention screens designed to minimise pressure drop. The design flow is 17 m^3/h (75 gpm). This capacity provides an adequate margin for processing a surge in the generation rate of this waste.

The media will be selected by the plant operator to optimise system performance according to prevailing plant conditions. Typically, the first bed will contain a cation exchange resin and the second two beds will contain mixed bed resins. Any of these vessels can be manually bypassed and the order of the last two can be interchanged, so as to provide complete usage of the ion exchange resin.

The ion exchange beds operate in the borated saturated mode. This means that the boric acid present in the reactor coolant effluent passes through the ion exchange beds without reduction in concentration.

The assumed decontamination factors (DFs) for the resin beds are shown in Table 3.4-4.

3.4.3.5 After Filter

This filter is provided downstream of the ion exchangers to collect particulate matter, such as resin fines. The unit is constructed of stainless steel and has disposable filter cartridges. The design filtration efficiency is 98 percent removal of 0.5µm (0.02 mils) particles.

3.4.3.6 Monitor Tanks

Treated effluent is discharged to the six monitor tanks located in the radwaste building. Information relating to the design of the monitor tanks and their secondary containment can be found in Tables 3.4-2 and 3.4-3, respectively.

Each tank has a capacity of 45 m^3 (12,000 gallons), giving a total storage capacity for treated effluent of 270 m^3 (72,000 gallons). This volume allows up to ~42 days storage during normal power operations when the average daily liquid radwaste release rate is ~8 m^3 (3 ft^3) per day (see Table 3.4-1). The storage period is reduced during the short periods associated with higher discharge rates resulting from boron dilution near the end of core life and during RCS heat-up following refueling.

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. If the effluent does not meet the permissible limits, it can be returned to a waste holdup tank or recirculated directly through the filters and ion exchangers.

A radiation monitor is located on the common discharge line downstream of the WLS monitor tanks. 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 discharge limits for radioactivity is pumped from the monitor tanks in a controlled fashion to the cooling water return from the CWS. The monitor tank pumps have a design flow rate of ~23 m^3/h (100 gallons/min). The once through cooling water flow rate is 136,000 m^3/h (600,000 gallons/min). It follows that the cooling water stream provides a substantial dilution of the discharged effluent before release to the environment.

3.4.3.7 Potential to By-pass Ion Exchange

Routine bypass of the WLS ion exchangers is not anticipated. However, the WLS is designed to be flexible and capable of handling a relatively wide range of inputs, including both high grade water (from reactor effluents) and low grade water (floor drains).

Each collection tank (effluent holdup tank, waste holdup tank) will typically be mixed and sampled prior to processing. Analysis of the sample from a specific batch of liquid will allow operator evaluation and determination of the optimum processing technique. The evaluation will include the chemistry of the batch, the radiological content of the batch, and the types and condition of the filters and ion exchange media in the WLS. This evaluation will be performed according to procedure.

When processing a batch of reactor effluents, all WLS ion exchangers and filters are anticipated to be in service. However, when processing floor drains from the waste holdup tanks, it is possible that some batches may be very low in radioactivity; an example would be following actuation of the fire water system in the radiologically controlled area of the plant, when a significant volume of uncontaminated fire water might be collected by the WLS. In this case, it may be acceptable and preferable to bypass one or more of the WLS ion exchangers, in order to maximise the life of the media, thereby minimising solid radwaste arisings and associated occupational radiation exposure.

The selection of WLS ion exchange vessels in and out of service is made through the alignment of manually-operated valves. These valves are opened and closed by an operator who works in a personnel access corridor, where the operator is well shielded from the radiological source of the demineralisers and filters. These valves are under administrative control to prevent an advertent bypass of demineralisers or sub-optimal treatment of waste.

In all cases, processed water downstream of the WLS ion exchanger/filter train is collected in a monitor tank, which is sampled prior to discharge to the environment. This sample will then be evaluated to ensure release goals are achieved, and serves as a final check that processing was appropriate.

3.4.3.8 Use of Mobile and Temporary Equipment

The WLS is designed to handle most liquid effluents and other anticipated events using installed equipment. However, for events occurring at a very low frequency, or producing effluents not compatible with the installed equipment, temporary equipment may be brought into the radwaste building mobile treatment facility truck bays. Any treatment of liquid waste by mobile or temporary equipment will be controlled and confirmed by plant procedures.

Connections are provided to and from various locations in the WLS to these mobile equipment connections. This allows the mobile equipment to be used in series with installed equipment, as an alternative to it with the treated liquids returned to the WLS, or as an ultimate disposal point for liquids that are to be removed from the plant site for disposal elsewhere.

The radwaste building truck bays and laydown space for mobile equipment, in addition to the flexibility of numerous piping connections to the WLS, allow the plant operator to incorporate mobile equipment in an integrated fashion.

Temporary equipment is also used to clean up the condensate storage tank if it becomes contaminated following steam generator tube leakage. This use of temporary equipment is similar to that just described, except that the equipment is used in the yard rather than in the radwaste building truck bays.

These provisions to utilise mobile and flexible technology properly interconnected with permanent systems allow for evolving state-of-the-art to be applied to waste processing throughout the life of the plant.

3.4.3.9 Detergent Waste

Detergent wastes are collected in the chemical waste tank. They have low concentrations of radioactivity and contain soaps and detergents not compatible with the ion exchange resins. If their activity is low enough, they 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.

The mobile equipment would comprise a concentration step (e.g., evaporation or cross-flow microfilter) in order to reduce the volume of waste to the extent necessary to allow an encapsulation plant to immobilise the concentrate in a cementitious grout. The encapsulation would typically take place in a 200l drum. The completed waste packages would be loaded into the half height ISO containers (HHISOs) for shipment to the Low Level Waste Repository (LLWR). After processing by the mobile equipment, the condensed evaporator distillate or filtrate would be transferred to a waste holdup tank for further processing in the WLS or transferred to a monitor tank for sampling and discharge. The commercial availability of mobile equipment using this type of volume reduction and waste stabilisation techniques has been confirmed through companies offering specialist contract services.

3.4.3.10 Chemical Waste

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 in subsection 3.4.3.9. If onsite processing capabilities are not suitable, processing can be performed using mobile equipment or the waste water can be shipped offsite for processing.

3.4.4 BAT Assessment for Liquid Radwaste Treatment

3.4.4.1 Ion Exchange vs. Evaporation

A comparison of typical flow sheets for evaporation and ion exchange is shown in Figure 3.4-2. The relative merits of ion exchange and evaporation has been evaluated by Westinghouse and the results are reported in Table 3.4-5.

In Europe, many nuclear reactors are located on major rivers and not on coastal sites. These locations have less capacity to accept discharges of borated effluent. It is common for these reactors to be equipped with evaporators to minimise the radioactive liquid and boric acid discharges. However, the standard **AP1000** NPP design does not have evaporators based on considerations shown in Table 3.4-5 and because it contradicts the **AP1000** NPP overriding principle of safety and simplicity.

Compared to the traditional evaporator-based WLS, 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 costs.

At Sizewell B, two evaporators were constructed; one for recycling boric acid from the RCS, and one for abatement of liquid radwaste. Evaporation of liquid for either purpose is not currently considered best practicable means (BPM) or ALARP, and the evaporators are not in use at Sizewell. 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 with only a minor impact on seawater boron concentrations and marine ecosystem dose rates (see Sections 4.2.5.2, 5.2, 5.3, and Reference 3-8).

It is concluded that, for the UK generic coastal site, the proposed WLS treatment system using ion exchange beds and filtration rather than evaporation is BAT.

3.4.4.2 Enriched Boric Acid vs. Natural Boric Acid

The **AP1000** NPP is designed not to require an enriched 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 percent B-10 compared to 20 percent 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.

The same amount of B-10 isotope is required as a moderator irrespective of whether it is provided in the enriched or natural form. This means that the production mechanism for tritium involving neutron activation of B-10 is similar for the enriched and natural forms of boric acid.

Another important formation mechanism for tritium is the neutron activation of lithium. In principle, the use of enriched boric acid reduces the amount of lithium hydroxide required for pH control by a factor of three. Such a reduction in lithium concentration would reduce the potential for tritium formation (see Section 3.2.7). 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 2.6.3 and 3.2.8)

- Use of lithium enriched in the Li-7 isotope rather than natural lithium in the lithium hydroxide for pH control. This substantially reduces the potential for tritium formation from neutron absorption by Li-6 present in natural lithium hydroxide (see Section 3.2.7).

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.

3.4.4.3 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 3.2). In particular, the use of mechanical shim control rather than chemical shim control during normal load follow operations substantially reduces the quantities of boron used 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 radwaste produced.

The use of natural boron rather than costly enriched boron (see subsection 3.4.4.2) 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 2700 mg/l (0.023 lb/gallon) of boron and that the effluent is discharged at 23 m^3/h (100 gpm) into in the seawater cooling return flow of 136,000 m^3/h (600,000 gpm), the boron concentration in the cooling return would be increased by 450 µg/l (3.8 lb x 10^{-6} lb/gallon). At an average liquid radwaste effluent flow rate of 8 m^3/d (2000 gallons/day), 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 7000 µg/l (6 x $10-^5$ lb/gallon) for the protection of saltwater life (Reference 3-9) and that discharge of boron to seawater meets BAT and ALARP criteria.

3.4.4.4 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 0.5 µm (0.02 mils) 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 (~8 m^3/day, 1.5 gpm) associated with the WLS. The filters are readily replaceable and treated as LLW.

Cross-flow filtration techniques of microfiltration and ultrafiltration potentially offer increasingly effective particulate removal efficiency (ranging from 0.1 µm to < 0.001 µm, 0.004 mils to < 0.00004 mils) compared to cartridge filtration. All these techniques use membrane processes that segregate a liquid that permeates through the membrane producing 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 an 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. Ceramic membranes are expensive.
- 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 radwaste 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.

3.4.5 Liquid Effluent Discharges

3.4.5.1 Annual Liquid Effluent Discharges

The annual release of radioactive effluents is quantified in Table 3.4-6. The data are based on proprietary calculations determined from the revised GALE Code (Reference 3-6). The emissions data are for annual average water discharges and take no account of short term variability of releases.

3.4.5.2 Monthly Liquid Effluent Discharges

Table 3.4-7 lists the total monthly discharges of radionuclides in liquid from an **AP1000** NPP during an 18-month operating or fuel cycle. Tables 3.4-8 through 3.4-18 list monthly discharges of tritium, non-tritium, C-14, Fe-55, Co-58, Co-60, Ni-63, Sr-90, Cs-137, Pu-241, and other particulates in the liquid, respectively. The discharged liquid is generated by the RCS and non-RCS sources.

As the fuel burn-up increases over the fuel cycle, less boron is needed in the reactor cooling water. This adjustment in boron concentration is achieved by bleeding borated water from the RCS and replacing it with unborated water. A larger volume of water needs to be removed each month; therefore, the radioactive discharges increase each month of the cycle. This results in the variability in activity in liquid discharges from the RCS shown in Tables 3.4-7 and 3.4-8 and Figure 3.4-3.

Non-RCS liquid volume comes from various sources in the plant:

- Inside containment
- Sample drains
- Reactor containment cooling
- Spent fuel pool liner leakage
- Miscellaneous drains
- Hot shower
- Hand wash
- Equipment and area decontamination
- Chemical wastes

The volume of liquid from non-RCS sources is expected to be almost constant during each month of the cycle; therefore, the radioactive non-RCS discharges are expected to be constant, as shown in Tables 3.4-7 and 3.4-8.

3.4.6 Comparison of AP1000 NPP Liquid Discharges with Other European Pressurised Water Reactors

The predicted liquid discharges from the **AP1000** NPP are compared in Tables 3.4-19 and 3.4-20 with published discharges from European nuclear reactors operating over the period 1995-1998 (Reference 3-10). The tritium data in Table 3.4-19 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-cooled reactors (AGRs), but higher than discharges from boiling water reactors (BWRs). In practice, it is very difficult to reduce discharges of tritium. The radiological impact of tritium is relatively small and the radiological impact of discharges is usually very low.

Table 3.4-20 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 3-10). 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 considerably lower than the average Magnox, AGR, BWR and Sizewell B discharges.

The liquid discharges from the **AP1000** NPP were compared with those from the following operating plants (Table 3.4-21):

- South Texas 1
- Braidwood 1
- D. C. Cook 1
- Vogtle 1
- Sizewell B

These plants were selected for comparison to the generic **AP1000** NPP because South Texas 1, Braidwood 1, D. C. Cook 1, and Vogtle 1 are more recently built Westinghouse PWRs in the United States; Sizewell B is a PWR in the UK. When the values are normalised to an annual basis and 1000 MW output, the **AP1000** NPP has lower discharges than the other plants.

3.5 Solid Radioactive Waste

3.5.1 Overview of the Integrated Waste Management Strategy

Management of radwaste is being planned with the expectation that the LLW, ILW, and spent fuel waste streams will be capable of being disposed in Nuclear Decommissioning Authority (NDA) facilities. Waste forms and treatment processes have been selected with this principle in mind. To ensure the waste packages are disposable, Radioactive Waste Management (RWM[2]) compliant containers have been designated.

Westinghouse has initiated discussions regarding the disposability of radwaste with the EA and the UK NDA, and will continue this dialogue. Westinghouse has provided the NDA with information relating to the wastes that are expected to arise over the lifetime of an **AP1000** NPP (Reference 3-11). In 2009, the NDA used this information as the basis for a disposability assessment report covering ILW and HLW generated by the **AP1000** NPP (Reference 3-12). This report stated:

"On the basis of the GDA Disposability Assessment for the **AP1000** [NPP], RWMD [currently referred to as RWM] has concluded that, compared with legacy wastes and existing spent fuel, no new issues arise that challenge the fundamental disposability of the wastes and spent fuel expected to arise from operation of such a reactor. This conclusion is supported by the similarity of the wastes to those expected to arise from the existing PWR at Sizewell B."

Uncertainties and risks relating to the achievement of this strategy will be identified as the strategy is implemented and managed by documenting and discussing them with the utility customers and the EA. The main uncertainty, risk, and assumptions in this strategy are associated with radioactive waste and spent fuel disposal in line with the NDA. At this time, the NDA is not able to provide information on the spent fuel packages they will accept; therefore, Westinghouse will assume that current practices for spent fuel packaging remain

[2] During the early part of the AP1000 NPP GDA process Radioactive Waste Management (RWM) was referred to as Radioactive Waste Management Directorate (RWMD). Except when included in a reference title or number, RWM is used throughout this revision, although the previous organization name may have been used when the activity being described took place.

acceptable once the **AP1000** NPP is built and operating. This includes container designs and sizes, and acceptable waste forms (spent fuel assemblies). Westinghouse is communicating with the NDA about these issues.

Nearby facilities, where and when available, will be used to the extent practical to minimise the environmental impact of transport. During site operations, communications will be maintained to assess onsite and offsite interdependencies; for example, those between the **AP1000** NPP and offsite disposal facilities.

Figure 3.5-1 is a pictorial representation of the **AP1000** NPP waste management strategy. This strategy is integrated to take into account all matters that might have a bearing on the management of radwaste and spent fuel, including the following:

- Waste minimisation
- Avoidance of unnecessary introduction of waste into the environment
- Waste characterisation and segregation
- Collection and retention of data on the waste and waste packages
- Consideration of options in a BAT assessment
- Communications with interfacing facilities and stakeholders
- Assurance that steps in the management of waste are compatible
- Characterisation of risks and uncertainties

3.5.1.1 Waste Minimisation

Waste minimisation is an inherent part of waste management. The basic **AP1000** NPP design principles minimise the creation of radwaste during operations and decommissioning. **AP1000** NPP was designed with fewer valves, pipes, and other components so less waste will be generated during maintenance activities (repair and replacement) and decommissioning.

Waste generation will be minimised in an **AP1000** NPP due to material selection. For example, the level of cobalt in reactor internal structures is limited to below 0.05 weight percent, and in primary and auxiliary materials to less than 0.2 weight percent (see Chapter 26 of the PCSR (Reference 1-10)). This limits the activation of the metal components. Surfaces, including steel wall and floor surfaces, will be sealed to prevent penetration and to facilitate decontamination. Also, during operation and maintenance, waste will be minimised by using best industry practices (for example, limiting the amount of material brought into containment).

At the end of the storage period, the radiological characteristics of ILW packages will be reviewed to assess if sufficient decay has occurred to allow them to be reclassified as LLW.

As described in subsection 3.5.1.7 laundry items will be reused, and therefore will not be considered waste.

3.5.1.2 Waste Generation

Waste inventory estimates have been developed based on operational experience with existing plants to ensure this strategy is consistent with the waste expected to be generated (see Sections 3.5.3 and 4.3).

Waste and discharges generated over the operational period of the **AP1000** NPP will be systematically identified. They will be managed, treated, handled, and stored onsite by designing appropriate facilities and demonstrating that they are compatible with the **AP1000** NPP (see Sections 3.5.7 to 3.5.10 and Section 4.3). Potential locations for the waste management facilities on the generic **AP1000** NPP site are provided in Section 2.3. Transportation and disposal of wastes will use the appropriate methods.

Management of wastes generated during decommissioning according to UK requirements is included (see Section 3.5.10).

3.5.1.3 Radioactive Waste Treatment

Waste will be categorised for treatment as conventional or radioactive, and within radioactive as LLW, ILW, and HLW. Solid waste will be characterised and segregated using equipment discussed in Section 3.5.7.

A range of appropriate options for waste treatment, such as evaporation, drying, incineration, and cement encapsulation, was considered at an optioneering workshop that included utility participation and its respective operational experiences managing radwaste. The results of this workshop were documented, and the chosen options substantiated; for example, cement encapsulation of solid ILW and compaction of compactable LLW (see Section 3.5.5). Each step in the management of radwaste will be compatible with all other steps, including pre-treatment, treatment, storage, disposal, handling, and onsite and offsite transport.

The design of **AP1000** NPP and the process used for selection of radwaste treatment equipment provide flexibility to the operator. The solid radwaste technologies considered are currently proven technologies that have been used in the UK and elsewhere; for example, a mobile cement encapsulation system for the treatment of ILW. However, this equipment is not required until 18 months after the plant becomes operational. Before that time, the utility could decide to reassess solid waste treatment to consider the latest developments in technology and different packaging arrangements. Also, because the cement encapsulation system is mobile and designed to be easily removed, the utility could choose to replace the equipment after operations have started and when newer technology has been developed.

Figure 3.5-2 is a pictorial representation of the **AP1000** NPP solid waste management.

3.5.1.4 On-Site Storage

The waste management strategy requires LLW to be shipped for disposal routinely according to schedules agreed to by the plant operator and the UK NDA. A facility will be available to store LLW during periods (up to 2 years) when waste cannot be received by the LLW disposal facility.

An onsite storage facility for arisings of ILW from operation of the plant has been designed (see subsection 3.5.8.2), because the disposal of ILW from any new nuclear power stations to a future geologic repository is unlikely to occur until late this century. The design of the storage facility addresses how the ILW is transported to the facility and how waste is handled within it. Waste storage procedures will be developed to ensure safety, transportability, stock control, and ability to retrieve waste packages are all in place prior to dispatching the first batch of ILW to the ILW store.

3.5.1.5 Waste Transportation

Westinghouse has discussed with the NDA the use of approved containers for transport and disposal of ILW, as well as LLW (see Section 3.5.9). The containers and packages with shielding will be acceptable for transport and disposal.

3.5.1.6 Spent Fuel Management

Spent fuel management is discussed in subsection 3.5.7.3. After spent fuel is removed from the reactor, it will be stored in the fuel storage pool. Details of the fuel storage pool can be found in Section 6.10 of the PCSR (Reference 1-10). Because spent fuel is not expected to be reprocessed, a facility for dry spent fuel storage for the operational period of the plant and beyond is being designed (see subsection 3.5.8.3).

Each step in the management of spent fuel will be compatible with all other steps, including storage, disposal, handling, and onsite and offsite transport. The spent fuel will be safely disposed, at appropriate times and in appropriate ways; Westinghouse has contacted the NDA to ensure spent fuel will be packaged in a manner acceptable to the NDA.

3.5.1.7 Laundry

There is no laundry facility in the standard **AP1000** NPP design. It is assumed that overalls and gloves would be laundered at offsite facilities. One example of such facilities is the Unitech laundry facility in Wales. It offers a lease programme wherein an operator would lease the garments, use them, package them in two categories based on contamination level, and ship them to the laundry facility. Unitech would supply clean replacement garments. The returned garments would be washed (and rewashed if necessary) before reuse. The leased garments would not form part of the LLW inventory, unless they became so contaminated that they could not be shipped to Unitech.

3.5.2 Radioactive Waste Classification

3.5.2.1 Low Level Waste

Low level radwaste is defined as "radioactive waste having a radioactive content not exceeding 4 GBq/te (0.1 Ci/ton) of alpha or 12 GBq/te (0.4 Ci/ton) of beta/gamma activity" (Reference 3-13).

Very Low Level Radioactive Waste (VLLW) is a sub-category of LLW. This is broken down into "low volume" and "high volume" VLLW, each of which has its own definition (Reference 3-14):

1. Low volume VLLW (LV-VLLW)

   In the case of low volumes ("dustbin loads") Low Volume Very Low Level Waste:

   "Radioactive waste which can be safely disposed of to an unspecified destination with municipal, commercial or industrial waste ("dustbin" disposal), each 0.1 m^3 (3.5 ft^3) of waste containing less than 400 kilobecquerels (kBq) (10.8 µC) of total activity or single items containing less than 40 kBq (1.08 µCi) of total activity." For waste containing carbon-14 or hydrogen-3 (tritium):

   - in each 0.1 m^3 (3.5 ft^3), the activity limit is 4,000 kBq (108 µCi) for carbon-14 and hydrogen-3 (tritium) taken together;

   - for any single item, the activity limit is 400 kBq (10.8 µCi) for carbon-14 and hydrogen-3 (tritium) taken together.

Controls for disposing of this material after removal from the premises where the wastes arose, are not necessary.

2. High Volume VLLW (HV-VLLW)

In the case of bulk disposals – High Volume Very Low Level Waste:

"Radioactive waste with maximum concentrations of four megabecquerels per tonne (MBq/t) [0.1 mCi/ton] of total activity which can be disposed of to specified landfill sites. For waste containing hydrogen-3 (tritium), the concentration limit for tritium is 40 MBq/t [1 mCi/ton]. Controls on disposal of this material, after removal from the premises where the wastes arose, will be necessary in a manner specified by the environmental regulators".

3.5.2.2 Intermediate Level Waste

ILW is radwaste with radioactivity levels exceeding the upper boundaries for LLW:

- Alpha emitters greater than 4 GBq/te (0.1 Ci/ton).
- Beta/gamma emitters greater than 12 GBq/te (0.4 Ci/ton).
- Waste that does not need radiological self-heating to be taken into account in the design of storage or disposal facilities. IAEA guidance is that ILW thermal power is below about 2 kW/m^3 (Reference 3-15).

3.5.2.3 High Level Waste

HLW is waste in which the temperature may rise significantly as a result of radioactivity, so that this factor has to be taken into account in designing storage or disposal facilities. IAEA guidance is that HLW thermal power exceeds about 2 kW/m^3 (Reference 3-15).

3.5.3 Radioactive Solid Waste Generation

The sources of solid waste generated in the **AP1000** NPP are summarised in Table 3.5-1. A detailed breakdown of the wastes can be found in Appendix A. The solid radioactive waste estimates are best, realistic estimates. A major source of information for their calculations was consultations with experienced personnel who have designed the **AP1000** NPP and worked on existing plants.

The annual solid radioactive waste production varies due to the 18-month fuel cycle and the different schedules for replacement consumables and equipment maintenance. However, the annual average waste volumes are presented in Table 3.5-2.

3.5.3.1 Low Level Waste

LLW includes dry active wastes, general trash, and mixed wastes as a result of normal plant operation, including anticipated operational occurrences. LLW waste will generally contain plastics, paper, metallic items, clothing, rubber, filters, redundant equipment, glass, and wood.

Under normal operations, LLW also includes Condensate Polishing System (CPS) resins which are non-radioactive. However, under abnormal conditions (e.g., steam generator tube failure), there may be a transfer for primary circuit activity into the secondary circuit. The amount of activity transferred will be very small and the activity of the CPS resin will not exceed the limits for LLW. If situations arise when the activity of the CPS resins exceed the threshold for LLW disposal, then they will be treated as ILW resins.

The quantities of LLW generated by the **AP1000** NPP are summarised in Table 3.5-1.

3.5.3.2 Intermediate Level Waste

ILWs are mainly comprised of spent ion exchange resins, activated carbon, and used filters. The production of theses wastes is intermittent and associated with replacement and maintenance procedures.

The quantities of ILW generated by the **AP1000** NPP are summarised in Table 3.5-1. The quantities of the ILW filter and resin wastes are based on contact with high coolant activity associated with 0.25 percent fuel failure. This is the bounding design case and, in reality, modern PWR fuel is considerably more reliable than this failure rate, and it is possible that the filter and resin activity levels will be low enough for them to be disposed as LLW. However, for the purpose of conservatism and to bound accident scenarios, the design case estimated volumes of ILW have been used.

Operating experience-based estimates on the quantities of actinides in dry solid radwaste is provided in Table 3.5-3. The actinides are of interest because they generally have long half lives. However, the activity due to actinides is at very low concentrations.

3.5.3.3 High Level Waste

The HLW produced by the **AP1000** NPP is spent fuel. Spent nuclear fuel is used fuel from a reactor that is no longer efficient in creating electricity, because its fission process has slowed. However, it is still thermally hot, highly radioactive, and potentially harmful.

Operational strategies can influence the amount of spent fuel and radioactivity of the spent fuel. The amount of spent fuel discharged as a function of time is primarily determined by the energy production rate (overall capacity factor including outages) and the discharge burn-up limit.

The reference 18-month equilibrium cycle feeds (and discharges) 64 fuel assemblies every 18 months. On average, this means that approximately 43 assemblies per year are discharged and stored in the spent fuel pool storage area. Each fuel assembly consists of 264 fuel rods in a 17 x 17 square array. The fuel rods consist of uranium dioxide ceramic pellets contained in cold-worked and stress relieved **ZIRLO** tubing, which is plugged and seal-welded at the ends to encapsulate the fuel.

The quantities of HLW generated by the **AP1000** NPP are summarised in Table 3.5-1.

3.5.4 Waste Minimisation

Waste minimisation, characterisation, and segregation are central to both establishing and updating a radwaste inventory and optimising waste management in line with the waste management hierarchy (Figure 3.1-1). Opportunities for waste minimisation, characterisation and segregation will be considered in all stages of waste management, including design, construction, commissioning, operation, decommissioning, storage, and disposal.

3.5.4.1 Low Level Waste

The basic **AP1000** NPP design principles minimise the creation of radwaste during operations and decommissioning:

- Good housekeeping
- Operating procedures
- Segregation
- Volume reduction
- Sealed surfaces (including steel wall and floor surfaces) to prevent penetration and to facilitate decontamination
- Limiting the amount of material brought into containment
- Training all staff allowed to enter radiation controlled areas (RCAs)
- Provision of waste facilities immediately outside of the RCAs, for the disposal of unnecessary packaging materials
- Provision of tool stores within the RCAs to prevent contamination of clean tools brought in from outside
- Testing filter performance to ensure filters are only replaced when necessary
- Provision of radwaste advice on radiation work permits

3.5.4.2 Intermediate Level Waste

On the **AP1000** NPP, ILW is minimised by the following activities:

- Optimum operation of the reactor in terms of power generation per tonne of fuel
- Select fuel with minimal potential for fuel defects, thereby minimising the radioactive isotope contamination of the primary cooling water circuit. This will reduce the load being treated by the ion exchange resin beds and hence the volume of ILW
- Fuel is received and carefully inspected for any imperfections
- Minimisation of plant shutdowns
- Use of gray rods for mechanical shim control
- Use of wet winding coolant pumps eliminates seal leaks and creation of radioactive waste water
- Selection of materials of construction with a composition low in cobalt
- Use of zinc addition for corrosion control

- Selections of ion exchange media to give an optimum DF, which will minimise the number of ion exchange media changes required and reduce the waste volume

- Flexibility in routing effluent through the different ion exchange beds to optimise resin uptake

- Monitoring differential pressure across the filters and filter performance to ensure that filters are only replaced when necessary

- Segregation procedures to prevent dilution of ILW streams by mixing them with LLW streams

- Formulation trials to determine an optimum blend ratio producing the optimum number of waste packages

- Operating procedures

3.5.4.3 High Level Waste

Westinghouse works with utilities to optimise the important characteristics of the fuel reload design in order to meet the utility needs. This is typically a balance between the economics of the fuel management and the amount of operationing margin available from the design, given any specific characteristics of the individual plant sites.

The fuel economics and the amount of spent fuel are closely correlated. Both are optimised when the fuel cycle is designed with fuel being discharged from the reactor as close as is reasonable to the licensed discharge burn-up limit. The current licensed limit for Westinghouse fuel is 62,000 MWD/MTU on the lead rod maximum burn-up. Considering inter-assembly power variations and variations of assembly power in assemblies within the same batch, this translates into a batch average burn-up of approximately 50,000 MWD/MTU.

The proposed operational regime will help to support the design intention of minimising spent fuel. The reference **AP1000** NPP equilibrium cycle design is an 18-month cycle. The cycle is based on an assumed 97 percent capacity factor and a 21-day refueling outage. This provides a cycle length of approximately 510 effective full power days. Many alternative cycling schemes are possible and have been studied to demonstrate the flexibility of the **AP1000** NPP design. However, the 18-month design is used as the reference for most work and provides close to the optimum (lowest) overall electrical production costs.

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 versus 43). For a plant lifecycle of 60 years, this translates to a generation of 2517 or 2653 spent fuel assemblies for an annual or 18-month cycle, respectively. However, depending on the cost of the extra outage every 3 years – combined with the cost of replacement power during the outage, the impact of outage length on average capacity factor, etc. – this may not be the most overall economically efficient operation of the core. Westinghouse works with its utility customers to determine the customer’s priorities before core loading patterns are defined. The vast majority of Westinghouse customers choose the longer fuel cycle.

3.5.5 BAT – LLW and ILW

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 (DBD), Westinghouse, and included representatives from several utilities (Reference 3-16).

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.

A prerequisite was that the options must comply with the following:

- Waste must be treated and handled in accordance with current LLW waste acceptance criteria (WAC) (Reference 3-14) and any future ILW repository WAC.

- ILW and LLW containers must meet RWM's recommendations on package design and the requirements of the RWM Generic Waste Package Specification (Reference 3-18).

3.5.5.1 Initial Option Screening

Initially, an optioneering process was carried out to identify a set of radwaste treatment options.

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 UK (a scale from one to five). This criterion is essential, because an option that is not fully tested in the UK 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 a 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 code based on these attributes (see Table 3.5-4). Red options were eliminated from further optioneering process if they do not meet the requirements from this initial screening. Amber options which show some potential, but are not necessarily proven for radwaste, would only be considered further if fully acceptable (green) options were not available. The outcome of the option screening is shown in Table 3.5-5.

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

3.5.5.2 Evaluation of Screened LLW Treatment Options

The initial option screening exercise for LLW identified the potential complete solution processes shown in Figure 3.5-3 (Reference 3-16).

The complete solutions comprise:

1. Sorting

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

2. Size Reduction

   Because the LLWs are a mixture of wastes, it is difficult to specify the best option at this stage of assessment. All of the size reduction options are of low cost technologies and are considered as potential approaches.

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 UK. In principle, "Controlled Oxidation" presents benefits in reducing the volume of wastes, 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, "Compaction," as the most suitable option.

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 3-14) for the LLW repository, the selected option is "No Immobilisation."

3.5.5.3 Evaluation of Screened ILW Organic Resin Treatment Options

The potential complete solutions that passed screening for ILW organic resin are shown in Figure 3.5-4 (Reference 3-16).

To evaluate these options further, a scoring workshop was held on 4th June 2008 with 21 attendees from Aker Solutions, DBD, Westinghouse, Rolls Royce, Vattenfall, RWE, Ulecia Endessa, and Iberdrola. Table 3.5-6 shows the set of agreed criteria for the scoring process which included the technical, safety, environmental, and economic aspects. 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 3-16). The results are shown in Table 3.5-7.

1. Dewatering Stage

Table 3.5-7 shows that no dewatering had the highest total weighted score, but also had the lowest primary waste score. The second highest score was 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.

2. Volume Reduction Stage

Table 3.5-7 shows that no compaction has the highest total weighted score for volume reduction. Compaction leads to higher cost and introduces additional safety hazards and operability issues. Hence, the option of "No Compaction" is selected.

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 in Table 3.5-7, is the selected option.

4. Immobilisation

In Table 3.5-7 the option of vitrification is eliminated as it emerges as the most costly and least beneficial option with the lowest total weighted score. 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 are cement encapsulation, which has the highest score, and polymer encapsulation, which has the second highest score. Cement encapsulation has the following advantages (Reference 3-17):

- This technology is widely used internationally and is well known as a practical and economic approach.
- Radwastes are transported safely.
- Meets requirements of the Nirex/RWM Generic Waste Package Specification (Reference 3-17).
- 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.

3.5.5.4 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 3-16). Figure 3.5-5 shows that simple encapsulation options require the lowest 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 3.5-6 shows that vitrification, wet oxidation, and controlled oxidation become more cost-effective when the predicted lifetime disposal costs are taken into consideration. This is because these technologies result in volume reduction rather than the volume increase 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 the ILW resins (organic) radwaste system are 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 future technologies. Mobile encapsulation facilities also enable the system to be moved to other locations, increasing its potential for utilisation.

3.5.5.5 Evaluation of Screened ILW Filter Treatment Options

The potential complete solutions for ILW filter treatment are shown in Figures 3.5-7 (Reference 3-16).

The complete solutions for ILW filter treatment comprise:

1. Size Reduction and Volume Reduction

   It was preferred that the treatment options for ILW filters be similar to the ILW organic resin treatment options. This is due to the low amount of wastes in this category and the capital and maintenance cost benefit associated with using common equipment. This led to the conclusion that neither size nor volume reduction are needed because the filters can be accommodated within the disposal package without size reduction.

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.

3.5.5.6 BAT ILW and LLW Radwaste Conclusion

Figure 3.5-8 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 (e.g., 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 available, 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.

The use of mobile systems for the processing functions permits the use of the latest technology and avoids the equipment obsolescence problems experienced with installed radwaste processing equipment. The most appropriate and efficient systems may be used as they become available.

3.5.5.7 Comparison with Other Practices

1. Sizewell B

   British Energy Generation Limited (BEGL) carried out a review of the control and impact of the discharge and disposal of radwaste at Sizewell B in 2005 (Reference 3-20). The review was prepared as a submission of information to the EA to enable their review of Radioactive Substances Act 1993 authorisations. In 2006, the EA published their decision document and authorisations regarding future regulation of disposals of radwaste at UK nuclear power stations (Reference 3-21). This review commented on the best practicable environmental option (BPEO) and BPM proposed by British Energy for the control of radwastes from Sizewell B.

   Table 3.5-8 presents the BPEO issues identified for solid wastes at Sizewell B and compares them with the practices proposed for the **AP1000** NPP. The table also provides a summary of the EA 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 EA. The **AP1000** NPP generic design does not have an on-site incinerator.

2. 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 3.5-9 identifies how LLW and ILW solid waste is handled at several European NPPs. More details of the European practices can be found in utility presentations attached in Appendix A of UKP-GW-GL-026, "**AP1000** Nuclear Power Plant BAT Assessment" (Reference 3-22).

   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 3.5.5.6. 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 RWM compliant waste packages (Reference 3-18 and 3-43). 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) or generic specifications 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 generic specifications.

3.5.6 BAT – HLW

Spent fuel created by nuclear power stations may either be disposed of or recycled by reprocessing to separate out the useful uranium and plutonium. Reprocessing of spent fuel has a number of advantages in that it maximises the recovery of the energy from the fuel, can improve energy security by providing a source of fresh fuel, and reduces the amount of HLW. However, there are a number of disadvantages including production of separated plutonium, which requires long-term storage, production of other waste streams, production of regulated effluent discharges, and the requirement to transport spent fuel and other nuclear materials.

The Government has concluded that, in the absence of any proposals from the industry, that any new nuclear power stations that might be built in the UK should proceed on the basis that spent fuel will not be reprocessed and that plans for, and financing of, waste management should proceed on this basis (Reference 1-2). Consistent with this approach, 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 (see subsection 3.5.7.3). The spent fuel pool has a capacity for 10 years of storage, which 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.

Fuel transfers and early storage are all carried out underwater; however, for long term storage of the fuel in canisters, it is preferred to store fuel under an inert gas atmosphere to minimise the corrosion issues associated with long term wet storage.

Underground dry storage has the advantage of providing greater levels of shielding and providing a more secure solution with respect to aircraft impact and other catastrophic events. The disadvantages of underground storage relate to control of groundwater issues and flood risk. However, these issues can be overcome by careful design of the storage system and evaluation of site-specific issues at the site-specific design stage.

For the generic site application, Westinghouse is proposing a dry spent fuel storage system to be stored inside an underground cylindrical cavity (see Holtec system in subsection 3.5.8.3).

3.5.7 Waste Treatment

The waste treatment methods for the generic site design are based on the waste minimisation and BAT assessment techniques described in Sections 3.5.4 to 3.5.6.

3.5.7.1 Low Level Waste

Incoming wastes from other radiation/contamination controlled areas will be brought into the facility and temporarily stored in a buffer/marshalling area within the radwaste building (see Figure 3.5-9). The wastes will then be sorted under controlled conditions (e.g., glove boxes).

This segregation is performed to ensure that no unnecessary waste is disposed of in the LLWR.

Contaminated material arising from equipment replacement parts, tools, and other metallic, plastic, or cloth parts from outage operations would normally be classified as LLW. However, in the event that they were initially classified as ILW, the **AP1000** NPP includes provisions for the decontamination of these types of materials to a LLW category, if feasible.

Within the radwaste building the segregated waste is dispatched to one of a number of possible treatments including:

On-site:

- Decontamination
- Cutting/shredding
- Compacting
- Immobilisation (if justified)

Off-site:

- Reconditioning and re-use (e.g., personal protective equipment)
- Incineration (e.g., oil)

Wherever possible, waste items will be decontaminated to the extent that allows free release and handling as conventional waste.

Compactable items will be sorted and compacted in drum to reduce packed volumes. Non-compactable items will be cut into pieces to allow packing into drums. Full drums will be assayed with a Low Resolution Gamma Spectroscope (LRGS) and placed into HHISO containers and when full, HHISO containers can be stored on site in the LLW buffer store prior to shipment to the national LLW repository. The treatment route for LLW is shown schematically in Figure 3.5-10. A preliminary safety statement (Reference 3-24) and hazard study (Reference 3-25) have been completed for the operations being carried out in the radwaste building.

Under normal circumstance, CPS resins are not radioactive and will be sent to licensed incineration facilities for disposal. However, under abnormal conditions, for example, steam generator tube failure, there may be a transfer of primary circuit activity into the secondary circuit. The amount of activity transferred will be very small and the activity of the CPS resin will not exceed the limits for LLW. Incineration will remain the disposal route so long as the activity of the CPS resin is within the WAC for the incineration facility. There may be situations when the activity of the CPS resin exceeds the WAC for incineration (although still LLW) and in these cases, the resin will be encapsulated into a compliant container, e.g., 220l drum, using mobile plant and equipment operated by specialist sub-contractors. The encapsulated packages are then placed into HHISO containers and are ultimately disposed of at the national LLW repository. The disposal route for CPS resin is shown schematically in Figure 3.5-11.

Waste oil will normally be non-radioactive. However, in the event of the oil becoming contaminated with radioactivity, it will be shipped to a suitable incinerator (e.g., the Tradebe Incinerator at Fawley) for incineration. Westinghouse have carried out a review of this contaminated oil against the conditions of acceptance of this incinerator and have shown that they can be met. Any waste oil that exceeds the radioactivity acceptance thresholds of the incinerator will be solidified by a mobile plant prior to disposal to the LLWR (see Figure 3.5-12).

The only large solid radwaste item that could be generated during the operation period of the **AP1000** NPP is the steam generators. It is expected that the steam generators will be radioactive but with an activity level that will fall into the LLW category. It is intended that these items are handled as they arise, are size-reduced, and are decontaminated to the extent practicable. To facilitate this disposal route, a temporary facility will be erected, for example, a tent with mobile HVAC equipment and connections to **AP1000** NPP power, water, and air systems, as necessary. The area currently allocated for temporary waste handling facilities for the GDA site is identified as Item 33 in Figure 2.3-2. Decontaminated pieces that are no longer radioactive will be released to conventional waste handling facilities for recycle or disposal. Decontaminated pieces that remain radioactive will be wrapped before placement into HHISO containers and sent for disposal at the LLWR.

The reactor pressure vessel head is only likely to be removed during decommissioning (see Section 3.5.10). However, if it needed to be replaced during the operating lifetime, it would be treated in a similar manner to steam generators.

3.5.7.2 Intermediate Level Waste

ILW will be treated and disposed according to the schematic in Figure 3.5-13. Further details can be found in Reference 3-3.

ILW will be segregated on an **AP1000** NPP site in the following ways:

- Ion exchange and spent activated carbon activity is monitored, and once the activity breakthrough level has been reached the material is transferred to spent resin tanks. Only ILW resins and activated carbon will be sent to spent resin tanks.

- Replacement filter cartridges will be tested for activity and any ILW filters will be placed in an RWM approved box. This segregation is an operationally-controlled procedure which will occur in the auxiliary building.

The spent ion exchange resin, or activated carbon, will be made passively safe by being immobilised in a cementitious grout formulation within an RWM approved drum (References 3-18 and 3-43). The spent filters will be immobilised in a cementitious grout formulation within an RWM approved box (Reference 3-3).

The waste encapsulation will be carried out using a Mobile Encapsulation Facility on a campaign basis (see Figure 3.5-14). The Encapsulation Facility will be stored in the radwaste building when not in use and moved to the **AP1000** NPP Auxiliary Building Railcar Bay for the campaign. Locating the mobile cement encapsulation plant within the auxiliary building allows the necessary extraction systems to be connected to the VAS which vents via the monitored Plant Vent. The encapsulation will take place by remote operation at three stations within the shielded Encapsulation Facility:

- Station 1 – Lidding and unlidding
- Station 2 – Grout capping /Curing/QA
- Station 3 – Fill/Grouting/Mixing

The spent ion exchange resin or activated carbon will be pumped to the fill station from the spent resin storage tank where it will be mixed with cementitious grout. Any water that is decanted from the ion exchange resins will be returned to the ion exchange resin tanks.

The grout formulation recipe will be determined during formulation trials. These trials will take into account the operational experience of similar plants close to the required date to ensure that a BAT solution is achieved and an approved and accepted formulation is used. The materials most commonly used to encapsulate UK ILW are hydraulic blends of Ordinary Portland Cement, with either Blast Furnace Slag or Pulverised Fuel Ash. The trials will confirm the optimum volume of ion exchange resin to minimise the number of waste packages, while maintaining the long term structural integrity of the waste package. It is assumed that the RWM package will contain 25 volume percent resin for the purpose of calculating the ILW waste volumes for storage.

Once mixed, the stabilised waste will be allowed to cure and the RWM containers will be subject to quality checks including:

- the Dartometer test to confirm the cement has set
- external swabbing to confirm that the waste package is free from contamination
- activity monitoring using a High Resolution Gamma Spectroscope (HRGS) to produce a "fingerprint" of the activity concentrations within the waste packages

The RWM containers are moved to and from the Mobile Encapsulation Facility by a self-propelled trailer and between stations of the Encapsulation Facility by a conveyor. Once the cement has cured and the containers have passed QA checks, they will then be transported to the ILW storage building.

The final specification for the processing plant including the grout mix and operating procedures will be prepared no later than during the first cycle of operation.

The encapsulation plant will be operational prior to the second plant outage as it may be decided by the operator to move resins from the resin tanks before the second outage. Thus, revised estimates of resin activity will be prepared during Cycle 1 based on early operational experience.

3.5.7.3 High Level Waste

HLW waste will be managed and disposed according to the schematic in Figure 3.5.15.

The spent fuel assemblies are initially stored in the spent fuel cooling pond to allow radioactive decay to occur and decay heat to be removed (see Section 6.10 of the PCSR, Reference 1-10). The cooling pond, containing borated water as a neutron absorber, is located in the auxiliary building.

The spent fuel assemblies are held in racks that contain integral neutron absorbing material, and are designed to ensure adequate spacing to ensure the appropriate degree of subcriticality is achieved. The spent fuel pool racks can hold approximately 617 fuel assemblies which equates to approximately 10 years of spent fuel.

The spent fuel pool cooling system is provided to remove decay heat which is generated by stored fuel assemblies from the water in the spent fuel pool. This is done by pumping the high temperature water from within the fuel pool through a heat exchanger, and then returning the water to the pool.

The spent fuel pool is equipped with a purification system which removes radioactive corrosion products, fission product ions, and dust to maintain low spent fuel pool activity levels during plant operation and to maintain water clarity during all modes. Two mixed bed type demineralisers are provided to maintain spent fuel pool purity, each one sized to accept the maximum purification flow from its respective cooling train. Downstream of the demineraliser in the purification branch lines, a spent fuel pool filter is provided to collect small particles and resin fines.

After the spent fuel has been stored to allow sufficient cooling, it will be transferred to dry cask storage.

3.5.8 Interim Storage

3.5.8.1 Low Level Waste Storage

There are a number of storage areas within the Radwaste Building, as shown in Figure 3.5-9. The ‘Waste Accumulation Room’ (room #50351) has a buffer/marshalling area in the North-East corner, used to store incoming LLW packages and bags. The ‘Mobile Systems Facility’ (room #50350), has areas on the North and South walls used to store clean, empty RWM 3 m^3 (100 ft^3) drums and boxes, secondary containment vessels (SCV) and 200 l (55 gallon) drums. The maximum storage capacities of these areas are:

- 10 off 3 m^3 (100 ft^3) packages (5 stacks of 2 high, mixture of boxes and drums)
- 2 off SCV (1 stack of 2)
- 27.3 m^3 (964 ft^3) for 200 l (55 gal) drums (7m (23 ft) L x 1 m (3 ft) W x 3.9 m (13 ft) H)
- 89.6 m^3 (3,160 ft^3) for incoming LLW (8 m (26 ft) L x 4 m (13 ft) W x 2.8 m (9.2 ft) H)

The buffer/marshalling area can also be used to store higher activity LLW, rejected from the LLW treatment processes. The waste will be stored for a temporary period, until:

- The activity decays to a level that allows handling with the installed equipment
- It is determined that the waste should be handled and treated as ILW

The LLW buffer store is a covered area comprising a concrete hard standing area with a steel-framed canopy. The buffer store is designed for HHISO containers that have been filled in the radwaste building. Standard handling machinery (fork truck) will be used to move the containers from the radwaste building to the buffer store.

The radwaste building will only store the HHISO containers currently being filled. All filled HHISO containers will be transported to the buffer store that provides storage for two years of waste arisings (Reference 3-3):

3.5.8.2 Intermediate Level Waste Storage

ILW (spent filters, spent ion exchange resin, and activated carbon) is stored within suitably contamination zoned and shielded areas within the auxiliary building prior to treatment in the mobile encapsulation plant. Once the ILW is encapsulated in RWM waste packages (3 m^3 boxes (100 ft^3) and 3 m^3 drums (100 ft^3)), the boxes and drums will be transported to an on-site ILW store where they will be stored until a national ILW repository becomes available. It is estimated that between 15 and 29 of these RWM waste packages will be produced each year (Reference 3-26). A total of 1116 RWM waste packages is predicted for the 60-year plant life (Reference 3-26).

The ILW store proposed for the generic site is a reinforced concrete structure with 1 m (3 ft) thick walls that can be extended at appropriate intervals to suit new ILW waste arisings (see Figure 3.5-16).

The ILW store incorporates a receipt area with waste package assay equipment and a shielded vault serviced by a certified nuclear crane. Office and administration space, and an equipment room housing HVAC and electrical and mechanical equipment are provided in an annex to the main store building.

The packages will be transferred into the receipt area via a shielded door and then transported by the crane to a position in the store vault determined using the tag information. The package position will be recorded in the control log for ease of future retrieval. The packages will be placed in the store layer by layer, to limit the potential topple height of stored packages. The layers will be constructed from the furthest point of the store working back to the receipt area. The chosen transfer path for placing/retrieving a package will be such to minimise the effective drop height. The store design and operation will be such to enable retrieval and visual examination of individual packages. Close circuit television (CCTV) within the import/inspection area will be used to facilitate this.

The first phase of construction will provide an ILW store suitable for 20 years of ILW production (372 No. x 3 m^3 (100 ft^3 RWM packages). Extensions to the store will be sized to suit future waste arisings and are expected to be added in 20-year increments. The ILW store will be designed for a total inventory of 60 years of operational waste arisings from one **AP1000** NPP unit. The ILW store has a 100-year design life and could be used to retain ILW after **AP1000** NPP is decommissioned and until the national ILW repository becomes available (Reference 3-3).

Every ILW waste package will be "finger printed" using an HRGS within the ILW store to monitor its activity level before it is transferred to the ILW Store Vault. Waste package inventory records will be completed according to the required regulations to maintain an inventory record of each waste package and its location within the ILW Store Vault. Because all waste packages sent to the store will be ILW and are expected to remain ILW, no segregation will be required within the store vault.

It is envisaged that when an ILW waste repository becomes available within the UK, the ILW waste packages will be removed from the store and monitored again with the HRGS before being sent to the repository along with its associated waste package inventory record. The same facilities used during placement of the ILW packages into the store will be used to ship the ILW packages, that is, the area at one end of the store building and the store building crane.

If the HRGS result of a package indicates the radionuclides in the package have decayed such that the package could be LLW, the package will be temporarily placed in an LLW storage area. The LLW disposal facility will be contacted to ensure the appropriate records are prepared for LLW disposal at that time.

All ILW packages will be visually inspected during handling and, if defects or external damage is found, the package will be flagged as "rogue" and placed/sealed within a SCV prior to storage. A "rogue" package might arise from:

- Overfill of a package during encapsulation, causing spillage and contamination to the outer surface.

- Malfunction during lidding, causing an unsealed package.

- Corrosion/damage to the package, resulting in a containment failure.

Any "rogue" package will be transferred to a SCV. The SCV is a container of similar design to the RWM packages that is sized to fit over the RWM 3 m^3 (100 ft^3) box/drum. If an ILW package is found to be "rogue" as a result of QA inspections, it will be inserted into an SCV, lidded, and positioned in the store as normal. A small batch of empty SCVs will be stored within the radwaste building until required.

3.5.8.3 High Level Waste Storage

The spent fuel system proposed for the generic site is a dry storage system and comprises:

- flask loading equipment within the **AP1000** NPP.
- suitable flask transportation vehicles and equipment.
- a seismically qualified below ground storage facility.

The flask handling equipment within the **AP1000** NPP can accommodate a variety of flask types.

Westinghouse is offering Holtec International's underground dry spent fuel dry storage system, the HI-STORM 100U System (see Figure 3.5-17), as an option for dry spent fuel storage management (References 3-27, 3-28, and 3-29). However, the spent fuel pool within the **AP1000** NPP provides sufficient capacity for up to 10 years of storage. This allows the **AP1000** NPP operator time to select other options for the spent fuel storage system or defer selection for a period to allow new techniques to be incorporated, if appropriate.

The HI-STORM 100U System is a vertical, ventilated dry spent fuel storage system. Holtec and Westinghouse have confirmed that the Holtec equipment can fit in the areas of the **AP1000** NPP that need to be traveled to transfer spent fuel from the spent fuel pool to the underground storage area. The system consists of three primary components:

1. HI-STORM 100U underground vertical ventilated module (VVM)

The VVM provides for storage of a multi-purpose canister (MPC) in a vertical configuration inside a subterranean cylindrical cavity entirely below the top-of-grade (Figure 3.5-18a). The principal function of the VVM structure is to provide the biological shield and cooling facility.

The MPC storage cavity is defined by the cavity enclosure container (CEC), consisting of the container shell integrally welded to the bottom plate. In the installed configuration, the CEC is interfaced with the surrounding subgrade for most of its height except for the top region where it is girdled by the top surface pad. The CEC is a closed bottom, open top, thick-walled cylindrical vessel, which has no penetrations or openings. Thus, groundwater has no path for intrusion into the interior space of the MPC storage cavity.

Corrosion mitigation measures commensurate with site-specific conditions are implemented on below-grade external surfaces of the CEC. All external and internal surfaces of the VVM are coated with an appropriate surface preservative. An optional concrete encasement around the coated external surface of the CEC may be added to control the pH at the CEC-to-subgrade interface. A corrosion allowance equal to 3 mm (1/8 in) on the external surfaces of the VVM in contact with the subgrade is nevertheless assumed in the structural evaluations.

The closure lid is a steel structure filled with shielding concrete and incorporates a specially designed air ventilation system (see Figure 3.5-18b).

2. MPC, which each contain 24 spent fuel assemblies

   The MPC and HI-TRAC in the HI-STORM 100U System are 100 percent identical to those in the Holtec aboveground system that has been in use for several years.

   The MPC is a single package equally suitable for onsite storage, transport, and permanent disposal in a future repository. The MPC is constructed entirely of stainless steel alloy materials with the exception of the Metamic, a fixed neutron absorber, which is contained within the canister for criticality control. The fuel assembly basket contained within the MPC is a honeycomb multi-flanged plate weldment that forms the square fuel cells in the basket. There is complete edge-to-edge continuity between the continuous cells that provides an uninterrupted heat transmission path, making the MPC an effective heat rejection device.

   The top end of the Holtec MPC uses a closure system that includes a lid equipped with vent and drain ports used to remove air and water and backfill the canister with inert gas [helium] (see Figure 3.5-17f) and a closure ring used to provide a redundant confinement boundary for the MPC lid. The vent and drain ports are covered, helium leak checked, and seal-welded before installing the closure ring. The closure ring is a circular ring that is edge welded to the canister outer shell and lid (see Figure 3.5-17g). The MPC lid provides sufficient structural capability to permit the loaded MPC to be lifted by threaded holes in the MPC lid.

   The heat from the fuel stored in the core region of the basket is removed by the thermosiphon (circulatory) action (Figure 3.5-18c). As a result, high heat rate fuel (gamma radiation emitted is proportional to the heat emission rate from the fuel) can be placed in the core region, surrounded by the cooler (and older) fuel in the periphery. This approach, known as "regionalised" storage, is extremely effective in mitigating the dose emitted from a basket in the lateral direction. The effectiveness of regionalised storage in reducing dose derives from the fact that almost 95 percent of the dose from the basket comes from the peripheral fuel; the inner region fuel is almost entirely noncontributory to the dose.

3. HI-TRAC transfer cask, which holds the MPC during loading operations

   HI-TRAC is the acronym for Holtec International transfer cask or "shuttle cask" for HI-STORM 100U. HI-TRAC is a slim cylindrical cask with removable bottom and top lids. HI-TRAC can be mounted on top of a HI-STORM 100U overpack to deliver or retrieve an MPC (see Figures 3.5-17i and 3.5-17k). HI-TRAC is a heavy-walled steel and lead cylinder with a water jacket attached to the exterior of the vessel. The main structural function of HI-TRAC is provided by carbon steel. Water and lead provide the main neutron and gamma shielding functions, respectively.

A total of 130 MPCs will be filled over the 60-year plant life of the plant containing 3104 fuel assemblies (Reference 3-26).

The spent fuel will remain within the HLW store for a determined period of time to enable the heat generating capacity of the spent fuel assemblies to reduce enough to meet the required standards for the national Geological Disposal Facility (GDF). At the proposed high burn-up rates, RWM has estimated that dry cask storage for up to 100 years may be necessary in order to allow it to cool sufficiently to be transferred to an approved RWM disposal canister for final disposal. However, Westinghouse expects the repository design may be reconsidered on the basis of current world-wide expectations from spent fuel characteristics which would allow for shorter dry cask storage periods.

3.5.9 Transportation and Disposal

3.5.9.1 Low Level Waste

LLW is bagged, collected manually, and transported to areas of the waste accumulation room. The waste is packaged in HHISO containers. A fork truck specifically designed for container handling will be used to move the HHISO containers from the radwaste building to the buffer store. HHISO containers will be handled within the radwaste building by the overhead crane.

The HHISO container will be transported by road to the LLWR at Drigg for final disposal. A preliminary application has been made to LLWR to confirm their acceptance of LLW from an **AP1000** NPP. The Acceptance in Principle (D1) forms obtained from the LLWR are presented in Reference 3-30 and contain activity estimates based on calculations from Reference 3-31.

636 HHISOs are predicted to be produced over the 60-year plant life (Reference 3-26).

3.5.9.2 Intermediate Level Waste

During transport to the ILW store, the waste package (drum or box) will be placed in an overpack which will provide shielding in order to limit exposure to operators or the public. A self-propelled trailer will be used to move the waste packages to the ILW store along designated routes. Inside the ILW store, the packages will be placed and recovered using an overhead crane (see Figure 3.5-16).

Shipment from the ILW store will only take place once a national repository is available. 1116 RWM waste packages are predicted to be produced over the 60-year plant life (Reference 3-26).

3.5.9.3 High Level Waste

Figure 3.5-19 shows a schematic of the process for the transportation and disposal of HLW.

HLW packages will be transported from the spent fuel cooling pond in their dry cask storage Holtec MPC canister to the HLW store using an appropriate transport vehicle. Two types of transport vehicle are proposed for the Holtec system: a low profile transporter and the HI-TRAC system (see Figures 3.5-17h and 3.5-17i, respectively).

The Holtec MPC is one option that can be used for the interim storage, transport, and final disposal of the **AP1000** NPP spent nuclear fuel. There are various other systems under development which could also be used for the interim storage, transport, and/or disposal. It may also be possible to place the **AP1000** NPP spent fuel directly into a RWMC disposal canister, once removed from the cooling pond, if the canister is designed for such a purpose.

Once the spent fuel assemblies have reached the acceptable limits for heat generation (typically 100 years) they will be transported from their dry cask storage to the national GDF once it is available. During transport, each waste package will be placed in an overpack to provide radiation shielding and also to ensure the integrity of the waste during a road accident. The total weight of the waste package will be within appropriate limits for transport on UK roads when necessary. It is envisaged that transport of packaged spent fuel would be undertaken using a Disposal Canister Transport Container. This is an RWM transport container concept that provides two layers of shielding (Reference 3-12):

- adjacent to the canister, a stainless steel gamma shield with a radial thickness of 140 mm and 50 mm (2 inch) at each end of the canister,

- surrounding the gamma shield, a 50 mm (2 inch) thick neutron shield made of high neutron capture material "Kobesh."

At present, there is no operator for the national GDF and RWM is acting as a "repository Licensee surrogate," providing input to government strategy for ILW and HLW management. The repository will need to be available for spent fuel at the end of operating life after the required cooling period.

The NDA have completed a disposability assessment of **AP1000** NPP spent fuel to satisfy the requirements of the GDA (Reference 3-12). This assessment assumed that spent fuel would be overpacked for disposal. Under this concept, spent fuel would be sealed inside durable, corrosion-resistant disposal canisters manufactured from suitable materials, which would provide long-term containment for the radionuclide inventory. The exact long-term disposal canister material and design remains to be confirmed, but candidates include copper or steel canisters with an additional cast-iron insert to provide additional mechanical strength. The current RWM disposal canisters do not allow ILW rod cluster control assemblies and certain other core components (e.g. burnable poisons and thimble plugs) to be included in the canisters. Westinghouse asked RWM to consider the option of disposing these within the spent fuel assemblies as practiced elsewhere in the world to minimise handling and to avoid production of orphan wastes.

Depending upon the features of the selected disposal canisters, it may be necessary to encapsulate **AP1000** NPP spent fuel in the preferred canister. The particular details of any required encapsulation plant are not fully determined due to uncertainties in the GDF requirements. However, developed processes exist in other countries, for example:

- In Germany, a pilot encapsulation plant has operated successfully repairing defective spent fuel casks for over 10 years (Reference 3-32). In this process, the spent fuel is removed from the CASTOR® cask used for interim storage and encapsulated in a Pollux® cask which is also used as the final repository cask. The plant has operated safely for a number of years and has successfully encapsulated spent fuel and the risks associated with these operations, and the required technologies are well understood.

- In Sweden, the disposal method determined is to encapsulate the spent fuel in copper canisters and embed the filled canisters within bentonite clay at a depth of 500 metres (1600 ft) in the crystalline bedrock of the GDF (Reference 3-33). This provides 3 separate environmental isolation barriers (canister, bentonite clay, and bedrock) for the spent fuel and prevents contamination from getting into groundwater.

For the GDA, it is assumed that any HLW conditioning facility in the UK will use similar technologies taking advantage of the experience gained at facilities elsewhere in the world. It is expected that the repackaging of spent fuel will take place at a central location as outlined in the Nirex repository concept (Reference 3-34).

The NDA disposability assessment (Reference 3-12) ultimately concluded that the characteristics of spent fuel from an **AP1000** NPP (with 65 GWd/TU burn-up) are consistent with those from the Sizewell B PWR. When compared with legacy wastes, there are no new issues that challenge the fundamental disposability of the wastes expected to arise from operation of such a reactor. The additional repository storage space required to accommodate spent fuel from the **AP1000** NPP is not excessive, and is not significantly affected by a specific reactor or fuel design.

3.5.10 Decommissioning Waste

The basic **AP1000** NPP design principles minimise the creation of radwaste during operations and decommissioning. The **AP1000** NPP was designed to have fewer valves, pipes, and other components so less waste will be generated during maintenance activities such as repair and replacement (see Figure 3.5-20). Also, less waste mass will be generated during decommissioning. As discussed in Chapters 26 and 27 of the PCSR (Reference 1-10), the level of cobalt in reactor internal structures is limited to below 0.05 weight percent, and in primary and auxiliary materials to less than 0.2 weight percent. This limits the activation of the metal components. Surfaces, including steel wall and floor surfaces, will be sealed to prevent penetration and to facilitate decontamination. Also, during operation and maintenance, waste will be minimised by using best industry practices, for example, by limiting the amount of material brought into containment.

More details on decontamination can be found in Ref. 3-40 and the decommissioning plan is described in more detail in Ref. 3-41.

The wastes generated during decommissioning comprise the following:

- Small volume components

  The types of small volume components at decommissioning are shown in Appendix A4 and include various electrical equipment, filters, electrodeionisation units, and skids. These wastes are classified as LLW.

- Large volume components

The quantities of ILW and LLW large volume component wastes generated during decommissioning are shown in Appendix A3 and summarised in Table 3.5-10. The waste includes the reactor head, which is not expected to require replacement during the operational period of the **AP1000** NPP.

It is assumed that the reactor vessel is disposed intact and the vessel is not decontaminated from its ILW classification. No benefit would be gained from decontaminating the reactor vessel since most of the dose would come from activation of vessel materials. Decontamination of the reactor systems could be accomplished with some piping modifications to bypass the reactor vessel, and the use of existing filter and demineraliser vessels.

- Waste from decontamination operations

  System decontamination operations produce ILW during the various system purification steps (e.g. spent resins and spent filter cartridges).

- Dry active waste

  The compactable dry active waste created during decommissioning operations (e.g. rags, overalls, gloves, and packaging) is LLW and is estimated to be 135 m^3 (4,770 ft^3) per year. This volume can be reduced five fold to 27 m^3 (950 ft^3) per year using a low force compactor. Assuming an accumulated decontamination operation of three years, the additional dry active LLW generated during decommissioning would be 81 m^3 (2,860 ft^3) (see Table 3.5-10). The use of super compaction could reduce this number further.

- Demolition waste

  The radioactive demolition waste (rubble) is identified in Appendix A6. Only a very small fraction of demolished wastes is likely to be considered LLW, with the majority being non-radioactive or made non-radioactive during the decontamination process.

  The estimate of LLW demolition waste includes 2165 m^3 (76,460 ft^3) of concrete and 158 m^3 (5,580 ft^3) of steel (see Appendix A6).

  The LLW demolition waste includes wastes from the demolition of five modules within containment (CA01 to CA05). The only significant civil structure with the potential to become activated is CA-04. The other four major modules in containment (CA-01, CA-02, CA-03, and CA-05) could become contaminated, but are unlikely to become activated. The modules are steel structures, some with concrete filled wall sections, which would be cut into transportable pieces with little volume increase. The modules are designed with exposed concrete and steel surface finishes that will prevent penetration and facilitate decontamination.

  The demolition waste also includes 199 m^3 (7,030 ft^3) of concrete surrounding module CA04 in the vicinity of the core. This represents a 1.5 m (5 ft) thickness of concrete around the reactor vessel cavity which may contain enough activation products to be treated as LLW. Beyond this thickness, the concrete is essentially VLLW (Reference 3-35). Since this is a solid chunk of concrete, the actual packaged LLW volume will be two to three times greater or about 400 to 600 m^3 (14,000 to 21,000 ft^3).

The total ILW decommissioning waste associated with large volume components and waste from decontamination operations is estimated to be about 800 m^3 (28,000 ft^3). The total

LLW waste associated with decommissioning large and small volume components, compactable dry active waste and demolition waste is estimated to be about 5500 - 6000 m^3 (194,000 ft^3 – 212,000 ft^3).

A typical schematic for the treatment of decommissioning waste is shown in Figure 3.5-21. The management of decommissioning waste is being planned with the expectation that the LLW, ILW, and spent fuel waste streams will be capable of being disposed in NDA facilities.

3.5.11 Comparison of Waste Volumes from the AP1000 NPP and Other UK NPPs

A comparison of the waste volumes generated by the **AP1000** NPP and other UK NPPs is presented in Table 3.5-11. The data for other plants is extracted from the UK 2007 radioactive waste inventory (Reference 3-36). The data is normalised to the annual electricity production.

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Table 3.1-1

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 e.g. 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 e.g., PPE/metal components, -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 (e.g., 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 RWM compliant packages e.g. 3 m^3 box/drum*	• Optimise both radioactive and non-radioactive impacts to reduce the environmental footprint of the facility *Use of* wet winding *coolant pumps eliminates seal leaks and creation of waste, Bunds, collection sumps are incorporated to locally retain leaks and spills, 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 are used to capture radionuclides from the soluble to the solid phase in a compact and energy efficient manner i.e. 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 their consequences for the environment *Sealed containment and shield around reactor pressure vessel, catalytic hydrogen recombiners in the containment ventilation system, trisodium phosphatein baskets use of* wet winding *coolant pumps eliminates seal leaks and creation of waste, bunds, collection sumps are incorporated 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 RWM compliant, ILW store incorporates handling equipment and package inspection bays allowing 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*

Table 3.3-1

AP1000 NPP ESTIMATED OPERATIONAL GASEOUS RADWASTE ARISING FROM SYSTEM OPERATIONS

System	Waste Description	Waste Level	Physical/Chemical Description	Estimated Quantities					
				Normal Volume		Maximum Volume		Volume for Life of Plant	
				cm^3/min	m^3/h	cm^3/min	m^3/h	ft^3	m^3
WGS	RCDT drains	LLW	Hydrogen, nitrogen, and fission gases	0	0	1.05	0.000063	1,170	33.14
WGS	CVS shim bleed (gas)	LLW	Hydrogen, nitrogen, and fission gases	45.7	0.002742	81.6	0.004896	50,904	1441

Table 3.3-2

ABATEMENT PROVISIONS FOR HVAC SYSTEMS

Area	Ventilation System	Area Classification [Ref 3-39]	Ventilation Abatement Provisions
Nuclear Island Nonradioactive	VBS	White	No filtration
Annex/Auxiliary Building Nonradioactive	VXS	White	No filtration
Diesel Generator Building	VZS	White	No filtration
Containment	VCS	N/A	No discharge outside containment.
Containment	VFS	Amber high risk	High efficiency and HEPA filtration
Health Physics and Hot Machine Shop - Gloveboxes	VHS	Red	HEPA filtration
Health Physics and Hot Machine Shop – Machine tools	VHS	Amber high risk	HEPA filtration
Health Physics and Hot Machine Shop – Remaining space	VHS	Green Low risk	No filtration but diversion to HEPA and charcoal filtered standby system (VFS) on High radiation signal*
Radwaste Building	VRS	Amber high risk	HEPA filtration*
Turbine Building – Bay 1 area	VTS	Green Low risk	No filtration
Turbine Building – Remaining space	VTS	White	No filtration
Auxiliary/Annex Building Radiologically Controlled Area – Fuel handling area	VAS	Amber Low risk	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	Amber Low risk	No filtration but diversion to HEPA and charcoal filtered standby system (VFS) on High radiation signal
Active and Passive Spent Fuel Pool Exhaust	VAS	Amber Low risk	HEPA filtration

Key to Area Classification [Ref. 3-39]:

WHITE means a clean area free from radioactive contamination, whether surface or airborne.

GREEN means an area which is substantially clean. Only in exceptional circumstances is airborne contamination such that provisions must be made for its control.

AMBER means an area in which some surface contamination is expected. In some cases, there will be a potential for airborne contamination such that provision must be made for its control.

RED means an area in which contamination levels are so high that there is normally no access without appropriate respiratory protection.

Table 3.3-3

SPECIFICATION OF CONTAINMENT FILTRATION SYSTEM ELEMENTS

Filter System Parameter	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)	10.7 m x 2.0 m x 1.7 m (35 ft x 6.5 ft x 5.6 ft)			
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 dioctyl phthalate (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 3.3-4

MAIN PLANT VENT RELEASE POINT DATA

Parameter	Value	Comments
Stack Height	74.926 m	Plant vent
Plume Rise	6.7 m	Under neutral atmospheric conditions
Release Height	81.626 m	Sum above
Vent Dimensions	2.025 m x 2.311 m	Rectangular stack
Volumetric Flow Rate	38.13 m^3s^{-1}	
Nominal Discharge Velocity	8.15ms^{-1}	
Exhaust Temperature	285 – 315 K	Depending on outside air temperature
Distance to Site Boundary	200 m	
Nearby Building Height	70 m	Reactor Building
Buildings Surface Area Wall	3000 m^2	Reactor Building

Table 3.3-5

CONDENSER AIR REMOVAL STACK RELEASE POINT DATA

Parameter	Value	Comments
Stack Height	38.4 m	Turbine building vent
Plume Rise	1.4m	Under neutral atmospheric conditions
Release Height	39.8m	Sum above
Vent Internal Diameter	0.3048m	Circular 12” diameter stack
Volumetric Flow Rate	0.6 m^3s^{-1}	Estimated
Nominal Discharge Velocity	8.2ms^{-1}	
Exhaust Temperature	285 – 315 K	Depending on outside air temperature
Distance to Site Boundary	200 m	
Nearby Building Height	70 m	Reactor building
Buildings Surface Area Wall	3000 m^2	Reactor building

Table 3.3-6

EXPECTED ANNUAL RELEASE OF AIRBORNE RADIOIODINES TO THE ATMOSPHERE

Nuclide	Activity Release, GBq/y					
	Waste Gas System	Building/Area Ventilation			Condenser Air Removal System	Total Release
		Containment Building	Auxiliary Building	Turbine Building		
I-131	7.4E-03	1.9E-02	1.8E-01	2.4E-03	9.6E-04	2.1E-01
I-133	1.1E-02	7.4E-02	2.6E-01	7.4E-04	3.0E-03	3.5E-01
Total Airborne Radioiodine						5.6E-01

Table 3.3-7

EXPECTED ANNUAL RELEASE OF RADIOACTIVE NOBLE GASES, TRITIUM, AND CARBON-14 TO THE ATMOSPHERE

Nuclide	Activity Release (1), GBq/y					
		Building/Area Ventilation			Condenser Air Removal System	Total Release
	Waste Gas System	Containment Building	Auxiliary Building	Turbine Building		
Kr-85m	4.6E-01	1.4E-01	1.6E+01	8.5E-04	7.8E+00	2.4E+01
Kr-85	3.0E+03	1.1E+01	5.2E+01	2.9E-03	2.6E+01	3.1E+03
Kr-87	negl.	4.4E-02	1.7E+01	2.6E-04	2.2E+00	1.9E+01
Kr-88	6.7E-03	1.0E-01	1.8E+01	9.6E-04	8.5E+00	2.7E+01
Xe-131m	1.1E+03	3.1E+01	1.8E+02	9.3E-03	8.1E+01	1.4E+03
Xe-133m	3.6E-02	6.7E+00	7.4E+01	4.1E-03	3.5E+01	1.1E+02
Xe-133	2.4E+02	8.9E+01	6.3E+02	3.3E-02	2.9E+02	1.3E+03
Xe-135m	negl.	6.7E-02	1.3E+02	7.0E-03	5.9E+01	1.9E+02
Xe-135	negl.	3.1E+00	1.7E+02	2.9E-02	2.6E+02	4.4E+02
Xe-137	negl.	negl.	3.4E+01	1.8E-03	1.6E+01	4.8E+01
Xe-138	negl.	2.9E-02	5.9E+01	3.3E-03	2.9E+01	8.9E+01
Total Noble Gas						6.7E+03

Tritium Release via Gaseous Pathway(2) (TBq/y) = 1.8
C-14 Released via Gaseous Pathway (TBq/y) = 0.606(3)
Ar-41 Released via Gaseous Pathway (TBq/y) = 1.3

Notes:
1. Values less than 1 microcurie (3.7E+4Bq) are considered to be negligible, but their values are included in the totals.
2. Tritium release based on Westinghouse TRICAL computer code.
3. C-14 from Westinghouse calculation APP-WLS-M3C-056, Rev. 0, 2009.

Table 3.3-8

EXPECTED ANNUAL RELEASE OF RADIOACTIVE PARTICULATES TO THE ATMOSPHERE

Nuclide	Activity Release (1), GBq/y				
	Waste Gas System	Building/Area Ventilation			Total
		Containment Building	Auxiliary Building	Fuel Handling Area	
Cr-51	negl.	negl.	1.2E-04	6.7E-05	2.3E-04
Mn-54	negl.	negl.	negl.	1.1E-04	1.6E-04
Co-57	negl.	negl.	negl.	negl.	negl.
Co-58	negl.	9.3E-05	7.0E-04	7.8E-03	8.5E-03
Co-60	negl.	negl.	1.9E-04	3.0E-03	3.2E-03
Fe-59	negl.	negl.	negl.	negl.	negl.
Sr-89	negl.	4.8E-05	2.8E-04	7.8E-04	1.1E-03
Sr-90	negl.	negl.	1.1E-04	3.0E-04	4.4E-04
Zr-95	negl.	negl.	3.7E-04	negl.	3.7E-04
Nb-95	negl.	negl.	negl.	8.9E-04	9.3E-04
Ru-103	negl.	negl.	negl.	negl.	negl.
Ru-106	negl.	negl.	negl.	negl.	negl.
Sb-125	negl.	negl.	negl.	negl.	negl.
Cs-134	negl.	negl.	2.0E-04	6.3E-04	8.5E-04
Cs-136	negl.	negl.	negl.	negl.	negl.
Cs-137	negl.	negl.	2.7E-04	1.0E-03	1.3E-03
Ba-140	negl.	negl.	1.5E-04	negl.	1.6E-04
Ce-141	negl.	negl.	negl.	negl.	negl.
Total Particulates					1.7E-02

Notes:

1. Values less than 1 microcurie (3.7E+4Bq) are considered to be negligible, but their values are included in the totals.

Table 3.3-9

RADIONUCLIDES LISTED IN *EU COMMISSION RECOMMENDATION* 2004/2/EURATOM FOR PWR NUCLEAR POWER REACTORS THAT WERE INCLUDED IN AP1000 NPP DESIGN BASIS ESTIMATES FOR MONTHLY DISCHARGES TO ATMOSPHERE

Noble gases			
Ar-41	Kr-87	Xe-133	Xe-135m
Kr-85	Kr-88	Xe-133m	Xe-137
Kr-85m	Xe-131m	Xe-135	Xe-138
Particulates (excluding iodines)			
Cr-51	Co-60	Nb-95	Ba-140
Mn-54	Sr-89	Sb-125	Ce-141
Co-58	Sr-90	Cs-134	
Fe-59	Zr-95	Cs-137	
Iodines			
I-131	I-133		
Tritium			
H-3			
Carbon-14			
C-14			

Table 3.3-10

MONTHLY DISCHARGES IN GAS, ALL RADIONUCLIDES

	RCS		NON-RCS		TOTAL	
Month	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	3.26	121	12.10	448	15.36	568
2	3.45	128	12.10	448	15.55	575
3	3.66	136	12.10	448	15.76	583
4	3.91	145	12.10	448	16.01	592
5	4.18	155	12.10	448	16.28	602
6	4.50	167	12.10	448	16.60	614
7	4.87	180	12.10	448	16.97	628
8	5.31	197	12.10	448	17.41	644
9	5.84	216	12.10	448	17.94	664
10	6.48	240	12.10	448	18.58	687
11	7.28	269	12.10	448	19.38	717
12	8.30	307	12.10	448	20.40	755
13	9.66	357	12.10	448	21.76	805
14	11.55	427	12.10	448	23.65	875
15	14.38	532	12.10	448	26.48	980
16	19.04	704	12.10	448	31.14	1152
17	28.27	1046	12.10	448	40.37	1494
18	56.19	2079	12.10	448	68.29	2527
Total	200.13	7405	217.80	8059	417.93	15463

Table 3.3-11

MONTHLY RADIOIODINE DISCHARGES IN GAS(1)

	RCS		NON-RCS										TOTAL	
			Fuel Handling Area		Containment Building		Auxiliary Building		Turbine Building		Condenser Air Removal System			
Month	curies	GBq	curies	GBq	curies	GBq	curies	GBq	curies	GBq	curies	GBq	curies	GBq
1	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-2	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
2	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E -02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
3	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
4	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
5	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
6	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
7	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
8	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
9	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
10	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
11	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
12	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
13	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
14	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
15	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02

Table 3.3-11 (cont.)

MONTHLY RADIOIODINE DISCHARGES IN GAS[1]

	RCS		NON-RCS Fuel Handling Area		NON-RCS Containment Building		NON-RCS Auxiliary Building		NON-RCS Turbine Building		NON-RCS Condenser Air Removal System		TOTAL	
Month	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**
16	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
17	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
18	negl.	negl.	4.1E-05	1.5E-03	2.9E-04	1.1E-02	9.9E-04	3.7E-02	7.1E-06	2.6E-04	8.8E-06	3.3E-04	1.34E-03	5.0E-02
Total	negl.	negl.	7.35E-04	2.72E-02	5.27E-03	1.95E-01	1.79E-02	6.62E-01	1.28E-04	4.74E-03	1.59E-04	5.88E-03	2.41E-02	8.92E-01

Notes:
1. Includes I-131 and I-133.

Table 3.3-12

MONTHLY NOBLE GAS DISCHARGES

	RCS		NON-RCS									
	Waste Gas System		Containment Building		Auxiliary Building		Turbine Building		Condenser Air Removal System		TOTAL	
Month	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	2.9E+00	1.1E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	8.07E+00	2.99E+02
2	3.1E+00	1.1E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	8.23E+00	3.05E+02
3	3.2E+00	1.2E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	8.42E+00	3.12E+02
4	3.5E+00	1.3E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	8.64E+00	3.20E+02
5	3.7E+00	1.4E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	8.88E+00	3.29E+02
6	4.0E+00	1.5E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	9.16E+00	3.39E+02
7	4.3E+00	1.6E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	9.49E+00	3.51E+02
8	4.7E+00	1.7E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	9.88E+00	3.66E+02
9	5.2E+00	1.9E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	1.03E+01	3.81E+02
10	5.7E+00	2.1E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	1.09E+01	4.03E+02
11	6.4E+00	2.4E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	1.16E+01	4.29E+02
12	7.3E+00	2.7E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	1.25E+01	4.62E+02
13	8.5E+00	3.1E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	1.37E+01	5.07E+02
14	1.0E+01	3.7E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	1.54E+01	5.70E+02
15	1.3E+01	4.8E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	1.79E+01	6.62E+02
16	1.7E+01	6.3E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	2.20E+01	8.14E+02
17	2.5E+01	9.2E+02	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	3.02E+01	1.12E+03
18	5.0E+01	1.8E+03	3.2E-01	1.2E+01	3.1E+00	1.1E+02	2.1E-04	7.8E-03	1.8E+00	6.7E+01	5.49E+01	2.03E+03
Total	1.77E+02	6.55E+03	5.70E+00	2.11E+02	5.57E+01	2.06E+03	3.77E-03	1.39E-01	3.20E+01	1.18E+03	2.70E+02	9.99E+03

Table 3.3-13

MONTHLY TRITIUM DISCHARGES IN GAS

	RCS		**NON-RCS**		**TOTAL**	
Month	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	0.18	6.7	3.37	125	3.56	132
2	0.19	7.0	3.37	125	3.57	132
3	0.21	7.8	3.37	125	3.58	132
4	0.22	8.1	3.37	125	3.59	133
5	0.24	8.9	3.37	125	3.61	134
6	0.25	9.2	3.37	125	3.63	134
7	0.27	10.0	3.37	125	3.65	135
8	0.30	11.1	3.37	125	3.67	136
9	0.33	12.2	3.37	125	3.70	137
10	0.37	13.7	3.37	125	3.74	138
11	0.41	15.2	3.37	125	3.78	140
12	0.47	17.4	3.37	125	3.84	142
13	0.54	20.0	3.37	125	3.92	145
14	0.65	24.0	3.37	125	4.02	149
15	0.81	30.0	3.37	125	4.18	155
16	1.07	39.6	3.37	125	4.45	165
17	1.59	58.8	3.37	125	4.97	184
18	3.17	117.3	3.37	125	6.54	242
Total	11.28	417.4	60.72	2247	72.00	2664

Table 3.3-14

MONTHLY C-14 DISCHARGES IN GAS

Month	RCS		NON-RCS		TOTAL	
	curies	GBq	curies	GBq	curies	GBq
1	0.06	2.32	1.15	42.64	1.22	44.97
2	0.07	2.46	1.15	42.64	1.22	45.10
3	0.07	2.61	1.15	42.64	1.22	45.25
4	0.08	2.78	1.15	42.64	1.23	45.43
5	0.08	2.98	1.15	42.64	1.23	45.62
6	0.09	3.21	1.15	42.64	1.24	45.85
7	0.09	3.47	1.15	42.64	1.25	46.12
8	0.10	3.78	1.15	42.64	1.25	46.43
9	0.11	4.16	1.15	42.64	1.26	46.80
10	0.12	4.62	1.15	42.64	1.28	47.26
11	0.14	5.18	1.15	42.64	1.29	47.83
12	0.16	5.91	1.15	42.64	1.31	48.56
13	0.19	6.88	1.15	42.64	1.34	49.53
14	0.22	8.23	1.15	42.64	1.38	50.88
15	0.28	10.24	1.15	42.64	1.43	52.89
16	0.37	13.57	1.15	42.64	1.52	56.21
17	0.54	20.15	1.15	42.64	1.70	62.79
18	1.08	40.04	1.15	42.64	2.23	82.69
Total	3.85	142.61	20.75	767.59	24.60	910.20

Table 3.3-15

MONTHLY AR-41 DISCHARGES IN GAS

Month	**RCS**		**NON-RCS**		**TOTAL**	
	curies	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	0.13	4.8	2.39	88.4	2.52	93.2
2	0.14	5.2	2.39	88.4	2.53	93.6
3	0.15	5.6	2.39	88.4	2.54	94.0
4	0.16	5.9	2.39	88.4	2.55	94.4
5	0.17	6.3	2.39	88.4	2.56	94.7
6	0.18	6.7	2.39	88.4	2.57	95.1
7	0.19	7.0	2.39	88.4	2.58	95.5
8	0.21	7.8	2.39	88.4	2.60	96.2
9	0.23	8.5	2.39	88.4	2.62	96.9
10	0.26	9.6	2.39	88.4	2.65	98.0
11	0.29	10.7	2.39	88.4	2.68	99.2
12	0.33	12.2	2.39	88.4	2.72	100.6
13	0.39	14.4	2.39	88.4	2.78	102.9
14	0.46	17.0	2.39	88.4	2.85	105.4
15	0.57	21.1	2.39	88.4	2.96	109.5
16	0.76	28.1	2.39	88.4	3.15	116.6
17	1.13	41.8	2.39	88.4	3.52	130.2
18	2.24	82.9	2.39	88.4	4.63	171.3
Total	7.99	295.6	43.01	1591	51.00	1887

Table 3.3-16

MONTHLY CO-60 DISCHARGES IN GAS

	RCS		NON-RCS							
	Waste Gas System		Containment Building		Auxiliary Building		Fuel Handling Area		TOTAL	
Month	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
2	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
3	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
4	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
5	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
6	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
7	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
8	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
9	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
10	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
11	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
12	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
13	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
14	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
15	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
16	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
17	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
18	negl.	negl.	negl.	negl.	4.3E-07	1.57E-05	6.8E-06	2.53E-04	7.26E-06	2.69E-04
Total	negl.	negl.	negl.	negl.	7.65E-06	2.83E-04	1.23E-04	4.55E-03	1.31E-04	4.83E-03

Table 3.3-17

MONTHLY KR-85 DISCHARGES IN GAS

	RCS		NON-RCS									
	Waste Gas System		Containment Building		Auxiliary Building		Turbine Building		Condenser Air Removal System		TOTAL	
Month	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	2.0E+00	7.32E+01	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	2.18E+00	8.06E+01
2	2.1E+00	7.75E+01	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	2.29E+00 -	8.49E+01
3	2.2E+00	8.23E+01	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	2.42E+00	8.97E+01
4	2.4E+00	8.77E+01	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	2.57E+00	9.51E+01
5	2.5E+00	9.40E+01	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	2.74E+00	1.01E+02
6	2.7E+00	1.01E+02	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	2.93E+00	1.08E+02
7	3.0E+00	1.09E+02	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	3.16E+00	1.17E+02
8	3.2E+00	1.19E+02	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	3.42E+00	1.27E+02
9	3.5E+00	1.31E+02	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	3.74E+00	1.38E+02
10	3.9E+00	1.45E+02	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	4.13E+00	1.53E+02
11	4.4E+00	1.63E+02	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	4.62E+00	1.71E+02
12	5.0E+00	1.86E+02	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	5.24E+00	1.94E+02
13	5.9E+00	2.17E+02	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	6.06E+00	2.24E+02
14	7.0E+00	2.60E+02	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	7.21E+00	2.67E+02
15	8.7E+00	3.23E+02	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	8.93E+00	3.30E+02
16	1.2E+01	4.28E+02	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	1.18E+01	4.35E+02
17	1.7E+01	6.35E+02	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	1.74E+01	6.42E+02
18	3.4E+01	1.26E+03	2.5E-02	9.25E-01	1.2E-01	4.32E+00	6.6E-06	2.44E-04	5.8E-02	2.13E+00	3.43E+01	1.27E+03
Total	1.22E+02	4.50E+03	4.5E-01	1.67E+01	2.10E+00	7.77E+01	1.19E-04	4.38E-03	1.04E+00	3.83E+01	1.26E+02	4.63E+03

Table 3.3-18

MONTHLY SR-90 DISCHARGES IN GAS

	RCS		NON-RCS							
	Waste Gas System		Containment Building		Auxiliary Building		Fuel Handling		TOTAL	
Month	curies	GBq	curies	GBq	curies	GBq	curies	GBq	curies	GBq
1	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
2	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
3	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
4	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
5	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
6	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
7	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
8	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
9	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
10	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
11	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
12	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
13	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
14	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
15	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
16	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
17	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
18	negl.	negl.	negl.	negl.	2.4E-07	8.94E-06	6.7E-07	2.47E-05	1.0E-06	3.70E-05
Total	negl.	negl.	negl.	negl.	4.35E-06	1.61E-04	1.20E-05	4.44E-04	1.80E-05	6.66E-04

Table 3.3-19

MONTHLY 1-131 DISCHARGES IN GAS

	RCS		NON-RCS									
	Waste Gas System		Containment Building		Auxiliary Building		Turbine Building		Condenser Air Removal System		TOTAL	
Month	curies	GBq	curies	GBq	curies	GBq	curies	GBq	curies	GBq	curies	GBq
1	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
2	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
3	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
4	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
5	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
6	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
7	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
8	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
9	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
10	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
11	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
12	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
13	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
14	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
15	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
16	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
17	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
18	negl.	negl.	4.3E-05	1.57E-03	4.0E-04	1.48E-02	5.4E-06	2.00E-04	2.2E-06	8.02E-05	4.67E-04	1.73E-02
Total	negl.	negl.	7.65E-04	2.83E-02	7.20E-03	2.66E-01	9.75E-05	3.61E-03	3.90E-05	1.44E-03	8.40E-03	3.11E-01

Table 3.3-20

MONTHLY XE-133 DISCHARGES IN GAS

	RCS		NON-RCS									
	Waste Gas System		Containment Building		Auxiliary Building		Turbine Building		Condenser Air Removal System		TOTAL	
Month	curies	GBq	curies	GBq	curies	GBq	curies	GBq	curies	GBq	curies	GBq
1	1.6E-01	5.97E+00	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.44E+00	9.01E+01
2	1.7E-01	6.31E+00	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.45E+00	9.05E+01
3	1.8E-01	6.71E+00	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.46E+00	9.09E+01
4	1.9E-01	7.15E+00	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.47E+00	9.13E+01
5	2.1E-01	7.66E+00	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.48E+00	9.18E+01
6	2.2E-01	8.24E+00	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.50E+00	9.24E+01
7	2.4E-01	8.92E+00	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.52E+00	9.31E+01
8	2.6E-01	9.72E+00	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.54E+00	9.39E+01
9	2.9E-01	1.07E+01	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.56E+00	9.49E+01
10	3.2E-01	1.19E+01	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.60E+00	9.60E+01
11	3.6E-01	1.33E+01	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.64E+00	9.75E+01
12	4.1E-01	1.52E+01	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.69E+00	9.94E+01
13	4.8E-01	1.77E+01	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.75E+00	1.02E+02
14	5.7E-01	2.11E+01	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.85E+00	1.05E+02
15	7.1E-01	2.63E+01	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	2.99E+00	1.10E+02
16	9.4E-01	3.48E+01	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	3.22E+00	1.19E+02
17	1.4E+00	5.17E+01	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	3.67E+00	1.36E+02
18	2.8E+00	1.03E+02	2.0E-01	7.40E+00	1.4E+00	5.24E+01	7.5E-05	2.78E-03	6.6E-01	2.44E+01	5.05E+00	1.87E+02
Total	9.90E+00	3.66E+02	3.60E+00	1.33E+02	2.55E+01	9.44E+02	1.35E-03	5.00E-02	1.19E+01	4.38E+02	5.10E+01	1.89E+03

Table 3.3-21

MONTHLY CS-137 DISCHARGES IN GAS

	RCS		NON-RCS							
	Waste Gas System		Containment Building		Auxiliary Building		Fuel Handling		TOTAL	
Month	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
2	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
3	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
4	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
5	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
6	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
7	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
8	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
9	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
10	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
11	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
12	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
13	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
14	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
15	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
16	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
17	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
18	negl.	negl.	negl.	negl.	6.0E-07	2.22E-05	2.3E-06	8.33E-05	3.00E-06*	negl.
Total	negl.	negl.	negl.	negl.	1.08E-05	4.00E-04	4.05E-05	1.50E-03	5.40E-05*	negl.

Table 3.3-22

MONTHLY DISCHARGES OF OTHER PARTICULATES

Month	RCS		NON-RCS						TOTAL	
	Waste Gas System		Fuel Handling Area		Containment Building		Auxiliary Building			
	curies	GBq	curies	GBq	curies	GBq	curies	GBq	curies	GBq
1	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
2	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
3	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
4	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
5	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
6	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
7	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
8	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
9	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
10	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
11	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
12	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
13	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
14	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
15	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
16	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
17	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
18	negl.	negl.	2.3E-05	8.54E-04	3.2E-07	1.17E-05	4.1E-06	1.51E-04	2.75E-05	1.02E-03
Total	negl.	negl.	4.16E-04	1.54E-02	5.70E-06	2.11E-04	7.37E-05	2.73E-03	4.95E-04	1.83E-02

Table 3.3-23

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

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

Table 3.4-1

AP1000 NPP ESTIMATED OPERATIONAL LIQUID RADWASTE ARISING FROM SYSTEM OPERATIONS

System	Waste Description	Waste Class	Physical/ Chemical Description	Estimated Quantities: Normal Daily Volume U.S. Gallons	Normal Daily Volume m^3	Maximum Daily Volume U.S. Gallons	Maximum Daily Volume m^3	Volume for Life of Plant U.S. Gallons	Volume for Life of Plant m^3	Radioactivity
BDS	Steam Generator blowdown	LLW	Secondary-side coolant	18.6	4.22	186	42.24	5.87E+08	2.22E+06	Included here as LLW but may be non-radioactive
CVS	Boron dilution near end-of–life (EOL)	LLW	Borated reactor coolant	1,663	6	6,980	26	6.65E+04	2.52E+02	100% reactor coolant
CVS	RCS heat up	LLW	Borated reactor coolant	22,440	85	44,880	170	1.08E+06	4.08E+03	100% reactor coolant without radiogas
WLS	CVS shim bleed (liquid)	LLW	Diverted reactor coolant/dilute boric acid	435	1.65	776	2.94	1.10E+07	4.17E+04	100% reactor coolant
WLS	Equipment leaks	LLW	Dilute boric acid	90	0.34	14400	54.5	2.84E+06	1.07E+04	100% reactor coolant
WLS	Floor drains (dirty wastes)	LLW	Dilute boric acid	1,200	4.54	5760	21.8	2.66E+07	1.01E+05	0.1% reactor coolant

Table 3.4-1 (cont.)

AP1000 NPP ESTIMATED OPERATIONAL LIQUID RADWASTE ARISING FROM SYSTEM OPERATIONS

System	Waste Description	Waste Class	Physical/ Chemical Description	Estimated Quantities: Normal Daily Volume		Estimated Quantities: Maximum Daily Volume		Estimated Quantities: Volume for Life of Plant		Radioactivity
				U.S. Gallons	m^3	U.S. Gallons	m^3	U.S. Gallons	m^3	
WLS	Sampling-system drains	LLW	Dilute boric acid	200	0.76	1000	3.79	4.56E+06	1.73E+04	100% reactor coolant
WLS	Hand wash/ hot shower	LLW	Grey water	200	0.76	2000	7.57	1.23E+07	4.64E+04	10E-7 µCi/cc (0.037 MBq/m^3)
WLS	Equipment and area decontamin-ation	LLW	Detergent waste	40	0.15	400	1.51	2.45E+06	9.28E+03	0.1% reactor coolant
WLS	Chemical waste	LLW	Spent samples containing analytical chemicals	7.14	0.03	14.28	0.05	1.88E+05	7.10E+02	≤ reactor coolant
WLS	Decontamin-ation fluids	LLW	Liquid with decontamination chemicals	0.62	0.0023	1.24	0.0047	1.63E+04	6.20E+01	1 µCi/cc (37,000 MBq/m^3)

Table 3.4-2

AP1000 NPP LIQUID RADWASTE STORAGE TANK INFORMATION

Parameter	Unit	Reactor Coolant Drain Tank	Effluent Holdup Tanks	Waste Holdup Tanks	Chemical Waste Tank	Monitor Tanks
Volume Each Tank	m^3	3.407 (1 tank)	106 (2 tanks) and 56.78 (1 tank)	56.78 (2 tanks)	29.15 (1 tank)	56.78 (6 tanks)
Dimensions Each Tank	mm	Length = 2184 Height = 1600 (horizontal tank)	Length = 11180 Height = 4114(2 horizontal tanks) and Diameter = 3657 Height = 6273 (1 vertical tank)	Diameter = 3657 Height = 6273	Diameter = 3645 Height = 1911 (Straight Shell Length)	Diameter = 3657 Height = 6273
Construction Material	–	Stainless Steel	Stainless Steel	Stainless Steel	Stainless Steel	Stainless Steel
Design Code	–	ASME III-3, Unstamped	ASME III-3, Unstamped	ASME III-3, Unstamped	ASME III-3, Unstamped	ASME III-3, Unstamped
Venting Arrangement	–	The vent is hard piped to the WGS because the tank gas space will contain hydrogen and trace fission gases.	The vent contains hydrogen monitoring instrumentation and is hard piped to the VAS. The tank is swept with air on a high hydrogen alarm. The gases are removed from the process stream before entering the tanks.	The tank is vented to the room because the water has been stored at atmospheric pressure prior to entering the tanks, so the gases have come out of solution prior to entering the tanks.	The tank is vented to the room because the gases have been removed prior to storage in the tank.	The tank is vented to the room because the gases have been removed prior to storage in the Monitor Tanks.
Overflow Arrangement	–	Overflow is through a relief valve. The relief valve discharge is hard piped to the containment sump.	Overflow is hard piped to a WRS floor drain.	Overflow is hard piped to a WRS floor drain.	Overflow discharge is hard piped to the auxiliary building sump.	Overflow is hard piped to a WRS floor drain.
Alarm Arrangement	–	There is a High-2 alarm associated with this tank. The alarm indicates that the tank is ready for discharge, which is automatic. The alarm will be displayed in both the MCR and on the local Liquid and Gaseous Radwaste Control Panel.	There are three alarms – High-2, High, and Low – associated with these tanks. The High-2 alarm indicates that the tank is full and ready for processing. The High alarm warns the operator that the tank is close to full and to be alert. The Low alarm tells the operator that the puma has been shut off. The Low alarm can be cleared once it is acknowledged. The alarms will be displayed in both the MCR and the local Liquid and Gaseous Radwaste Control Panel.	There are three alarms – High-2, High, and Low – associated with these tanks. The High-2 alarm indicates that the tank is full and ready for processing. The High alarm warns the operator that the tank is close to full and to be alert. The Low alarm tells the operator that the pump has been shut off. The Low alarm can be cleared once it is acknowledged. The alarms will be displayed in both the MCR and the local Liquid and Gaseous Radwaste Control Panel.	There are three alarms – High-2, High, and Low – associated with these tanks. The High-2 alarm indicates that the tank is full and ready for processing. The High alarm warns the operator that the tank is close to full and to be alert. The Low alarm tells the operator that the pump has been shut off. The Low alarm can be cleared once it is acknowledged. The alarms will be displayed in both the MCR and the local Liquid and Gaseous Radwaste Control Panel.	There are three alarms – High-2, High, and Low – associated with these tanks. The High-2 alarm indicates that the tank is full and ready for discharge. The High alarm warns the operator that the tank is close to full and to be alert. The Low alarm tells the operator that the pump has been shut off. The Low alarm can be cleared once it is acknowledged. The alarms will be displayed in both the MCR and the local Liquid and Gaseous Radwaste Control Panel.
Tank Content Mixing	–	Mixing is not necessary for this tank; however, the tank contents can be recirculated through a sparger located at the bottom of the tank. This will be done when the contents of the tank are too hot for efficient gas removal in the WLS degasifier column.	The pump suction is taken from the bottom of the tank and returned to the top as quickly as possible to minimise the time needed for mixing. The tank will be mixed prior to sampling. Sampling will be done periodically to determine the effectiveness of gas removal by the degasifier column.	The pump suction is taken from the bottom of the tank and returned to the top as quickly as possible to minimise the time needed for mixing. The tank will be mixed prior to sampling. Sampling is done prior to discharge for processing to determine if the chemistry of the contents needs to be adjusted to increase the effectiveness of processing.	The pump suction is taken from the bottom of the tank and returned to the top as quickly as possible to minimise the time needed for mixing. The tank will be mixed prior to sampling. Sampling is done prior to discharge to determine the appropriate method of processing.	The pump suction is taken from the bottom of the tank and returned to the top as quickly as possible to minimise the time needed for mixing. The tank will be mixed prior to sampling. Sampling is done prior to discharge to the environment.
Sampling Arrangement	–	Local sample taken on the discharge line outside of containment. The samples are taken to the chemistry lab for analysis.	A local grab sample is taken from the recirculation line. The samples are taken to the chemistry lab for analysis.	A local grab sample is taken from the recirculation line. The samples are taken to the chemistry lab for analysis.	A local grab sample is taken from the recirculation line. The samples are taken to the chemistry lab for analysis.	A local grab sample is taken from the recirculation line. The samples are taken to the chemistry lab for analysis.

Table 3.4-3

LIQUID RADWASTE STORAGE TANK SECONDARY CONTAINMENT INFORMATION

Parameter	Unit	Reactor Coolant Drain Tank	Effluent Holdup Tanks	Waste Holdup Tanks	Chemical Waste Tank	Monitor Tanks
Volume	m^3	N/A (Containment Shell)	N/A (Auxiliary Building)	N/A (Auxiliary Building)	N/A (Auxiliary Building)	Tanks A, B, C located in Monitor Tank Room A Tanks D, E, F located in Monitor Tank Room B
Dimensions	m	N/A (Containment Shell)	N/A (Auxiliary Building)	N/A (Auxiliary Building)	N/A (Auxiliary Building)	N/A (Monitor Tank Rooms A & B)
Construction Material	–	Concrete	Concrete/Steel	Concrete/Steel	Concrete/Steel	Concrete/Steel
Drainage Arrangements	–	The floor drains are routed to theWLS containment sump.	The floor drains are routed to the auxiliary building sump via the Radwaste Drain System.	The floor drains contain valves which are normally closed because the Waste Holdup Tank rooms are designed to be water tight. If necessary, the floor drains can be opened to drain the room.	The floor drains are normally plugged because the contents of the tank may not be compatible with the normal floor drain wastes or installed demineralisers. The floor drain plugs can be removed to drain the room if necessary.	The floor drains contain valves which are normally closed because the Monitor Tank rooms are designed to be water tight. If necessary, the floor drains can be opened to drain the rooms to the auxiliary building sump via the Radwaste Drain System.
Sump Arrangements	–	The water is collected and on High level is pumped to the WLS Waste Holdup tanks for processing.	The water is collected in the Auxiliary Building sump and on High level is pumped to the WLS Waste Holdup tanks for processing.	The water is collected in the Auxiliary Building sump and on High level is pumped to the WLS Waste Holdup Tanks for processing.	The water is collected in the Auxiliary Building sump and on High level is pumped to the WLS Waste Holdup Tanks for processing.	The water is collected and on High level is pumped to the WLS Waste Holdup tanks for processing.
Leak Detection	–	The tanks have Low level alarms. If the tank level is low and the pump is not running, then the tank has failed.	The tanks have Low level alarms. If the tank level is low and the pump is not running, then the tank has failed.	Each tank room contains a level instrument to detect the water level in the room. The tank level instrument can be used in association with the room level instrument to determine a tank failure.	The room contains a level instrument to detect the overflow or failure of the chemical waste tank.	The tanks have Low level alarms. If the tank level is low and the pump is not running, then the tank has failed.

Table 3.4-4

ASSUMED DECONTAMINATION FACTORS FOR LIQUID RADWASTE ION EXCHANGE BEDS

Resin Type / Component	Iodine	Cs/Rb	Other
Zeolite/deep bed filter(1)	1	100	1
Cation/waste ion exchanger 1	1	10	10
Mixed/waste ion exchanger 2	100	2(2)	100
Mixed/waste ion exchanger 2	10	10(2)	10(2)

Notes:
1. This component is not included in NUREG-0017. DFs are based upon "Reduction of Caesium and Cobalt Activity in Liquid Radwaste Processing Using Clinoptilolite Zeolite at Duke Power Company," by O.E. Ekechokwu, et al., Proc. Waste Management '92, Tucson, Arizona, March 1992, University of Arizona, Tucson.
2. Credit for this DF is not taken in determination of anticipated annual releases.

Table 3.4-5

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 12 wt% "slurry" which is drummed or solidified	Evaporator removes all solids in waste stream. Concentrates to 12 wt% "slurry" (or higher) which is drummed or solidified	Ion exchange process removes activity from fluid. Non-specified solids (e.g., concrete dust, sand) and boric acid pass through to discharge.
Effectiveness	Acceptable DF 100-500	Good DF 100-1000	Good to excellent with appropriate usage. DF 100 -400 for single vessel, higher for multiple vessels in series.
Flexibility	Poor – many inputs can upset evaporator (e.g., detergents, oil)	Excellent – same process for all inputs	Excellent, but requires intelligent control: • Oils must be segregated because they 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

Table 3.4-5 (cont.)

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

	Natural Circulation Evaporators	Forced Circulation Evaporators	Ion Exchange
Operating Cost	Moderate – steam/energy	Moderate – steam/energy	High – resins Low – equipment – much less equipment and less active equipment to maintain
Safety	Excellent	Excellent	Excellent
Reliability	Poor – 12 wt% 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 Filter Cartridges Evaporator Bottoms Chemical Wastes Total	6 m^3/y 0.5 m^3/y 102 m^3/y 1 m^3/y 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

Table 3.4-5 (cont.)

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

	Natural Circulation Evaporators	**Forced Circulation Evaporators**	**Ion Exchange**
Tritium	Increased transfer of tritium from water to air. Impact of tritium dose is more significant in air than water.		Greater proportion of tritium in water than air
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 3.4-6

EXPECTED ANNUAL RELEASE OF RADIOACTIVE EFFLUENT DISCHARGES

Nuclide	Activity Release(1) GBq/y			
	Shim Bleed + Equip. Drains	Miscellaneous Wastes	Turbine Building	Total Release
C-14	3.3E+00(2)	negl.	negl.	3.3E+00(2)
Na-24	3.5E-02	2.3E-04	2.8E-03	3.8E-02
Cl-36	negl.	negl.	negl.	negl.
Cr-51	4.5E-02	1.3E-04	2.8E-04	4.6E-02
Mn-54	3.2E-02	7.2E-05	1.4E-04	3.2E-02
Fe-55	4.8E-01	1.1E-03	2.1E-03	4.9E-01
Fe-59	4.9E-03	negl.	negl.	5.0E-03
Co-58	4.1E-01	1.0E-03	2.0E-03	4.1E-01
Co-60	2.2E-01	5.0E-04	9.4E-04	2.3E-01
Ni-63	5.3E-01	1.2E-03	2.1E-03	5.4E-01
Zn-65	1.0E-02	negl.	4.5E-05	1.0E-02
Nb-94	negl.	negl.	negl.	negl.
W-187	2.8E-03	negl.	1.7E-04	3.0E-03
U-234	negl.	negl.	negl.	negl.
U-235	negl.	negl.	negl.	negl.
U-238	negl.	negl.	negl.	negl.
Np-237	negl.	negl.	negl.	negl.
Pu-238	negl.	negl.	negl.	negl.
Pu-239	negl.	negl.	negl.	negl.
Pu-240	negl.	negl.	negl.	negl.
Pu-241	8.0E-05	negl.	negl.	8.0E-05
Pu-242	negl.	negl.	negl.	negl.
Am-241	negl.	negl.	negl.	negl.
Am-243	negl.	negl.	negl.	negl.
Cm-242	negl.	negl.	negl.	negl.
Cm-244	negl.	negl.	negl.	negl.
As-76	negl.	negl.	negl.	negl.

Table 3.4-6 (cont.)

EXPECTED ANNUAL RELEASE OF RADIOACTIVE EFFLUENT DISCHARGES

Nuclide	Activity Release(1) GBq/y			
	Shim Bleed + Equip. Drains	Miscellaneous Wastes	Turbine Building	Total Release
Br-82	negl.	negl.	negl.	negl.
Rb-86	negl.	negl.	negl.	negl.
Rb-88	3.9E-04	negl.	negl.	3.9E-04
Sr-89	2.4E-03	negl.	negl.	2.4E-03
Sr-90	2.5E-04	negl.	negl.	2.5E-04
Y-91	9.0E-05	negl.	negl.	9.1E-05
Zr-95	6.8E-03	negl.	negl.	6.9E-03
Nb-95	6.1E-03	negl.	negl.	6.1E-03
Mo-99	1.9E-02	1.1E-04	5.3E-04	1.9E-02
Tc-99m	1.8E-02	1.1E-04	3.8E-04	1.8E-02
Tc-99	negl.	negl.	negl.	negl.
Ru-103	1.2E-01	3.1E-04	6.6E-04	1.2E-01
Ru-106	negl.	negl.	negl.	negl.
Ag-110m	2.6E-02	5.8E-05	1.1E-04	2.6E-02
Sn-117m	negl.	negl.	negl.	negl.
Sb-122	negl.	negl.	negl.	negl.
Sb-124	negl.	negl.	negl.	negl.
Sb-125	negl.	negl.	negl.	negl.
I-129	negl.	negl.	negl.	negl.
I-131	1.5E-02	6.3E-05	2.5E-04	1.5E-02
I-132	1.9E-02	9.1E-05	8.5E-04	2.0E-02
I-133	2.6E-02	1.7E-04	2.7E-03	2.9E-02
I-134	5.8E-03	3.9E-05	negl.	5.9E-03
Cs-134	7.5E-03	negl.	Negl.	7.6E-03
I-135	2.0E-02	1.3E-04	3.2E-03	2.4E-02
Cs-136	9.2E-03	negl.	8.5E-05	9.3E-03
Cs-137	2.3E-02	5.0E-05	1.1E-04	2.3E-02
Ba-140	1.3E-02	4.6E-05	1.1E-04	1.4E-02

Table 3.4-6 (cont.)

EXPECTED ANNUAL RELEASE OF RADIOACTIVE EFFLUENT DISCHARGES

	Activity Release[1] GBq/y			
Nuclide	**Shim Bleed + Equip. Drains**	**Miscellaneous Wastes**	**Turbine Building**	**Total Release**
La-140	1.8E-02	6.6E-05	2.0E-04	1.8E-02
Ce-144	7.9E-02	1.8E-04	3.4E-04	8.0E-02
Pr-144	7.9E-02	1.8E-04	3.4E-04	8.0E-02
All Others	negl.	negl.	negl.	negl.
Total	5.7E+00	6.3E-03	2.1E-02	5.8E+00

Tritium Release in Liquid Effluents[3] (TBq/y) = 33.4

Notes:
1. Values less than 1 microcurie (3.7E+4Bq) are considered to be negligible, but their values are included in the totals.
2. C-14 from (Ref. 3-37).
3. Tritium Release based on Westinghouse TRICAL computer code.

Table 3.4-7

MONTHLY DISCHARGES IN LIQUID, ALL RADIONUCLIDES

	RCS		NON-RCS		TOTAL	
Month	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	3.46	128	63.39	2345	66.85	2473
2	3.66	135	63.39	2345	67.05	2481
3	3.89	144	63.39	2345	67.27	2489
4	4.14	153	63.39	2345	67.53	2499
5	4.44	164	63.39	2345	67.82	2510
6	4.77	177	63.39	2345	68.16	2522
7	5.17	191	63.39	2345	68.56	2537
8	5.63	208	63.39	2345	69.02	2554
9	6.19	229	63.39	2345	69.58	2574
10	6.87	254	63.39	2345	70.26	2600
11	7.72	285	63.39	2345	71.11	2631
12	8.80	326	63.39	2345	72.19	2671
13	10.24	379	63.39	2345	73.63	2724
14	12.25	453	63.39	2345	75.64	2799
15	15.25	564	63.39	2345	78.63	2909
16	20.19	747	63.39	2345	83.58	3092
17	29.98	1109	63.39	2345	93.37	3455
18	59.59	2205	63.39	2345	122.98	4550
Total	212.22	7852	1141.01	42217	1353.24	50070

Table 3.4-8

MONTHLY TRITIUM DISCHARGES IN LIQUID

Month	RCS		NON-RCS		TOTAL	
	curies	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	3.45	128	63.39	2345	66.84	2473
2	3.65	135	63.39	2345	67.04	2480
3	3.88	144	63.39	2345	67.27	2489
4	4.14	153	63.39	2345	67.53	2499
5	4.43	164	63.39	2345	67.82	2509
6	4.77	176	63.39	2345	68.16	2523
7	5.16	191	63.39	2345	68.55	2536
8	5.63	208	63.39	2345	69.02	2553
9	6.18	229	63.39	2345	69.57	2574
10	6.86	254	63.39	2345	70.25	2599
11	7.71	265	63.39	2345	71.10	2631
12	8.79	325	63.39	2345	72.18	2671
13	10.23	379	63.39	2345	73.62	2724
14	12.24	453	63.39	2345	75.63	2798
15	15.23	564	63.39	2345	78.62	2909
16	20.17	746	63.39	2345	83.56	3092
17	29.95	1108	63.39	2345	93.33	3453
18	59.52	2202	63.39	2345	122.91	4548
Total	211.99	7844	1141.01	42217	1353.00	50061

Table 3.4-9

MONTHLY NON-TRITIUM DISCHARGES IN LIQUID

Month	RCS		NON-RCS		TOTAL	
	curies	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	3.8E-03	1.4E-01	6.2E-05	2.31E-03	3.87E-03	1.43E-01
2	4.0E-03	1.5E-01	6.2E-05	2.31E-03	4.10E-03	1.52E-01
3	4.3E-03	1.6E-01	6.2E-05	2.31E-03	4.35E-03	1.61E-01
4	4.6E-03	1.7E-01	6.2E-05	2.31E-03	4.63E-03	1.71E-01
5	4.9E-03	1.8E-01	6.2E-05	2.31E-03	4.95E-03	1.83E-01
6	5.3E-03	1.9E-01	6.2E-05	2.31E-03	5.33E-03	1.97E-01
7	5.7E-03	2.1E-01	6.2E-05	2.31E-03	5.76E-03	2.13E-01
8	6.2E-03	2.3E-01	6.2E-05	2.31E-03	6.27E-03	2.32E-01
9	6.8E-03	2.5E-01	6.2E-05	2.31E-03	6.89E-03	2.55E-01
10	7.6E-03	2.8E-01	6.2E-05	2.31E-03	7.64E-03	2.83E-01
11	8.5E-03	3.1E-01	6.2E-05	2.31E-03	8.57E-03	3.17E-01
12	9.7E-03	3.6E-01	6.2E-05	2.31E-03	9.77E-03	3.61E-01
13	1.1E-02	4.2E-01	6.2E-05	2.31E-03	1.14E-02	4.20E-01
14	1.4E-02	5.0E-01	6.2E-05	2.31E-03	1.36E-02	5.02E-01
15	1.7E-02	6.2E-01	6.2E-05	2.31E-03	1.69E-02	6.24E-01
16	2.2E-02	8.2E-01	6.2E-05	2.31E-03	2.23E-02	8.26E-01
17	3.3E-02	1.2E+00	6.2E-05	2.31E-03	3.31E-02	1.23E+00
18	6.6E-02	2.4E+00	6.2E-05	2.31E-03	6.58E-02	2.43E+00
Total	2.34E-01	8.7E+00	11.25E-04	4.16E-02	2.35E-01	8.70E+00

Table 3.4-10

MONTHLY C-14 DISCHARGES IN LIQUID

Month	RCS		NON-RCS		TOTAL	
	curies	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	2.2E-03	8.13E-02	negl.	negl.	2.20E-03	8.13E-02
2	2.3E-03	8.61E-02	negl.	negl.	2.33E-03	8.61E-02
3	2.5E-03	9.14E-02	negl.	negl.	2.47E-03	9.14E-02
4	2.6E-03	9.75E-02	negl.	negl.	2.63E-03	9.75E-02
5	2.8E-03	1.04E-01	negl.	negl.	2.82E-03	1.04E-01
6	3.0E-03	1.12E-01	negl.	negl.	3.04E-03	1.12E-01
7	3.3E-03	1.22E-01	negl.	negl.	3.29E-03	1.22E-01
8	3.6E-03	1.33E-01	negl.	negl.	3.58E-03	1.33E-01
9	3.9E-03	1.46E-01	negl.	negl.	3.94E-03	1.46E-01
10	4.4E-03	1.62E-01	negl.	negl.	4.37E-03	1.62E-01
11	4.9E-03	1.82E-01	negl.	negl.	4.91E-03	1.82E-01
12	5.6E-03	2.07E-01	negl.	negl.	5.60E-03	2.07E-01
13	6.5E-03	2.41E-01	negl.	negl.	6.52E-03	2.41E-01
14	7.8E-03	2.88E-01	negl.	negl.	7.79E-03	2.88E-01
15	9.7E-03	3.59E-01	negl.	negl.	9.70E-03	3.59E-01
16	1.3E-02	4.75E-01	negl.	negl.	1.28E-02	4.75E-01
17	1.9E-02	7.06E-01	negl.	negl.	1.91E-02	7.06E-01
18	3.8E-02	1.40E+00	negl.	negl.	3.79E-02	1.40E+00
Total	1.35E-01	5.00E+00	negl.	negl.	1.35E-01	5.00E+00

Table 3.4-11

MONTHLY FE-55 DISCHARGES IN LIQUID

Month	RCS		NON-RCS		TOTAL	
	curies	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	3.2E-04	1.18E-02	7.2E-06	2.65E-04	3.25E-04	1.20E-02
2	3.4E-04	1.24E-02	7.2E-06	2.65E-04	3.43E-04	1.27E-02
3	3.6E-04	1.32E-02	7.2E-06	2.65E-04	3.64E-04	1.35E-02
4	3.8E-04	1.41E-02	7.2E-06	2.65E-04	3.88E-04	1.43E-02
5	4.1E-04	1.51E-02	7.2E-06	2.65E-04	4.15E-04	1.53E-02
6	4.4E-04	1.62E-02	7.2E-06	2.65E-04	4.46E-04	1.65E-02
7	4.7E-04	1.76E-02	7.2E-06	2.65E-04	4.82E-04	1.78E-02
8	5.2E-04	1.91E-02	7.2E-06	2.65E-04	5.25E-04	1.94E-02
9	5.7E-04	2.10E-02	7.2E-06	2.65E-04	5.76E-04	2.13E-02
10	6.3E-04	2.34E-02	7.2E-06	2.65E-04	6.38E-04	2.36E-02
11	7.1E-04	2.62E-02	7.2E-06	2.65E-04	7.16E-04	2.65E-02
12	8.1E-04	2.99E-02	7.2E-06	2.65E-04	8.16E-04	3.02E-02
13	9.4E-04	3.48E-02	7.2E-06	2.65E-04	9.48E-04	3.51E-02
14	1.1E-03	4.17E-02	7.2E-06	2.65E-04	1.13E-03	4.19E-02
15	1.4E-03	5.18E-02	7.2E-06	2.65E-04	1.41E-03	5.21E-02
16	1.9E-03	6.86E-02	7.2E-06	2.65E-04	1.86E-03	6.89E-02
17	2.8E-03	1.02E-01	7.2E-06	2.65E-04	2.76E-03	1.02E-01
18	5.5E-03	2.03E-01	7.2E-06	2.65E-04	5.48E-03	2.03E-01
Total	1.95E-02	7.22E-01	1.29E-04	4.77E-03	1.96E-02	7.26E-01

Table 3.4-12

MONTHLY CO-58 DISCHARGES IN LIQUID

	RCS		NON-RCS		TOTAL	
Month	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	2.7E-04	9.94E-03	6.8E-06	2.53E-04	2.76E-04	1.02E-02
2	2.8E-04	1.05E-02	6.8E-06	2.53E-04	2.91E-04	1.08E-02
3	3.0E-04	1.12E-02	6.8E-06	2.53E-04	3.09E-04	1.14E-02
4	3.2E-04	1.19E-02	6.8E-06	2.53E-04	3.29E-04	1.22E-02
5	3.4E-04	1.28E-02	6.8E-06	2.53E-04	3.52E-04	1.30E-02
6	3.7E-04	1.37E-02	6.8E-06	2.53E-04	3.78E-04	1.40E-02
7	4.0E-04	1.49E-02	6.8E-06	2.53E-04	4.09E-04	1.51E-02
8	4.4E-04	1.62E-02	6.8E-06	2.53E-04	4.45E-04	1.65E-02
9	4.8E-04	1.78E-02	6.8E-06	2.53E-04	4.88E-04	1.81E-02
10	5.3E-04	1.98E-02	6.8E-06	2.53E-04	5.41E-04	2.00E-02
11	6.0E-04	2.22E-02	6.8E-06	2.53E-04	6.07E-04	2.24E-02
12	6.8E-04	2.53E-02	6.8E-06	2.53E-04	6.91E-04	2.56E-02
13	8.0E-04	2.95E-02	6.8E-06	2.53E-04	8.03E-04	2.97E-02
14	9.5E-04	3.52E-02	6.8E-06	2.53E-04	9.59E-04	3.55E-02
15	1.2E-03	4.39E-02	6.8E-06	2.53E-04	1.19E-03	4.41E-02
16	1.6E-03	5.81E-02	6.8E-06	2.53E-04	1.58E-03	5.83E-02
17	2.3E-03	8.62E-02	6.8E-06	2.53E-04	2.34E-03	8.65E-02
18	4.6E-03	1.71E-01	6.8E-06	2.53E-04	4.64E-03	1.72E-01
Total	1.65E-02	6.11E-01	1.2E-04	4.55E-03	1.66E-02	6.15E-01

Table 3.4-13

MONTHLY CO-60 DISCHARGES IN LIQUID

	RCS		NON-RCS		TOTAL	
Month	**curies**	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	1.5E-04	5.51E-03	3.3E-06	1.2E-04	1.52E-04	5.63E-03
2	1.6E-04	5.84E-03	3.3E-06	1.2E-04	1.61E-04	5.96E-03
3	1.7E-04	6.20E-03	3.3E-06	1.2E-04	1.71E-04	6.32E-03
4	1.8E-04	6.61E-03	3.3E-06	1.2E-04	1.82E-04	6.73E-03
5	1.9E-04	7.08E-03	3.3E-06	1.2E-04	1.94E-04	7.20E-03
6	2.1E-04	7.62E-03	3.3E-06	1.2E-04	2.09E-04	7.74E-03
7	2.2E-04	8.24E-03	3.3E-06	1.2E-04	2.26E-04	8.36E-03
8	2.4E-04	8.99E-03	3.3E-06	1.2E-04	2.46E-04	9.11E-03
9	2.7E-04	9.87E-03	3.3E-06	1.2E-04	2.70E-04	9.99E-03
10	3.0E-04	1.10E-02	3.3E-06	1.2E-04	2.99E-04	1.11E-02
11	3.3E-04	1.23E-02	3.3E-06	1.2E-04	3.36E-04	1.24E-02
12	3.8E-04	1.40E-02	3.3E-06	1.2E-04	3.83E-04	1.42E-02
13	4.4E-04	1.63E-02	3.3E-06	1.2E-04	4.45E-04	1.65E-02
14	5.3E-04	1.95E-02	3.3E-06	1.2E-04	5.31E-04	1.97E-02
15	6.6E-04	2.43E-02	3.3E-06	1.2E-04	6.61E-04	2.44E-02
16	8.7E-04	3.22E-02	3.3E-06	1.2E-04	8.74E-04	3.23E-02
17	1.3E-03	4.78E-02	3.3E-06	1.2E-04	1.30E-03	4.79E-02
18	2.6E-03	9.51E-02	3.3E-06	1.2E-04	2.57E-03	9.52E-02
Total	9.15E-03	3.39E-01	5.9E-05	2.16E-03	9.21E-03	3.41E-01

Table 3.4-14

MONTHLY NI-63 DISCHARGES IN LIQUID

Month	RCS		NON-RCS		TOTAL	
	curies	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	3.4E-04	1.27E-02	7.5E-06	2.78E-04	3.50E-04	1.29E-02
2	3.6E-04	1.34E-02	7.5E-06	2.78E-04	3.69E-04	1.37E-02
3	3.8E-04	1.42E-02	7.5E-06	2.78E-04	3.92E-04	1.45E-02
4	4.1E-04	1.52E-02	7.5E-06	2.78E-04	4.17E-04	1.54E-02
5	4.4E-04	1.62E-02	7.5E-06	2.78E-04	4.46E-04	1.65E-02
6	4.7E-04	1.75E-02	7.5E-06	2.78E-04	4.80E-04	1.78E-02
7	5.1E-04	1.89E-02	7.5E-06	2.78E-04	5.19E-04	1.92E-02
8	5.6E-04	2.06E-02	7.5E-06	2.78E-04	5.65E-04	2.09E-02
9	6.1E-04	2.27E-02	7.5E-06	2.78E-04	6.20E-04	2.29E-02
10	6.8E-04	2.51E-02	7.5E-06	2.78E-04	6.87E-04	2.54E-02
11	7.6E-04	2.82E-02	7.5E-06	2.78E-04	7.71E-04	2.85E-02
12	8.7E-04	3.22E-02	7.5E-06	2.78E-04	8.78E-04	3.25E-02
13	1.0E-03	3.75E-02	7.5E-06	2.78E-04	1.02E-03	3.78E-02
14	1.2E-03	4.49E-02	7.5E-06	2.78E-04	1.22E-03	4.51E-02
15	1.5E-03	5.58E-02	7.5E-06	2.78E-04	1.52E-03	5.61E-02
16	2.0E-03	7.39E-02	7.5E-06	2.78E-04	2.01E-03	7.42E-02
17	3.0E-03	1.10E-01	7.5E-06	2.78E-04	2.97E-03	1.10E-01
18	5.9E-03	2.18E-01	7.5E-06	2.78E-04	5.90E-03	2.18E-01
Total	2.10E-02	7.77E-01	1.4E-04	5.00E-03	1.35E-04*	7.82E-01

Table 3.4-15

MONTHLY SR-90 DISCHARGES IN LIQUID

Month	RCS		NON-RCS		TOTAL	
	curies	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	1.6E-07	5.97E-06	negl.	negl.	1.61E-07	1.61E-07
2	1.7E-07	6.31E-06	negl.	negl.	1.71E-07	1.71E-07
3	1.8E-07	6.71E-06	negl.	negl.	1.81E-07	1.81E-07
4	1.9E-07	7.15E-06	negl.	negl.	1.93E-07	1.93E-07
5	2.1E-07	7.66E-06	negl.	negl.	2.07E-07	2.07E-07
6	2.2E-07	8.24E-06	negl.	negl.	2.23E-07	2.23E-07
7	2.4E-07	8.92E-06	negl.	negl.	2.41E-07	2.41E-07
8	2.6E-07	9.72E-06	negl.	negl.	2.63E-07	2.63E-07
9	2.9E-07	1.07E-05	negl.	negl.	2.89E-07	2.89E-07
10	3.2E-07	1.19E-05	negl.	negl.	3.20E-07	3.20E-07
11	3.6E-07	1.33E-05	negl.	negl.	3.60E-07	3.60E-07
12	4.1E-07	1.52E-05	negl.	negl.	4.11E-07	4.11E-07
13	4.8E-07	1.77E-05	negl.	negl.	4.78E-07	4.78E-07
14	5.7E-07	2.11E-05	negl.	negl.	5.72E-07	5.72E-07
15	7.1E-07	2.63E-05	negl.	negl.	7.11E-07	7.11E-07
16	9.4E-07	3.48E-05	negl.	negl.	9.42E-07	9.42E-07
17	1.4E-06	5.17E-05	negl.	negl.	1.40E-06	1.40E-06
18	2.8E-06	1.03E-04	negl.	negl.	2.78E-06	2.78E-06
Total	9.90E-06	3.66E-04	negl.	negl.	1.00E-05	3.66E-04

Table 3.4-16

MONTHLY CS-137 DISCHARGES IN LIQUID

Month	RCS		NON-RCS		TOTAL	
	curies	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	1.5E-05	5.51E-04	3.6E-07	1.33E-05	1.53E-05	5.65E-04
2	1.6E-05	5.84E-04	3.6E-07	1.33E-05	1.61E-05	5.97E-04
3	1.7E-05	6.20E-04	3.6E-07	1.33E-05	1.71E-05	6.33E-04
4	1.8E-05	6.61E-04	3.6E-07	1.33E-05	1.82E-05	6.74E-04
5	1.9E-05	7.08E-04	3.6E-07	1.33E-05	1.95E-05	7.21E-04
6	2.1E-05	7.62E-04	3.6E-07	1.33E-05	2.09E-05	7.75E-04
7	2.2E-05	8.24E-04	3.6E-07	1.33E-05	2.26E-05	8.38E-04
8	2.4E-05	8.99E-04	3.6E-07	1.33E-05	2.46E-05	9.12E-04
9	2.7E-05	9.87E-04	3.6E-07	1.33E-05	2.70E-05	1.00E-03
10	3.0E-05	1.10E-03	3.6E-07	1.33E-05	3.00E-05	1.11E-03
11	3.3E-05	1.23E-03	3.6E-07	1.33E-05	3.36E-05	1.24E-03
12	3.8E-05	1.40E-03	3.6E-07	1.33E-05	3.83E-05	1.42E-03
13	4.4E-05	1.63E-03	3.6E-07	1.33E-05	4.45E-05	1.65E-03
14	5.3E-05	1.95E-03	3.6E-07	1.33E-05	5.32E-05	1.97E-03
15	6.6E-05	2.43E-03	3.6E-07	1.33E-05	6.61E-05	2.45E-03
16	8.7E-05	3.22E-03	3.6E-07	1.33E-05	8.74E-05	3.23E-03
17	1.3E-04	4.78E-03	3.6E-07	1.33E-05	1.30E-04	4.80E-03
18	2.6E-04	9.51E-03	3.6E-07	1.33E-05	2.57E-04	9.52E-03
Total	9.15E-04	3.39E-02	6.5E-06	2.26E-04	9.21E-04	3.41E-02

Table 3.4-17

MONTHLY PU-241 DISCHARGES IN LIQUID

Month	RCS		NON-RCS		TOTAL	
	curies	GBq	curies	GBq	curies	GBq
1	5.4E-08	1.99E-06	negl.	negl.	5.37E-08	1.99E-06
2	5.7E-08	2.10E-06	negl.	negl.	5.69E-08	2.10E-06
3	6.0E-08	2.24E-06	negl.	negl.	6.04E-08	2.24E-06
4	6.4E-08	2.38E-06	negl.	negl.	6.44E-08	2.38E-06
5	6.9E-08	2.55E-06	negl.	negl.	6.90E-08	2.55E-06
6	7.4E-08	2.75E-06	negl.	negl.	7.42E-08	2.75E-06
7	8.0E-08	2.97E-06	negl.	negl.	8.04E-08	2.97E-06
8	8.8E-08	3.24E-06	negl.	negl.	8.76E-08	3.24E-06
9	9.6E-08	3.56E-06	negl.	negl.	9.62E-08	3.56E-06
10	1.1E-07	3.95E-06	negl.	negl.	1.07E-07	3.95E-06
11	1.2E-07	4.44E-06	negl.	negl.	1.20E-07	4.44E-06
12	1.4E-07	5.06E-06	negl.	negl.	1.37E-07	5.06E-06
13	1.6E-07	5.89E-06	negl.	negl.	1.59E-07	5.89E-06
14	1.9E-07	7.05E-06	negl.	negl.	1.91E-07	7.05E-06
15	2.4E-07	8.77E-06	negl.	negl.	2.37E-07	8.77E-06
16	3.1E-07	1.16E-05	negl.	negl.	3.14E-07	1.16E-05
17	4.7E-07	1.72E-05	negl.	negl.	4.66E-07	1.72E-05
18	9.3E-07	3.43E-05	negl.	negl.	9.27E-07	3.43E-05
Total	3.30E-06	1.22E-04	negl.	negl.	3.30E-06	1.22E-04

Table 3.4-18

MONTHLY OTHER PARTICULATE DISCHARGES IN LIQUID

Month	RCS		NON-RCS		TOTAL	
	curies	**GBq**	**curies**	**GBq**	**curies**	**GBq**
1	4.1E-04	1.5E-02	3.4E-05	1.26E-03	4.40E-04	1.63E-02
2	4.3E-04	1.6E-02	3.4E-05	1.26E-03	4.63E-04	1.71E-02
3	4.6E-04	1.7E-02	3.4E-05	1.26E-03	4.90E-04	1.81E-02
4	4.9E-04	1.8E-02	3.4E-05	1.26E-03	5.20E-04	1.92E-02
5	5.2E-04	1.9E-02	3.4E-05	1.26E-03	5.55E-04	2.05E-02
6	5.6E-04	2.1E-02	3.4E-05	1.26E-03	5.94E-04	2.20E-02
7	6.1E-04	2.2E-02	3.4E-05	1.26E-03	6.40E-04	2.37E-02
8	6.6E-04	2.4E-02	3.4E-05	1.26E-03	6.95E-04	2.57E-02
9	7.3E-04	2.7E-02	3.4E-05	1.26E-03	7.60E-04	2.81E-02
10	8.1E-04	3.0E-02	3.4E-05	1.26E-03	8.40E-04	3.11E-02
11	9.1E-04	3.3E-02	3.4E-05	1.26E-03	9.39E-04	3.48E-02
12	1.0E-03	3.8E-02	3.4E-05	1.26E-03	1.07E-03	3.95E-02
13	1.2E-03	4.4E-02	3.4E-05	1.26E-03	1.24E-03	4.57E-02
14	1.4E-03	5.3E-02	3.4E-05	1.26E-03	1.47E-03	5.45E-02
15	1.8E-03	6.6E-02	3.4E-05	1.26E-03	1.82E-03	6.74E-02
16	2.4E-03	8.8E-02	3.4E-05	1.26E-03	2.40E-03	8.89E-02
17	3.5E-03	1.3E-01	3.4E-05	1.26E-03	3.55E-03	1.31E-01
18	7.0E-03	2.6E-01	3.4E-05	1.26E-03	7.03E-03	2.60E-01
Total	2.49E-02	9.2E-01	6.15E-04	2.28E-02	2.55E-02	9.44E-01

Table 3.4-19

COMPARISON OF AP1000 NPP LIQUID RADIOACTIVE DISCHARGES OF TRITIUM WITH EUROPEAN NUCLEAR POWER PLANTS BETWEEN 1995 AND 1998

	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 3.4-20

COMPARISON OF AP1000 NPP LIQUID RADIOACTIVE DISCHARGES OF RADIONUCLIDES OTHER THAN TRITIUM WITH EUROPEAN NUCLEAR POWER PLANTS BETWEEN 1995 AND 1998

	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

Note:

Data in Tables 3.4-19 and 3.4-20 for other nuclear plants extracted from measured data reported in "Implementation of PARCOM Recommendation 91/4 on Radioactive Discharges," OSPAR Commission 2003 (Reference 3-10). **AP1000** NPP data based upon estimated monthly discharge calculations.

Table 3.4-21

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

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

Table 3.5-1

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 3.5-1 (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 – cannedpump	LLW	Once/60 yr	0.06		0.06
Resin transfer screw pump	LLW	Once/10 yr	0.003		0.02

Table 3.5-2

ANNUAL AVERAGE SOLID RADWASTE PRODUCTION[1]

Waste Classification	Raw Waste Volume m^3/y	Treated Waste Volume m^3/y	Internal Packaged Waste Volume[2] m^3/y	Container Type	No. of Waste Containers No./y	Volume of Containers m^3/y
ILW Total	10.26	40.86[3]	41.8	RWM $3m^3$ (100 ft^3) drums or boxes	19	68.9[6]
LLW non-maintenance	161.85	59.05[4]	59.2	200 litre (55 gal) drums or HHISO	413[5] [9] or 12[5] [9]	108.6[7] [9] or 234[8] [9]
LLW maintenance	13.75	13.68[5] [9]	23.38[9]			
LLW Total	175.6	72.73	82.58			

Note:

1. Reference 3-26
2. Treated waste volume + void space associated with partially filled containers
3. Volume following encapsulation
4. Volume following compaction
5. Each HHISO container holds thirty-nine 200 litre (55 gal) drums
6. Based on RWM box with external dimensions 1.72 m x 1.72 m x 1.225 m (5.64 ft x 5.64 ft x 4.019 ft)
7. Based on 200 L (55 gal) drum with external dimensions 0.615 m (2.02 ft) diam (with lid clamp fitted) x 0.886 m (2.91 ft) high
8. Based on HHISO external volume of 19.5 m^3 (688 ft^3)
9. This is based on one HEPA filter per 200 L (55 gal) drum. This gives a filter loading of 57% and hence a void space of 43% of the drum. This volume could be reduced by packaging other LLW in the same drum as the HEPA filters and thus reducing the void space per drum. Also, compaction may be used to reduce volume. Utility operators may adopt techniques and develop procedures to improve packaging efficiencies.

Table 3.5-3

ACTINIDES IN DRY ACTIVE SOLID RADWASTE

Actinide	Activity (Bq/g)	Activity[1] (GBq/y)
Pu-238	1.26E-03	2.11E-05
Pu-239	3.0E-03	5.03E-05
Pu-240	2.8E-04	4.69E-06
Pu-241	3.6E-01	6.04E-03
Am-241	5.4E-03	9.05E-05
Cm-242	9.4E-04	1.58E-05
Cm-243	4.8E-04	8.05E-06
Cm-244	3.2E-04	5.37E-06

Note:

1. Activity assumes 16.767 tonnes/y of dry active solid waste is produced. These numbers are conservative estimates, because fuel performance is improving; the above values assume previous fuel designs, not the RFA fuel design.

Table 3.5-4

COLOUR CODING REPRESENTATIONS FOR INITIAL SCREENING OF RADWASTE TREATMENT OPTIONS

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

Table 3.5-5

INITIAL RADWASTE TREATMENT OPTION SCREENING RESULTS (REFERENCE 3-16)

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. It is a potential viable process for hollow items (e.g., tubes, canisters, but not for valves and solid items).

Table 3.5-5 (cont.)

INITIAL RADWASTE TREATMENT OPTION SCREENING RESULTS (REFERENCE 3-16)

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, it 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.
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.

Table 3.5-5 (cont.)

INITIAL RADWASTE TREATMENT OPTION SCREENING RESULTS (REFERENCE 3-16)

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	
Vitrification	Y	Y	Y	N	N	Y	4	2	Single UK application on liquid HLW, several applications 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 (i.e., encapsulation). No full scale nuclear application 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 Oxidation	Y	N	Y	N	N	N	4	n/a	One UK licensed mobile plant. Partial solution only – requires further treatment.

*Note that ILW resins (organic and inorganic) and ILW charcoal will be treated via the same waste stream.

Table 3.5-6

MAPPING OF SCORING REQUIREMENTS AGAINST BAT CRITERIA (REFERENCE 3-16)

Criterion		Weight	Score 1	Score 2	Score 3	Score 4	Score 5	Description
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 licence position.	Novel concept which has undergone a significant amount of development to underpin its feasibility. Little/no evidence of full scale deployment either in UK 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 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 (i.e. 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.

Table 3.5-6 (cont.)

MAPPING OF SCORING REQUIREMENTS AGAINST BAT CRITERIA (REFERENCE 3-16)

Criterion		Weight	Score					Description
			1	2	3	4	5	
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.
	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 noumber 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 (i.e., 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 noumber of or high consequence potential accident scenarios. Significant reliance on managerial control and protective measures.	Medium number of or medium consequence potential accident scenarios. Moderate reliance on managerial control & 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 (temperature pressure, height, confined space, moving machinery, etc.). 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.

Table 3.5-6 (cont.)

MAPPING OF SCORING REQUIREMENTS AGAINST BAT CRITERIA (REFERENCE 3-16)

Criterion		Weight	Score 1	Score 2	Score 3	Score 4	Score 5	Description
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 whilst 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 (volume).
	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 a 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.L.G. waste streams including new liabilities and consumables (e.g., filters or other media). Does not consider generic effluents (e.g., washdown that are 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 (e.g., with respect to public inquiry) and is particularly relevant to options such as incinerators.

Table 3.5-6 (cont.)

MAPPING OF SCORING REQUIREMENTS AGAINST BAT CRITERIA (REFERENCE 3-16)

Criterion		Weight	Score 1	Score 2	Score 3	Score 4	Score 5	Description
	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 L.o.C. 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 RWM by meeting their CFA for the ILW Repository: immobilised, free of water, homogeneous, radiologically stable, chemically passive (i.e., zero gas generation), characterised, voids minimised (Ref. 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, and energy. Does not consider the 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.

Table 3.5-6 (cont.)

MAPPING OF SCORING REQUIREMENTS AGAINST BAT CRITERIA (REFERENCE 3-16)

Criterion		Weight	Score					Description
			1	2	3	4	5	
Economic	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 and 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 3.5-7

Mapping of Scoring Requirements Against BAT Criteria (Reference 3-16)														
		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**	**Implementation 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

Table 3.5-7 (continued)

Mapping of Scoring Requirements Against BAT Criteria (Reference 3-16)														
		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**	**Implementation Time**	**Process Technology Costs**	
Immobi lisation	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 3.5-8

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

BPEO Issue (Reference 3-20)	Options Considered in Sizewell B BPEO Study (Reference 3-20)	Sizewell B Recommendation following BPEO Study (Reference 3-20)	Environment Agency Comment (Reference 3-21)	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 LLW repository	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 LLW repository

Table 3.5-8 (cont.)

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

BPEO Issue (Reference 3-20)	Options Considered in Sizewell B BPEO Study (Reference 3-20)	Sizewell B Recommendation following BPEO Study (Reference 3-20)	Environment Agency Comment (Reference 3-21)	AP1000 NPP Comparison
Trash solid LLW	Volume reduction and disposal to the LLW repository Recycling Decontamination	Disposal to the LLW repository	Disposal to the LLW repository with volume reduction as a matter for BPM consideration	Disposal to the LLW repository
LLW resins	Encapsulation in cement, disposal to the LLW repository	Encapsulation in cement, disposal to the LLW repository	Considered encapsulation in cement, disposal to the LLW repository to be suitable	Accumulation until end of generation cycle Commercial (off-site) incinerator or encapsulation, storage on-site, disposal to LLW repository
LLW sludge – oily	Disposal to the LLW repository	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 LLW repository, following pre-treatment Incineration	Disposal to the LLW repository, following pretreatment subject to BPM constraints	Agreed disposal to the LLW repository, following pretreatment, if necessary	Compaction, disposal to the LLW repository

Table 3.5-8 (cont.)

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

BPEO Issue (Reference 3-20)	Options Considered in Sizewell B BPEO Study (Reference 3-20)	Sizewell B Recommendation following BPEO Study (Reference 3-20)	Environment Agency Comment (Reference 3-21)	AP1000 NPP Comparison
Wet ILW	Early encapsulation, long-term storage, disposal to national repository Pyrolysis* Acid Digestion*	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:
* Novel applications with limited relevant information on which to base an assessment.

Table 3.5-9

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 3-38)**	**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 (58 gal)) compaction Non-compactable: • Placement in 220 litre (58 gal) 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 dose rate level and content of nuclides.

Table 3.5-9 (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 3-38)**	**EDF**	**RWE – arious**	**Iberdrola – Various**	**Ringhals**
ILW Resins	• Dewatering • Transfer to 200 litre (55 gal) drums • Transfer to drying / heating (thermal oil) vessel • Transfer to metal press drums and lidded • High energy compaction • Pellets packed in 200 (55 gal) litre drums • Interim store or final repository	• Dewatering • Cement or Polymer immobilisation • Concrete cask	• In-package drying • Vapour condensate treatment • Storage in MOSAIK cask	• Dewatering • Cement immobilisation in 220 litre (58 gal) 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/RWM Generic Waste Package Specification	No	Yes	No	Yes	Yes

Table 3.5-10

SUMMARY OF MAIN RADWASTE ARISINGS FROM DECOMMISSIONED PROCESS EQUIPMENT

Waste Description	Waste Level	Volume[(1)]		Mass		Notes
		cubic feet	cubic metres	pounds	tonnes	
Reactor Vessel and Pressuriser tanks	ILW	13703	388	936,846	426	
Pumps – various	ILW	3540	100	719488	327	
Reactor System Internals	ILW	*	*	321969	146	*individual pieces
Process Equipment Internals – various	ILW	6542	185	641,094	291	
Heat Exchangers	ILW	332	9	21,280	10	
Filters – various	ILW	203	6	18,200	8	
Pressuriser Heaters	ILW	2	0.05	358	0.16	
Waste from system decontamination operations (e.g. spent resins and spent filter cartridges)	ILW	~3002	~85	~225	~102	Mass based on average density 1200kg/m^3
Steam Generators / Heat Exchangers – various	LLW	52,732	1,493	2,818,560	1,281	
Reactor Integrated Head Package	LLW	7,917	224	258,676	118	
Tanks	LLW	28,510	808	205,899	94	
Ion exchange systems	LLW	532	15.1	60,998	28	
Pumps	LLW	1,135	32	35489	16	
Fasteners	LLW	133	4	33,900	15	
Insulation	LLW	765	22	18,400	8	

HVAC Filters – various	LLW	1,224	35	17,900	8	

Table 3.5-10 (cont).

SUMMARY OF MAIN RADWASTE ARISINGS FROM DECOMMISSIONED PROCESS EQUIPMENT

Waste Description	**Waste Level**	**Volume(1)**		**Mass**		**Notes**
		cubic feet	**cubic metres**	**pounds**	**tonnes**	
Adsorbers	LLW	252	7	9,245	4	
Small vessels	LLW	0.53	0.02	51	0.02	
Compacted dry active waste generated during decontamination operations	LLW	2860	81	267	121	Mass based on average density 1500kg/m^3

Notes:
1. Volume basis may include protruding appendages such as nozzles and brackets allowing the component to fit into an overpack for transport without modification.

Table 3.5-11

COMPARISON OF AP1000 NPP ILW/LLW PRODUCTION AGAINST OTHER TYPES OF UK NPPS

Reactor Type	ILW and LLW (m^3 per GW(e)-y)[1]
Magnox	1800[(2, 3)]
AGR	890.3[(2)]
PWR	430[(2)]
AP1000 NPP	102[(4)]

Notes:
1. Volumes are for wastes packaged for long-term management based on the probable conditioning method and container type. Station operational and decommissioning wastes are included.
2. Data source Reference 3-32.
3. Operating stations only.
4. Estimated operational waste only.

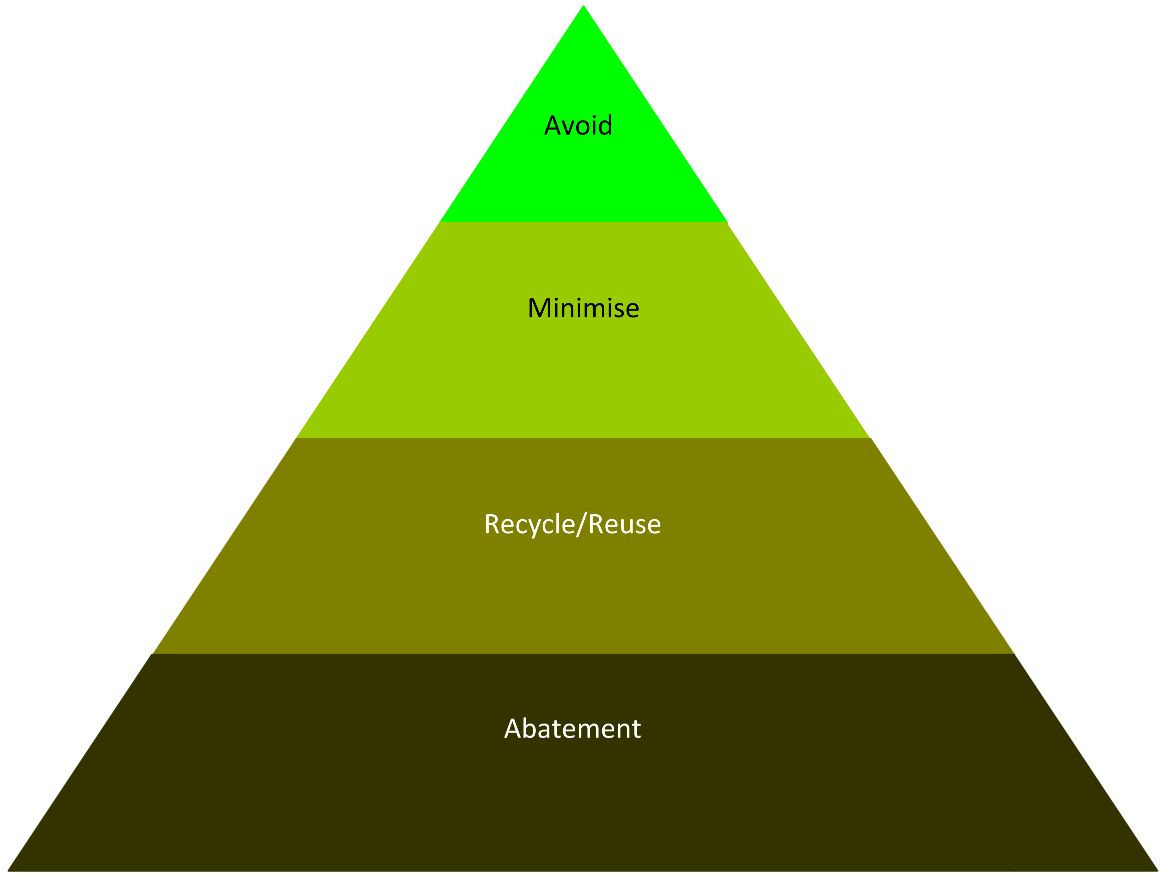


Figure 3.1-1. Waste Management Hierarchy

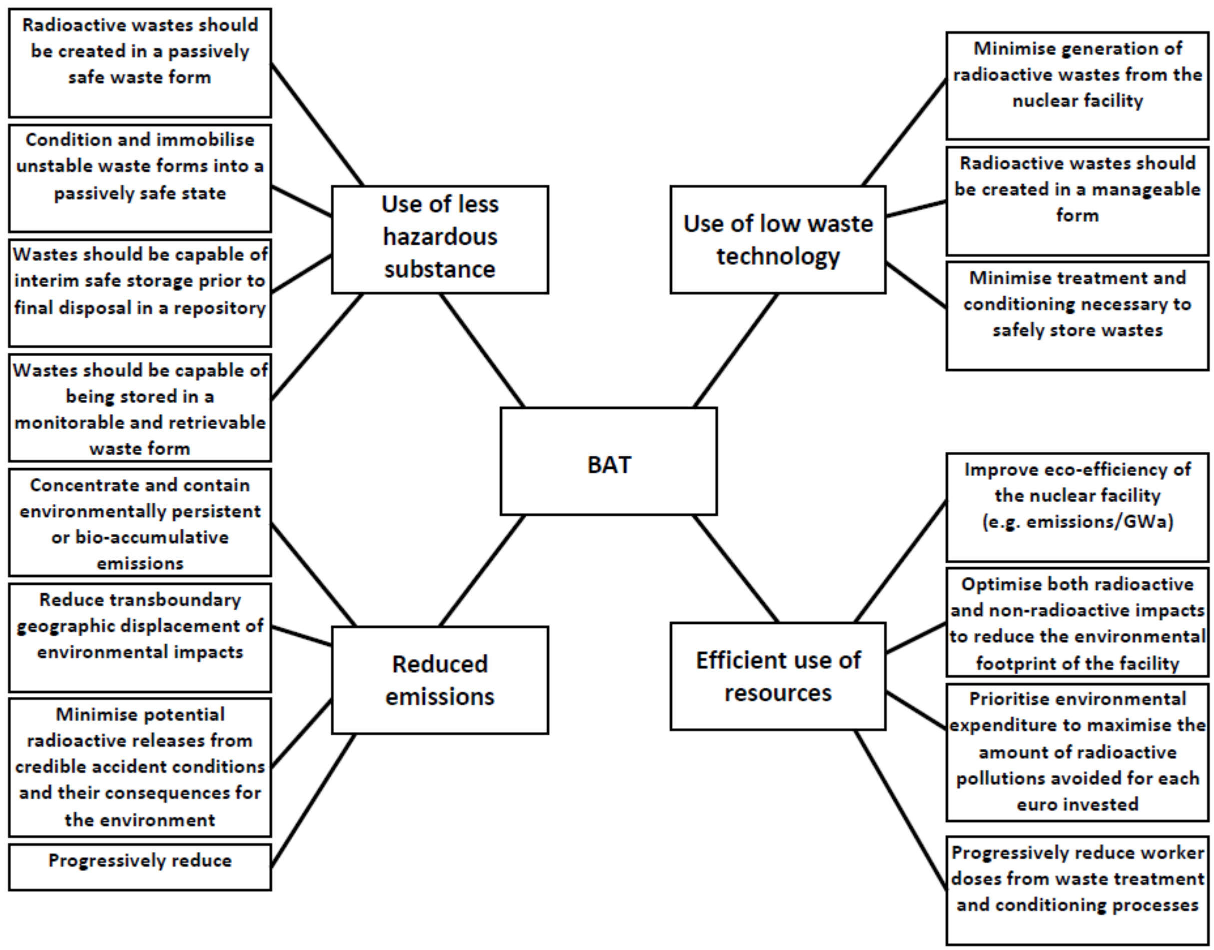


Figure 3.1-2. Nuclear BAT Management Factors for Optimisation of Releases from Nuclear Facilities (Reference 3-10)

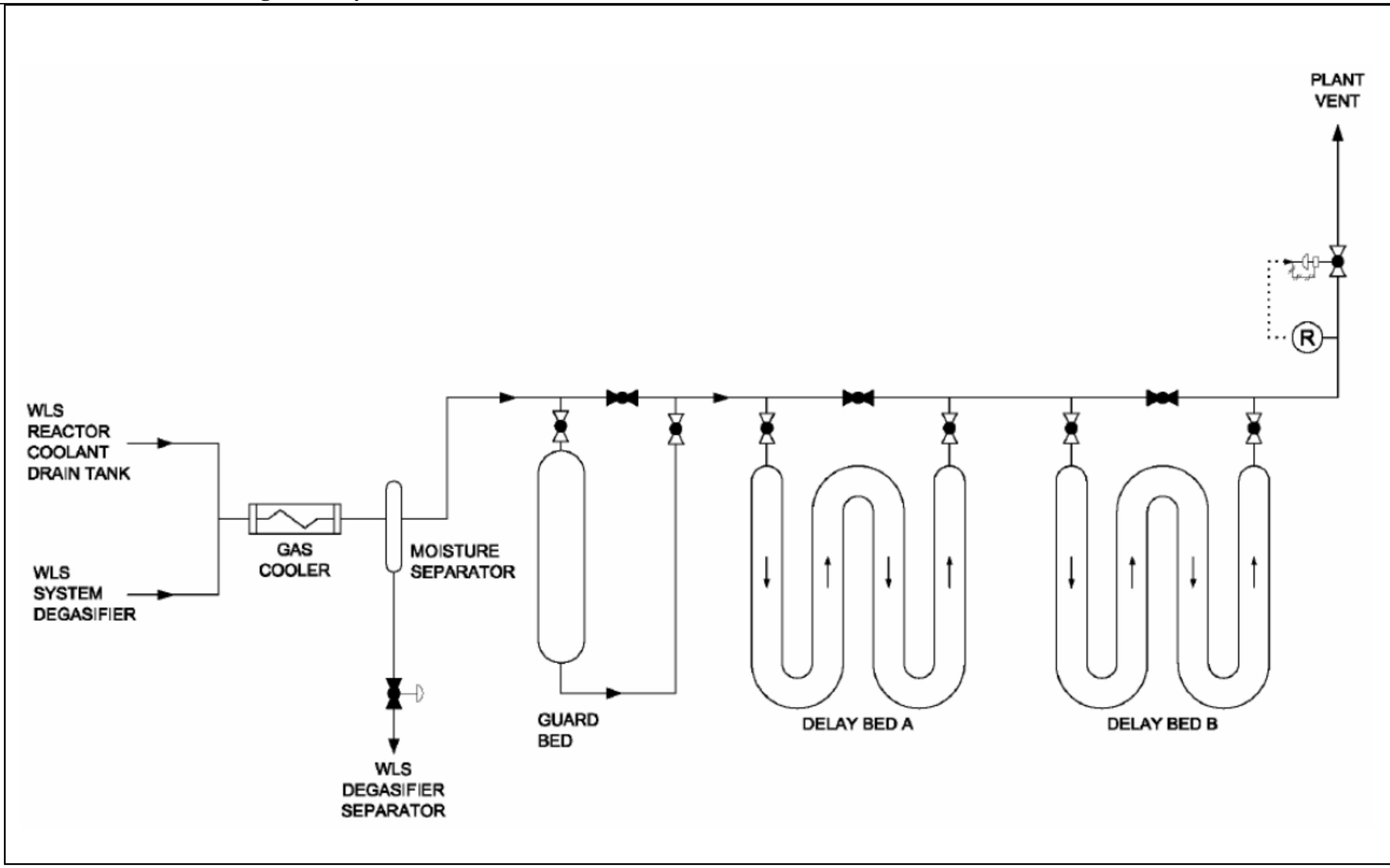
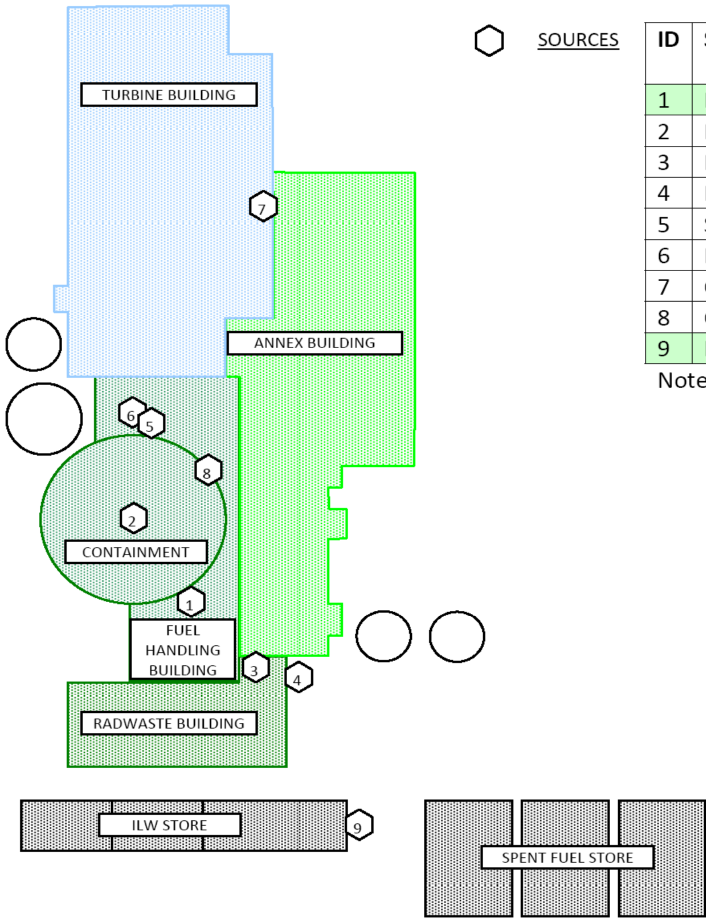


Figure 3.3-1. AP1000 NPP Gaseous Radwaste System



ID	Source Description	Release Elevation[1] (m)
1	Plant Vent [2]	55.7
2	PCS Air Diffuser [3]	69.8
3	Fuel Building Blowout Panel [3]	17.4
4	Fuel Building Rail Bay Door [3]	1.5
5	Steam Vent[3]	17.1
6	PORV/Safety Valves [3]	19.2
7	Condenser Air Removal Stack [3]	38.4
8	Containment Shell (Diffuse Area Source) [3]	Variable
9	ILW Store Stack [2]	TBD

Notes: (1) All elevations relative to grade at 0.0m
(2) Main radioactive emission sources during normal operations
(3) Radioactive emissions only during abnormal conditions

Figure 3.3-2. AP1000 NPP Air Emission Point Source Locations

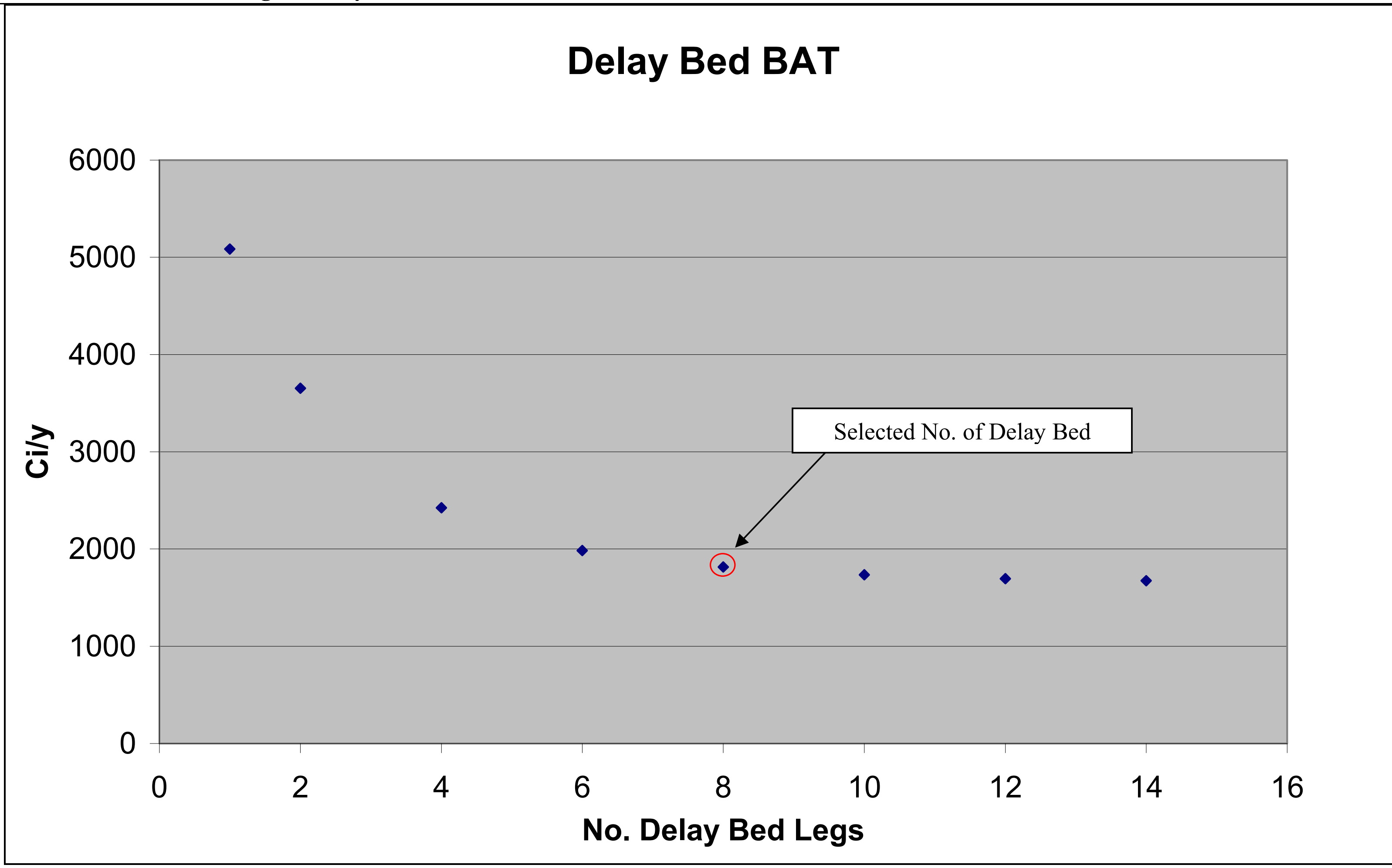


Figure 3.3-3. BAT Sizing of WGS Delay Beds

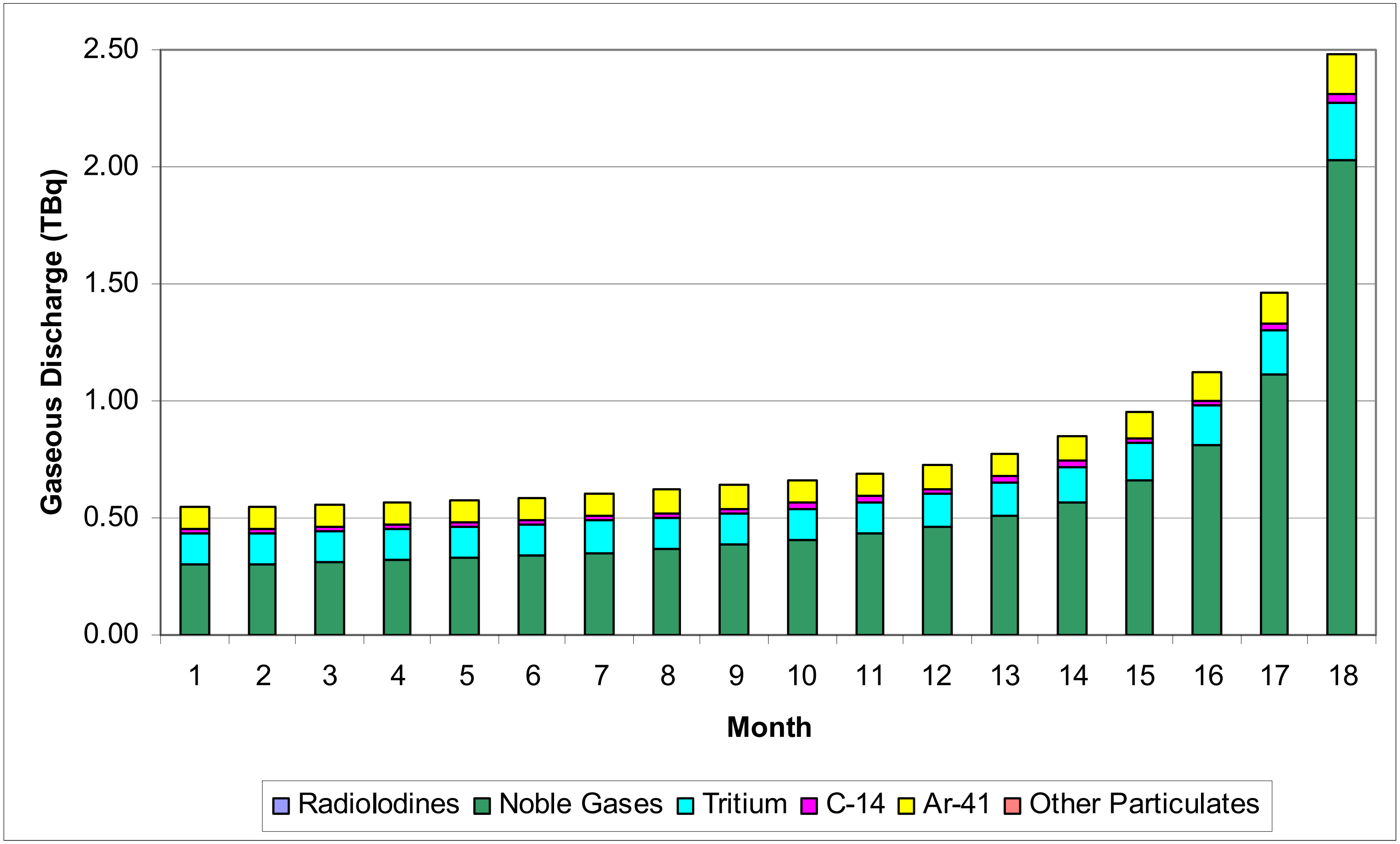


Figure 3.3-4. AP1000 NPP Monthly Gaseous Radioactive Emissions

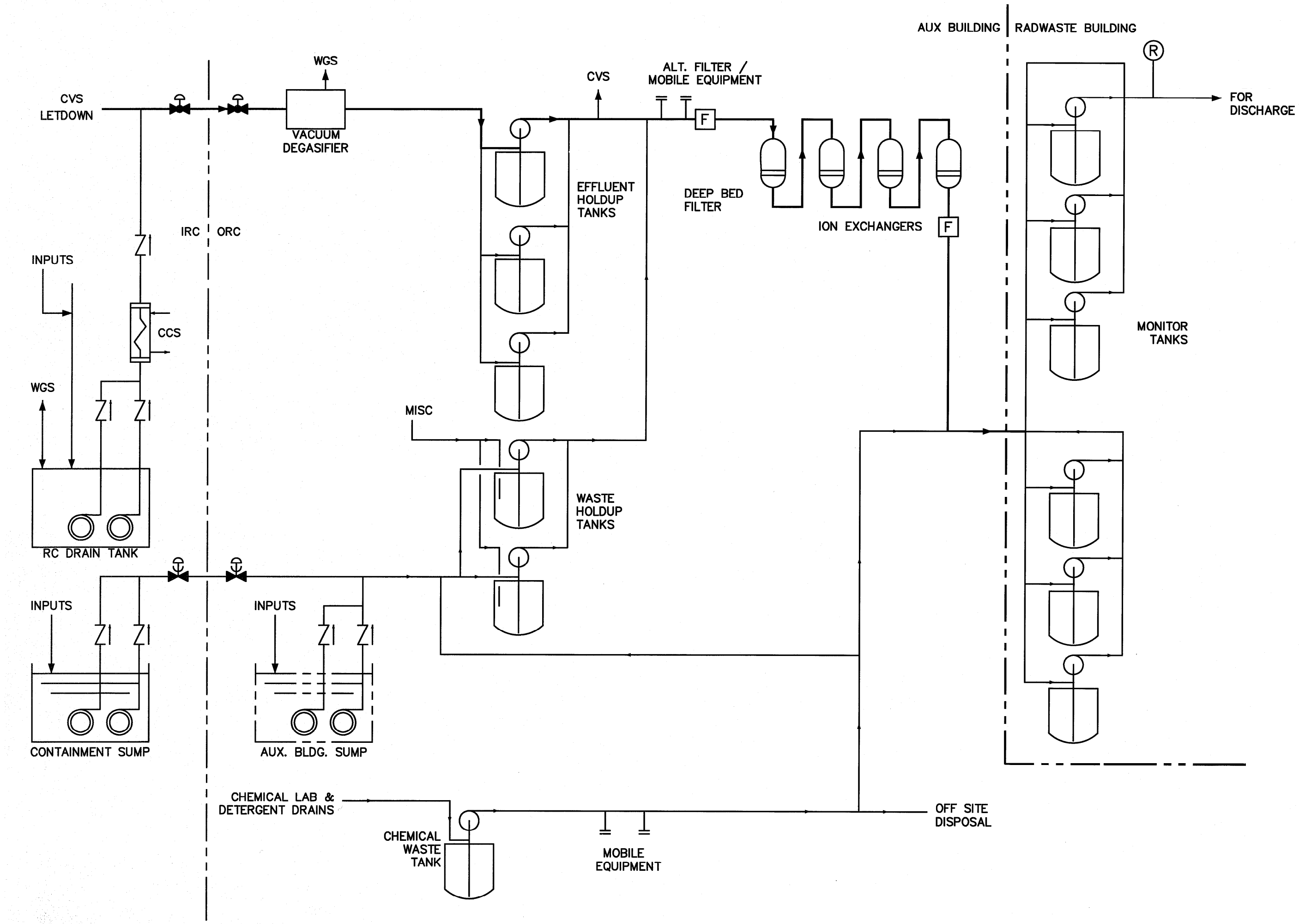
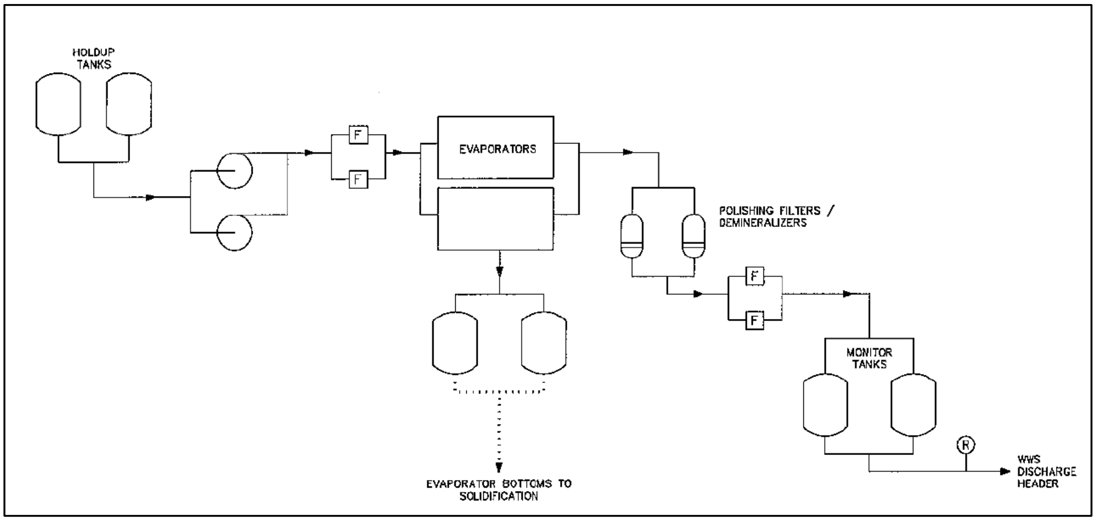


Figure 3.4-1. AP1000 NPP Liquid Radwaste System

EVAPORATOR APPROACH



ION EXCHANGE APPROACH

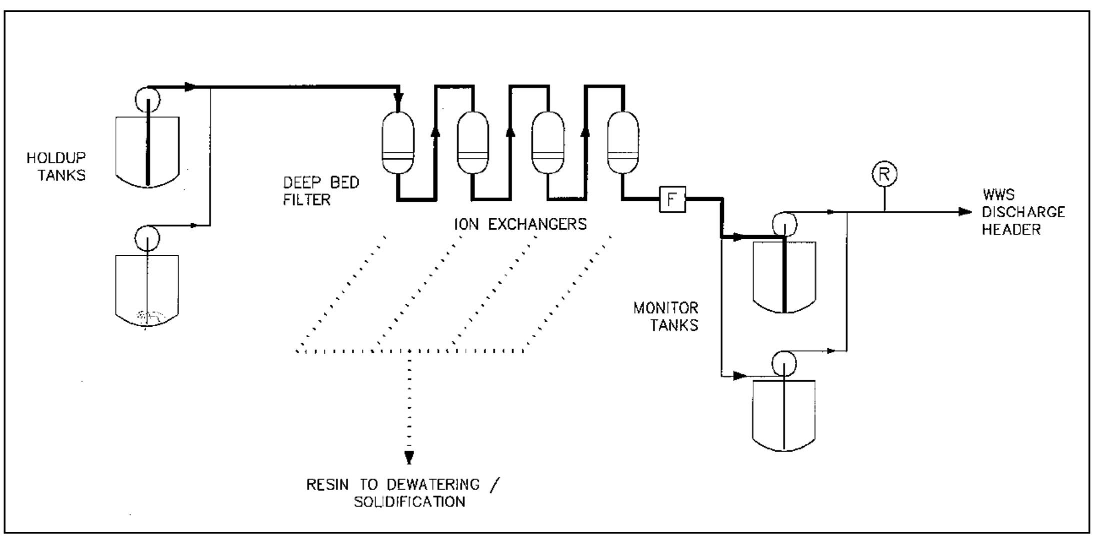
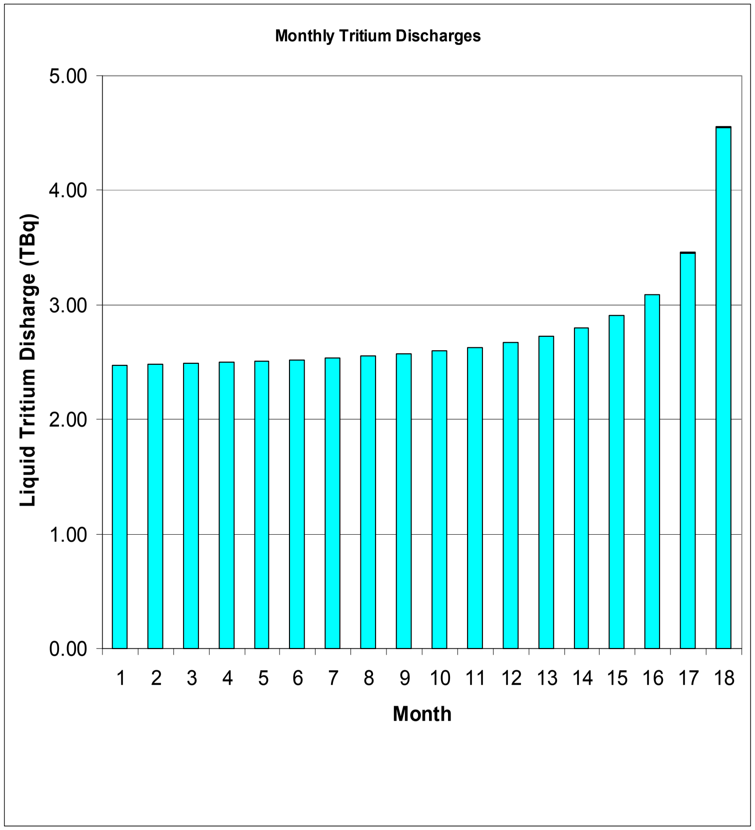
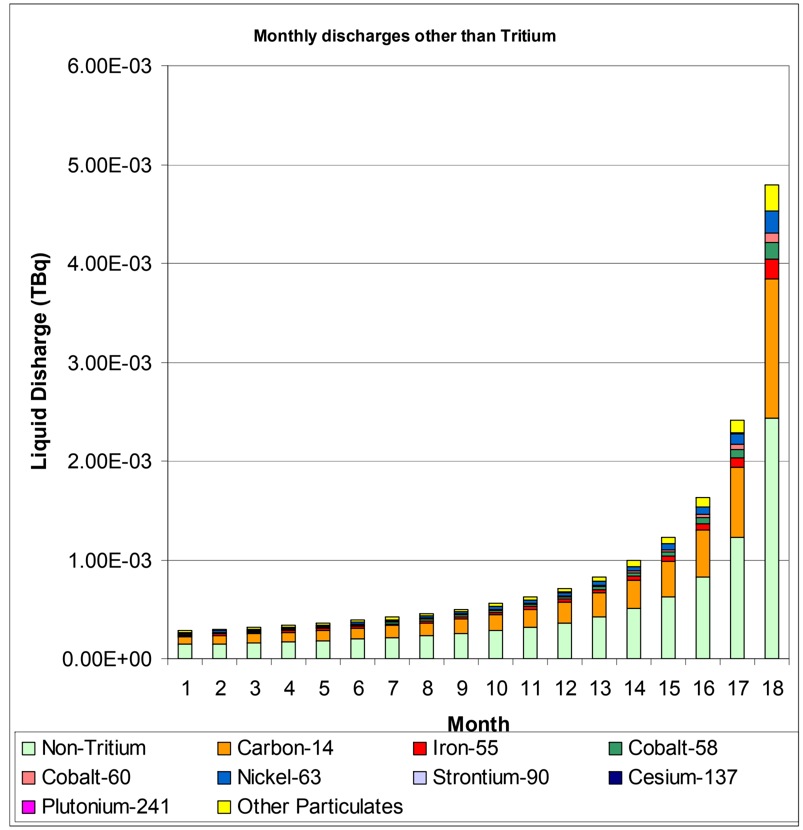


Figure 3.4-2. Comparison of Evaporator and Ion Exchange Flow Sheets for Liquid Radwaste Treatment



TRITIUM



NON-TRITIUM ISOTOPES

Figure 3.4-3. AP1000 NPP Monthly Liquid Radioactive Discharges

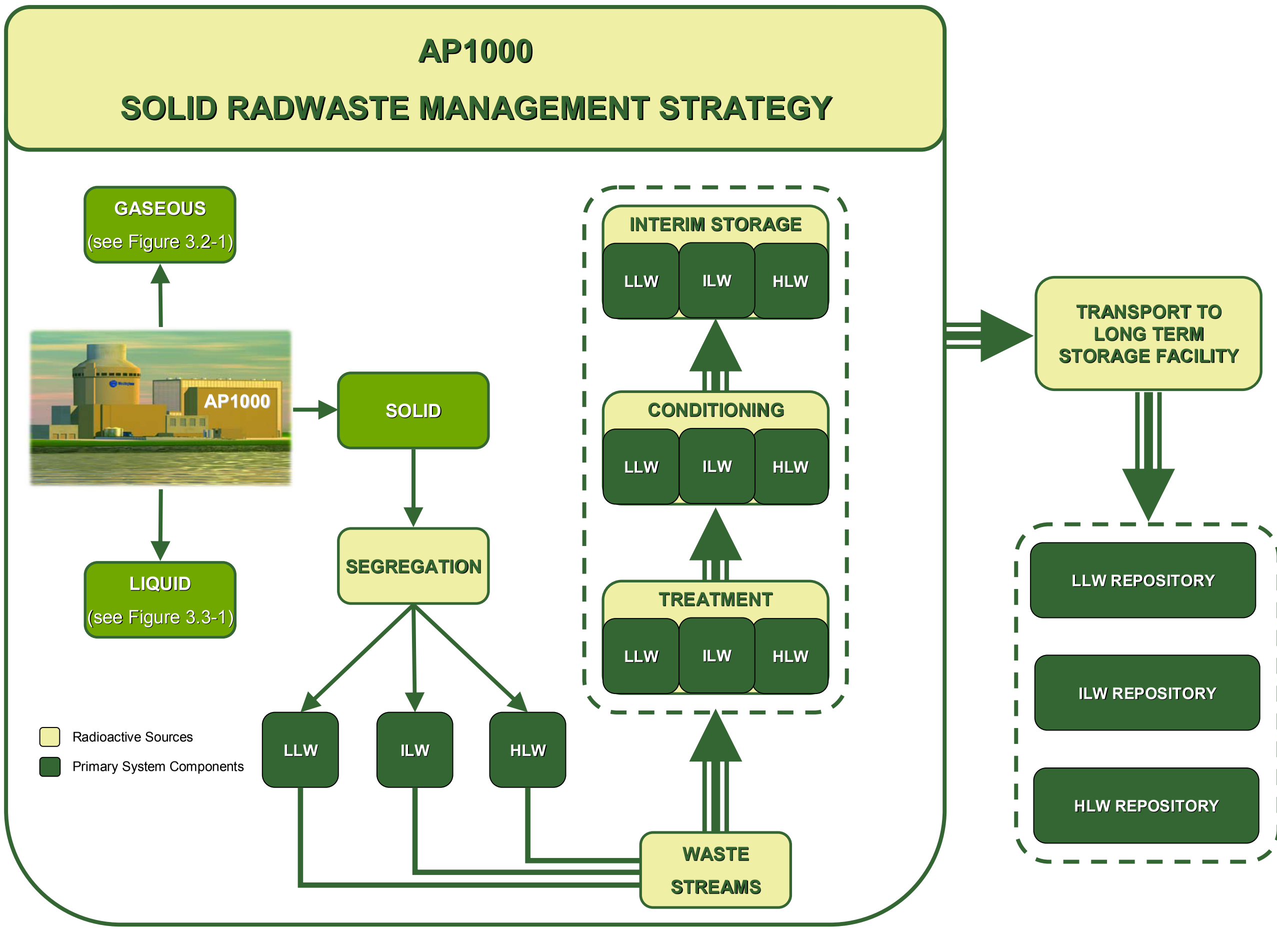


Figure 3.5-1. AP1000 NPP Solid Radwaste Management Strategy

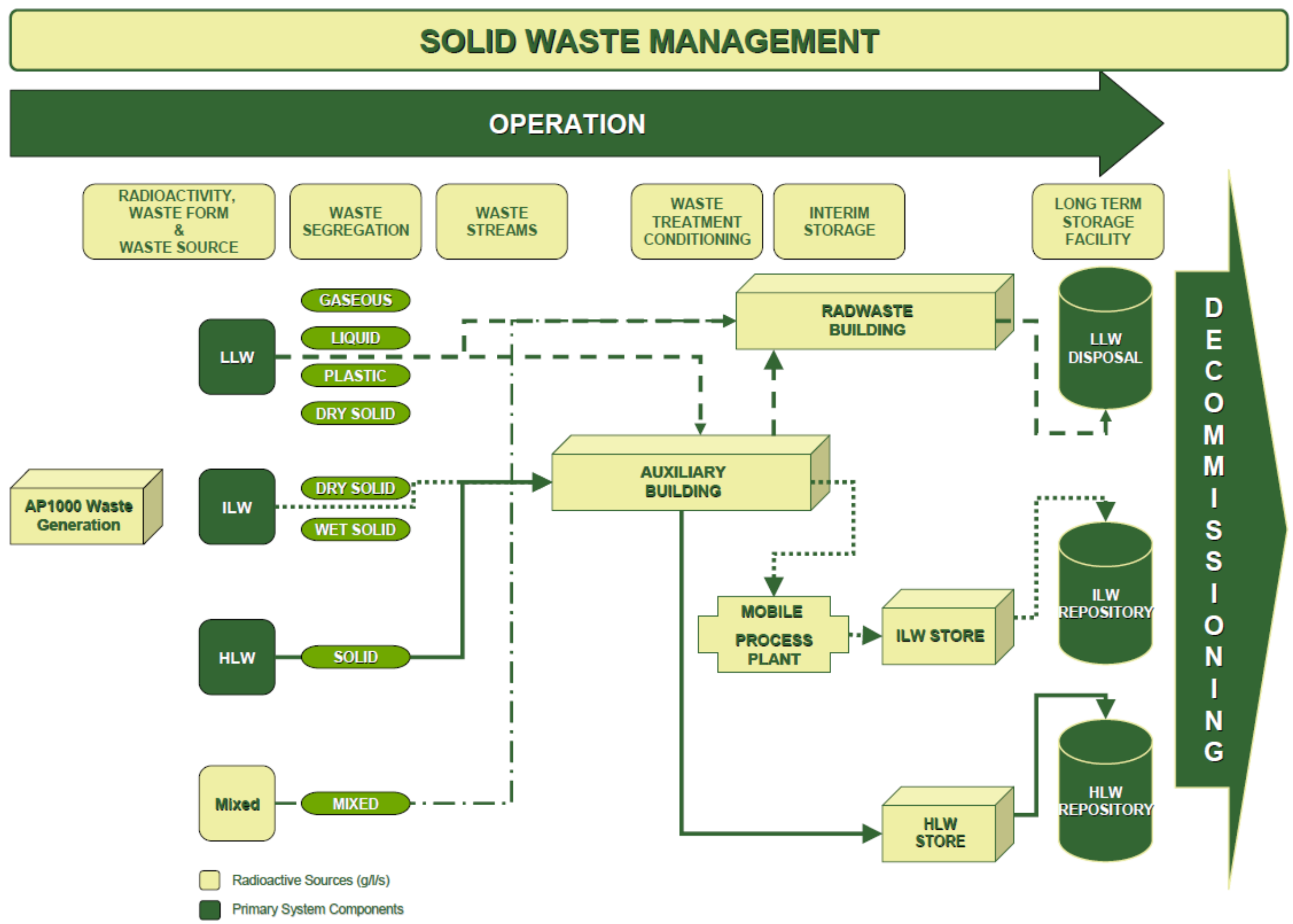


Figure 3.5-2. Solid AP1000 NPP Waste Management

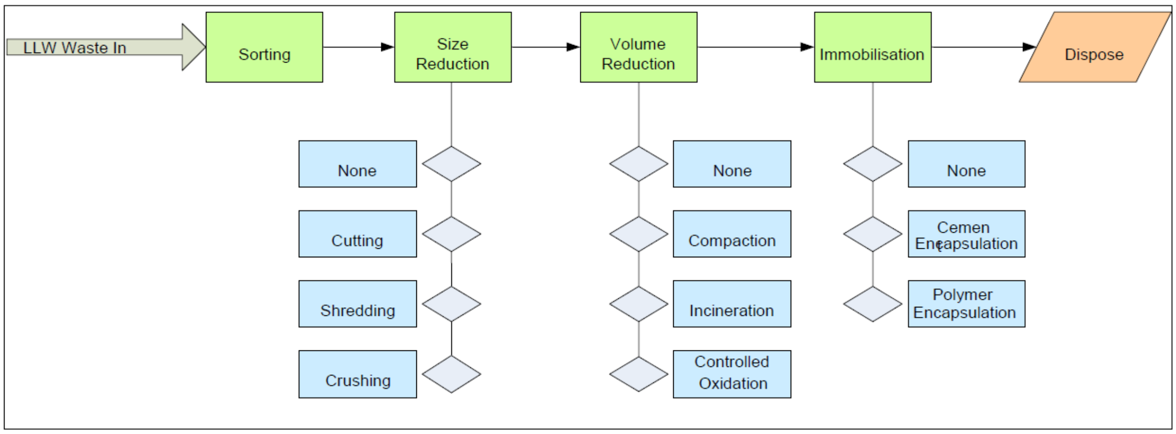


Figure 3.5-3. Low Level Waste Options (Reference 3-16)

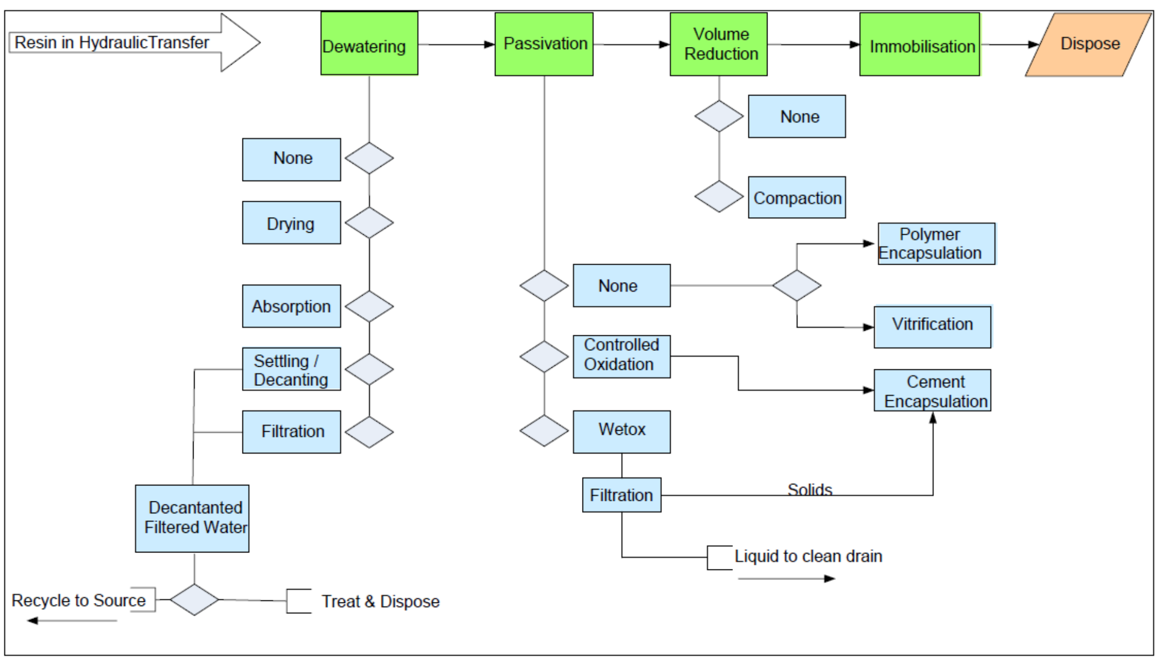


Figure 3.5-4. Intermediate Level Waste Organic Resin Treatment Options (Ref. 3-16)

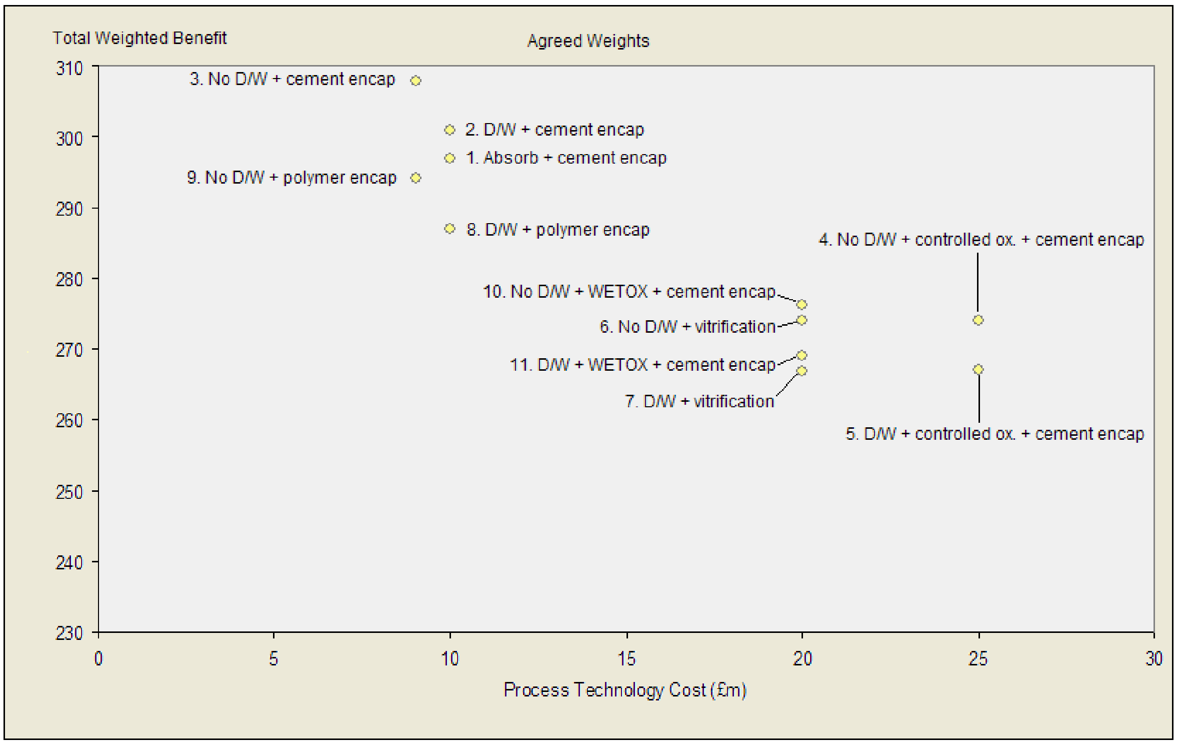


Figure 3.5-5. Total Weighted Benefit versus Cost of Process Technology (Ref. 3-16)

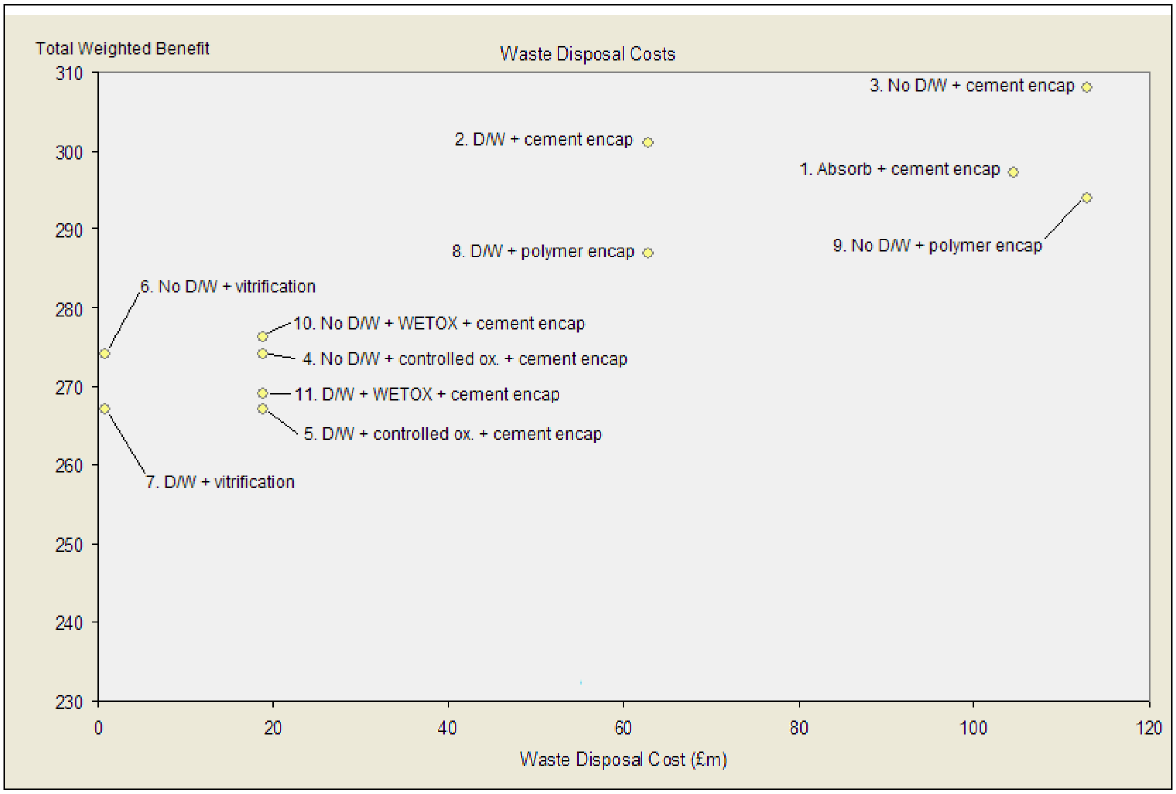


Figure 3.5-6. Total Weighted Benefit versus Cost of Waste Disposal (Ref. 3-16)

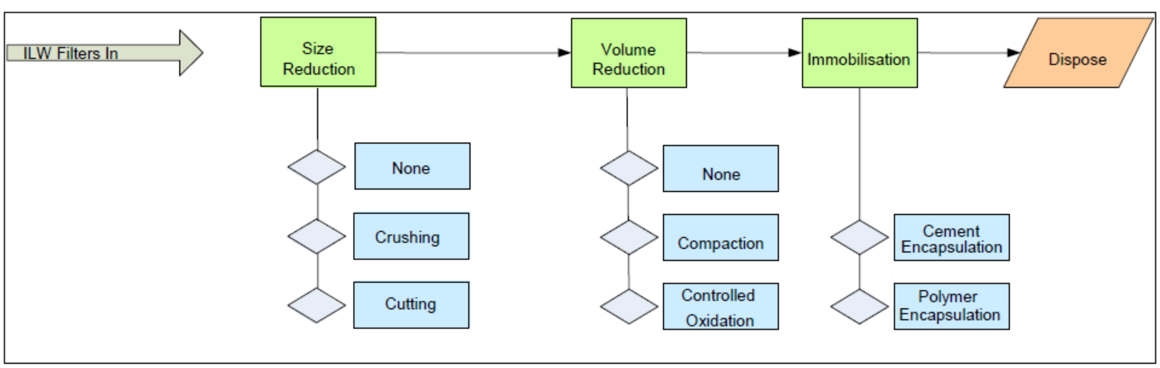


Figure 3.5-7. ILW Filter Treatment Options (Ref. 3-13)

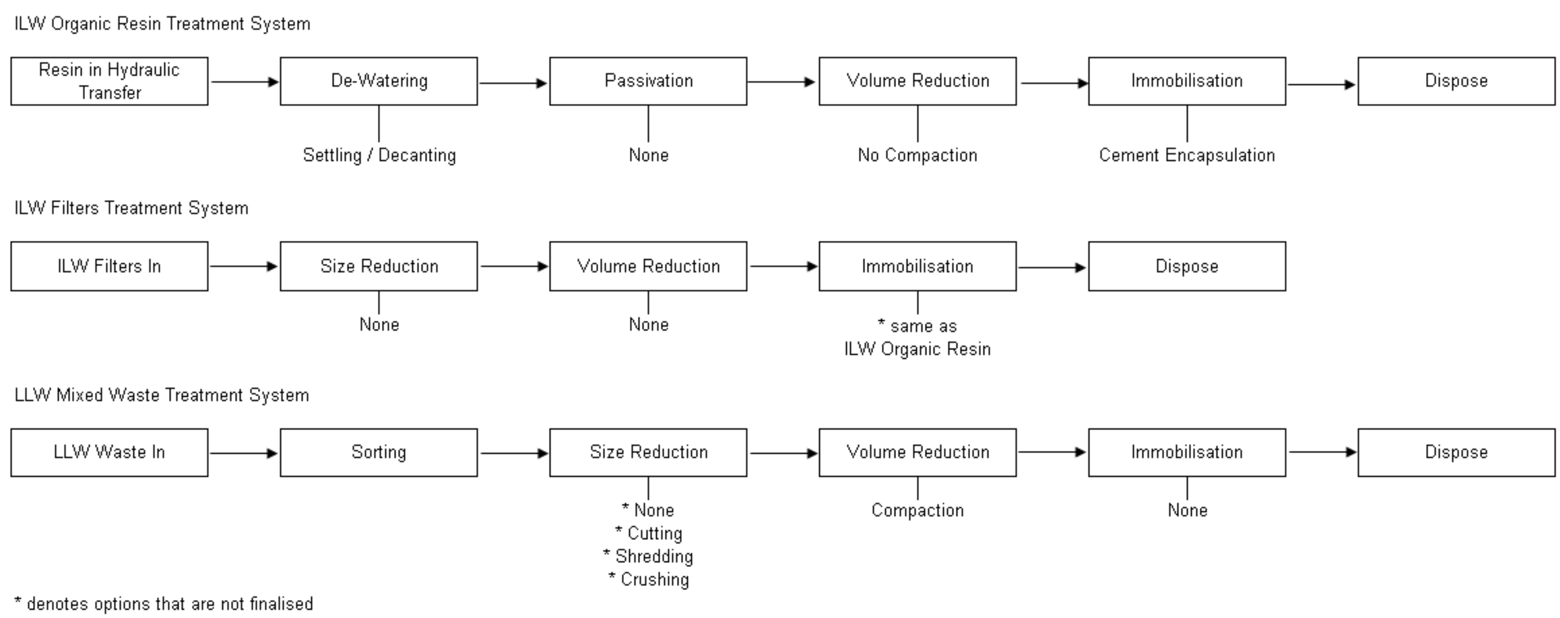


Figure 3.5-8. Summary of Selected BAT for ILW and LLW Radwaste

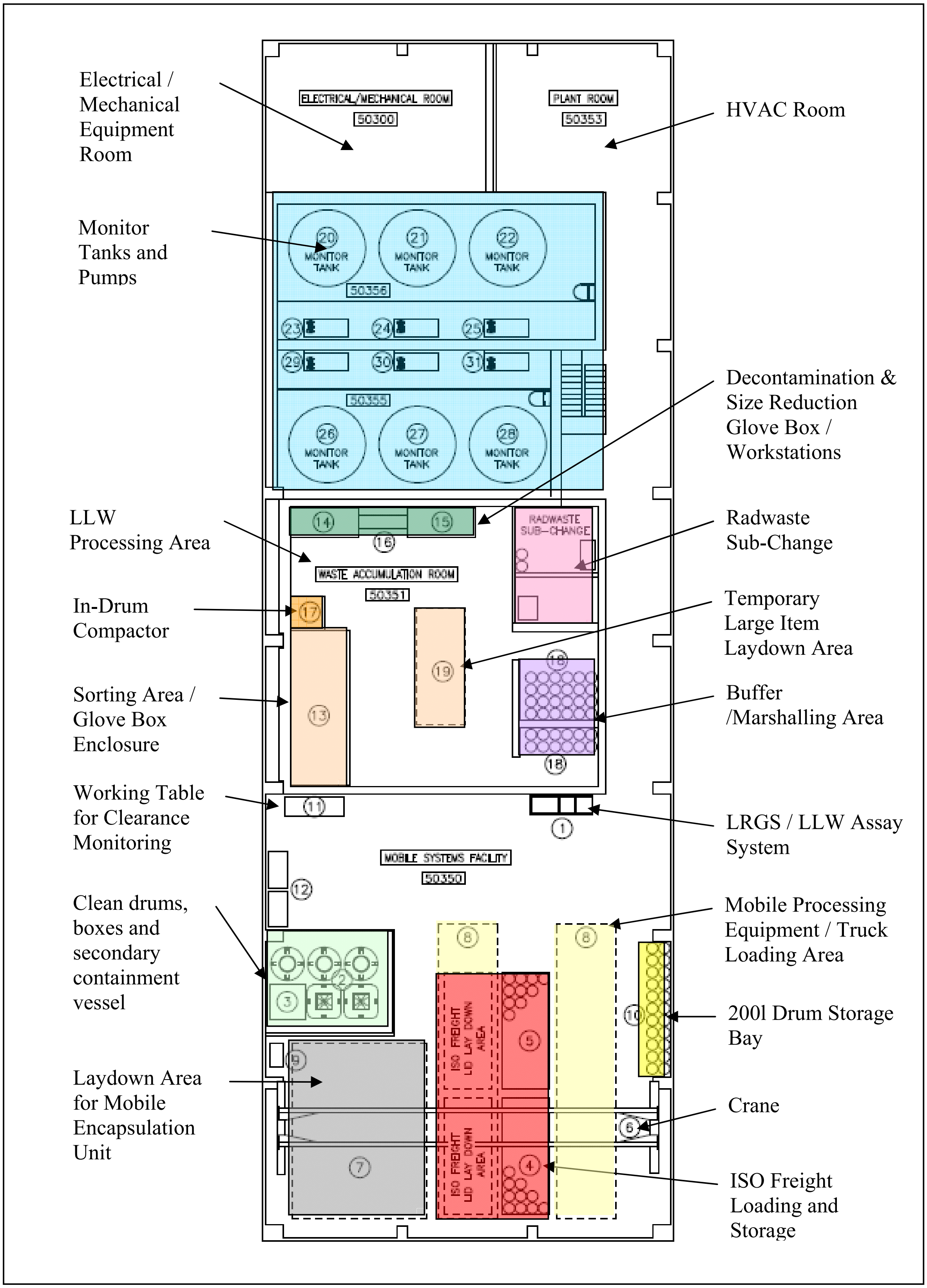


Figure 3.5-9. LLW Processing in Radwaste Building

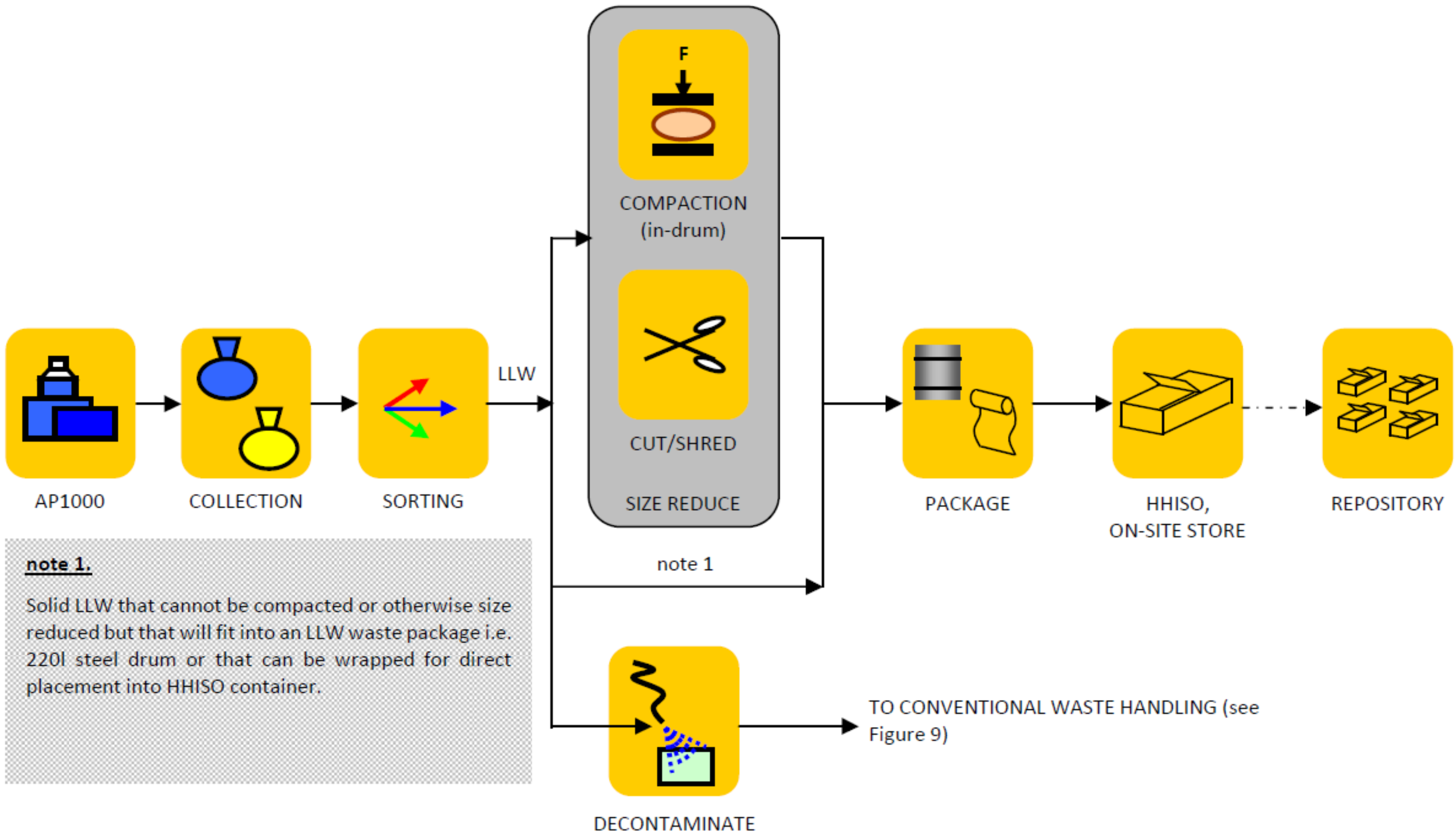


Figure 3.5-10. Solid LLW Disposal Routes

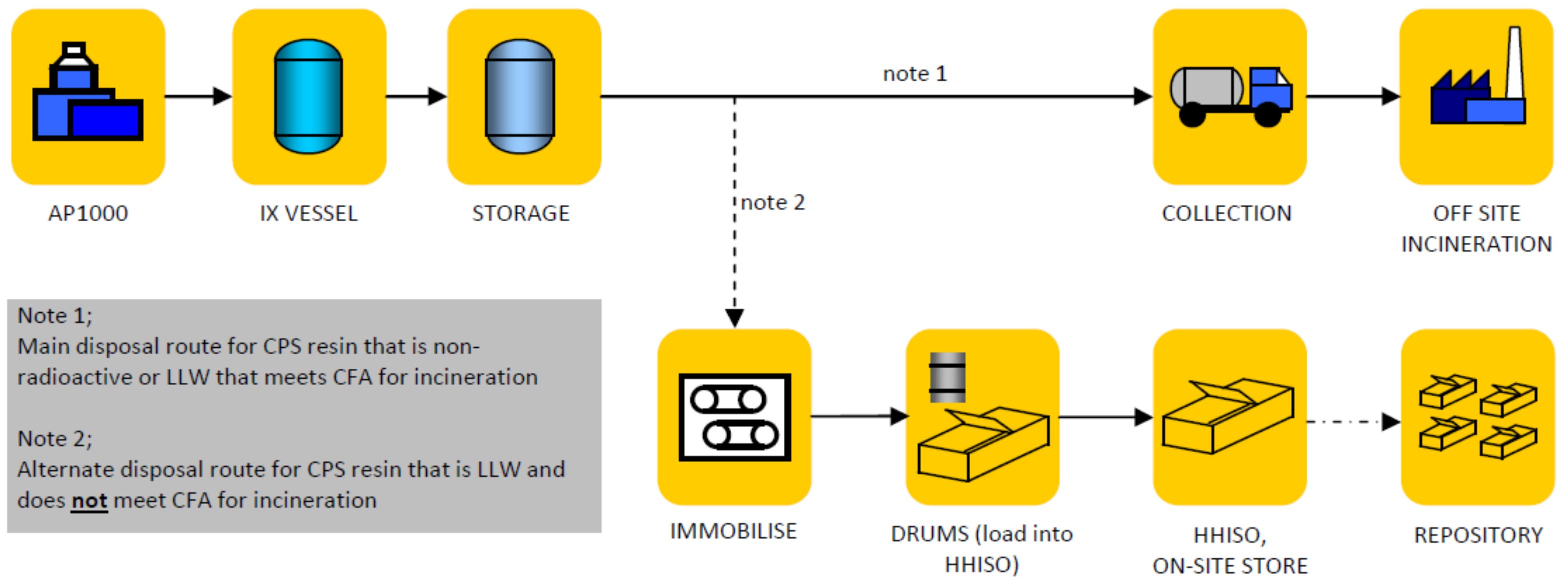


Figure 3.5-11. CPS Resin Disposal

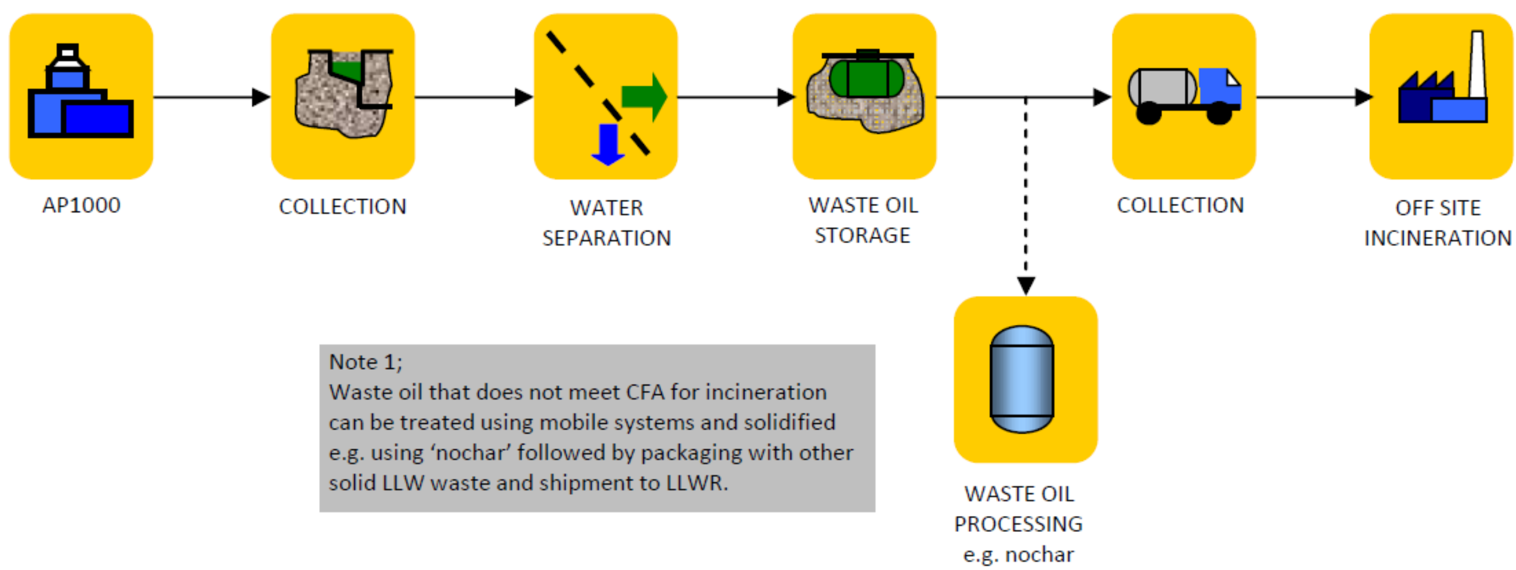


Figure 3.5-12. Solid LLW Waste Oil Disposal Route

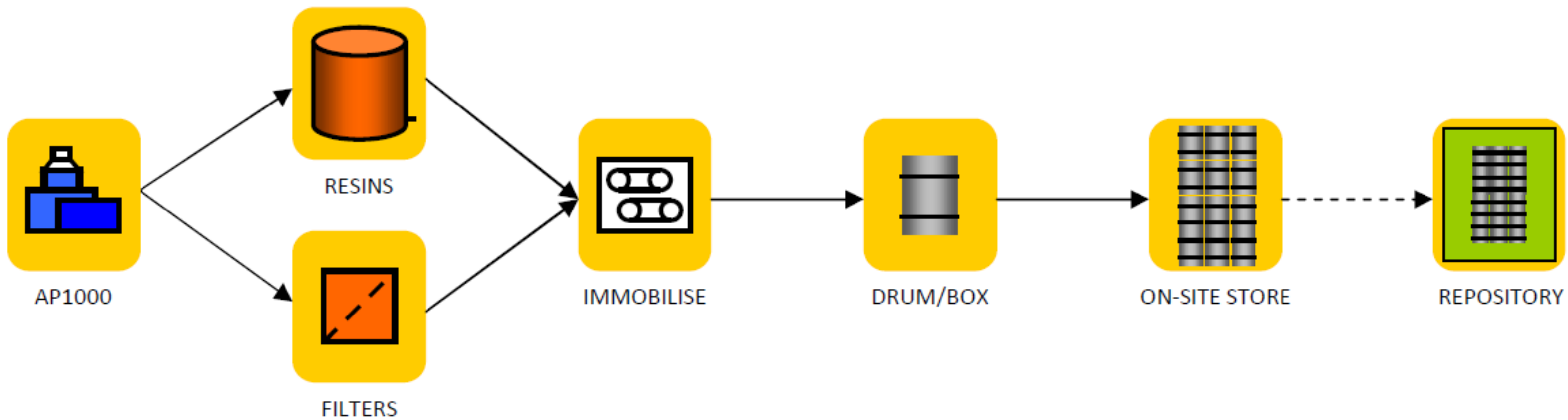


Figure 3.5-13. Solid ILW Treatment and Disposal

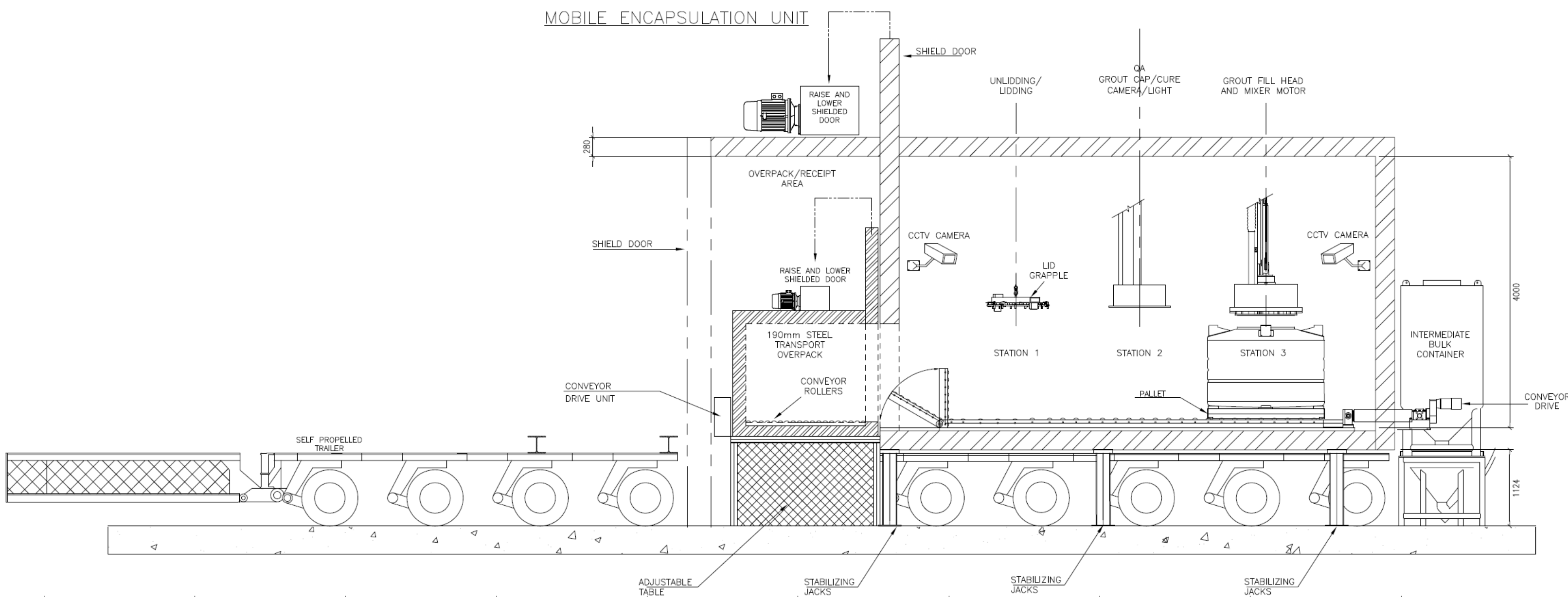


Figure 3.5-14. ILW Mobile Encapsulation Plant

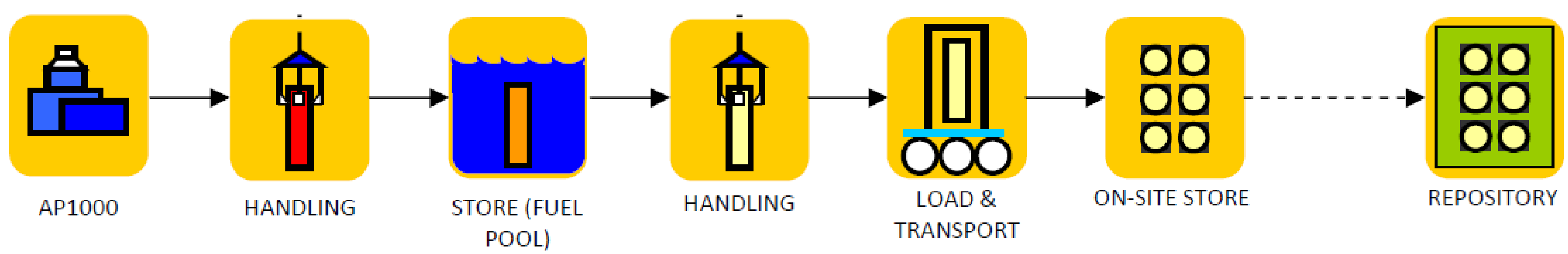


Figure 3.5-15. Solid HLW Treatment and Disposal

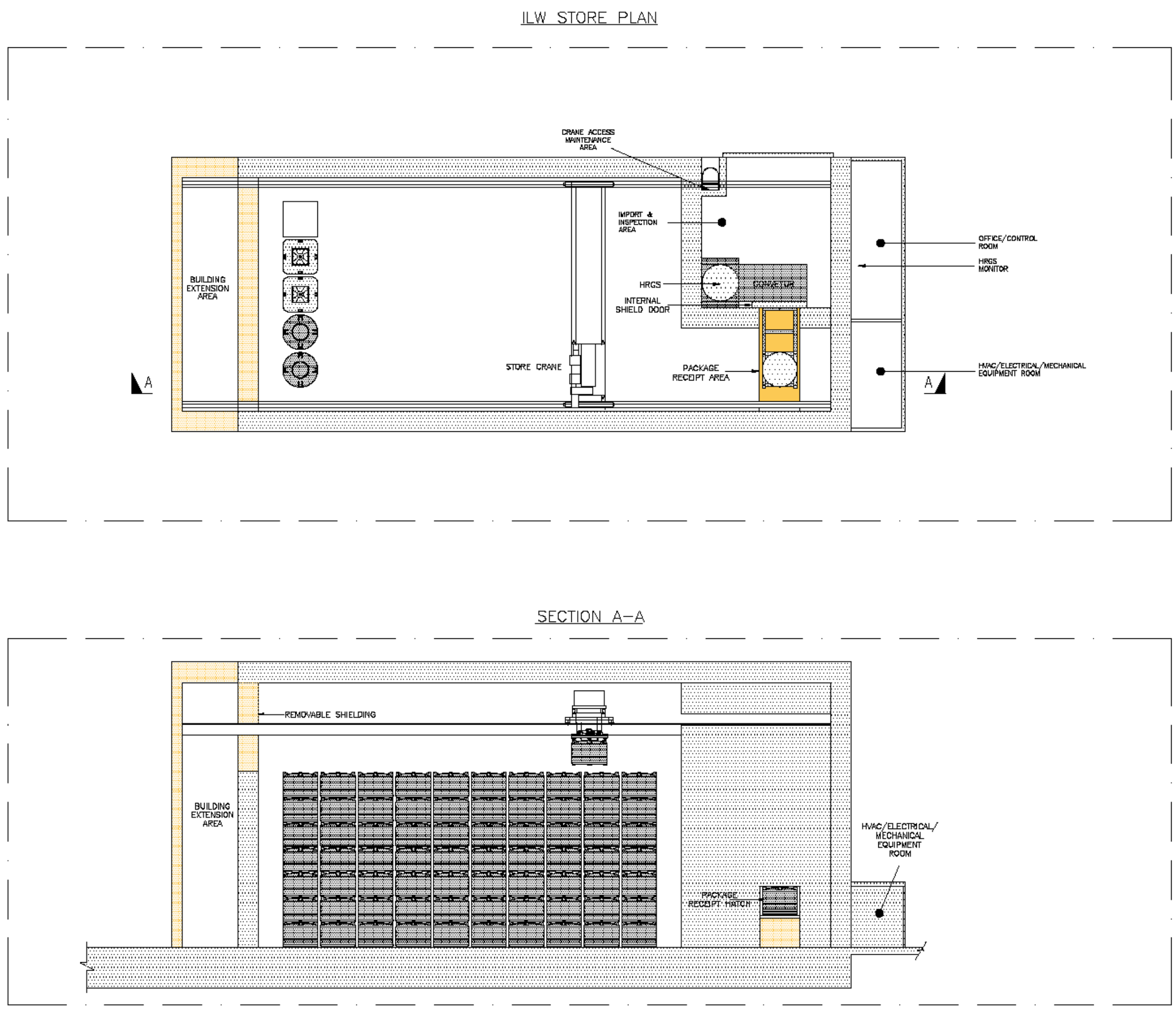
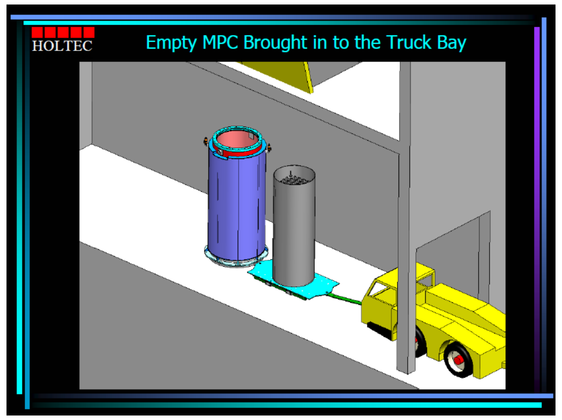


Figure 3.5-16. ILW Store



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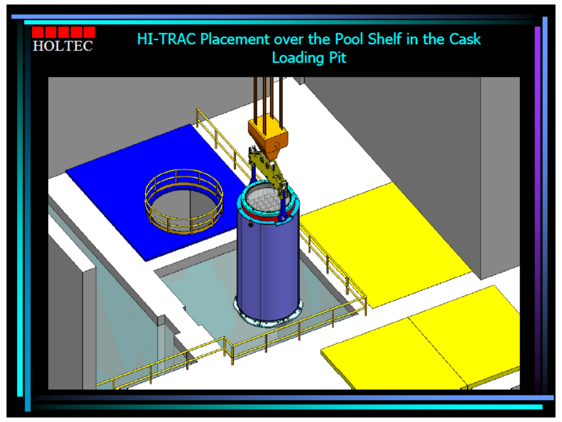

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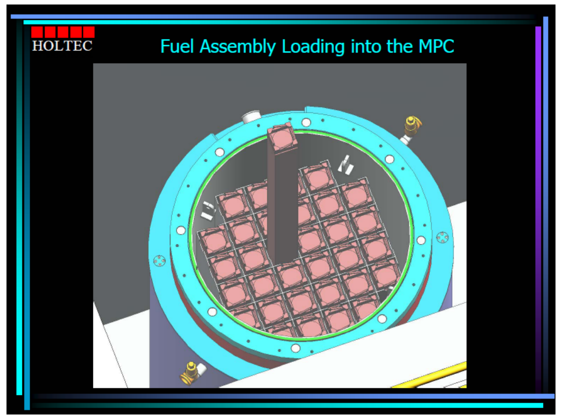

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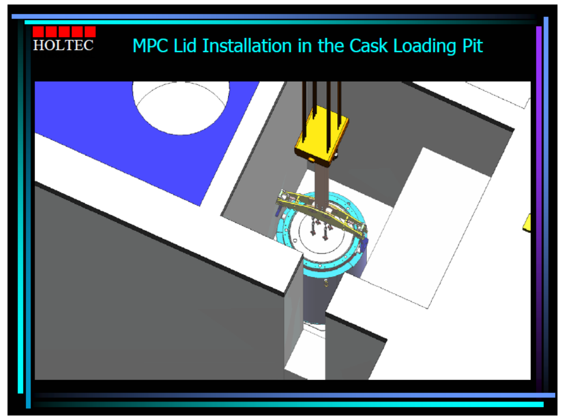

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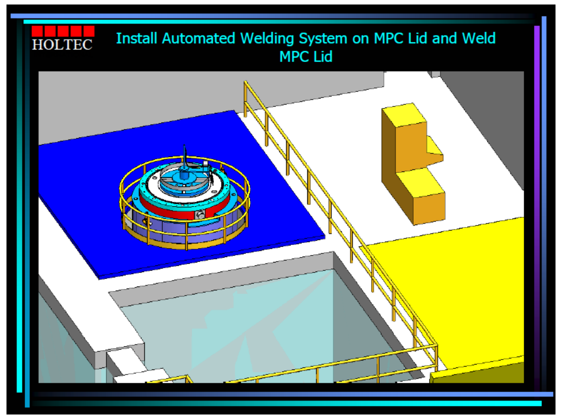
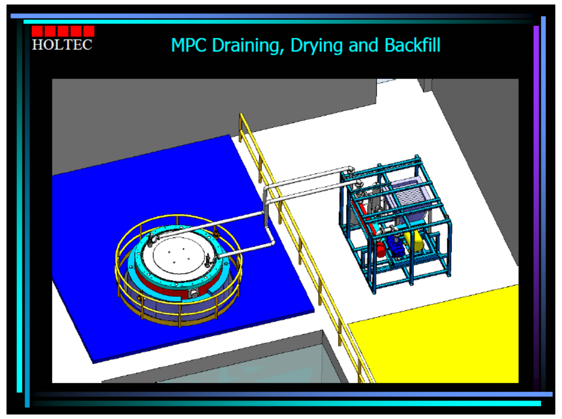


e)



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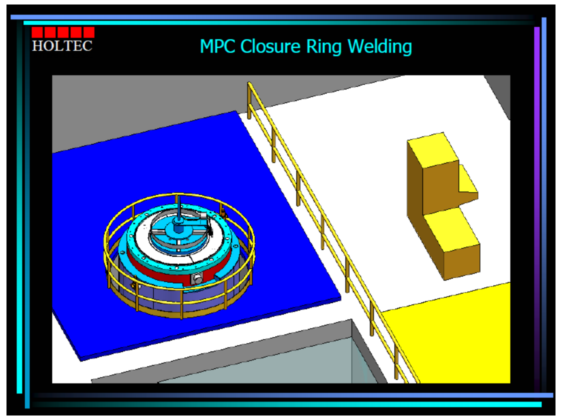

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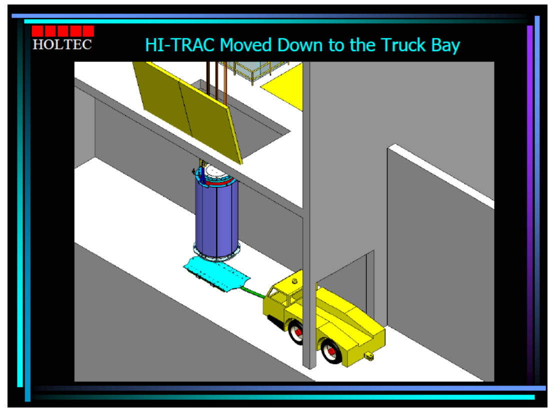

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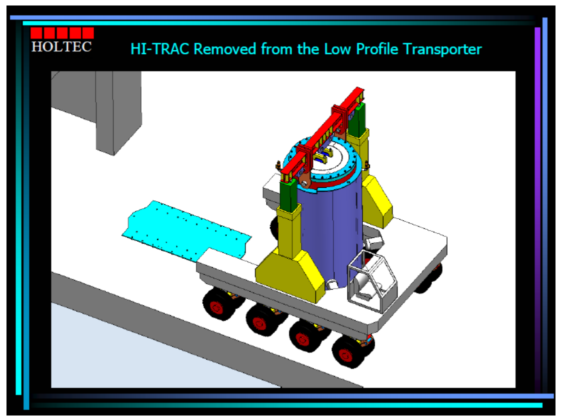

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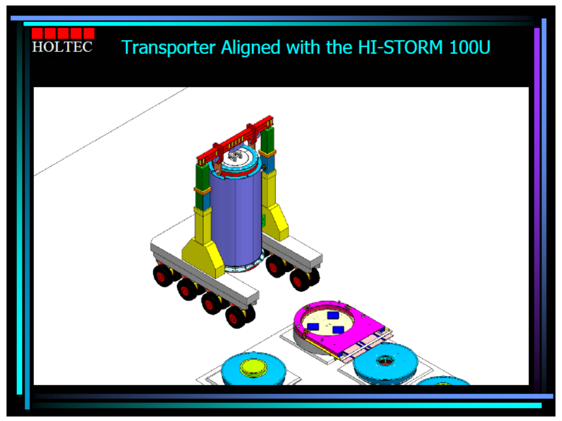

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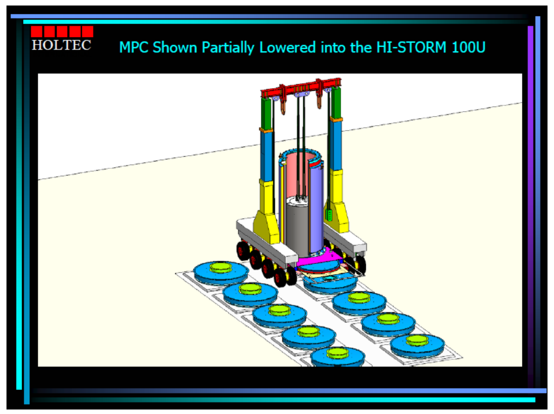

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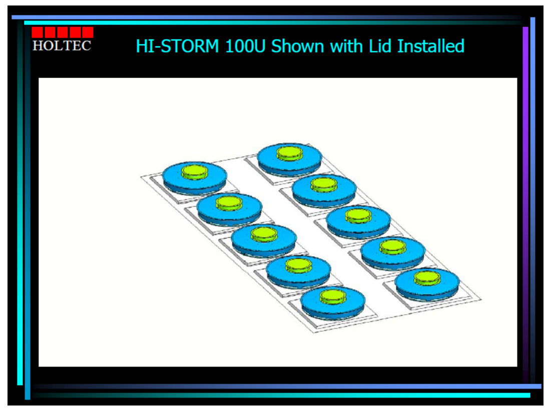
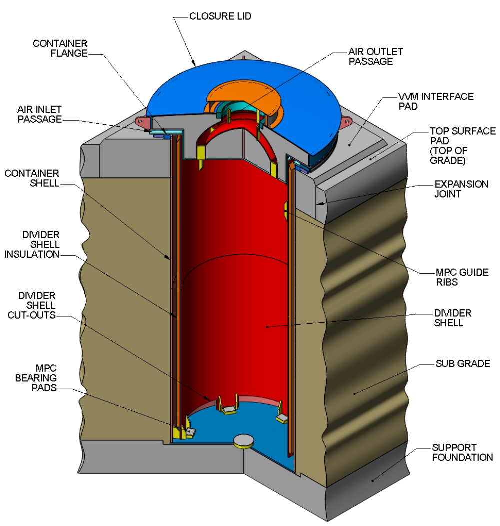
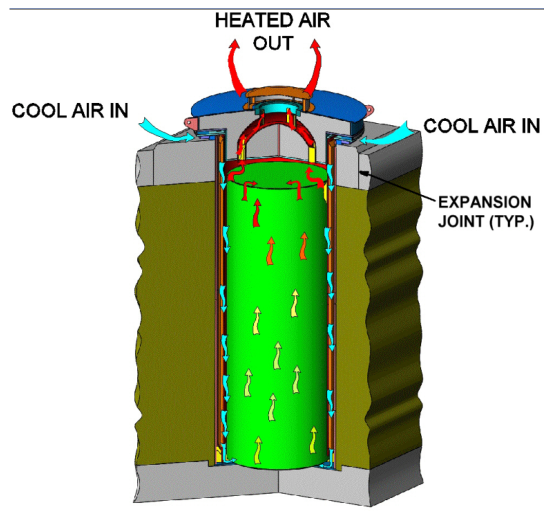


l)

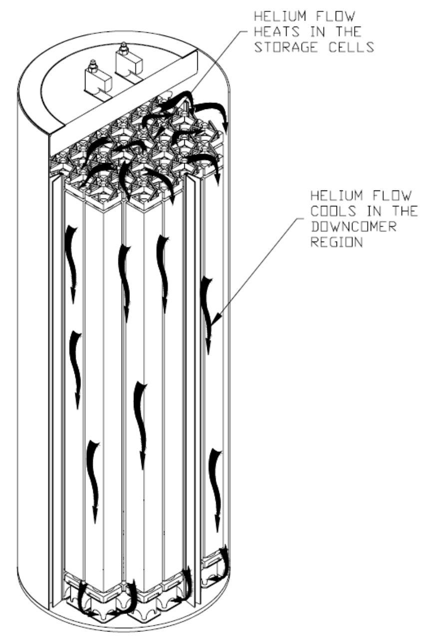
Figure 3.5-17.
Holtec Spent Fuel Storage System



a) Cutaway View of HI-STORM 100U VVM



b) HI-STORM 100U System Air Flow Pattern



c) Heat Rejection in a Holtec MPC through Thermosiphon Action

Figure 3.5-18. Holtec Spent Fuel Storage System

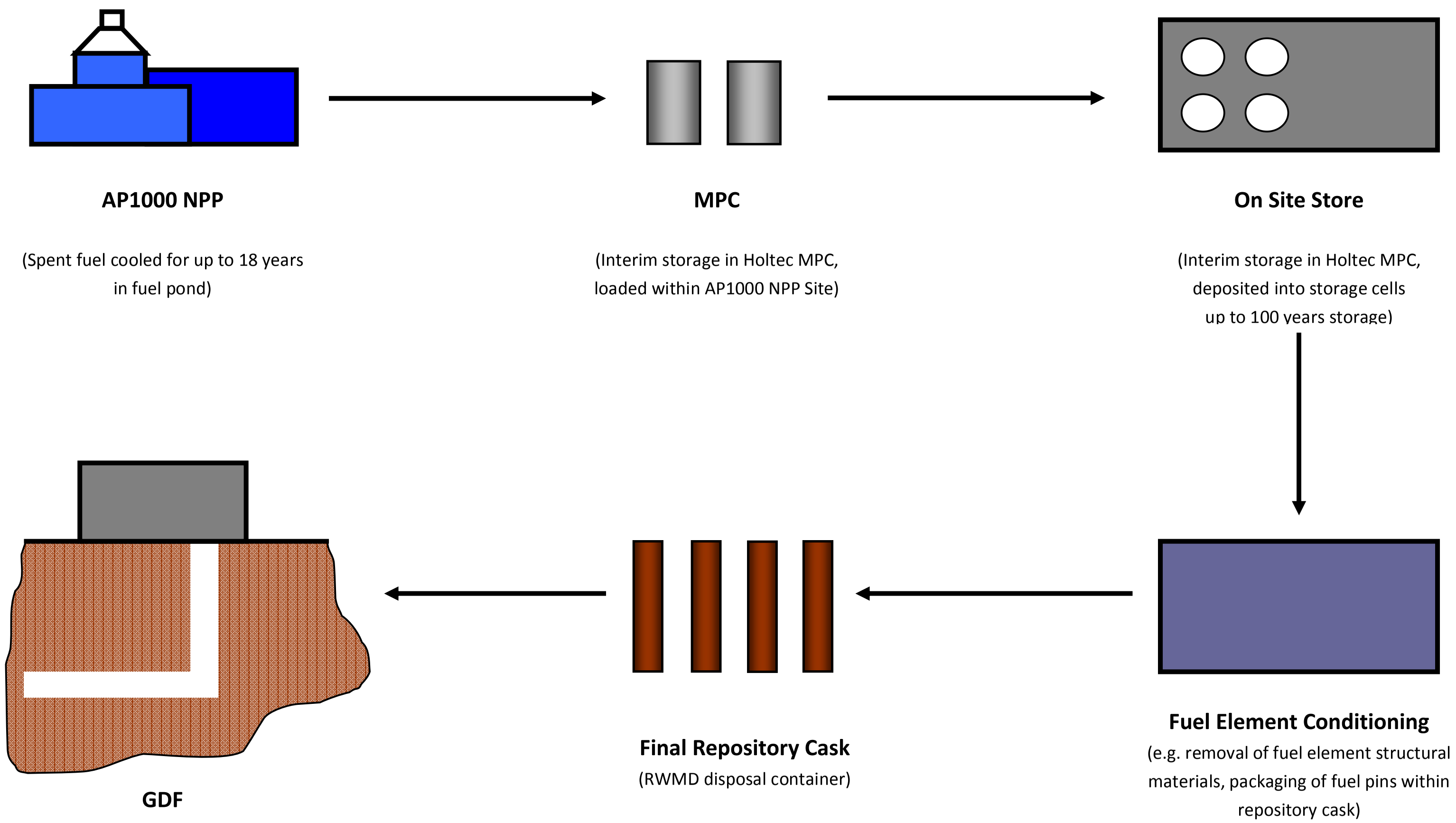


Figure 3.5-19. Spent Fuel Interim Storage, Transportation and Disposal Strategy

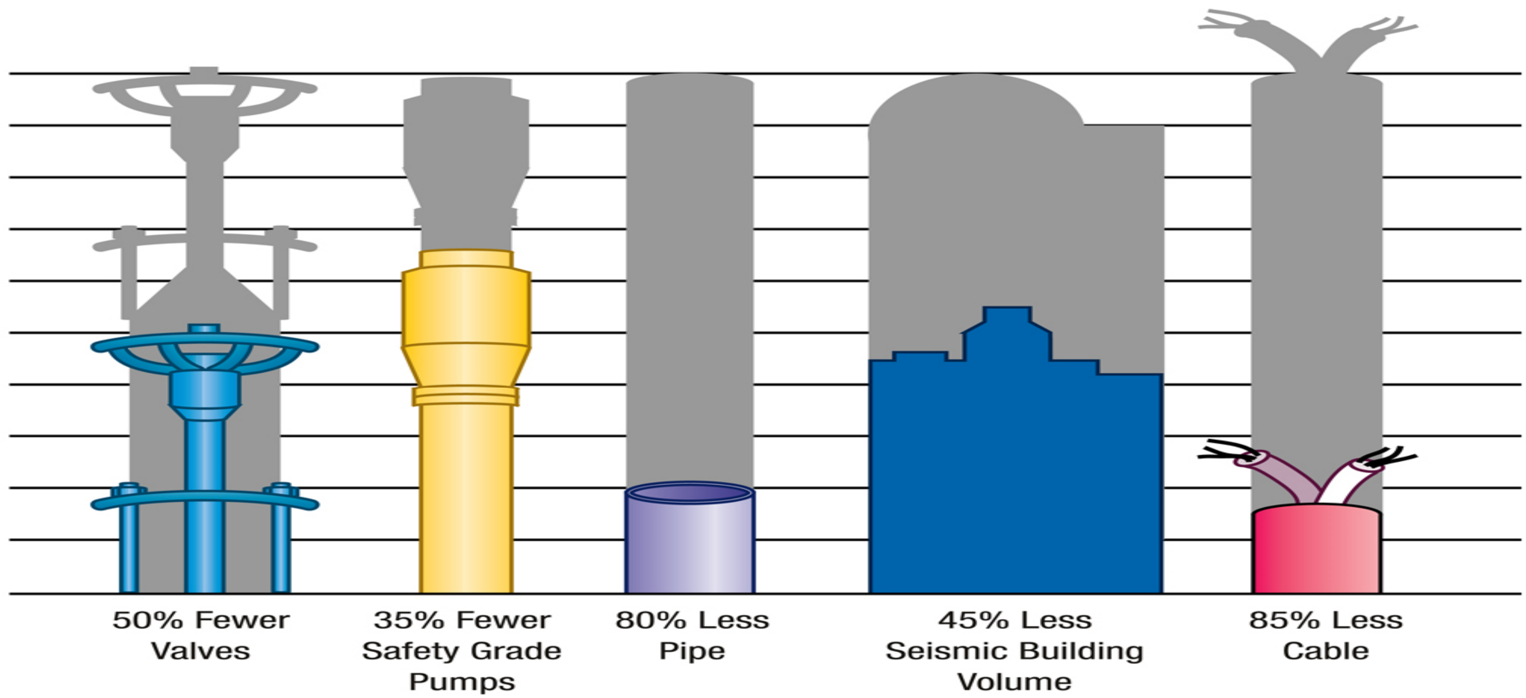


Figure 3.5-20. Minimisation of Equipment and Materials

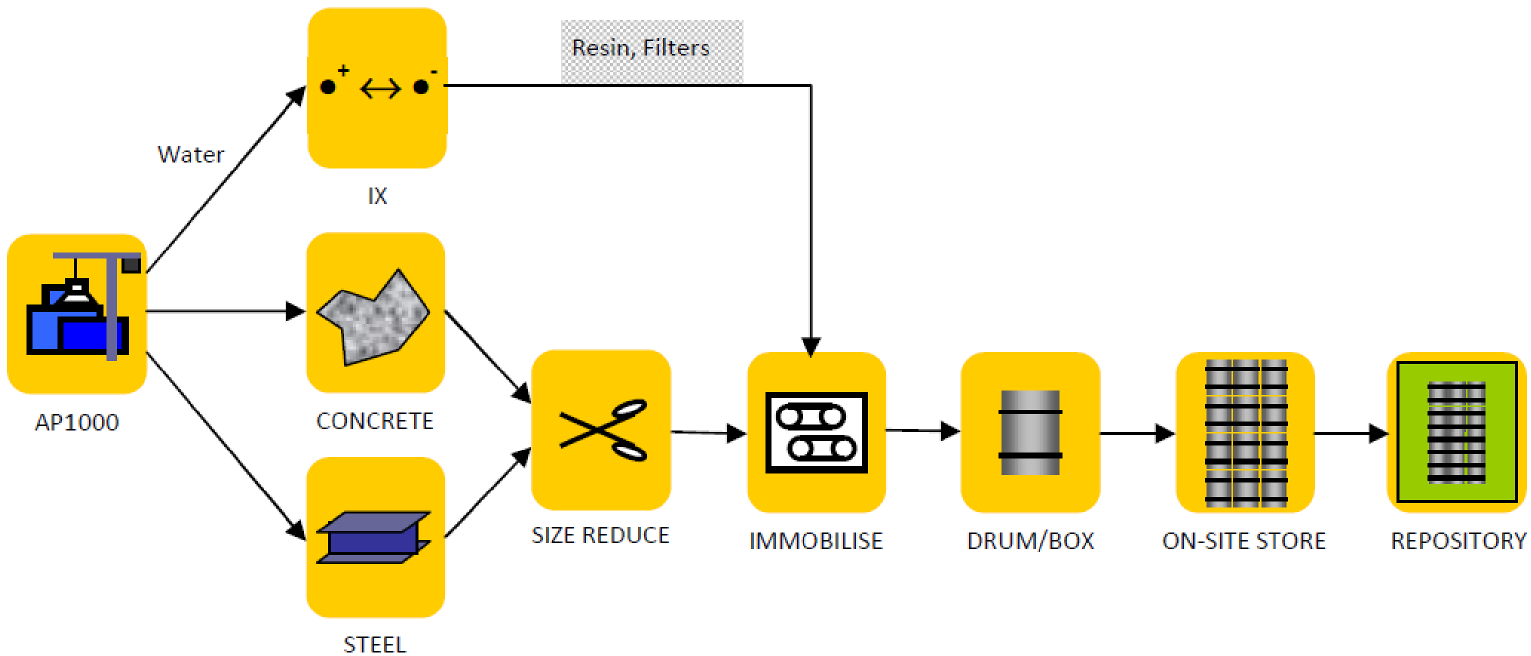


Figure 3.5-21. Decommissioning Waste Treatment and Disposal

4.0 NON-RADIOACTIVE WASTE MANAGEMENT SYSTEMS

4.1 Gaseous Non-Radioactive Waste

4.1.1 Emission Sources

4.1.1.1 Mobile Encapsulation Plant

The mobile encapsulation plant stabilises ILW by mixing with cementitious grout. The availability of locally generated premixed grout will be evaluated on a site-specific basis. Use of premixed grout is likely to be the simpler, therefore, preferred method of cement addition. However, in the absence of a suitable source of premixed grout, it will be necessary to mix the cementitious grout on-site from bagged, dry, powdered materials. If powdered grout materials are handled on-site, then local extraction systems and bag filters will be provided to reduce dust emissions. The grout mixing will take place within the rail car bay of the auxiliary building which has its own ventilation system (see Section 3.3.3).

4.1.1.2 Standby Generators

There are four diesel generators on the **AP1000** NPP:

- Two on-site standby diesel generators, output rated at 5200 kW (on a 50 Hz plant)
- Two ancillary diesel generators, output rated at 80 kW (on a 50 Hz plant)

The maximum thermal rated input of each standby generator is 12.8[3] MW. At this level, the diesel generators fall below the threshold of combustion devices that are subject to permitting under Schedule 1.1 of the Environmental Permitting (England & Wales) Regulations 2010 (References 4-1 and 4-2).

During operation, the diesel generators will emit combustion gases including sulfur dioxide, nitrogen dioxide, carbon monoxide, and particulates (see Table 4.1-1). However, these generators will only operate for a few hours per year during mains power failure or during testing.

The ancillary generators are rated two percent of the power of the standby generators. The contribution of these generators to the overall thermal rated input and air pollution is minimal.

4.1.1.3 HVAC Systems for Non-Radioactive Areas

The **AP1000** NPP design uses the following HVAC Systems to extract air from non-radioactively controlled areas and operate either in recirculation mode or exhaust to atmosphere without abatement. These areas are as follows:

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3 The maximum thermal rated input is "the rate at which fuel can be burned at the maximum continuous rating of the appliance multiplied by the gross calorific value of the fuel and expressed as megawatts thermal". It is therefore depending on which fuel is used. The fuel to be used in the UK will be selected during site specific licensing.

- Nuclear Island Nonradioactive Ventilation System (VBS) – The VBS serves the main control room, control support area, 1E electrical spaces, and the Passive Containment Cooling System (PCS) valve room. These areas do not have sources of activity present during normal operation or during fault conditions.

- Annex/Auxiliary Building Nonradioactive Ventilation System (VXS) – The VXS serves the office areas, switchgear rooms, locker rooms, battery rooms, computer rooms, toilets, and other similar spaces. These areas do not typically have sources of radioactive contamination present during normal operation. A drain line from the Steam Generator Blowdown System (BDS) to the Liquid Radwaste System (WLS) passes through the areas served by the VXS. The drain line can be contaminated by the BDS system if there are 1) fuel leaks, 2) steam generator leakage, and 3) a radiation monitor failure or an isolation valve failure. Since the drain system has no valves or connections within the area served by the VXS, and it is gravity drained, the chance of leakage into the VXS area is negligible as there would need to be a pre-existing leak coincident with a fault. It is not reasonably foreseeable for activity to be present in the areas served by the VXS.

- Diesel Generator Building Heating and Ventilation System (VZS) – The VZS supplies air to and exhausts from the diesel generator building through a roof vent to atmosphere. The diesel generator building is a physically separate building and there is no credible source or fault which would result in a radioactive release from the diesel generator building.

- Turbine Building HVAC System (VTS) – The VTS serves all areas of the turbine building. The HVAC systems serving the switchgear rooms, rectifier room, security rooms, and plant control system cabinet rooms do not have a credible source of radioactive contamination. The general area of the turbine building is ventilated using about 850 m^3s^{-1} (1,800,000 cfm) exhausted through roof ventilators without abatement. The Bay 1 area of the turbine building contains the reactor coolant pump variable-speed drives, CCS equipment (a non-radioactive system), and the BDS. The BDS may be contaminated in the very unlikely event of concurrent fuel defects, steam generator leak, radiation monitor or BDS isolation failure, and a BDS leak. The HVAC systems serving the switchgear rooms, rectifier room, security rooms, Bay 1 areas and plant control system cabinet rooms are recirculation systems. The sizing for these recirculation systems will not been determined until later in the design process.

4.2 Liquid Non-Radioactive Wastes

4.2.1 Non-Radioactive Waste Water Systems with Off-Site Release

There are no direct or indirect discharges to ground or groundwater. The non-radioactive waste water systems are described below. The discharge flow rates are presented in Table 4.2-1.

4.2.1.1 Waste Water System (WWS)

The WWS 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 is pumped to a waste water retention basin (WWRB) for settling 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.

4.2.1.2 Sanitary Drainage System (SDS)

The SDS is designed to collect the site sanitary waste (from plant restrooms and locker room facilities in the turbine building, auxiliary building, and annex building) for treatment, dilution and discharge. The SDS does not service facilities in radiologically-controlled areas. The SDS transports sanitary waste to either an on-site or off-site waste treatment plant. The selection of the waste treatment plant option is site-specific and is outside the scope of the generic site **AP1000** NPP application.

4.2.2 Systems Discharging to the Waste Water System

4.2.2.1 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 DWS. The treatment system comprises cartridge filters, two reverse osmosis units, clean in place unit, germicidal irradiation, non-regenerable mixed bed ion exchangers, sample panel unit, and electrodeionisation systems. The reject flow or brine from the first reverse osmosis unit is discharged to the WWS. A pH adjustment chemical is added from the CFS to maintain the system within the operating range of the reverse osmosis membranes to inhibit scaling and corrosion. A dilute anti-scalant, which is chemically compatible with the pH adjustment chemical feed, is metered into the reverse osmosis influent water to increase the solubility of salts (decrease scale formation on the membranes).

4.2.2.2 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 by removing impurities which are concentrated in the steam generator. The BDS consists of two blowdown trains, one for each steam generator. The BDS accepts water from each steam generator and processes the water as required. If significant radioactivity is detected in secondary side systems, blowdown is re-directed 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 electrodeionisation demineralising unit to remove impurities from the blowdown flow. The blowdown fluid is then normally recovered for reuse in the CDS. Blowdown with high levels of impurities can be discharged directly to the WWS. A small waste stream from the electrodeionisation system may also be directed to the WWS or the WLS.

4.2.2.3 Condensate System (CDS)

The CDS provides feedwater at the required temperature, pressure, and flow rate to the deaerator. Condensate is pumped from the main condenser hotwell by the condensate pumps and passes through the low-pressure feedwater heaters to the deaerator. During startup, the condensate is treated by ion exchange resin in the CPS to ensure the condensate and feedwater system (FWS) water chemistry meets specifications. Upon removal of the exhausted resin from the polisher vessel, the vessel is rinsed and the new resin is placed in the vessel using the resin addition hopper and eductor. Prior to plant startup, a new resin bed is rinsed and resin performance is verified, with flow through the vessel discharged to the WWS.

4.2.3 Seawater Cooling Systems

The two seawater cooling systems are described below. The discharge flow rates are presented in Table 4.2-1.

4.2.3.1 Circulating Water System (CWS)

The CWS supplies cooling water to remove heat from the main condensers, the TCS heat exchangers, and the condenser vacuum pump seal water heat exchangers.

The cooling water system is a site-specific design. However, for the generic coastal site it is assumed that a once through seawater cooling system will be used with warm reject seawater being discharged directly to the sea via the cooling water return. A once through seawater cooling system will be dosed with sodium hypochlorite to control biofouling when seawater temperatures exceed 10°C (50°F) (Reference 4-3).

Key mitigation measures for control of cooling water impacts are as follows:

- Design and location of the abstraction point to minimise impact on habitats and entrainment of fish;
- Modelling, design, and location of the discharge point to minimise impacts on sensitive species and habitats;
- Minimising the need for conditioning of the cooling water by best practice design and choice of materials;
- Best practice design and monitoring of the cooling water treatment system;
- Blending of chlorinated and un-chlorinated streams to reduce residual oxidant to a minimum.

4.2.3.2 Service Water System (SWS)

The SWS supplies cooling water to remove heat from the non-Class 1 CCS heat exchangers in the turbine building. Like the CWS, it is assumed that a once through seawater cooling system will be used for a generic coastal site (although the option for the use of cooling towers has been retained based on specific site requirements). The SWS uses ~4% of the seawater cooling flow of the CWS. This will be dosed with sodium hypochlorite to control biofouling when seawater temperatures exceed 10°C (50°F).

4.2.3.3 Thermal Discharges

The CWS will be designed to remove 2210 kilowatts (7,540 million Btu per hour) of heat with a seawater cooling flow of 136,275 m^3/h (600,000 U.S. gpm). Cooling water will be discharged from the cooling water system approximately 14°C warmer than the intake (Reference 4-3). This heat will be dissipated as rapidly as possible by suitable design and location of the discharge point at each site.

By comparison, the SWS is small being designed to remove 101 kilowatts (346 million Btu per hour) of heat with a seawater cooling flow of 4769 m^3/h (21,000 U.S. gpm). The SWS will discharge at a temperature differential of up to 18.3°C (25.5°F).

The CWS and SWS discharge will be blended in the seawater return sump. The combined CWS and SWS discharge temperature differential will be 14.15°C (25.5°F).

Once discharged, the cooling water will start to mix with the ambient water body. Based on a temperature in the discharge water at 14°C (25°F) above ambient, the following dilution factors would be required to achieve lower rises in temperature (Reference 4-3):

- 7 x dilution for a 2°C temperature difference
- 10 x dilution for a 1.5°C temperature difference
- 14 x dilution for a 1°C temperature difference

A mixing zone will be proposed around the point of discharge such that the differential temperature beyond the mixing zone does not exceed 1 or 2°C (2 to 4°F). The mixing zone requires a site-specific definition and impact evaluation.

4.2.4 Closed Loop Cooling Systems

Closed loop cooling systems do not normally result in discharges to the waste water system. Discharges only arise in the event of maintenance or as a result of blowdown to maintain water chemistry or leakage. The closed loop cooling systems are described below:

4.2.4.1 Component Cooling Water System

The CCS is a non-Class 1, closed loop cooling system that transfers heat from various plant components to the SWS during normal phases of operation. Cooling medium is provided by the SWS. 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 a high level in the surge tank.

4.2.4.2 Central Chilled Water System

The VWS supplies chilled water to the HVAC systems and is functional during reactor full-power and shutdown operation. It also supplies chilled water to the WLS, WGS, secondary sampling system, and the temporary air supply units of the containment leak rate test system. The chemical feed tanks and associated piping are used to add chemicals to each chilled water subsystem stream to maintain proper water quality.

4.2.4.3 Turbine Building Closed Cooling Water System (TCS)

The TCS is a closed loop system which provides chemically treated, demineralised cooling water for the removal of heat from non-Class 1 heat exchangers in the turbine building and rejects the heat to the CWS. The cooling water is treated with a corrosion inhibitor and uses demineralised water for makeup.

4.2.5 Chemicals Discharged with Liquid Effluents

The chemicals used in the **AP1000** NPP has been identified in Section 2.9.2 and Tables 2.9-1 and 2.9-2. Some of these chemicals are released with liquid effluent discharges into the seawater cooling return.

The normal flow rate of once through seawater cooling is approximately two thousand times the normal flow rate of the non-radioactive effluent discharges (see Table 4.2-1). It follows that the seawater cooling return provides a substantial dilution of the normal plant effluent discharges.

The concentrations of chemicals present at the seawater outfall are estimated in Table 4.2-2.

4.2.5.1 Sodium Hypochlorite and Halogenated By-Products

The use of biocides is essential to prevent biofouling cooling water systems. Sodium hypochlorite is used as a biocide in the **AP1000** NPP cooling water systems. The level of sodium hypochlorite dosing will be minimised by using the BAT design of the cooling water system (see Table 4.2-3) to minimise the potential for biofouling. Good design of the dosing and monitoring systems will reduce the level of hypochlorite dosing required further. These two factors will minimise the discharge of both residual oxidant and chlorination by-products to the receiving waters.

The total residual chlorine discharge is expected to be ~0.2mg/l (see Table 4.2-2) between May to November when biofouling is prevalent because seawater temperatures exceed 10°C (50°F). This is based on a hypochlorite dose rate of 0.2 mg/l (1 ppm) which is the BAT concentration reported for once through seawater cooling systems (Reference 4-4). Levels as low as 50 µg/l (50 ppb) may be achievable based on best practice dosing regimes and mixing of chlorinated and unchlorinated streams prior to discharge (Reference 4-3). The required dosing rate is subject to site-specific differences in water temperature, particulate loading, and organic matter.

Table 4.2-2 indicates that the predicted discharge of total residual chlorine will exceed the environmental quality standard of 10 µg/l (10 ppb) at the point of discharge to the sea. However, the concentration is expected to rapidly decrease on contact with coastal water due to chlorine demand, exposure to sunlight, and dilution. Further dilution occurs in the mixing zone around the outlet of the cooling water discharge in the coastal water. There is minimal risk that the environmental quality standard for residual chlorine would be exceeded at the edge of the mixing zone.

In principle, it is feasible to remove residual chlorine to below the environmental quality standard limits prior to discharge by addition of dechlorination chemicals (e.g., sulfite). This may be considered for a specific site with very sensitive marine environments. However, for the generic site, it is not considered BAT to incur the chemical cost and transportation cost associated with dosing another chemical to reduce the residual chlorine to below the environmental quality standard limits prior to discharge.

The chlorination of seawater can give rise to the formation of halogenated by-products. A mixture of chlorinated and brominated compounds is formed due to the reaction of the chlorine with bromide. Brominated species normally predominate and, whilst generally more toxic, they tend to breakdown more rapidly in the environment. Trihalomethanes and halogenated acetic acids are the most common by-products formed. The quantities of by-products formed will be site-specific. Factors affecting the by-product formation include the applied chlorine dose, the concentration of organic carbon in the water, temperature, pH, and contact time. The concentrations of halogenated byproducts can be minimised by the use of good design of the dosing and monitoring systems to minimise the sodium hypochlorite dose rates and residual chlorine discharges. A summary of common by-products of chlorination in seawater is listed in Table 4.2-4 (Reference 4-3).

4.2.5.2 Boric Acid

The **AP1000** NPP discharges borated reactor coolant water in the radioactive liquid waste discharges. The boron is not removed by the ion exchange beds in the WLS because these operate in a boron saturated mode. The concentration of boric acid in the WLS discharge is a maximum of 2700 mg/l (2700 ppm) and this level declines over the 18-month fuel cycle.

Table 4.2-2 indicates that the predicted contribution to boron concentrations from the **AP1000** NPP discharge is ≤ 1.1 µg/l (1.1 ppb) compared to an annual average environmental quality standard for seawater of 7000 µg/l (7000 ppb). This discharge compares to typical seawater boron concentrations of ~4500 µg/l (4500 bbp) (Reference 4-5). It is concluded that the boron discharge can be considered negligible as it is less than 1% of the environmental quality standard.

4.2.5.3 Trace Metals

Zinc acetate is dosed into the RCS to reduce corrosion. The zinc dose rate is typically 10 ppb (+/- 5 ppb). When reactor coolant water is letdown via the CVS, a small amount of zinc acetate will be released. The zinc will be removed by passage through the WLS ion exchange resins, so no release of zinc is expected in the **AP1000** NPP liquid discharges from this source.

Some trace metal impurities may be present in the bulk chemical dosed into the various **AP1000** NPP water systems. For a worst case calculation, it is assumed that the trace metal impurity of all chemicals may be 1 ppm. The trace metal discharges associated with this level of bulk chemical impurities would result in a metal discharge of 0.0026 µg/l (0.0026 ppb) (see Table 4.2-2). Table 4.2-2 also shows the environmental quality standards for various trace metals in the salt water environment. The worst case metal concentration predicted in the **AP1000** NPP effluent discharge is less than 1% of any environmental quality standard for trace metals. As such, it is concluded that trace metal discharge associated with chemical impurity is negligible.

The non-radioactive metal discharges associated with corrosion products have not been predicted for the **AP1000** NPP. However, the presence of iron, nickel, copper, and chromium might all be expected in trace quantities.

4.2.5.4 Other Chemicals

Other chemical discharges identified in Table 4.2-2 include ammonium hydroxide, ammonium chloride, monoethanolamine, and lithium7 hydroxide. The discharge concentrations predicated are presented in Table 4.2-2. There are no relevant environmental quality standards to compare these discharge concentrations against.

4.2.6 Treatment and Disposal of Non-Radioactive Effluent

Details of the WWS can be found in Chapter 26 of the PCSR (Reference 1-10). The block flow diagram in Figure 2.7-1 shows the WWS comprising sumps, oil-water separator, and WWRB.

The non-radioactive waste water during normal plant operation and during plant outages is handled by the WWS. Wastes from the turbine building floor and equipment drains (which include laboratory and sampling sink drains, oil storage room drains, the transformers area, the main steam isolation valve compartment, auxiliary building penetration area, and the auxiliary building HVAC room) are collected in the two turbine building sumps. Drainage from the auxiliary building sump north (a non-radioactive sump), and the annex building sump is also collected in the turbine building sumps. The turbine building sumps provide a temporary storage capacity and a controlled source of fluid flow to the oil separator.

A radiation monitor located on the common discharge piping of the sump pumps provides an alarm upon detection of radioactivity in the waste water. In the event radioactivity is present in the turbine building sumps, the waste water is diverted from the sumps to the WLS for processing and disposal. The radiation monitor also trips the sump pumps on detection of radioactivity to isolate the contaminated waste water. Provisions are included for sampling the sumps.

The turbine building sump pumps route the waste water from either of the two sumps to the oil separator for removal of oily waste. The diesel fuel oil area sump pump and the transformers sump pump also discharge waste water to the oil separator. The oil separator removes oily waste from the waste water stream which flows by gravity to the waste oil storage tank. It contains an oil holdup tank, sampling provisions are included on the oil holdup tank to confirm that the oil does not require handling and disposal as a hazardous waste. A sampling connection is also provided at the discharge of the oil separator.

The oil separator has the capacity to process approximately 12.6 l/sec (200 gpm) of oil contaminated water. The oil separator is designed to remove free-oil droplets > 20 microns (0.8 mils) from the process stream. At an inlet concentration of 0.1 percent, consisting of non-permanent, mechanically emulsified oil, grease, or petroleum hydrocarbons, the oil separator will discharge water with a maximum hydrocarbon concentration of 10 mg/l (10 ppm). The waste water from the oil separator is pumped to the WWRB.

The waste oil storage tank provides temporary storage prior to removal by truck for offsite disposal. A bypass line allows for the oil separator to be out of service for maintenance. The bypass line will be normally closed. During maintenance of the oil separator, waste water can be retained in the turbine building sumps or, if necessary, in the WWRB.

The WWRBs allow for retention of waste water, settling of suspended solids and treatment, if required, prior to discharge. The detailed design and configuration of the plant WWRB and associated discharge piping, including piping design pressure, basin transfer pump size, basin size, and location of the retention basins will be made according to site-specific conditions.

Waste water that complies with discharge limits will be released intermittently via the seawater cooling return sump for final discharge via the plant outfall to the sea. The maximum design flow rate from the WWRB is ~408 m^3/h (108,000 gallons/hour) (see Figure 2.7-1). The once through cooling water flow rate is 136,275 m^3/h (36,000,000 gallons/hour). It follows that the cooling water stream provides a substantial dilution of the discharged effluent before release to the environment.

4.2.6.1 Containment of Unplanned Emissions

All discharges to the WWS are released via the WWRB. The **AP1000** NPP design will have sufficient containment within the WWRB to retain unplanned emissions of effluents and spillages. The quality of these discharges can be ascertained by sampling and analysis from the WWRB to determine whether direct discharge is acceptable. If the water quality is unacceptable, then a mobile treatment system can be brought in to deal with the effluent or vacuum tankers can be used to remove the off-specification effluent to a licensed treatment plant.

4.2.7 Storm Water

Storm water falling on hard standing outdoor areas will drain to a storm water pond or sustainable urban drainage system (SUDS) to minimise the risk of contamination or flooding of receiving waters. An oil water separator will be incorporated to prevent oily spillages on roads and loading bays from being carried over to discharge. The details of the storm water management will be developed in the site-specific design.

4.2.8 Fire Water

Fire water from internal fire fighting systems will be retained within the lower levels of the buildings. Fire water used externally will fall on hard standing areas and be collected in the storm water pond or SUDS system. If the water quality is unacceptable, then a mobile treatment system can be brought in to deal with the effluent or vacuum tankers can be used to remove the off-specification effluent to a licensed treatment plant.

4.3 Solid Non-Radioactive Waste

4.3.1 Sources

The sources of non-radwaste are identified in Appendix A and summarised in Table 4.3-1.

4.3.2 Waste Minimisation

The **AP1000** NPP incorporates features of the waste management hierarchy (see Figure 3.1-1), realising the intent to avoid the generation of waste, minimise the generation of waste, reuse or recycle waste wherever possible.

The conventional waste management strategy will ensure, to the extent practicable, that techniques will be used to prevent or minimise the production of wastes. Examples include:

- Re-using receptacles, hoses, and other plant consumables, where it is safe to do so
- Use of appropriate signage to remind staff of the need to reduce waste
- Reuse of HEPA filter boxes
- Provision of waste collection facilities in the RCA outside of contamination-controlled areas from which waste can be monitored and, if it is not contaminated, removed from the RCA for disposal as non-radwaste
- Learning from operator experience shared between the sites and externally from other organisations in the UK and overseas

4.3.3 Treatment and Disposal

Use will be made of recycling and recovery techniques, where appropriate. For example, items that can be recycled include lube oil, batteries, metals, glass, and paper.

Items that need to be disposed will be transferred to licensed waste disposal facilities by a registered waste disposal contractor. Nearby facilities, where and when available, will be used to the extent practical to minimise the environmental impact of transport.

A schematic showing the proposed treatment and disposal of non-radwaste is shown in Figure 4.3-1.

4.4 References

4-1 UKP-GW-GL-036, Rev. 3, "Applicability of the Environmental Permitting (England and Wales) Regulations 2010 to **AP1000**," Westinghouse Electric Company LLC, December 2016.

4-2 "The Environmental Permitting (England and Wales) Regulations 2010," The Office of Public Sector Information, SI 2010 No. 675.

4-3 UKP-GW-GL-034, Rev. 1, "Generic Assessment of the Impacts of Cooling Options for the Candidate Nuclear Power Plant **AP1000**," Westinghouse Electric Company LLC, February 2010.

4-4 "Reference Document on the Application of Best Available Techniques to Industrial Cooling Systems," European Commission, December 2001.

4-5 Summerhayes, C. P., and S. A. Thorpe, "Oceanography: An Illustrated Guide," Chapter 11, 165-181, 1996.

Table 4.1-1

ANNUAL EMISSIONS FROM DIESEL GENERATORS

Pollutant Discharged	Two 5200 kW Standby Diesel Generators[1]		Two 80 kW Ancillary Diesel Generators[1]	
	lb	**kg**	**lb**	**kg**
Particulates	<800	<363	<10	<4.54
Sulfur Oxides	<2,500	<1134	<5	<2.23
Carbon Monoxide	<1,000	<454	<30	<13.61
Hydrocarbons	<600	<272	<11	<4.99
Nitrogen Oxides	<12,000	<5443	<140	<63.5

Notes:
1. Emissions are based on 4 hours per month operation for each of the generators

Table 4.2-1

AP1000 NPP ESTIMATED OPERATIONAL LIQUID CONVENTIONAL WASTE FROM SYSTEM OPERATIONS

			Estimated Quantities		
			Normal Volume	Max Volume	Annual Average
System	Waste Description	Physical/Chemical Description	m^3/h	m^3/h	m^3/y
CWS	Circulating water system cooling water	Non-contact seawater cooling (once through)	136,275	136,275	1.194E+09
SWS	Service water system cooling water		2385	4770	2.09E+07
BDS	Steam generator blowdown	Secondary side coolant	4	42	37,000
BDS	Condensate demineraliser startup bypass flow	Off specification demineraliser water	26	82	329,000
DTS	Reverse osmosis and electrodeionisation reject		13	41	164,400
Multiple	Turbine island waste water	Demineraliser water with minor solids	18	74	257,000
CWS	Strainer backwash		2	413	18,900
SWS	Strainer backwash		1	681	8,360
BDS	Fire testing drains		0.1	170	756
CPS	Condensate polisher rinse		0.01	466	103
WWS	Condensate demin rinses and backwashes		0.01	466	103
CDS	Condenser water box drain		0	250	2.8

Table 4.2-2

AP1000 NPP ESTIMATED DISCHARGE OF CHEMICALS WITHIN THE LIQUID EFFLUENT STREAMS

Chemical (Chemical Use)	System	Liquid Effluent Stream	System Concentration	Discharge Rate Typical	Chemical Quantity Discharged	Annual Average Discharge Concentration (at controlled waters)	Environmental Quality Standard (saltwater)
			mg/l	m^3/y	kg/y	µg/l	µg/l
Sodium hypochlorite (biocide)	CWS	Circulating water system cooling water	≤0.2[1,2]	1.194E+09	119400[2]	≤200[2]	TRO 10[3] MAC
	SWS	Service water system cooling water	≤0.2[1,2]	2.09E+07	≤2090[2]		
Ammonium chloride (algaecide)	SWS	Service water system cooling water	0.3		6270	≤11[4]	–
Ammonium hydroxide (pH control)	CDS	Condenser water box drain	≤100	2.8	≤0.28		
	Multiple	Turbine island waste water	≤100	257,000	≤25700		
	BDS	Steam generator blowdown	≤100	37,000	≤3700		
Hydrazine (oxygen scavenger)	BDS	Steam generator blowdown	10		370	0.3	–
	CDS	Condenser water box drain	10	2.8	0.03		

Table 4.2-2 (cont.)

AP1000 NPP ESTIMATED DISCHARGE OF CHEMICALS WITHIN THE LIQUID EFFLUENT STREAMS

Chemical (Chemical Use)	System	Liquid Effluent Stream	System Concentration	Discharge Rate Typical	Chemical Quantity Discharged	Annual Average Discharge Concentration (at controlled waters)	Environmental Quality Standard (saltwater)
			mg/l	m^3/y	kg/y	µg/l	µg/l
Monoethanolamine	BDS	Steam generator blowdown	3	37000	111	0.09	–
	CDS	Condenser water box drain	3	2.8	0.008		
Boric Acid (chemical shim control)	WLS	Borated reactor coolant	≤2700	2920(6)	≤7884	≤1.1(5)	B 7000 AT
Lithium7hydroxide (pH control and tritium reduction)	WLS	Borated reactor coolant	≤2.2		≤6.4	≤0.005	–
Zinc Acetate (corrosion inhibitor)	WLS	Borated reactor coolant	<0.04		<1.2(7)	<3.4E-05(7,8)	Zn 40 AD

Table 4.2-2 (cont.)

AP1000 NPP ESTIMATED DISCHARGE OF CHEMICALS WITHIN THE LIQUID EFFLUENT STREAMS

Chemical (Chemical Use)	System	Liquid Effluent Stream	System Concentration	Discharge Rate Typical	Chemical Quantity Discharged	Annual Average Discharge Concentration (at controlled waters)	Environmental Quality Standard (saltwater)
			mg/l	m^3/y	kg/y	µg/l	µg/l
Trace metals impurities from chemicals	All	All	Assume 1ppm contamination in all chemicals	1.25E+09	3.3^(9)	0.0027^(9)	Hg 0.3 AD Cd 2.5 AD As 25 AD Pb 25 AD Cr 15 AD Zn 40 AD Cu 5 AD Ni 30 AD Fe 1000 AD

Notes:
1. Concentrations reported as total residual chlorine
2. Hypochlorite is dosed into seawater only when seawater temperature >10C, assumed to be 6 months of the year
3. Total residual oxidant
4. Ammonium concentrations reported as nitrogen
5. Concentrations reported as total dissolved boron
6. Flow rate converted from life of plant borated water discharge data in Table 3.4-1
7. Zinc removed by WLS ion exchange resins
8. Concentration reported as total zinc
9. Calculation based on 1ppm metal contamination present in all chemicals

    AD Annual average dissolved
    AT Annual average total
    MAC Maximum allowable concentration

Table 4.2-3

BAT APPROACH FOR THE COOLING WATER SYSTEM

Criterion	BAT Approach	Remarks
Coastal Area	Once through systems for large capacity units >10MWth	Avoid mixing local thermal plume near intake point, e.g., by deep water extraction below mixing zone using temperature stratification.
Location of abstraction.	The cooling water abstraction point is situated in a location where it will minimise any potential impacts on habitats.	It is designed to avoid disturbing sediments and hence avoids adverse impacts on the cooling water system and the mobilisation of pollutants.
Location of discharge point.	Design and location of discharge point is normally based on extensive modelling taking into account the location of sensitive habitats and fish migration pathways. Design is made to ensure that the maximum dispersion of the thermal load whilst minimising the impact on ecology.	Two approaches to maximise the dissipation of heat; Aim of rapid initial mixing and dilution, typically using diffusers to create a large volume of slightly warmed water. Allow plume (typically buoyant due to lower density of warm water) to rise virtually unmixed to the surface and spread horizontally where it will lose heat to the atmosphere and slowly mix downwards.
Reduction of fouling and corrosion which reduces the requirement for chemical additives.	Intake design to minimise the entrainment of fish, debris, organic and inorganic material including suspended solids. Stagnant zones and turbulence should be avoided and flow velocities maintained at a high enough level to avoid fixation of organic organisms. Velocity of flow should be more than the critical velocity. Use smooth surfaces and non-toxic coatings and paints to reduce fixation of the organisms, to reinforce the velocity effect and to facilitate cleaning. On-line or off-line cleaning.	This approach will reduce the concentrations of both residual oxidant and by-products being discharged to the receiving waters.

Table 4.2-3 (cont.)

BAT APPROACH FOR THE COOLING WATER SYSTEM

Criterion	BAT Approach	Remarks
Biofouling treatment system.	Targeted dosing at locations with a high fouling risk such as the heat exchanger inlet and outlet boxes. Optimisation of chemical monitoring and controlled (automatic) dosing to ensure the minimum required dose. Since the applied hypochlorite concentration will decrease through the cooling water system, chemical monitors ensure effective concentrations in the system. Pulse-alternating chlorination, which is an optimum anti-fouling treatment with the minimum usage of chlorine. Monitoring of biofouling.	Current industry practice is to use biocide (sodium hypochlorite) which has the advantage over other biocides. The aim is to prevent biofouling from occurring, as once it does large doses for long periods are required.
Reduction of chemical application.	Monitoring and control of cooling water chemistry.	
Use of less hazardous chemicals.	Avoid using prohibited substances (List I substances) such as chromium compounds, mercury compounds, organometallic compounds, mercaptobenzothiazole, and shock treatment substances other than chlorine, bromine, ozone, and H_2O_2. Limitation of certain substances (List II substances) such as chlorine dioxide, chlorine and bromine, adsorbable organically bound halogens (AOX), chemical oxide demand (COD), zinc, and phosphorus compounds.	Sodium hypochlorite (biocide) is recognised as a List II substance.

Table 4.2-3 (cont.)

BAT APPROACH FOR THE COOLING WATER SYSTEM

Criterion	BAT Approach	Remarks
Re-circulating of dangerous substances.	Constant monitoring programme to detect dangerous substances in system.	
Risk of bioaccumulation causing scale and corrosion.	Reduce algae formation by reducing light energy reaching the cooling water. Reduce biological growth by avoiding stagnant zones (design) and optimised chemical treatment. Mechanical and chemical cleaning after outbreak. Control of pathogens via periodic monitoring.	
Re-use of waste heat.	Consider alternative uses for waste heat to decrease the thermal impact on the receiving water as well as optimising overall energy savings. This approach is considered as good environmental and energy management.	This can only be assessed during site-specific analysis. This approach may not be beneficial if the waste heat is of low-grade heat.

Table 4.2-4

HALOGENATED BY-PRODUCTS OF CHLORINATION IN SEAWATER

Chemical	CAS No
Haloamines	
monobromamine	14519-10-9
dibromamine	14519-03-0
tribromamine	–
Haloacetonitriles	
Bromochloroacetonitrile	83463-62-1
Bromoacetonitrile	590-17-0
Dibromoacetonitrile	3252-43-5
Haloacids	
Bromoacetic acid	79-08-3
Dibromoacetic acid	631-64-1
Tribromoacetic acid	75-96-7
Bromochloroacetic acid	5589-96-8
Bromodichloroacetic acid	71133-14-7
Chlorodibromoacetic acid	5278-95-5
Halogenated phenols	
2, 4-Dibromophenol	615-58-7
3, 5-Dibromophenol	626-41-5
2, 4, 6-Tribromophenol	118-79-6
Haloketones	
Bromopropanone	867-54-9
3-bromo-2-butanone	814-75-5
Trihalomethanes	
Bromodichloromethane	75-27-4
Bromoform	75-25-2
Chlorodibromomethane	124-48-1

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	360	700
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

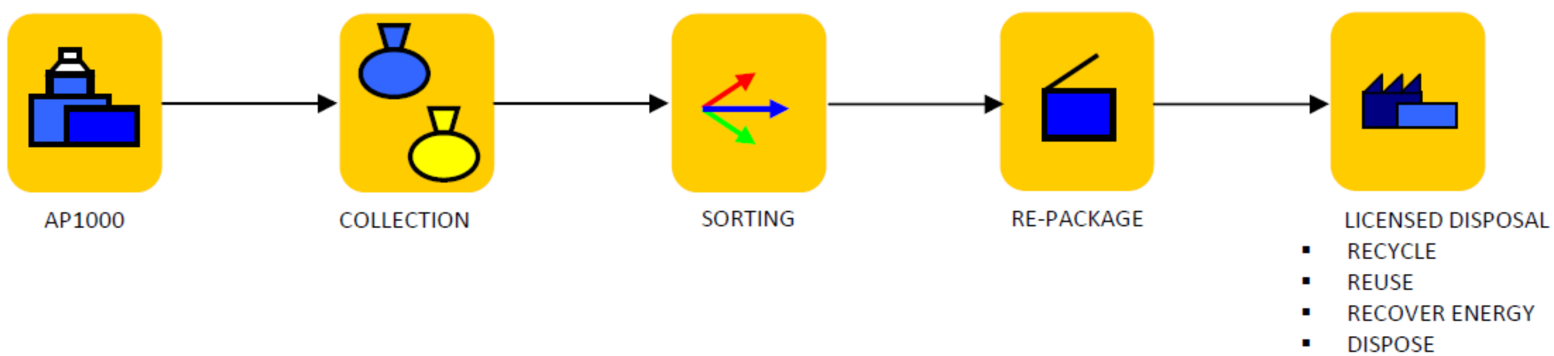

Figure 4.3-1. Conventional Solid Waste Treatment and Disposal Route

5.0 ENVIRONMENTAL IMPACT

5.1 Characteristics of the Generic Site

Much of the **AP1000** NPP design information presented in this document is independent of the location chosen for its construction. However, some assumptions about the characteristics of the plant's environment must be considered in developing the design of certain safety and environmentally-related features. In the absence of the selection of a specific site and in order to allow early assessment of the proposed reactor designs, it is assumed that the site has characteristics of the generic coastal site established in Reference 5-1.

The generic site characteristics are based on information obtained from five coastal nuclear power stations around the UK. These power stations are Dungeness (A), Hartlepool (B), Heysham (C), Hinkley (D), and Sizewell (E). These sites are considered typical of the range of nuclear coastal sites in the UK. The sites are located around the English coast (see Figure 5.1-1).

The information obtained in this section has been largely derived from the government's on-line geographical information system (Reference 5-2).

Maps have been generated from the generic site data gathered in Reference 5-1 and, although not unique solutions to the generic site, are consistent with the information and help to visualise the generic site. Figure 5.1-2 shows the population centres for the generic design case. Figure 5.1-3 shows the land use and habitat areas within 5 km (3 miles) of the **AP1000** NPP located on the generic site. Figure 5.1-4 shows the sites of special interest within 5 km (3 miles) of the site.

5.1.1 Human Population

Analysis has been carried out on the centres of population within 20 km (12 miles) of the five coastal power stations used as a basis of this assessment (Reference 5-1). It has been assumed that the generic site has the 80th percentile number of population centres within a given distance. For the purpose of the generic site, it is assumed that the population centres of a given size are located at the nearest distance recorded for the five existing nuclear power stations.

The assumed number of population centres within 2 km (1 mile), 10 km (6 miles), and 20 km (12 miles) of the generic site are shown in Table 5.1-1. The table also shows the number of individual farms and properties within 1 km (0.6 mile) and 2 km (1 mile) of the site, and the nearest population centre of a given size.

5.1.1.1 Exposed Population Groups

Two exposure groups are considered at a generic coastal site – the local resident family and the fisherman family.

The local resident family is selected to represent the exposure pathways associated with atmospheric releases from the **AP1000** NPP point sources. The habit data which includes food consumption, breathing rates, and occupancy fraction for this group is described in Table 5.1-2.

The fisherman family is selected to represent the exposure pathways associated with the discharges from the **AP1000** NPP point to the coastal environment. The habit data associated with this group is described in Table 5.1-3. It is assumed that the members of the fisherman family exposure group consume fish, molluscs and crustaceans at higher consumption rates than the local resident family. A fisherman family may be exposed to radiation through the following pathways:

- internal irradiation from the consumption of seafood contaminated with radionuclides;
- external radiation from radionuclides in beach and shore sediment during bait collection.

5.1.2 Reference Organisms

It is assumed that various terrestrial and marine reference organisms are located within the vicinity of the plant (see Table 5.1-4). The reference organisms have been selected to be representative of all protected species within Europe (Reference 5-3). The reference organisms have precisely defined anatomical, physiological, and life history properties that can be used for the purposes of relating exposure to dose and dose to effects for other organisms with similar taxonomy.

5.1.3 Meteorology

The meteorological data for the generic site is summarised in Table 5.1-5. The data set has been derived from the worst case maximum and minimum data and the average data from the five nuclear sites described in Reference 5-1.

5.1.3.1 Atmospheric Conditions

For the purpose of the generic site it is assumed that the atmospheric conditions are as shown in Table 5.1-6 (Reference 5-4). The Pasquil Stability Category is a measurement of atmospheric turbulence; A = unstable and G = extremely stable.

5.1.3.2 Atmospheric Disposition Coefficients

For the purpose of human health risk assessment, the deposition coefficients (a measure of the rate of transfer of pollution from the air to the earth's surface) are proposed as follows (Reference 5-5):

Dry deposition velocity:	Default:	0.001 m/s (0.003 ft/s).
	Inorganic forms of iodine isotopes:	0.01 m/s (0.03 ft/s.
	Noble gases:	zero
Washout coefficient:	Default:	0.0001 s^{-1}.
	Noble gases:	zero

5.1.4 Terrestrial Environment

5.1.4.1 Topography

The highest ground elevations within 2 km (1 mile) and 10 km (6 miles) of the generic site are shown in Table 5.1-7.

5.1.4.2 Land Cover/Surface Roughness

The main land cover within 5 km (3 miles) of the sites is listed in Tables 5.1-8. A surface roughness of 0.3 m (1 ft) is assumed for a typical rural location.

5.1.4.3 Geology and Hydrogeology

It is assumed that the land is stable and the presence of faults is minimal. It is assumed that the superficial geology is glacial clays with sands and gravel lenses. Discontinuous, perched groundwater is assumed to be 2 m (7 ft) below the site surface. It is also assumed that the site overlies a major aquifer with groundwater level at 20 m (66 ft) below ground level.

The **AP1000** NPP is designed for a normal groundwater elevation to within 0.6 m (2 ft) of the plant grade elevation.

5.1.4.4 Seismology

The following information has been obtained from the British Geological Survey website (Reference 5-6).

Twenty to thirty earthquakes are felt by people every year in the UK. Most of these are very small and cause no damage. However, some British earthquakes have caused some damage, although nothing like the devastation caused by large earthquakes in other parts of the world.

A magnitude 4 earthquake on the Richter scale happens in Britain roughly every two years. A magnitude 5 earthquake in the UK occurs roughly every 10 to 20 years. The largest recorded earthquake in the UK had a magnitude of 6.1 and occurred 100 km (60miles) off the Yorkshire coast beneath the North Sea. Research suggests that the largest possible earthquake in the UK is around magnitude 6.5. For the purpose of the generic site characterisation, it is assumed that the site has the potential to experience a magnitude 6.5 earthquake.

The **AP1000** NPP safe shutdown earthquake design is for a peak ground acceleration of 0.3g. This ground acceleration would be typical of a magnitude 6.7-8 earthquake. This design exceeds the largest recorded earthquake in the UK.

5.1.4.5 Natural Habitat/Nearest Sensitive Sites

The characteristic semi-natural habitats within 5 km (3 miles) of the site are identified in Table 5.1-9. The nearest sensitive sites to the generic site are assumed to be as shown in Table 5.1-10.

5.1.5 Coastal Environment

5.1.5.1 Tidal Range/Volumetric Exchange Rate

The assumed tidal range for the generic site is shown in Table 5.1-11 (Reference 5-1). The volumetric exchange rate for the generic site is 130 m^3/s (430 ft/s) (Reference 5-7). The generic site has been allocated the most conservative (lowest) exchange rate of the five nuclear coastal sites evaluated.

5.1.5.2 Intertidal Tidal Zone

It is assumed that the generic site may have a wide range of intertidal substrates within 10 km (6 miles) of the site. These may include sand, gravel, sand and gravel, rock platform, mud, sand and mud, and made ground.

5.1.5.3 Bathymetry

The assumed water depths off the coast of the generic site are shown in Table 5.1-12. The depths are based on Admiralty Chart datum.

5.1.5.4 Marine Biology

The marine biological features assumed to be within 10 km (6 miles) of the generic site are listed in Table 5.1-13.

5.2 Prospective Human Dose Assessment

5.2.1 Approach

The following assessments of the effects of the **AP1000** NPP aerial emissions and liquid discharges on members of the public have been made:

- annual individual dose to the most exposed members of the public for liquid discharges;
- annual individual dose to the most exposed members of the public for gaseous discharges;
- annual dose to the most exposed members of the public for all discharges from the facility;
- annual dose from direct radiation to the most exposed member of the public;
- annual dose to the critical group for the facility;
- a comparison of the calculated doses with the relevant dose constraints;
- potential short-term doses;
- collective dose for liquid discharges;
- collective dose for gaseous discharges; and
- an assessment of the build-up of radionuclides in the local environment.

The assessment of annual individual doses has been made following the Initial Assessment Method, provided by the Environment Agency (Reference 5-7). This consists of three assessment stages. Firstly, a conservative scoping assessment is carried out applying default data (stage 1). If the resulting dose exceeds 20 $\mu Sv\ y^{-1}$ (2 millirems/yr) then the assessment is refined by applying more appropriate data (stage 2). If the resulting dose still exceeds 20 $\mu Sv\ y^{-1}$ (2 millirems/yr) then a detailed assessment is carried out (stage 3).

5.2.2 Initial Assessment of Doses Stage 1

5.2.2.1 Stage 1 Doses from Liquid Discharges

Liquid discharges to the marine environment are based on data from Table 6.1-6. For the purpose of the dose assessment, cerium-144 was assigned as surrogate radionuclide for the 'other radionuclides' category. Doses were calculated for annual representative discharges as well as for calculated annual limit discharges. Representative radionuclides and liquid discharge rates used for the dose assessment are shown in Table 5.2-1.

The EA's initial assessment methodology (Reference 5-7) was used to assess the potential doses from liquid discharges of the **AP1000** NPP. It is based on generic Dose Per Unit Release (DPUR) values. For liquid discharges into the sea, the relevant exposure group is "fisherman family." This group is assumed to be exposed to radioactive releases from the proposed **AP1000** NPP through the following pathways:

- internal irradiation from the consumption of seafood contaminated with radionuclides;
- external irradiation from radionuclides in beach and shore sediments during bait digging.

Detailed information on these pathways and associated habit data such as seafood consumption rates and sediment occupancy rates are listed in Table 5.1-3. Dose per unit intake factors for ingestion are shown in Table 5.2-2. Doses are assessed in the 50th year of discharge, the only available integration time option. The DPUR values used in this assessment are shown in Table 5.2-3.

The results are shown in Table 5.2-4 for representative discharges and Table 5.2-5 for limit discharges by radionuclide and pathway. The total dose from liquid discharges is 3.0 µSv y^{-1} 0.3 millirems/yr for representative discharges and 4.8 µSv y^{-1} (0.48 millirems/yr) for calculated annual limit discharges. The doses are dominated by carbon-14 which contributes 67 percent to the annual doses from liquid discharges, followed by cobalt-60 which contributes 29 percent.

5.2.2.2 Stage 1 Doses from Gaseous Discharges

Atmospheric discharges were taken from Table 6.1-5. For the purpose of the dose assessment representative surrogate radionuclides were assigned to the following categories:

- Iodine-133 for "other iodines," taken to be all radioiodines apart from iodine-131 for which doses have been assessed individually;
- Krypton-85 for "other noble gasses," taken to be all isotopes of krypton and xenon apart from krypton-85 and xenon-133 for which doses have been assessed individually;
- Cobalt-58 for "other particulates," taken to be all particulates apart from cobalt-60, strontium-90, and caesium-137, for which doses have been assessed individually.

Doses were calculated for annual representative discharges as well as for annual limit discharges. Representative radionuclides and gaseous discharge rates used for the dose assessment are shown in Table 5.2-6.

The EA's initial assessment methodology (Reference 5-7) was used to assess the potential doses from atmospheric discharges of the **AP1000** NPP. For atmospheric discharges, the relevant exposure group is "local resident family." This group is assumed to be exposed to radioactive releases from the proposed **AP1000** NPP through the following pathways:

- inhalation of radionuclides in the effluent plume at a distance of 100 m (330 ft);
- internal irradiation from the consumption of terrestrial foodstuffs incorporating radionuclides deposited to the ground at a distance of 500 m (1640 ft); and
- external irradiation from radionuclides in the effluent plume and deposited to the ground at a distance of 100 m (330 ft).

Detailed information on these pathways and associated habit data such as terrestrial food consumption rates, inhalation rates, building shielding factors, and occupancy times are listed in Table 5.1-2. The meteorological data applied is referred to as '50% stability category D', as shown in Table 5.1-6. Dose per unit intake factors for ingestion and inhalation are shown in Table 5.2-7. Doses are assessed in the 50th year of discharge, the only available integration time option. Similarly, the location distance of 100 m (330 ft) for the local resident family is the only distance available as part of the EA's initial assessment methodology. This distance leads to a conservative dose assessment in relation to the generic site distance to receptor of 280 m (200 m to site boundary + 80 m) (920 ft (660 ft to site boundary +260 ft)) to nearest residential property, see Tables 3.3-4 and 5.1-1). The DPUR values used in this assessment are shown in Table 5.2-8.

The results are shown in Table 5.2-9 for representative discharges and Table 5.2-10 for limit discharges by radionuclide and pathway. The total dose from aerial discharges is 51 μSv y^{-1} (5.1 millirem/yr) for representative discharges and 79 μSv y^{-1} (7.9 millirem/yr) for limit discharges. The doses are dominated by carbon-14 which contributes 86 percent to the annual doses from aerial discharges, followed by argon-41 at 8 percent.

5.2.2.3 Direct Radiation Doses

Exposure to external radiation due to direct radiation from the **AP1000** NPP design will result in a very small dose to members of the public. A direct radiation dose typically arises from a number of sources on a nuclear site, including the main reactor building and any waste processing or storage plants. Its magnitude varies greatly according to the distance and angle between these sources and the receptor. For the purposes of estimating dose to members of the public, a direct radiation dose is not normally modelled. Rather, a number of measurements are made around the site perimeter fence and these are used to estimate a public dose.

The closest comparable design currently in operation in the UK is the Sizewell B PWR, which is based on an older Westinghouse design. The measured direct shine dose at the Sizewell B perimeter fence was 4 (0.4 millirems/yr) μSv in 2007 (Reference 5-8). These measurements were taken after the Sizewell A station ceased power generation in December 2006 and before decommissioning activities commenced.

Based on the existing Sizewell data, an annual dose contribution to the **AP1000** NPP design critical group of 4 μSv y^{-1} (0.4 millirems/yr^{-1}) has been applied here.

5.2.2.4 Total Stage 1 Doses

The total assessment stage 1 dose is the sum of the dose resulting from liquid discharge, gaseous discharges and direct radiation. The total stage 1 dose is 58 μSv y^{-1} (5 millirems/yr) for representative discharges and 88 μSv y^{-1} (8.8 millirems/yr) for discharges at the proposed discharge limit.

These doses are higher than 20 μSv y^{-1} (2 millirems/yr). As a result a stage 2 assessment was carried out.

5.2.3 Initial Assessment of Doses Stage 2

5.2.3.1 Stage 2 Doses from Liquid Discharges

For the refined initial assessment for liquid discharges, the volumetric exchange rate of water between the local and regional marine compartments was changed from the default value of 100 m^3 s^{-1} (330 ft/s) to 130 m^3 s^{-1} (430 ft/s) in accordance with the generic site information (see Section 5.1.5.1). The revised DPUR values used in this assessment are shown in Table 5.2-11.

The results are shown in Table 5.2-12 for representative discharges and Table 5.2-13 for limit discharges by radionuclide and pathway. The total dose from liquid discharges is 2.3 μSv y^{-1} (0.23 millirem/yr) for representative discharges and 3.7 μSv y^{-1} (0.37 millirem/yr) for calculated annual limit discharges. The doses are dominated by carbon-14 which contributes 68 percent to the annual doses from liquid discharges, followed by cobalt-60 which contributes 29 percent.

5.2.3.2 Stage 2 Doses from Gaseous Discharges

For the refined initial assessment for gaseous discharges, the release height was changed from the default value of 0 m (0 ft) to the effective release height of the main plant vent of the **AP1000** NPP. The effective release height is a function of the physical stack height, an initial rise of the plume due to its momentum at the point of release and the characteristics of any nearby buildings. The presence of buildings can alter the structure of the wind field that the plume enters and can lead to the plume being entrained into the building wake, thus resulting in higher activity concentrations at ground level closer to the source. For the purpose of this assessment it has been assumed that all releases occur from the main plant vent.

In order to derive an effective release height the atmospheric dispersion model ADMS (Reference 5-9) was applied. A distribution of meteorological conditions of '50% stability category D' was applied, as shown in Table 5.1-6. Firstly, the model was run using the in-built plume rise and building modules. The relevant parameters used are shown in Table 5.2-14. With this set-up the model was run for a number of wind directions. Thus it was determined that the highest downwind activity concentrations at ground level occur when the bulk of the building is positioned upwind from the stack. In this configuration the plume becomes partially entrained in the building wake. Then an equivalent effective stack height was determined by re-running ADMS without the plume rise and building options but for a range of different physical stack heights. Resulting downwind activity concentrations were compared with those determined for the 'building upwind' configuration. For the distances of interest (100 m (330 ft) and 500 m (1600 ft)) the lowest equivalent stack height is 40 m (130 ft). As a result an effective release height of 40 m (130 ft) was used in the stage 2 dose assessment.

The following release height scaling factors were applied to the dose per unit discharge factors (Figure 2 of Reference 5-7):

- food dose scaling factor of 0.15;

- inhalation and external dose scaling factor of 0.007.

The revised DPUR values used in this assessment are shown in Table 5.2-15.

The results are shown in Table 5.2-16 for representative discharges and Table 5.2-17 for limit discharges by radionuclide and pathway. The total dose from aerial discharges is 3.6 μSv y^{-1} (0.36 millirem/yr) for representative discharges and 5.6 μSv y^{-1} (0.56 millirem/yr) for limit discharges. The doses are dominated by carbon-14 which contributes 93 percent to the annual doses from aerial discharges, followed by iodine-131 at 4 percent.

5.2.3.3 Total Stage 2 Doses

The total dose for assessment stage 2 is the sum of the stage 2 doses resulting from liquid discharge, gaseous discharges and direct radiation. The direct radiation dose has been taken to be as determined for stage 1 (see Section 5.2.2.3). The total stage 2 dose is 9.8 μSv y^{-1} (0.98 millirem/yr) for representative discharges and 13 μSv y^{-1} (1.3 millirem/yr) for discharges at the proposed discharge limit.

These doses are lower than 20 μSv y^{-1} (2 millirem/yr). As a result a stage 3 assessment was not carried out.

5.2.4 Total Individual Doses for Comparison with the Discharge Limit and Discharge Constraint

In order to derive the annual dose from all discharges to the most exposed members of the public, the following have to be taken into account:

- Annual dose from liquid discharges (Section 5.2.3.1)
- Annual dose from atmospheric discharges (Section 5.2.3.2)

From the results of the dose assessments for the fisherman family (exposed to liquid discharges from the proposed facility) and for the local resident family (exposed to aerial discharges), it can be seen that the local resident family potentially receives the highest dose. As a result, the local resident family is the individual exposure group receiving the highest dose in this assessment.

However, it can not be ruled out at this stage that the fisherman family is not exposed to the aerial discharges from the facility and vice versa for the local resident family. Therefore, it is prudent to add the contribution from both discharge streams. Thus, the annual dose to the critical group for all continuous discharges from the **AP1000** NPP design is 5.8 μSv y^{-1} (0.58 millirem/yr) for representative discharges and 9.2 μSv y^{-1} (0.92 millirem/yr) for limit discharges.

In order to assess the total annual dose to the critical group, direct radiation also needs to be taken into account. Adding the direct radiation dose of 4 μSv y^{-1} (0.4 millirem/yr) (Section 5.2.2.3) to the dose from all discharges gives a total dose of 9.8 μSv y^{-1} (0.98 millirem/yr) for representative discharges and 13.2 μSv y^{-1} (1.32 millirem/yr) for limit discharges. These represents maximum critical group doses as it is assumed that members of this group are exposed to both discharge streams as well as the direct radiation dose expected at the site boundary fence.

This value can be compared with the dose constraint of 300 μSv y^{-1} (30 millirem/yr) which is applicable to any single new source in the UK (Reference 5-10). The doses lie well below the dose constraint, by a factor of over 20.

The total dose to the Sizewell critical group in 2007 was < 5 μSv (0.5 millirem/yr) (Reference 5-8). It should be noted that this dose has been derived using different methods from those applied here; for example, radionuclide levels in terrestrial and marine foodstuffs were based on monitoring data rather than computer model calculations.

5.2.5 Potential Short-Term Doses

5.2.5.1 Approach

When doses from routine discharges are assessed, it is normally assumed that these discharges occur continuously and uniformly over a year. However, during normal operations at nuclear sites, it is possible to have short-term enhanced releases, e.g., during routine maintenance operations of the plant. An assessment of the potential impact of short-term doses is typically carried out as part of a detailed stage 3 assessment. Although a full stage 3 dose assessment of operational discharges of the **AP1000** NPP was not undertaken, an assessment of the potential impact of short-term doses was carried to determine if the impact is significant in comparison with the doses from routine releases.

Discharging radioactive material to the atmosphere over the short-term may lead to doses that are higher than would be expected if it were assumed that the same discharge took place uniformly over a year. This is mainly due to the fact that short-term releases can lead to peak activity concentrations in air and foodstuffs, which, combined with seasonal agricultural practices and variation in habit data, can lead to higher doses.

For liquid discharges to the marine environment, effects from short-term releases are deemed to be much lower. This is mainly due to limited pumping capacity from discharge tanks, making it unfeasible to assume that a month's liquid discharge volume can be released into the marine environment over a period of a few hours. Also, for marine discharges via a pipeline which discharges into the sea away from the near shore, the timescale of the release is less important than for atmospheric, marine near shore, or freshwater releases. This is because travel times to potential exposure locations are much longer. As a result, only the effects from short-term aerial releases are assessed here.

A methodology published by the Health Protection Agency (HPA) has been adapted to assess the impact of short-term atmospheric releases from the **AP1000** NPP design (Reference-5-13).

5.2.5.2 Atmospheric Dispersion Modelling for Short-Term Release

The short-term discharges have been grouped into the same radionuclide categories used for the routine assessment (see Section 5.2.2.2). The maximum short-term planned discharge is taken to be the highest planned discharge in a single month, based on Table 6.1-3. The discharge period is conservatively set at 0.5 hours, the shortest period recommended in the HPA methodology. The discharge is data used in this assessment is shown in Table 5.2-18.

The atmospheric dispersion model ADMS (Reference 5-9) was used to derive activity concentrations in air, deposition rates, and cloud gamma doses. The model set up was based on HPA's recommendation to give a cautious estimate of atmospheric concentrations and deposition on the ground close to the discharge location (Reference 5-11). Details of meteorological conditions and deposition rates used are given in Table 5.2-19.

To derive activity concentrations in air and deposition rates ADMS was run with the building and plume rise options enabled. The relevant model set up parameters are shown in Table 5.2-14. To derive cloud gamma dose factors ADMS was run using an effective release height, as the building module can not be run at the same time as the gamma dose module. The effective release height for short-term releases was derived similarly as described in Section 5.2.3.2 but using the meteorological data given in Table 5.2-19.

The predicted activity concentrations in air and deposition rates per unit release for each radionuclide from the ADMS model run are shown in Table 5.2-20. For the cloud gamma pathway cloud gamma dose factors from gamma emitting radionuclide daughters were aggregated into the results for each parent radionuclide. The cloud gamma dose factors are shown in Table 5.2-21.

5.2.5.3 Dose Calculation for Short-Term Release

The dose pathways and exposure locations assumed for these short-term releases were the same as for the atmospheric dose methodology for routine releases, described in Section 5.2.2.2.

The doses to adults, children, and infants were calculated separately for each pathway. The methodology used to calculate the dose from the modelled activity concentrations for each pathway is the same as the HPA's. More information on the input data is given below:

- The breathing rates were taken from Table 4 in the HPA report (Reference 5-11), shown here in Table 5.2-22.

- The same dose per unit intake values for ingestion and inhalation were used as for the assessment for routine releases described in Section 5.2.2.2 and shown in Table 5.2-7.

- The crop and animal uptake concentrations per unit deposition were derived from runs of the FARMLAND module which is part of the PC CREAM08 modelling suite (Reference 5-12). The values were adjusted for a single, instantaneous deposition. That is:

  The concentration *C* in the food at time *T* following a continuous deposition rate of 1 Bq/m^2/s (2.51×10^{-6} (μCi/ft^2)/s) is given by

  $$C(T) = \int_0^T c(T - t)dt$$

  where c(τ) is the concentration in the food at time τ after an instantaneous unit deposition.

  The average concentration over a period from an instantaneous deposition is the integrated concentration over the year divided by the length of period, i.e.:

  $$\overline{c(T)} = \frac{1}{T}\int_0^T c(t)dt$$

  Reversing the integration limits (i.e., substitute t'=T-t) shows that this is equivalent to:

$$\overline{c(T)} = \frac{1}{T}\int_0^T c(T - t')dt'$$

Therefore, the average concentration in food over the year after an instantaneous deposition is:

$$\overline{c(T)} = \frac{C(T)}{T} = \frac{C(T)}{365 \times 24 \times 3600}$$

That is, the average concentration in food over one year from an effectively instantaneous 1 Bq/m^2 (3×10^{-6} µCi/ft^2) deposition is the Farmland output concentration following a continuous release for one year divided by 3.2 x 10^7. For tritium and carbon-14, the values are for a continuous air concentration of 1 Bq m^{-3}) (8×10^7 µcuries/ft^3), and a similar argument applies.

The final values are shown in Table 5.2-23. Note that the tritium and carbon-14 crop uptakes are based on air concentrations, rather than ground deposition rates.

- The critical and average food intake rates are taken from Reference (5-22). Note that critical group intakes were only used for the two food groups giving the highest dose for each person; average intakes were used for other foods. The two foods leading to the highest ingestion dose are root vegetables and milk. The ingestion rates used for the short-term dose assessment are shown in Table 5.2-24.

- The annual external dose per unit surface deposition rates were taken from the ORNL Radiological Toolbox (Reference 5-11) and adjusted for radioactive decay over a year. The final dose rates are shown in Table 5.2-25.

- The shielding and occupancy factors were taken from Table 5 in the HPA report (Reference 5-11) and are reproduced here in Table 5.2-26.

The results are shown in Table 5.2-27 for adults, Table 5.2-28 for children and Table 5.2-29 for infants by radionuclide and pathway. The highest dose to the local inhabitant exposure group from a single atmospheric short-term discharge release is 4.9 µSv (0.49 millirem).

5.2.6 Collective Dose Assessment

5.2.6.1 Assessment Approach

Radionuclides discharged into the environment have the potential to disperse, allowing exposure of wider populations, albeit at much lower levels of individual exposure than to the individuals within the general population who would be expected to receive the highest doses (the critical group). This collective effective dose is defined as the sum of all the exposures from a given source to a defined group of people and has units of man-sieverts (manSv).

The assessment of collective dose was carried out following guidance provided in Reference 5-10.

Collective doses will arise from discharges of radionuclides to atmosphere and from liquid discharges to the marine environment.

Collective doses were derived by combining discharge rates and collective dose per unit discharge factors. The collective dose per unit discharge factors for radionuclides discharged to atmosphere and sea were calculated using the PC CREAM08 model (Reference 5-12). The collective dose assessment is based on population and food production grids for the atmospheric assessment and seafood catch data for the marine assessment. PC CREAM also contains a model to estimate doses from the global circulation of released radionuclides.

Some radionuclides, owing to the magnitude of their radioactive half-lives and their behaviour in the environment, may become globally dispersed and act as a long-term source of irradiation of both the regional and world populations. Such exposures would be in addition to the irradiation of the populations exposed during the initial dispersion of these radionuclides from their points of discharge. Of the radionuclides discharged by the **AP1000** NPP design, tritium, carbon-14, and krypton-85 are affected by this characteristic. As a result, the UK and Europe doses presented here are the sum of the so-called “first pass” dispersion dose and the global circulation component for these radionuclides. Collective doses to the world population from atmospheric discharges were assumed to be equivalent to the first pass dispersion dose for Europe plus the global circulation component for the world population for tritium, carbon-14, and krypton-85. Where krypton-85 was used as surrogate radionuclide for “other noble gasses,” the global circulation part of the dose was not included, as all noble gasses listed under that category have half lives of a few days at most.

Collective doses were calculated for UK, European, and World populations, truncated at 500 years.

Collective dose assessments from routine discharges tend to be site-specific as they rely on grids of population distribution, agricultural food production, and seafood catches that have been established for individual sites. In the absence of a specific site, the collective dose assessment was carried out for an **AP1000** NPP located at each of the five representative coastal sites used to determine the generic site characteristics – Dungeness, Hartlepool, Heysham, Hinkley Point, and Sizewell (see Section 5.1).

5.2.6.2 Collective Doses from Liquid Discharges

Liquid discharges to the marine environment are based on data from Table 6.1-6. For the purpose of the dose assessment, cerium-144 was assigned as surrogate radionuclide to the ‘other radionuclides’ category. Doses were calculated for annual representative discharges as well as for calculated annual limit discharges. Representative radionuclides and liquid discharge rates used for the dose assessment are shown in Table 5.2-1.

Default PC CREAM model settings were applied.

5.2.6.3 Collective Doses from Gaseous Discharges

Atmospheric discharges were taken from Table 6.1-5. For the purpose of the dose assessment representative surrogate radionuclides were assigned to the following categories:

- Iodine-133 for “other iodines,” taken to be all radioiodines apart from iodine-131 for which doses have been assessed individually;

- Krypton-85 for “other noble gasses,” taken to be all isotopes of krypton and xenon apart from krypton-85 and xenon-133 for which doses have been assessed individually;

- Cobalt-58 for "other particulates," taken to be all particulates apart from cobalt-60, strontium-90, and caesium-137, for which doses have been assessed individually.

Doses were calculated for annual representative discharges as well as for annual limit discharges. Representative radionuclides and gaseous discharge rates used for the dose assessment are shown in Table 5.2-6.

Default PC CREAM model settings were applied with the following exceptions:

- An effective release height of 40 m (130 ft) was applied, as described in Section 5.2.2.2.

- For the atmospheric assessment, an atmospheric stability distribution of 50 percent category D was applied (see Table 5.1-6).

5.2.6.4 Collective Dose Assessment Results

The collective doses per year of discharge and truncated to 500 years from the **AP1000** NPP are shown in Tables 5.2-30 and 5.2-31 for discharges to atmosphere and in Tables 5.2-32 and 5.2-33 for liquid discharges to the marine environment. These tables provide maximum, averaged, and minimum summary statistics of the results obtained for the five sites evaluated.

For representative discharges to the atmosphere the maximum collective dose is 0.23 manSv (23 rem) for the UK population, 1.5 manSv (150 rem) for the European population and 8.8 manSv (880 rem) for the World population.

For limit discharges to the atmosphere the maximum collective dose is 0.36 manSv for the UK population, 2.3 manSv (230 rem) for the European population and 14 manSv (1400 rem) for the World population.

For representative liquid discharges the collective dose is 0.001 manSv (0.1 rem) for the UK population, 0.0046 manSv (0.46 rem) for the European population and 0.033 manSv (3.3 rem) for the World population.

For limit liquid discharges the collective dose is 0.0017 manSv (0.17 rem) for the UK population, 0.0072 manSv (0.72 rem) for the European population and 0.053 manSv (5.3 rem) for the World population.

The total collective doses are dominated by doses from atmospheric discharges by more than two orders of magnitude.

5.2.7 Build-up of Radionuclides in the Environment

The prospective build-up of radionuclides discharged from the facility in the local environment has been evaluated. For liquid discharges into the marine environment, the build-up in marine coastal sediments has been assessed. For aerial discharges, the build-up in undisturbed soil has been assessed.

In order to take account of accumulation of radionuclides over the plant's operational life span, the plant's anticipated licensing period should be taken into account (Reference 5-10). Therefore it was assumed that build-up occurs over 60 years.

For discharges to the atmosphere, predicted environmental concentrations of radionuclides in soil were obtained with the PLUME and FARMLAND modules of the PC CREAM08 model (Reference 5-12). For the build-up in the environment, radionuclides present in the discharge list have been assessed individually, rather than grouped into categories. The build-up of noble gases in soil and sediments was assumed to be negligible.

Tritium and carbon-14 are not included in FARMLAND. The transfer of tritium and carbon-14 between the atmosphere and the terrestrial environment is more complex than that for other radionuclides, since hydrogen and carbon are fundamental to biological systems. A relatively simple "specific activity" approach is widely used in terrestrial foodchain modelling for these radionuclides. It is assumed that all foodstuffs come into rapid equilibrium with atmospheric carbon-14 and tritium in atmospheric water vapour, and thus, it is implicitly assumed that build-up does not occur.

Discharge data used are based on data in Tables 3.3-6 to 3.3-8. These are average annual discharges based on average releases over the 18 month fuel cycle. The following assumptions were made for calculating soil concentrations as a result of aerial releases:

- Effective stack height of 67 m (220 ft) (derived for a distance of 500 m (1600 ft) using the method described in Section 5.2.3.2)
- Distance from source 500 m (1600 ft)
- Uniform windrose
- Atmospheric stability category distribution of 50% D

For discharges to the marine environment, predicted environmental activity concentrations of radionuclides in coastal sediments were obtained with the DORIS module of the PC CREAM08 model (Reference 5-12). As far as possible, radionuclides present in the discharge list have been assessed individually. Discharge data are based on the average annual data in Table 3.4-6. Any radionuclides in the discharge list for which model parameters were not available have not been assessed. These include rubidium-88, molybdenum-99 and tungsten-187.

The following assumptions were made for calculating coastal sediment concentrations as a result of liquid discharges:

- Volumetric exchange rate of water between the local and adjacent regional compartments: 130 m^3 s^{-1} (34,000 gallons/s).

Activity concentrations in soil after 60 years of atmospheric discharges from the **AP1000** NPP design are shown in Table 5.2-34 for the radionuclides in the discharge list and their daughters. Activity concentrations in the soil range from 2.9×10^{-20} Bq kg^{-1} (3.6×10^{-31} Ci lb^{-1}) for niobium-95m to 1.1×10^{-12} Bq kg^{-1} (1.3×10^{-23} Ci lb^{-1}) for caesium-137.

Activity concentrations in local coastal sediment after 60 years of liquid discharges from the **AP1000** NPP design are shown in Table 5.2-35 for the radionuclides in the discharge list and their daughters. The activity concentration for tritium is 14 Bq kg^{-1} (1.7×10^{-10} Ci lb^{-1}) and that of carbon-14 is 1.1 Bq kg^{-1} (1.3×10^{-11} Ci lb^{-1}). The activity concentrations for all other radionuclides in the sediment range from 3.4×10^{-19} Bq kg^{-1} (4.2×10^{-30} Ci lb^{-1}) for daughters of plutonium-241 to (0.54 Bq kg^{-1}) (6.6×10^{-12} Ci lb^{-1}) for nickel-63. Interpretation of these activity concentrations is not clear at present as no widely accepted guidance is available. However, the radiological impact of these is already covered in the dose assessments in Sections 5.2.2 to 5.2.4. The dose assessments also incorporate the build-up of the radionuclides in the environment as the doses have been assessed in the 60th year of discharge.

5.3 Radiological (Non-Human Dose Assessment)

An assessment of the likely impact of radioactive discharges from the Westinghouse **AP1000** NPP on non-human species has been made in UKP-GW-GL-033, "Assessment of Radioactive Discharges on Non-Human Species" (Reference 5-14). This report is summarised in this section.

The predicted radioactive emissions and discharges from the **AP1000** NPP are input into the ERICA (February 2007) tool (Reference 5-3) to determine the impact on the various reference organisms identified in Table 5.1-4. The Wildlife Dose Assessment Spreadsheet Version 1.20 (Reference 5-15) was used to address the impact of emissions of the inert gas isotopes – argon, krypton, and xenon.

5.3.1 ERICA Assessment

The ERICA tool provides a recognised methodology for assessing the environmental exposure, effects, and risks from ionising radiation on ecosystems (Reference 5-3). The ERICA tool functions at three levels:

- Tier 1 – a concentration screening level

  The Tier 1 output is expressed as a risk quotient (RQ) where:

  $$RQ = M/EMCL$$

  where

  M = Estimated or measured activity concentration for a given radionuclide in Bq l^{-1} for water, Bq kg^{-1} dry wt for soil/sediment, or Bq m^{-3} for isotopes of C, H, P, and S within the terrestrial environment;

  EMCL = Screening Dose Rate (10 µGy h^{-1} (1 mRad/hr))/F

  where

  F = the dose rate that a given organism will receive for a unit concentration of a given radionuclide in an environmental medium (µGy h^{-1} per Bq l^{-1} or kg^{-1} (dry weight) or m^{-3} of medium). The value of F depends upon the reference organism type, its position(s) within habitat, and the radionuclide. It is calculated by the ERICA tool.

  For the terrestrial environment, EMCL values always refer to soil activity concentrations, except for isotopes of H, C, S, and P that refer to air concentrations.

  Tier 1 compares emissions and discharges against a default incremental dose rate of 10 µGy h^{-1} (1 mRad/hr) for all ecosystems and organisms. The overall RQ is the sum of the RQs for the most limiting reference organism for each radionuclide.

  If the sum of the RQs for all nuclides is less than one, there is a very low probability that the absorbed dose rate to any organism exceeds the screening dose rate, and the situation may be considered to be of negligible radiological concern. If the ratio exceeds unity, further Tier 2 assessment is required.

- Tier 2 – a dose rate screening level

  Tier 2 is a dose rate screening level. In Tier 2, the ERICA screening dose rate of 10 µGy h^{-1} (1 mRad/hr) is compared directly to the total estimated whole body absorbed dose rate for each individual organism. The RQ is calculated as follows:

$$RQ = \frac{\text{Whole Body Absorbed Dose Rate}}{\text{Screening Level Dose Rate}}$$

  In Tier 2, RQ for a given organism equals the sum of the radionuclide-specific RQs for that organism.

  Two RQs are reported in Tier 2 for every organism selected in the assessment; the expected RQ and the conservative RQ. The expected RQ uses the best estimate values for the input data and the parameters. The conservative RQ uses the 95th or 99^{th} percentile input values to determine the 5 percent or 1 percent probability of exceeding the dose screening value.

  The significance of the Tier 2 output is determined based on the values of the expected RQ and the conservative RQ for each individual organism (see Table 5.3-1). The results are categorised as "negligible," "insufficient confidence," and "of concern."

  For results in the "insufficient confidence" and "of concern" category, it may be necessary to make more qualified judgments and/or to refine model inputs to complete the Tier 2 assessment. Alternatively, an in-depth Tier 3 assessment is required.

- Tier 3 – site-specific probabilistic level which is beyond the scope of work carried out in Reference 5-14.

The ERICA assessment was carried out at the Tier 1 level for terrestrial organisms and at the Tier 1 and Tier 2 level for marine organisms.

5.3.1.1 Terrestrial Organisms

The input data for the ERICA tool is shown in Tables 5.3-2 to 5.3-3 for the terrestrial ecosystem assessment that is impacted by air emissions.

The Tier 1 output of the ERICA model is shown in Table 5.3-4. The Tier 1 results show that the sum of the RQs is substantially below unity indicating that the values are below the ERICA screening dose rate of 10 µGyh^{-1} (1 mRad/hr).

This indicates that no further analysis at Tier 2 or Tier 3 is required.

Sensitivity testing of the of the terrestrial ERICA assessment was carried out to determine whether changing wind speed, distance to receptor and stack height would affect the overall RQ. The results of the sensitivity tests are shown in Table 5.3-5 and are described below:

- Scenario A is a default scenario which matches the results in Table 5.3-4.
- Scenario B identifies the effect on the RQ if all emissions were released from the lower turbine vent stack. In practice, no more than 12 percent of the radioactivity emitted to the atmosphere from the plant is potentially released from the condenser air removal stack (see Table 3.3-7), so this is highly conservative.
- Scenarios C-F show the impact of changing the wind speed from 1 ms^{-1} (3.3 ft/s) to 10 ms^{-1} (33 ft/s). The RQ reduces as the wind speed increases.
- Scenarios G-J show the effect of changing the distance to receptor from 50 m (164 ft) to 300 m (980 ft). The RQ reduces as the distance to receptor increases.
- Scenario K is a worst case scenario assuming emissions are from the lower stack height, the lowest wind speed, and the nearest receptor distance. Under these conditions, the overall RQ is 0.291 indicating that the screening dose rate of 10 μGyh^{-1} (1 mRad/hr) is not exceeded.

It is concluded from the Tier 1 ERICA assessment that there is negligible risk to terrestrial organisms from the **AP1000** NPP radioactive air emissions.

5.3.1.2 Marine Organisms

The input data for the ERICA tool is shown in Tables 5.3-6 and 5.3-7 for the marine ecosystem assessment that is impacted by coastal discharges.

The Tier 1 output of the ERICA model is shown in Table 5.3-8. The results indicate that the sum of the RQs exceeds 1. The screening dose to polychaete worms warrants a Tier 2 assessment.

A Tier 2 analysis was carried out with all isotope data sets to predict total dose rates for each reference organism. The Tier 2 results for the ERICA model are presented in Table 5.3-9.

Based on the definition in Section 3.2.2, the Tier 2 results indicate "negligible risk" to wading birds, zooplankton, pelagic fish, and phytoplankton at distances greater than 100 m (330 ft) from the point of release. The results for mammals and reptiles indicate "insufficient confidence" to be sure that there will be negligible effect on these organisms which live within the sediment. The results for polychate worms, macroalgae, sea anemonies or true corals – polyps and colonies, benthic molluscs, vascular plants, benthic fish, and crustaceans show that the ERICA screening dose rate of 10 μGyh^{-1}(1 mRad/hr) is exceeded. The maximum predicted dose rate for all organisms is 25.2 μGyh^{-1} (2.52 mRad/hr) for polychaete worms.

Table 5.3-9 shows that the isotopes responsible for the ERICA screening dose rate being exceeded either Fe-59 or Fe-55. Iron partitions strongly into the sediment phase and the organisms experiencing dose rates greater than the ERICA screening dose rate of 10 μGyh^{-1} (1 mRad/hr) are the ones that have high occupancy factors in the sediment or at the sediment-water interface.

The sensitivity of the Tier 2 results to changes in input parameters has been investigated for variations in water depth, distance between release point and shore, distance between release point and receptor, and coastal current. The results are shown in Table 5.3-10 and are described below:

- Scenario A represents the default scenario which matches the results in Tables 5.3-10 and 5.3-9.

- Scenarios A-D show the effect of changing water depth. The dose rate decreases with increasing water depth. At a depth of 7m, only polychaete worms dose date exceeds the ERICA screening dose rate of 10 μGyh^{-1} (1 mRad/hr) (Scenario C). At a depth of 13 m (43 ft), the dose rate to polychaete worms drops below the ERICA screening dose rate and changes to a “insufficient confidence” condition, where the conservative RQ exceeds 1.

- Scenarios A and E-G show that changing the distance between the release point and the shore has no effect on dose rates on organisms 100 m (330 ft) from the discharge point at sea.

- Scenarios A and H-K show the effect of changing the distance between the release point and the receptor. The ERICA screening dose rate is exceeded for at least one organism at all distances between the release point and receptor up to 220 m (720 ft) (Scenario J). At 220 m (720 ft), the most sensitive organism, polychate worms, falls into the “insufficient confidence” category together with seven other organisms. At 560 m (1840 ft) (Scenario K) and beyond, the dose rates for all organisms fall into the “negligible” category indicating negligible risk.

- Scenarios A and L-N show the effect of changing the coastal current. The predicted dose rates decrease slightly as the coastal current decreases from 0.5 ms^{-1} (1.6 ft/sec) to 0.05 ms^{-1} (0.16 ft/sec). The number of organisms receiving dose rates above the ERICA screening level falls from eight to six as the coastal current decreases.

- Scenario O shows the worst case combination of variables from the scenarios selected above (i.e., water depth 2 m (6.6 ft), the distance between the release point and the shore 150m (490 ft), the distance between the release point and the receptor 50 m (160 ft), and the coastal current 0.5 ms^{-1} (1.6 ft/sec)). The results predict that the ERICA screening dose rate is exceeded for eleven organisms with the highest predicted dose rate of 191 μGyh^{-1} (19.1 mRad/hr) being experienced by polychaete worms.

- Scenario P shows the best case combination of variables from those selected above (i.e., water depth 13m, the distance between the release point and the shore 150 m (490 ft), the distance between the release point and the receptor 560 m (1840 ft), and the coastal current 0.05 ms^{-1} (0.16 ft/sec)). The results produce a “negligible” risk condition for all organisms.

The sensitivity analysis confirms that polychaete worms are the most vulnerable organism, experiencing the highest dose rates. This is because polychaete worms are the only organism that resides entirely within the sediment, and the dominant source of radioactive dose is from Fe-59 and Fe-55 (see Table 5.3-9), which partitions strongly into sediments. The range of dose rates predicted for polychaete worms in this sensitivity analysis is 3.27 μGyh^{-1} (0.327 mRad/hr) to 191 μGyh^{-1} (19.1 mRad/hr).

Organisms with occupancy factors of 100% at the sediment water interface (benthic molluscs, crustacean, macroalgae, benthic fish, sea anemones, or true corals – colony, polyps, and vascular plants) experience the next highest dose rates ranging from 0.5 μGyh^{-1} (0.05 mRad/hr) to 103 μGyh^{-1} (10.3 mRad/hr).

Mammals, wading birds, and reptiles have 100% occupancy factors in water, but food sources may be within the sediment or at the sediment – water interface. These organisms receive a range of dose rates under the scenarios considered of 0.1 μGyh^{-1} (0.01 mRad/hr) to 26 μGyh^{-1} (2.6 mRad/hr).

The organisms which experience the lowest dose rates fall within the "negligible" risk category for all scenarios. These are pelagic fish, phytoplankton, and zooplankton which have 100 percent occupancy factors in water with no habitation within the sediment surface or sediment zones. These organisms receive a range of dose rates under the scenarios considered of 0.001 μGyh^{-1} (0.0001 mRad/hr) to 0.2 μGyh^{-1} (0.02 mRad/hr).

The predicted dose rates have been compared with reported effects on organisms in Reference 5-14. However, it is notable that the dose rates are well below the 400 μGyh^{-1} (40 mRad/hr), which is a benchmark derived from the IAEA (1992) and UNSCEAR (1996) reports below, which populations are unlikely to be significantly harmed based on reviews of the scientific literature (References 5-16 and 5-17). This also corresponds to the U.S. Department of Energy (DOE) dose limit of 10 $mGyd^{-1}$ (1 Rad/day) ($\approx$ 400 μGyh^{-1}) (40 mRad/hr) for native aquatic animals (Reference 5-18).

The prediction that the ERICA screening dose rate of 10 μGyh^{-1} (1 mRad/hr) is exceeded for several organisms and the sensitivity analysis carried out indicates the importance of undertaking site-specific studies to optimise the selection of the effluent discharge point and minimise the impact on benthic organisms.

5.3.2 Wildlife Dose Assessment Spreadsheet

The Wildlife Dose Assessment Spreadsheet (Reference 5-15) was developed in the UK and formed an important building block of the ERICA tool. This spreadsheet comes in three forms; the freshwater, marine, and terrestrial versions. The terrestrial spreadsheet was used to supplement the analysis carried out by the ERICA tool, because it enables the impact of emissions of Argon-41 and Krypton-85 to be assessed. These inert gases contribute approximately 80 percent of the radioactive releases from the **AP1000** NPP to the atmosphere (see Tables 3.3-6 to 3.3-8).

The input data for the Wildlife Dose Assessment Spreadsheet includes the ground level concentrations of argon-41 and krypton-85 in Bqm^{-3} predicted at the assumed receptor distance of 200 m (656 ft). These have been calculated using the emissions data identified in Table 5.3-2 and the same Gaussian plume model equations as those built into the ERICA tool for an emission point that is in the lee of a building inside the wake zone (Reference 5-8). The ground level concentrations are shown in Table 5.3-11 together with other input data used in the Wildlife Dose Assessment Spreadsheet.

The weighted output of the Wildlife Dose Assessment Spreadsheet for the air emissions of Argon-41 and Krypton-85 is shown in Table 5.3-12. For Argon-41 and Krypton-85, the total dose rate occurs entirely from external rather than internal dose rates. The highest total dose rate occurs for fungi and is 0.00027 μGyh^{-1} (0.000027 mRad/hr). This is negligible when compared to the ERICA screening dose rate of 10 μGyh^{-1} (1 mRad/hr). This implies that the dose rate associated with noble gas emissions on terrestrial organisms is insignificant.

5.4 Accidental Release of COMAH Chemicals

The **AP1000** NPP site is an upper tier COMAH site because of storage of hydrazine (see subsection 2.9.2.1). The potential for a major accident to the environment (MATTE) resulting from the storage of hydrazine is minimal.

5.4.1 Causes of Hydrazine Spills

Hydrazine is stored on the Turbine Island in tote tanks (see subsection 2.9.3.1 and Figures 2.9-2 and 2.9-3).

Overflow from over filling the lower permanent storage tank from the tote container is not possible since there is no overflow connection. The worst case of misalignment results in filling vent pipes to the level of the hydrazine in the tank assembly. These vent pipes are directed up to a high elevation and cannot overflow.

There is no manual handling of drums or carboys that can be overturned or running drum pumps that can be left unattended.

The only credible cause of a major hydrazine spill is the puncturing of a tote container or lower permanent storage tank by a fork truck tine. The tote containers are rugged and have passed severe drop and vibration tests. Should a tine penetrate the protective outer shielding and tough inner wall, the leak rate would be limited by the size of the puncture. For a punctured tote container, the truck operator will be able to back up and tip the tote so that the hole is on the top, stopping the leak. A punctured permanent tank will drain the level of the hole in its side. The leak from a permanent storage tank will be contained within the spill barrier. However, the tough stainless steel walls of both the tote container and permanent tank are likely to bend and deform before they can be pierced.

5.4.2 Accidental Hydrazine Spills within the Turbine Building

During tote container handling, it is possible for a tote container to be dropped by a fork truck. These containers are certified for retention of contents by having been tested to ensure unacceptable leakage does not occur when the container is full and dropped in a way that is most likely to cause damage. Dropping a tote container inside or outside the turbine building is expected to be so rare an event that it will not happen during the life of the plant. Failure of the container to permit release of its contents is not probable. A feature at most plants is to have a temporary secondary containment established at the truck delivery/loading point. This should be large enough to contain the contents of a dropped tote. The **AP1000** NPP intends to follow the best practices developed by the nuclear industry.

Should there be a large spill in the turbine building during transport of the tote to the chemical storage area, the hydrazine will run into floor drains and flow to the building sump, part of the WWS. The wetted floor areas can be flushed with bleach (for example dilute calcium hypochlorite ($Ca(OCl)_2$) solution) to neutralise and wash the remaining hydrazine into the floor drain system (Reference 5-19). Additional bleach can then be added by pouring it into the floor drains. The sump is automatically pumped based on level into the WWS WWRB where waste water is held prior to its release to the plant outfall. The WWRB water is sampled and the chemical adjustment is made until it complies with discharge consents. When samples show the hydrazine has been neutralised, the WWRB contents are disposed. Therefore, a liquid chemical release from the turbine building through the WWS to the plant out fall is not a credible scenario.

5.4.3 Accidental Hydrazine Spills outside the Turbine Building

The worst case conceivable spill within the **AP1000** NPP site is the release of the contents of a hydrazine tote container (400 gallons, 1.51 m^3), due to overturning a trailer during transportation on the plant site roadway.

In the event that a large spill occurs outside of the turbine building, the liquid should be contained within a temporary spill barrier. If a large amount of hydrazine is involved, the wetted area should be sowed with solid calcium hypochlorite and washed with bleach dilute sodium hypochlorite solution (Reference 5-19). A small spill should be treated with bleach only. Using solid calcium hypochlorite results in an insoluble gooey paste like semi-solid that will present problems for subsequent cleanup.

An unmitigated spill unto an impermeable surface can be expected to behave in the same fashion as the Air Force hydrazine test samples that were left outside exposed to the environment. Hydrazine solutions evaporate in a similar fashion as does water when exposed (Reference 5-19). There is an initial high evaporation rate followed by a much slower evaporation rate. The density of the solution increases as hydrazine reacts with atmospheric carbon dioxide and the resulting product passes through stages described as oily, a syrupy consistency, and finally, a white semi-solid.

Vapor-phase hydrazine is degraded in the atmosphere by reaction with photochemically-produced hydroxyl radicals and ozone with estimated half-lives of about 6 and 9 hours, respectively.

A spill onto soil will result in rapid neutralisation of hydrazine due to reactions with organic carbon and clay. Spills onto sand will penetrate more deeply and take longer to neutralise.

5.4.4 Hydrazine in the Marine Environment

Hydrazine is dangerous for the environment and is toxic for aquatic organisms (Reference 5-20). Depending upon the species of fish, the toxicity levels and reproductive effects are likely to occur after a long exposure (24 to 96 hours) of hydrazine with concentrations on the order of 0.1 mg/litre and higher. Other marine organisms showed a wider range of tolerance to hydrazine than fish.

In the event of accidental release of the contents of a hydrazine tote container (, 1.51 m^3, 400 gallons) on the plant site roadway, the hydrazine will tend to drain toward the storm water drainage system or into the WWRB. The design of the storm water drainage system is site- and utility operator-specific. Provisions for temporarily blocking flow from this drainage system into the sea are also site and utility operator-specific.

The portion of the large spill that works its way through the storm water drainage system into the sea will be promptly diluted to well below the toxic concentrations listed in the World Health Organisation (WHO) report (Reference 5-20). In the event of such a large spill, there is a risk of the hydrazine deoxygenating a quantity of water because oxygen is a neutralising agent for hydrazine. Deoxygenation of sea water is hazardous to marine life. However, this risk is reduced because hydrazine will be in contact with the air above sea level with attendant vaporisation, decomposition, and oxidation limiting the amount that reaches the sea.

If some or the entire spill goes onto unpaved ground, research has shown that sand, soil, clay, and organic substances aid in the consumption of hydrazine. This reflects the results of two general processes; the chemical decomposition of hydrazine, and its reaction with soil components (Reference 5-21).

The theoretical quantity of deoxygenated seawater due to mixing with 1.51 m^3 (400 gallons) of hydrazine is 66,000 m^3 (17,400,000 gallons), assuming a seawater oxygen concentration of 8.1 mg/l (10 ppm) at 15°C (59°F). This assumes a thorough dispersion of hydrazine throughout the body of water and reaction only with dissolved oxygen. It does not take into account the competing reduction due to dissolved cation catalytic decomposition and reactions of hydrazine with dissolved and suspended solids. Experimental studies have shown a slow decrease of very dilute hydrazine in seawater. Hydrazine can persist in seawater for days. Wave action at the seacoast produces a rapid rate of hydrazine dilution. Therefore, it is unlikely that the entire quantity of a tote container's hydrazine or the turbine building's hydrazine storage will spill into the sea and the consequential deoxygenation effect will be much less than the theoretical maximum. The potential effects on aquatic organisms will be of a minor, limited spatial extent, for a short duration, and local to the release point.

5.4.5 Summary of Accidental Hydrazine Release Control Measures

In summary, the protective measures to prevent total release of hydrazine are:

- Primary containment within the turbine building
- Secondary containment within the turbine building
- Spill collection within the turbine building sumps
- Manual intervention can neutralise hydrazine spills in secondary containment, turbine building sumps, or the WWRB to prevent releases to the environment.
- WWRB
- Temporary spill barriers for outdoor spills

A realistic spill is limited to an insignificant portion of the total hydrazine and the protective measures will be effective to prevent release into the sea.

5.5 References

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Table 5.1-1

NUMBER OF POPULATION CENTRES WITHIN 2KM OF THE GENERIC SITE

Population	<1km	<2km	<10km	<20km	Closest to Site Boundary
>100000	–	0	0	1	8.5km
>20000	–	1	3	5	5.2km
>5000	–	1	1	6	3.0km
>1000	–	0	3	14	3.5km
≤1000	–	0	0	0	–
Farms/Properties	50	100	–	–	80m

Table 5.1-2

HABIT DATA OF LOCAL RESIDENT FAMILY EXPOSURE GROUP

Food Consumption Rates (kg/y)	Infant (1y)	Child (10y)	Adult
Green vegetables	15	35	80
Root vegetables	45	95	130
Fruit	35	50	75
Sheep meat	3	10	25
Sheep liver	2.75	5	10
Cow meat	10	30	45
Cow liver	2.75	5	10
Milk	320	240	240
Breathing Rates (m3/h)	0.22	0.64	0.92
Occupancy at Habitation (h/y)	8760	8760	8760
Fraction of Time Spent Indoors	0.9	0.8	0.5
Cloud Shielding Factor	0.2	0.2	0.2
Shielding Factor for Deposited Radionuclides	0.1	0.1	0.1

Table 5.1-3

HABIT DATA OF LOCAL FISHERMAN FAMILY EXPOSURE GROUP

Food Consumption Rates (kg/y)	Infant (1y)	Child (10y)	Adult
Fish	5	20	100
Crustaceans	0	5	20
Molluscs	0	5	20
Occupancy on the Beach (h/y)	30	300	2000

Table 5.1-4

REFERENCE ORGANISMS CONSIDERED IN THE VICINITY OF THE GENERIC SITE

Terrestrial	Marine
Amphibian (frog)	(Wading) bird (duck)
Bird (duck)	Benthic fish (flat fish)
Bird egg (duck egg)	Bivalve mollusc
Detritivorous invertebrate	Crustacean (crab)
Flying insect (bee)	Macroalgae (brown seaweed)
Gastropod	Mammal
Grasses and herbs (wild grass)	Pelagic fish
Lichen and bryophytes	Phytoplankton
Mammal (rat, deer)	Polychaete worm
Reptile	Reptile
Shrub	Sea anemones/true corals
Soil invertebrate (earthworm)	Vascular plant
Tree (pine tree)	Zooplankton

Table 5.1-5

METEOROLOGICAL DATA FOR THE GENERIC SITE

Parameter		Unit	Generic Site Value
Temperature	Max	C	37.7
	Min	C	-6.9
	Avg	C	11.8
Dew Point	Max	C	21.7
	Min	C	-12.3
	Avg	C	7.3
Humidity	Max	%	100.0
	Min	%	12.0
	Avg	%	76.2
Wind Speed	Max	km/h	93.3
	Min	km/h	8.7
	Avg	km/h	18
Wind Speed	Gust	km/h	127.8
Wind Direction	Avg	Deg	200.5
	Gust	Deg	241.9
	Maximum fraction of time in any one 30^{o} sector	–	0.25
Rainfall	Max	mm/y	998.5

Table 5.1-6

ATMOSPHERIC CONDITIONS

Pasquill Stability Category	Frequency of Occurrence (%)	Wind Speed at 10m Height (ms^{-1})
A	1	1
B	9	2
C	21	5
D	50	5
E	8	3
F	10	2
G	2	1

Table 5.1-7

HIGHEST GROUND ELEVATIONS AROUND THE GENERIC SITE

Distance from Site	Highest Ground Elevation
2km	30m
10km	358m

Table 5.1-8

MAIN LAND COVER WITHIN 5KM

Main Land Cover	Generic Site
Arable and grassland	+
Arable, some grassland	+
Dune vegetation	+
Grassland and arable, some woodland	+

+ (plus sign indicates the cover is included in the generic site.

Table 5.1-9

CHARACTERISTIC SEMI-NATURAL HABITATS WITHIN 5KM

Characteristic Semi-Natural Habitats	Generic Site
Lowland seasonally wet pastures and woodland	+
Sand dune vegetation ranging from pioneer dune vegetation through to low shrub	+
Wet brackish coastal flood meadows and grazing marsh	+

+ (plus sign) indicates the habitat is included in the generic site.

Table 5.1-10

NEAREST SENSITIVE SITES

Type	Distance (m)
Forestry Commission Woodland (England)	5460
Grassland Inventory (England)	700
Green Belt (England)	7000
Important Bird Areas (England)	250
Local Nature Reserves (England)	850
Lowland Grazing Marsh (England)	340
National Parks (England)	13000
Ramsar Sites (England)	290
RSPB Reserves (England)	1042
Sites of Special Scientific Interest (England)	180
Special Areas of Conservation (England)	330
Special Protection Areas (England)	300

Table 5.1-11

TIDAL RANGE

Tide	Generic Site
Highest Astronomical Tide	11.17m
Mean High Water Springs	10.06m
Mean High Water Neaps	7.75m
Mean Low Water Springs	1.72m
Mean Low Water Neaps	0.67m
Lowest Astronomical Tide	-0.06m

Table 5.1-12

BATHYMETRY

Distance from Site	Depth (Max/Min)	Depth*
1km	Max	5m
	Min	-15m
2km	Max	5m
	Min	-15m
10km	Max	15m
	Min	-15m

Note:
* Admiralty Chart Datum

Table 5.1-13

MARINE BIOLOGICAL FEATURES WITHIN 10 KM (6.2 MILES)

Parameter	Generic Site
Biosphere Reserves	0
Grey Seal Colonies	0
Harbour Porpoise	0
Minke Whales	0
Seabird Nesting Colonies	6
Sensitive Fish Areas	1
Waders and Wildfowl Areas	1
White Beaked Dolphin	0

Table 5.2-1

ANNUAL LIQUID DISCHARGES APPLIED FOR THE HUMAN DOSE ASSESSMENTS

Radionuclide	Representative liquid discharges	Calculated annual limit liquid discharges
	Bq/y	**Bq/y**
Tritium	3.5E+13	6.0E+13
Carbon-14	4.4E+09	7.0E+09
Iron-55	6.4E+08	1.0E+09
Cobalt-58	5.4E+08	9.0E+08
Cobalt-60	3.0E+08	5.0E+08
Nickel-63	6.9E+08	1.0E+09
Strontium-90	3.2E+05	5.0E+05
Caesium-137	3.0E+07	5.0E+07
Plutonium-241	1.1E+05	2.0E+05
Cerium-144[(1)]	1.1E+09	2.0E+09

Notes:
1. Surrogate radionuclide for ‘other radionuclides’ category

Table 5.2-2

DOSE PER UNIT INTAKE FACTORS FOR THE ASSESSMENT OF DOSES FROM LIQUID DISCHARGES

Radionuclide	Dose per unit intake by ingestion (Sv/Bq)		
	Adult	Child	Infant
Tritium	1.8E-11	2.3E-11	4.8E-11
Carbon-14	5.8E-10	8.0E-10	1.6E-09
Iron-55	3.3E-10	1.1E-09	2.4E-09
Cobalt-58	7.4E-10	1.7E-09	4.4E-09
Cobalt-60	3.4E-09	1.1E-08	2.7E-08
Nickel-63	1.5E-10	2.8E-10	8.4E-10
Strontium-90	2.8E-08	6.0E-08	7.3E-08
Caesium-137	1.3E-08	1.0E-08	1.2E-08
Plutonium-241	4.8E-09	5.1E-09	5.7E-09
Cerium-144	5.2E-09	1.1E-08	3.9E-08

Table 5.2-3

STAGE 1 DOSE PER UNIT RELEASE FACTORS FOR ANNUAL LIQUID DISCHARGES

Radionuclide	Dose per unit release factor (µSv/y per Bq/y)		
	External	**Ingestion**	**Total**
Tritium	0.0E+00	8.9E-16	8.9E-16
Carbon-14	1.6E-16	4.6E-10	4.6E-10
Iron-55	0.0E+00	3.0E-13	3.0E-13
Cobalt-58	5.4E-11	1.5E-11	6.9E-11
Cobalt-60	2.7E-09	7.5E-11	2.8E-09
Nickel-63	0.0E+00	3.6E-12	3.6E-12
Strontium-90	1.0E-15	6.1E-12	6.1E-12
Caesium-137	1.2E-10	2.8E-11	1.5E-10
Plutonium-241	2.4E-13	3.2E-11	3.2E-11
Cerium-144	1.4E-11	1.3E-12	1.5E-11

Table 5.2-4

STAGE 1 DOSE TO FISHERMAN FAMILY EXPOSURE GROUP FROM REPRESENTATIVE ANNUAL LIQUID DISCHARGES

Radionuclide	Annual dose from representative discharges (µSv/y)		
	External	Ingestion	Total
Tritium	0.0E+00	3.1E-02	3.1E-02
Carbon-14	7.0E-07	2.0E+00	2.0E+00
Iron-55	0.0E+00	1.9E-04	1.9E-04
Cobalt-58	2.9E-02	8.1E-03	3.7E-02
Cobalt-60	8.1E-01	2.3E-02	8.4E-01
Nickel-63	0.0E+00	2.5E-03	2.5E-03
Strontium-90	3.2E-10	2.0E-06	2.0E-06
Caesium-137	3.6E-03	8.4E-04	4.5E-03
Plutonium-241	2.6E-08	3.5E-06	3.5E-06
Cerium-144(1)	1.5E-02	1.4E-03	1.7E-02
Total	8.6E-01	2.1E+00	3.0E+00

Notes
1. Surrogate radionuclide for ‘other radionuclides’ category

Table 5.2-5

STAGE 1 DOSE TO FISHERMAN FAMILY EXPOSURE GROUP FROM CALCULATED ANNUAL LIMIT LIQUID DISCHARGES

Radionuclide	Annual dose from limit discharges (µSv/y)		
	External	Ingestion	Total
Tritium	0.0E+00	5.3E-02	5.3E-02
Carbon-14	1.1E-06	3.2E+00	3.2E+00
Iron-55	0.0E+00	3.0E-04	3.0E-04
Cobalt-58	4.9E-02	1.4E-02	6.2E-02
Cobalt-60	1.4E+00	3.8E-02	1.4E+00
Nickel-63	0.0E+00	3.6E-03	3.6E-03
Strontium-90	5.0E-10	3.1E-06	3.1E-06
Caesium-137	6.0E-03	1.4E-03	7.5E-03
Plutonium-241	4.8E-08	6.4E-06	6.4E-06
Cerium-144[1]	2.8E-02	2.6E-03	3.0E-02
Total	1.4E+00	3.3E+00	4.8E+00

Notes
1. Surrogate radionuclide for 'other radionuclides' category

Table 5.2-6

ANNUAL GASEOUS DISCHARGES APPLIED FOR THE HUMAN DOSE ASSESSMENTS

Radionuclide	Representative gaseous discharges Bq/y	Calculated annual limit gaseous discharges Bq/y
Tritium	1.9E+12	3.0E+12
Carbon-14	6.4E+11	1.0E+12
Argon-41	1.3E+12	2.0E+12
Cobalt-60	3.2E+06	5.0E+06
Krypton-85	4.1E+12	7.0E+12
Strontium-90	4.4E+05	7.0E+05
Iodine-131	2.1E+08	3.0E+08
Xenon-133	1.3E+12	2.0E+12
Caesium-137	1.3E+06	2.0E+06
Iodine-133(1)	3.9E+08	7.0E+08
Krypton-85(2)	2.7E+12	4.0E+12
Cobalt-58(3)	1.2E+07	2.0E+07

Notes:
1. Surrogate radionuclide for ‘other radioiodines’ category
2. Surrogate radionuclide for ‘other noble gasses’ category
3. Surrogate radionuclide for ‘other particulates’ category

Table 5.2-7

DOSE PER UNIT INTAKE FACTORS FOR THE ASSESSMENT OF DOSES FROM GASEOUS DISCHARGES

Radionuclide	Dose per unit intake by ingestion (Sv/Bq)			Dose per unit intake by inhalation (Sv/Bq)		
	Adult	**Child**	**Infant**	**Adult**	**Child**	**Infant**
Tritium	1.8E-11	2.3E-11	4.8E-11	1.8E-11	2.3E-11	4.8E-11
Carbon-14	5.8E-10	8.0E-10	1.6E-09	2.0E-09	2.8E-09	6.6E-09
Argon-41	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Cobalt-60	3.4E-09	1.1E-08	2.7E-08	1.0E-08	1.5E-08	3.4E-08
Krypton-85	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Strontium-90	2.8E-08	6.0E-08	7.3E-08	3.6E-08	5.1E-08	1.1E-07
Iodine-131	2.2E-08	5.2E-08	1.8E-07	7.4E-09	1.9E-08	7.2E-08
Xenon-133	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Caesium-137	1.3E-08	1.0E-08	1.2E-08	4.6E-09	3.7E-09	5.4E-09
Iodine-133	4.3E-09	1.0E-08	4.4E-08	1.5E-09	3.8E-09	1.8E-08
Krypton-85	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Cobalt-58	7.4E-10	1.7E-09	4.4E-09	1.6E-09	2.4E-09	6.5E-09

Table 5.2-8

STAGE 1 DOSE PER UNIT RELEASE FACTORS FOR ANNUAL GASEOUS DISCHARGES

Radionuclide	Dose per unit release factor (µSv/y per Bq/y)			
	Ingestion	**Total external**	**Inhalation**	**Total**
Tritium	2.7E-13	0.0E+00	6.9E-13	9.6E-13
Carbon-14	3.3E-11	6.4E-17	3.5E-11	6.8E-11
Argon-41	0.0E+00	3.2E-12	0.0E+00	3.2E-12
Cobalt-60	5.3E-11	1.1E-08	2.2E-10	1.2E-08
Krypton-85	0.0E+00	1.3E-14	0.0E+00	1.3E-14
Strontium-90	6.4E-10	3.2E-15	8.0E-10	1.4E-09
Iodine-131	4.1E-09	3.8E-11	3.9E-10	4.5E-09
Xenon-133	0.0E+00	7.0E-14	0.0E+00	7.0E-14
Caesium-137	3.8E-10	6.5E-09	1.0E-10	7.0E-09
Iodine-133	7.2E-11	7.6E-12	9.7E-11	1.8E-10
Krypton-85	0.0E+00	1.3E-14	0.0E+00	1.3E-14
Cobalt-58	4.4E-12	2.7E-10	3.6E-11	3.1E-10

Table 5.2-9

STAGE 1 DOSE TO LOCAL RESIDENT FAMILY EXPOSURE GROUP FROM REPRESENTATIVE ANNUAL GASEOUS DISCHARGES

Radionuclide	Annual dose from representative discharges (µSv/y)			
	Ingestion	Total external	Inhalation	Total
Tritium	5.1E-01	0.0E+00	1.3E+00	1.8E+00
Carbon-14	2.1E+01	4.1E-05	2.2E+01	4.4E+01
Argon-41	0.0E+00	4.2E+00	0.0E+00	4.2E+00
Cobalt-60	1.7E-04	3.5E-02	7.0E-04	3.8E-02
Krypton-85	0.0E+00	5.3E-02	0.0E+00	5.3E-02
Strontium-90	2.8E-04	1.4E-09	3.5E-04	6.2E-04
Iodine-131	8.6E-01	8.0E-03	8.2E-02	9.5E-01
Xenon-133	0.0E+00	9.1E-02	0.0E+00	9.1E-02
Caesium-137	4.9E-04	8.5E-03	1.3E-04	9.1E-03
Iodine-133(1)	2.8E-02	3.0E-03	3.8E-02	7.0E-02
Krypton-85(2)	0.0E+00	3.5E-02	0.0E+00	3.5E-02
Cobalt-58(3)	5.3E-05	3.2E-03	4.3E-04	3.7E-03
Total	2.3E+01	4.4E+00	2.4E+01	5.1E+01

Notes:
1. Surrogate radionuclide for ‘other radioiodines’ category
2. Surrogate radionuclide for ‘other noble gasses’ category
3. Surrogate radionuclide for ‘other particulates’ category

Table 5.2-10

STAGE 1 DOSE TO LOCAL RESIDENT FAMILY EXPOSURE GROUP FROM CALCULATED ANNUAL LIMIT GASEOUS DISCHARGES

Radionuclide	Annual dose from limit discharges (µSv/y)			
	Ingestion	Total external	Inhalation	Total
Tritium	8.1E-01	0.0E+00	2.1E+00	2.9E+00
Carbon-14	3.3E+01	6.4E-05	3.5E+01	6.8E+01
Argon-41	0.0E+00	6.4E+00	0.0E+00	6.4E+00
Cobalt-60	2.7E-04	5.5E-02	1.1E-03	6.0E-02
Krypton-85	0.0E+00	9.1E-02	0.0E+00	9.1E-02
Strontium-90	4.5E-04	2.2E-09	5.6E-04	9.8E-04
Iodine-131	1.2E+00	1.1E-02	1.2E-01	1.4E+00
Xenon-133	0.0E+00	1.4E-01	0.0E+00	1.4E-01
Caesium-137	7.6E-04	1.3E-02	2.0E-04	1.4E-02
Iodine-133(1)	5.0E-02	5.3E-03	6.8E-02	1.3E-01
Krypton-85(2)	0.0E+00	5.2E-02	0.0E+00	5.2E-02
Cobalt-58(3)	8.8E-05	5.4E-03	7.2E-04	6.2E-03
Total	3.5E+01	6.8E+00	3.7E+01	7.9E+01

Notes:
1. Surrogate radionuclide for 'other radioiodines' category
2. Surrogate radionuclide for 'other noble gasses' category
3. Surrogate radionuclide for 'other particulates' category

Table 5.2-11

STAGE 2 DOSE PER UNIT RELEASE FACTORS FOR ANNUAL LIQUID DISCHARGES

Radionuclide	Dose per unit release factor (µSv/y per Bq/y)		
	External	**Ingestion**	**Total**
Tritium	0.0E+00	6.8E-16	6.8E-16
Carbon-14	1.2E-16	3.5E-10	3.5E-10
Iron-55	0.0E+00	2.3E-13	2.3E-13
Cobalt-58	4.2E-11	1.2E-11	5.3E-11
Cobalt-60	2.1E-09	5.8E-11	2.1E-09
Nickel-63	0.0E+00	2.8E-12	2.8E-12
Strontium-90	7.7E-16	4.7E-12	4.7E-12
Caesium-137	9.2E-11	2.2E-11	1.1E-10
Plutonium-241	1.8E-13	2.5E-11	2.5E-11
Cerium-144	1.1E-11	1.0E-12	1.2E-11

Table 5.2-12

STAGE 2 DOSE TO FISHERMAN FAMILY EXPOSURE GROUP FROM REPRESENTATIVE ANNUAL LIQUID DISCHARGES

Radionuclide	Annual dose from representative discharges (µSv/y)		
	External	Ingestion	Total
Tritium	0.0E+00	2.4E-02	2.4E-02
Carbon-14	5.4E-07	1.6E+00	1.6E+00
Iron-55	0.0E+00	1.5E-04	1.5E-04
Cobalt-58	2.2E-02	6.2E-03	2.9E-02
Cobalt-60	6.2E-01	1.7E-02	6.4E-01
Nickel-63	0.0E+00	1.9E-03	1.9E-03
Strontium-90	2.5E-10	1.5E-06	1.5E-06
Caesium-137	2.8E-03	6.5E-04	3.4E-03
Plutonium-241	2.0E-08	2.7E-06	2.7E-06
Cerium-144(1)	1.2E-02	1.1E-03	1.3E-02
Total	6.6E-01	1.6E+00	2.3E+00

Notes
1. Surrogate radionuclide for ‘other radionuclides’ category

Table 5.2-13

STAGE 2 DOSE TO FISHERMAN FAMILY EXPOSURE GROUP FROM CALCULATED ANNUAL LIMIT LIQUID DISCHARGES

Radionuclide	Annual dose from limit discharges (µSv/y)		
	External	Ingestion	Total
Tritium	0.0E+00	4.1E-02	4.1E-02
Carbon-14	8.6E-07	2.5E+00	2.5E+00
Iron-55	0.0E+00	2.3E-04	2.3E-04
Cobalt-58	3.7E-02	1.0E-02	4.8E-02
Cobalt-60	1.0E+00	2.9E-02	1.1E+00
Nickel-63	0.0E+00	2.8E-03	2.8E-03
Strontium-90	3.8E-10	2.3E-06	2.3E-06
Caesium-137	4.6E-03	1.1E-03	5.7E-03
Plutonium-241	3.7E-08	4.9E-06	5.0E-06
Cerium-144(1)	2.2E-02	2.0E-03	2.4E-02
Total	1.1E+00	2.6E+00	3.7E+00

Notes
1. Surrogate radionuclide for ‘other radionuclides’ category

Table 5.2-14

PARAMETERS APPLIED FOR PLUME RISE AND BUILDING WAKE MODELLING

Parameter	Value
Height of reactor building	70 m
Equivalent footprint of reactor building (rectangular shape assumed for modelling purposes)	43 m by 43 m
Position of release point relative to reactor building	On the roof, 7 m from the edge
Physical stack height	75 m
Equivalent diameter of the plant vent (circular shape assumed for modelling purposes)	2.44 m
Release temperature	Ambient
Release velocity	8.15 m/s
Volumetric flow rate	38.13 m^3/s

Table 5.2-15

STAGE 2 DOSE PER UNIT RELEASE FACTORS FOR ANNUAL GASEOUS DISCHARGES

Radionuclide	Dose per unit release factor (µSv/y per Bq/y)			
	Ingestion	Total external	Inhalation	Total
Tritium	4.1E-14	0.0E+00	4.8E-15	4.5E-14
Carbon-14	5.0E-12	4.5E-19	2.5E-13	5.2E-12
Argon-41	0.0E+00	2.2E-14	0.0E+00	2.2E-14
Cobalt-60	8.0E-12	7.7E-11	1.5E-12	8.6E-11
Krypton-85	0.0E+00	9.1E-17	0.0E+00	9.1E-17
Strontium-90	9.6E-11	2.2E-17	5.6E-12	1.0E-10
Iodine-131	6.2E-10	2.7E-13	2.7E-12	6.2E-10
Xenon-133	0.0E+00	4.9E-16	0.0E+00	4.9E-16
Caesium-137	5.7E-11	4.6E-11	7.0E-13	1.0E-10
Iodine-133	1.1E-11	5.3E-14	6.8E-13	1.2E-11
Krypton-85	0.0E+00	9.1E-17	0.0E+00	9.1E-17
Cobalt-58	6.6E-13	1.9E-12	2.5E-13	2.8E-12

Table 5.2-16

STAGE 2 DOSE TO LOCAL RESIDENT FAMILY EXPOSURE GROUP FROM REPRESENTATIVE ANNUAL GASEOUS DISCHARGES

Radionuclide	Annual dose from representative discharges (µSv/y)			
	Ingestion	Total external	Inhalation	Total
Tritium	7.7E-02	0.0E+00	9.2E-03	8.6E-02
Carbon-14	3.2E+00	2.9E-07	1.6E-01	3.3E+00
Argon-41	0.0E+00	2.9E-02	0.0E+00	2.9E-02
Cobalt-60	2.5E-05	2.5E-04	4.9E-06	2.8E-04
Krypton-85	0.0E+00	3.7E-04	0.0E+00	3.7E-04
Strontium-90	4.2E-05	9.9E-12	2.5E-06	4.5E-05
Iodine-131	1.3E-01	5.6E-05	5.7E-04	1.3E-01
Xenon-133	0.0E+00	6.4E-04	0.0E+00	6.4E-04
Caesium-137	7.4E-05	5.9E-05	9.1E-07	1.3E-04
Iodine-133(1)	4.2E-03	2.1E-05	2.6E-04	4.5E-03
Krypton-85(2)	0.0E+00	2.5E-04	0.0E+00	2.5E-04
Cobalt-58(3)	7.9E-06	2.3E-05	3.0E-06	3.4E-05
Total	3.4E+00	3.1E-02	1.7E-01	3.6E+00

Notes:
1. Surrogate radionuclide for ‘other radioiodines’ category
2. Surrogate radionuclide for ‘other noble gasses’ category
3. Surrogate radionuclide for ‘other particulates’ category

Table 5.2-17

STAGE 2 DOSE TO LOCAL RESIDENT FAMILY EXPOSURE GROUP FROM CALCULATED ANNUAL LIMIT GASEOUS DISCHARGES

Radionuclide	Annual dose from limit discharges (µSv/y)			
	Ingestion	Total external	Inhalation	Total
Tritium	1.2E-01	0.0E+00	1.4E-02	1.4E-01
Carbon-14	5.0E+00	4.5E-07	2.5E-01	5.2E+00
Argon-41	0.0E+00	4.5E-02	0.0E+00	4.5E-02
Cobalt-60	4.0E-05	3.9E-04	7.7E-06	4.3E-04
Krypton-85	0.0E+00	6.4E-04	0.0E+00	6.4E-04
Strontium-90	6.7E-05	1.6E-11	3.9E-06	7.1E-05
Iodine-131	1.8E-01	8.0E-05	8.2E-04	1.9E-01
Xenon-133	0.0E+00	9.8E-04	0.0E+00	9.8E-04
Caesium-137	1.1E-04	9.1E-05	1.4E-06	2.1E-04
Iodine-133(1)	7.6E-03	3.7E-05	4.8E-04	8.1E-03
Krypton-85(2)	0.0E+00	3.6E-04	0.0E+00	3.6E-04
Cobalt-58(3)	1.3E-05	3.8E-05	5.0E-06	5.6E-05
Total	5.3E+00	4.7E-02	2.6E-01	5.6E+00

Notes:
1. Surrogate radionuclide for ‘other radioiodines’ category
2. Surrogate radionuclide for ‘other noble gasses’ category
3. Surrogate radionuclide for ‘other particulates’ category

Table 5.2-18

DISCHARGES APPLIED FOR THE SHORT-TERM DOSE ASSESSMENT

Radionuclide	Maximum monthly discharge (Bq)	Discharge rate assuming 30 minute release (Bq/s)
Tritium	2.4E+11	1.3E+08
Carbon-14	8.3E+10	4.6E+07
Argon-41	1.7E+11	9.5E+07
Cobalt-60	2.7E+05	1.5E+02
Krypton-85	1.3E+12	7.1E+08
Strontium-90	3.7E+04	2.1E+01
Iodine-131	1.7E+07	9.6E+03
Xenon-133	1.9E+11	1.0E+08
Caesium-137	1.1E+05	6.2E+01
Iodine-133(1)	3.2E+07	1.8E+04
Krypton-85(2)	5.8E+11	3.2E+08
Cobalt-58(3)	1.0E+06	5.7E+02

Notes:
1. Surrogate radionuclide for ‘other radioiodines’ category
2. Surrogate radionuclide for ‘other noble gasses’ category
3. Surrogate radionuclide for ‘other particulates’ category

Table 5.2-19

PARAMETERS APPLIED FOR SHORT-TERM RELEASE MODELLING

Parameter	Value
Wind direction	Single wind direction, from source to receptor
Wind speed (at a height of 10 m)	3 m/s
Rainfall rate	0.1 mm/h (continuous)
Atmospheric stability condition	Neutral
Boundary layer height	800 m
Surface roughness length	0.3 m
Wet deposition rate for noble gasses, tritium and carbon-14	0 s^{-1}
Wet deposition rate for particulates	0.0001 s^{-1}
Dry deposition rate for noble gasses, tritium and carbon-14	0 m/s
Dry deposition rate for iodine isotopes	0.01 m/s
Dry deposition rate for other particulates	0.001 m/s

Table 5.2-20

ACTIVITY CONCENTRATION IN AIR AND DEPOSITED TO THE GROUND PER UNIT RELEASED FOR A SHORT-TERM RELEASE

Radionuclide	Integrated air: Bq s m^{-3} per Bq released Deposited: Bq m^{-2} per Bq released			
	Integrated air at 100 m	**Deposited at 100 m**	**Integrated air at 500 m**	**Deposited at 500 m**
Tritium	2.0E-06	0.0E+00	7.9E-06	0.0E+00
Carbon-14	2.0E-06	0.0E+00	7.9E-06	0.0E+00
Argon-41	2.0E-06	0.0E+00	7.9E-06	0.0E+00
Cobalt-60	1.9E-06	5.5E-07	7.9E-06	1.2E-07
Krypton-85	2.0E-06	0.0E+00	7.9E-06	0.0E+00
Strontium-90	1.9E-06	5.5E-07	7.9E-06	1.2E-07
Iodine-131	1.8E-06	5.7E-07	7.2E-06	1.9E-07
Xenon-133	2.0E-06	0.0E+00	7.9E-06	0.0E+00
Caesium-137	1.9E-06	5.5E-07	7.9E-06	1.2E-07
Iodine-133	1.8E-06	5.7E-07	7.2E-06	1.9E-07
Krypton-85	2.0E-06	0.0E+00	7.9E-06	0.0E+00
Cobalt-58	1.9E-06	5.5E-07	7.9E-06	1.2E-07

Table 5.2-21

CLOUD GAMMA DOSE FACTORS FOR A SHORT-TERM RELEASE

Radionuclide	**External dose (Sv per Bq released)**
Tritium	0.0E+00
Carbon-14	0.0E+00
Argon-41	1.8E-18
Cobalt-60	3.4E-18
Krypton-85	2.0E-21
Strontium-90	2.4E-30
Iodine-131	3.5E-19
Xenon-133	6.7E-20
Caesium-137	4.0E-19
Iodine-133	5.5E-19
Krypton-85	2.0E-21
Cobalt-58	7.2E-19

Table 5.2-22

BREATHING RATES FOR SHORT-TERM RELEASE ASSESSMENT

	Breathing rate (m^3/h)		
	Adult	**Child**	**Infant**
Inhalation rate	3.0	0.87	0.31

Table 5.2-23

ACTIVITY CONCENTRATIONS IN FOOD ONE YEAR AFTER A SHORT-TERM RELEASE

Radionuclide	H-3 and C-14: Bq/kg per Bq m^{-3} s air concentration Other radionuclides: Bq/kg per Bq m^{-2} instantaneous deposition							
	Green vegetables	**Root vegetables**	**Sheep meat**	**Sheep offal**	**Cow meat**	**Cow offal**	**Cow milk**	**Fruit**
Tritium	3.2E-06	3.2E-06	2.8E-06	2.8E-06	2.8E-06	2.8E-06	3.6E-06	3.2E-06
Carbon-14	1.7E-05	1.7E-05	2.5E-05	2.5E-05	2.5E-05	2.5E-05	8.5E-06	1.7E-05
Cobalt-60	3.5E-03	1.2E-05	9.7E-05	9.7E-03	6.1E-05	6.1E-03	9.4E-05	7.0E-04
Strontium-90	3.6E-03	3.3E-05	3.6E-04	3.6E-04	2.9E-04	2.9E-04	1.4E-03	7.3E-04
Iodine-131	1.3E-03	2.7E-04	1.0E-03	1.0E-03	7.8E-04	7.8E-04	1.8E-03	9.3E-04
Caesium-137	4.2E-03	3.9E-03	4.6E-02	4.6E-02	2.5E-02	2.5E-02	4.9E-03	1.1E-02
Iodine-133	2.0E-04	1.5E-06	2.2E-05	2.2E-05	3.5E-05	3.5E-05	1.2E-04	1.9E-05
Cobalt-58	2.9E-03	6.4E-06	3.6E-05	3.6E-03	2.2E-05	2.2E-03	8.1E-05	4.7E-04

Table 5.2-24

INGESTION RATES FOR SHORT-TERM RELEASE ASSESSMENT

	Ingestion rate (kg/y)		
	Adult	**Child**	**Infant**
Green vegetables	30	10	5
Root vegetables(1)	130	95	45
Sheep meat	3	1.5	0.6
Sheep offal	1	0.5	0.2
Cow meat	15	10	3
Cow offal	1	0.5	0.2
Cow milk(1)	240	240	320
Fruit	15	15	7.5

Notes:
1. High rate consumption assumed

Table 5.2-25

ANNUAL EXTERNAL GAMMA DOSE PER UNIT SURFACE DEPOSITION

Radionuclide	External dose (Sv/y per Bq/m^2 initial deposit)
Cobalt-60	6.80E-08
Strontium-90	2.77E-08
Iodine-131	3.81E-10
Caesium-137	8.41E-07
Iodine-133	2.98E-09
Cobalt-58	3.81E-07

Table 5.2-26

OCCUPANCY AND SHIELDING FACTORS FOR SHORT-TERM DOSE ASSESSMENT

Parameter	Value
Indoor occupancy assumed for cloud gamma pathway	0
Indoor occupancy assumed for deposited gamma pathway (all ages)	0.9
Deposited gamma indoor shielding factor	0.1

Table 5.2-27

DOSE TO ADULTS FROM SHORT-TERM GASEOUS DISCHARGE

Radionuclide	Dose from short-term discharges to adults (μSv)			
	Ingestion	Total external	Inhalation	Total
Tritium	5.1E-02	0.0E+00	7.1E-03	5.8E-02
Carbon-14	2.1E+00	0.0E+00	2.7E-01	2.4E+00
Argon-41	0.0E+00	3.1E-01	0.0E+00	3.1E-01
Cobalt-60	1.8E-05	1.9E-03	4.3E-06	1.9E-03
Krypton-85	0.0E+00	2.6E-03	0.0E+00	2.6E-03
Strontium-90	5.9E-05	1.1E-04	2.1E-06	1.7E-04
Iodine-131	3.9E-02	7.2E-04	2.0E-04	4.0E-02
Xenon-133	0.0E+00	1.3E-02	0.0E+00	1.3E-02
Caesium-137	4.6E-04	9.8E-03	8.2E-07	1.0E-02
Iodine-133(1)	9.4E-04	1.0E-02	7.4E-05	1.1E-02
Krypton-85(2)	0.0E+00	1.2E-03	0.0E+00	1.2E-03
Cobalt-58(3)	1.1E-05	4.1E-02	2.6E-06	4.1E-02
Total	2.2E+00	3.9E-01	2.8E-01	2.8E+00

Notes:
1. Surrogate radionuclide for ‘other radioiodines’ category
2. Surrogate radionuclide for ‘other noble gasses’ category
3. Surrogate radionuclide for ‘other particulates’ category

Table 5.2-28

DOSE TO CHILDREN FROM SHORT-TERM GASEOUS DISCHARGE

Radionuclide	Dose from short-term discharges to children (µSv)			
	Ingestion	Total external	Inhalation	Total
Tritium	5.6E-02	0.0E+00	2.6E-03	5.9E-02
Carbon-14	2.3E+00	0.0E+00	1.1E-01	2.4E+00
Argon-41	0.0E+00	3.1E-01	0.0E+00	3.1E-01
Cobalt-60	2.8E-05	1.9E-03	1.9E-06	1.9E-03
Krypton-85	0.0E+00	2.6E-03	0.0E+00	2.6E-03
Strontium-90	1.1E-04	1.1E-04	8.8E-07	2.1E-04
Iodine-131	8.5E-02	7.2E-04	1.5E-04	8.6E-02
Xenon-133	0.0E+00	1.3E-02	0.0E+00	1.3E-02
Caesium-137	2.9E-04	9.8E-03	1.9E-07	1.0E-02
Iodine-133(1)	1.9E-03	1.0E-02	5.5E-05	1.2E-02
Krypton-85(2)	0.0E+00	1.2E-03	0.0E+00	1.2E-03
Cobalt-58(3)	1.3E-05	4.1E-02	1.1E-06	4.1E-02
Total	2.4E+00	3.9E-01	1.1E-01	2.9E+00

Notes:
1. Surrogate radionuclide for ‘other radioiodines’ category
2. Surrogate radionuclide for ‘other noble gasses’ category
3. Surrogate radionuclide for ‘other particulates’ category

Table 5.2-29

DOSE TO INFANTS FROM SHORT-TERM GASEOUS DISCHARGE

Radionuclide	Dose from short-term discharges to infants (μSv)			
	Ingestion	**Total external**	**Inhalation**	**Total**
Tritium	1.2E-01	0.0E+00	2.0E-03	1.3E-01
Carbon-14	4.0E+00	0.0E+00	9.1E-02	4.1E+00
Argon-41	0.0E+00	3.1E-01	0.0E+00	3.1E-01
Cobalt-60	5.1E-05	1.9E-03	1.5E-06	2.0E-03
Krypton-85	0.0E+00	2.6E-03	0.0E+00	2.6E-03
Strontium-90	1.6E-04	1.1E-04	6.8E-07	2.6E-04
Iodine-131	3.6E-01	7.2E-04	2.0E-04	3.6E-01
Xenon-133	0.0E+00	1.3E-02	0.0E+00	1.3E-02
Caesium-137	3.2E-04	9.8E-03	1.0E-07	1.0E-02
Iodine-133(1)	1.1E-02	1.0E-02	9.2E-05	2.1E-02
Krypton-85(2)	0.0E+00	1.2E-03	0.0E+00	1.2E-03
Cobalt-58(3)	2.5E-05	4.1E-02	1.1E-06	4.1E-02
Total	4.5E+00	3.9E-01	9.4E-02	4.9E+00

Notes:
1. Surrogate radionuclide for ‘other radioiodines’ category
2. Surrogate radionuclide for ‘other noble gasses’ category
3. Surrogate radionuclide for ‘other particulates’ category

Table 5.2-30

COLLECTIVE DOSE STATISTICS FOR AP1000 NPP REPRESENTATIVE DISCHARGES TO ATMOSPHERE

	UK (manSv)			Europe (manSv)			World (manSv)		
Radionuclide	**min**	**average**	**max**	**min**	**average**	**max**	**min**	**average**	**max**
Tritium	1.5E-03	2.0E-03	2.4E-03	4.8E-03	5.2E-03	5.7E-03	5.3E-03	5.8E-03	6.3E-03
Carbon-14	1.8E-01	2.1E-01	2.3E-01	1.2E+00	1.3E+00	1.5E+00	8.5E+00	8.6E+00	8.8E+00
Argon-41	3.8E-05	9.3E-05	1.9E-04	4.3E-05	9.0E-05	1.9E-04	4.3E-05	9.0E-05	1.9E-04
Cobalt-60	3.7E-06	5.2E-06	7.0E-06	6.9E-06	7.8E-06	8.7E-06	6.9E-06	7.8E-06	8.7E-06
Krypton-85	2.4E-05	2.8E-05	3.1E-05	9.8E-05	1.0E-04	1.1E-04	1.1E-03	1.1E-03	1.1E-03
Strontium-90	9.7E-07	1.4E-06	1.6E-06	4.7E-06	6.1E-06	7.9E-06	4.7E-06	6.1E-06	7.9E-06
Iodine-131	8.5E-05	1.9E-04	3.3E-04	1.1E-04	1.4E-04	1.9E-04	1.1E-04	1.4E-04	1.9E-04
Xenon-133	1.3E-05	1.6E-05	1.9E-05	3.1E-05	3.4E-05	3.9E-05	3.1E-05	3.4E-05	3.9E-05
Caesium-137	2.7E-06	3.7E-06	4.3E-06	1.1E-05	1.3E-05	1.6E-05	1.1E-05	1.3E-05	1.6E-05
Other radioiodines(1)	4.6E-05	8.3E-05	1.4E-04	6.6E-05	9.1E-05	1.4E-04	6.6E-05	9.1E-05	1.4E-04
Other noble gasses(2)	1.2E-05	1.4E-05	1.7E-05	3.3E-05	3.7E-05	4.0E-05	3.3E-05	3.7E-05	4.0E-05
Other particulates(3)	4.8E-07	6.4E-07	8.4E-07	8.7E-07	9.9E-07	1.1E-06	8.7E-07	9.9E-07	1.1E-06
Total	1.8E-01	2.1E-01	2.3E-01	1.2E+00	1.3E+00	1.5E+00	8.5E+00	8.6E+00	8.8E+00

Notes:
1. All radioiodines apart from iodine-131, assessed as iodine-133
2. All noble gasses apart from argon-41, krypton-85 and xenon-133, assessed as krypton-85
3. All particulates apart from cobalt-60, strontium-90, and caesium-137, assessed as cobalt-58

Table 5.2-31

COLLECTIVE DOSE STATISTICS FOR AP1000 NPP CALCULATED ANNUAL LIMIT DISCHARGES TO ATMOSPHERE

Radionuclide	UK (manSv)			Europe (manSv)			World (manSv)		
	min	average	max	min	average	max	min	average	max
Tritium	2.4E-03	3.2E-03	3.9E-03	7.6E-03	8.4E-03	9.2E-03	8.6E-03	9.4E-03	1.0E-02
Carbon-14	2.8E-01	3.2E-01	3.5E-01	1.9E+00	2.1E+00	2.3E+00	1.3E+01	1.4E+01	1.4E+01
Argon-41	5.7E-05	1.4E-04	2.9E-04	6.5E-05	1.4E-04	2.8E-04	6.5E-05	1.4E-04	2.8E-04
Cobalt-60	5.8E-06	8.1E-06	1.1E-05	1.1E-05	1.2E-05	1.4E-05	1.1E-05	1.2E-05	1.4E-05
Krypton-85	4.1E-05	4.7E-05	5.4E-05	1.7E-04	1.8E-04	1.9E-04	1.9E-03	1.9E-03	1.9E-03
Strontium-90	1.5E-06	2.2E-06	2.6E-06	7.4E-06	9.6E-06	1.2E-05	7.4E-06	9.6E-06	1.2E-05
Iodine-131	1.2E-04	2.8E-04	4.8E-04	1.6E-04	2.0E-04	2.8E-04	1.6E-04	2.0E-04	2.8E-04
Xenon-133	2.0E-05	2.5E-05	2.9E-05	4.6E-05	5.1E-05	5.8E-05	4.6E-05	5.1E-05	5.8E-05
Caesium-137	4.1E-06	5.6E-06	6.5E-06	1.6E-05	1.9E-05	2.4E-05	1.6E-05	1.9E-05	2.4E-05
Other radioiodines(1)	8.3E-05	1.5E-04	2.6E-04	1.2E-04	1.6E-04	2.6E-04	1.2E-04	1.6E-04	2.6E-04
Other noble gasses(2)	1.7E-05	2.1E-05	2.5E-05	4.9E-05	5.4E-05	6.0E-05	4.9E-05	5.4E-05	6.0E-05
Other particulates(3)	7.9E-07	1.1E-06	1.4E-06	1.4E-06	1.6E-06	1.8E-06	1.4E-06	1.6E-06	1.8E-06
Total	2.9E-01	3.3E-01	3.6E-01	1.9E+00	2.1E+00	2.3E+00	1.3E+01	1.4E+01	1.4E+01

Notes:
1. All radioiodines apart from iodine-131, assessed as iodine-133
2. All noble gasses apart from argon-41, krypton-85, and xenon-133, assessed as krypton-85
3. All particulates apart from cobalt-60, strontium-90, and caesium-137, assessed as cobalt-58

Table 5.2-32

COLLECTIVE DOSE STATISTICS FOR AP1000 NPP REPRESENTATIVE DISCHARGES TO SEA

Radionuclide	UK (manSv)			Europe (manSv)			World (manSv)		
	min	average	max	min	average	max	min	average	max
Tritium	8.3E-06	1.2E-05	1.9E-05	4.6E-05	7.0E-05	9.4E-05	1.2E-03	1.2E-03	1.3E-03
Carbon-14	2.7E-04	5.9E-04	1.0E-03	1.3E-03	3.1E-03	4.4E-03	2.7E-02	3.0E-02	3.2E-02
Iron-55	9.0E-07	3.2E-06	7.8E-06	1.7E-06	1.8E-05	3.3E-05	1.7E-06	2.0E-05	3.7E-05
Cobalt-58	1.2E-07	4.5E-07	1.2E-06	2.5E-07	2.2E-06	3.5E-06	2.8E-07	2.7E-06	4.6E-06
Cobalt-60	6.3E-07	1.8E-06	4.2E-06	1.3E-06	7.8E-06	1.3E-05	1.4E-06	9.6E-06	1.6E-05
Nickel-63	1.1E-07	2.0E-07	3.3E-07	3.1E-07	1.0E-06	2.1E-06	3.9E-07	1.5E-06	3.0E-06
Strontium-90	2.9E-11	1.3E-10	2.6E-10	9.4E-11	5.8E-10	9.5E-10	1.3E-10	9.6E-10	1.5E-09
Caesium-137	5.3E-08	2.0E-07	3.6E-07	1.8E-07	8.4E-07	1.3E-06	2.4E-07	1.5E-06	2.4E-06
Plutonium-241	2.0E-10	8.4E-10	1.6E-09	1.0E-09	4.7E-09	1.1E-08	1.1E-09	5.2E-09	1.2E-08
Other radionuclides(1)	5.8E-08	2.2E-07	6.3E-07	1.1E-07	1.1E-06	1.8E-06	1.2E-07	1.2E-06	2.0E-06
Total	2.8E-04	6.1E-04	1.0E-03	1.4E-03	3.2E-03	4.6E-03	2.8E-02	3.1E-02	3.3E-02

Notes:
1. All other radionuclides apart from those already assessed, assessed as cerium-144

Table 5.2-33

COLLECTIVE DOSE STATISTICS FOR AP1000 NPP CALCULATED ANNUAL LIMIT DISCHARGES TO SEA

Radionuclide	UK (manSv)			Europe (manSv)			World (manSv)		
	min	average	max	min	average	max	min	average	max
Tritium	1.4E-05	2.1E-05	3.2E-05	7.9E-05	1.2E-04	1.6E-04	2.0E-03	2.1E-03	2.2E-03
Carbon-14	4.3E-04	9.3E-04	1.6E-03	2.1E-03	4.9E-03	7.0E-03	4.3E-02	4.8E-02	5.1E-02
Iron-55	1.4E-06	5.0E-06	1.2E-05	2.6E-06	2.8E-05	5.2E-05	2.7E-06	3.1E-05	5.7E-05
Cobalt-58	1.9E-07	7.4E-07	1.9E-06	4.1E-07	3.7E-06	5.8E-06	4.6E-07	4.5E-06	7.6E-06
Cobalt-60	1.0E-06	2.9E-06	7.1E-06	2.1E-06	1.3E-05	2.1E-05	2.4E-06	1.6E-05	2.7E-05
Nickel-63	1.6E-07	2.8E-07	4.8E-07	4.5E-07	1.5E-06	3.1E-06	5.6E-07	2.1E-06	4.3E-06
Strontium-90	4.5E-11	2.0E-10	4.0E-10	1.5E-10	9.0E-10	1.5E-09	2.0E-10	1.5E-09	2.3E-09
Caesium-137	8.8E-08	3.3E-07	6.1E-07	2.9E-07	1.4E-06	2.2E-06	4.0E-07	2.5E-06	3.9E-06
Plutonium-241	3.7E-10	1.6E-09	2.9E-09	1.9E-09	8.7E-09	1.9E-08	2.0E-09	9.7E-09	2.2E-08
Other radionuclides(1)	1.1E-07	4.1E-07	1.2E-06	2.1E-07	2.0E-06	3.3E-06	2.2E-07	2.2E-06	3.7E-06
Total	4.5E-04	9.6E-04	1.7E-03	2.2E-03	5.0E-03	7.2E-03	4.5E-02	5.0E-02	5.3E-02

Notes:
All other radionuclides apart from those already assessed, assessed as cerium-144

Table 5.2-34

ACTIVITY IN SOIL IN 60TH YEAR OF ATMOSPHERIC DISCHARGES

Radionuclide	Activity Concentration at 500 m (Bq/kg)
Chromium-51	8.1E-16
Manganese-54	7.0E-15
Cobalt-58	8.6E-14
Cobalt-60	7.1E-13
Strontium-89	7.6E-15
Strontium-90	3.5E-13
Yttrium-90 (daughter of strontium-90)	3.4E-20
Zirconium-95	3.3E-15
Niobium-95 (daughter of zirconium-95)	1.0E-19
Niobium-95m (daughter of zirconium-95)	2.9E-20
Niobium-95	4.3E-15
Iodine-131	7.9E-13
Iodine-133	1.4E-13
Caesium-134	8.1E-14
Caesium-137	1.1E-12
Barium-137m (daughter of caesium-137)	7.8E-20
Barium-140	2.5E-16
Lanthanum-140 (daughter of barium-140)	1.1E-19

Table 5.2-35

ACTIVITY CONCENTRATION IN COASTAL SEDIMENTS IN 60TH YEAR OF LIQUID DISCHARGES

Radionuclide	Activity Concentration in Local Compartment (Bq/kg)
Tritium	1.4E+01
Carbon-14	1.1E+00
Sodium-24	6.6E-09
Chromium-51	1.9E-04
Manganese-54	2.1E-03
Iron-55	8.7E-02
Iron-59	3.8E-05
Cobalt-58	5.7E-03
Cobalt-60	7.4E-02
Nickel-63	5.4E-01
Zinc-65	4.3E-04
Strontium-89	4.5E-06
Strontium-90	3.5E-05
Yttrium-91	1.0E-06
Zirconium-95	8.6E-05
Niobium-95 (daughter of zirconium-95)	9.7E-05
Niobium-95	3.7E-05
Technetium-99m	9.5E-10
Technetium-99 (daughter of technetium-99m)	1.1E-12
Ruthenium-103	7.0E-05
Silver-110m	2.7E-04
Iodine-131	3.2E-07
Xenon-131m (daughter of iodine-131)	3.8E-07
Iodine-132	1.0E-10

Table 5.2-35 (continued)

ACTIVITY CONCENTRATION IN COASTAL SEDIMENTS IN 60TH YEAR OF LIQUID DISCHARGES

Radionuclide	Activity Concentration in Local Compartment (Bq/kg)
Iodine-133	1.1E-08
Xenon-133 (daughter of iodine-133)	4.4E-08
Iodine-134	4.3E-12
Iodine-135	9.7E-10
Xenon-135 (daughter of iodine-135)	2.1E-09
Caesium-135 (daughter of iodine-135)	3.3E-12
Caesium-134	4.7E-04
Caesium-136	6.0E-06
Caesium-137	7.2E-03
Barium-140	1.1E-05
Lanthanum -140 (daughter of barium-140)	1.4E-05
Lanthanum-140	8.9E-07
Cerium-144	4.9E-03
Promethium-144	2.3E-10
Plutonium-241	4.7E-05
Americium-241 (daughter of plutonium-241)	1.4E-06
Neptunium-237 (plutonium-241 decay chain)	6.1E-12
Uranium-233 (plutonium-241 decay chain)	3.2E-16
Thorium-229 (plutonium-241 decay chain)	3.4E-19
Actinium-225 (plutonium-241 decay chain)	3.4E-19
Bismuth-213 (plutonium-241 decay chain)	3.4E-19
Lead-209 (plutonium-241 decay chain)	3.4E-19

Table 5.3-1 **ERICA TOOL TIER 2 OUTPUT CLASSIFICATION**		
Level of Concern	**Expected RQ**	**Conservative RQ**
Negligible	<1	<1
Insufficient Confidence	<1	>1
Of Concern	>1	>1

Table 5.3-2

AIR EMISSION DATA USED IN THE ERICA TOOL

Isotope	Westinghouse Predicted Operating Data [c.f. Table 3.3-8] Bqs^{-1}	Value Used in ERICA Tool Bqs^{-1}	Value Used in Wildlife Dose Assessment Spreadsheet Bqs^{-1}
Tritium	5.71E+04	5.71E+04	–
Carbon-14	1.92E+04	1.92E+04	–
Argon-41	3.99E+04	–	3.99E+04
Chromium-51	7.29E-03	–	–
Manganese-54	5.07E-03	5.07E-03	–
Cobalt-58	2.70E-01	2.70E-01	–
Cobalt-60	1.01E-01	1.01E-01	–
Krypton-85	1.73E+05 [1]	–	2.12E+05 [1,2]
Krypton-85m	7.61E+02	–	–
Strontium-89	3.49E-02	3.49E-02	–
Strontium-90	1.40E-02	1.40E-02	–
Zirconium-95	1.17E-02	1.17E-02	–
Niobium-95	2.95E-02	2.95E-02	–
Iodine-131	6.66E+00	6.02E+00	–
Iodine-133	1.11E+01	9.83E+00	–
Xenon-133	4.12E+04	–	[2]
Caesium-134	2.70E-02	2.70E-02	–
Caesium-137	4.12E-02	4.12E-02	–
Barium-140	5.07E-03	–	–

Notes:
1. Krypton-85 value includes emissions of isotopes Krypton-87, Krypton-88, Xenon-131 m, Xenon-133 m, Xenon-135, Xenon-135m, Xenon-137 and Xenon-138
2. Xenon-133 included with Krypton-85

Table 5.3-3

INPUT DATA FOR THE ERICA TIER 1 ASSESSMENT OF AIR EMISSIONS

Parameter	Input	Comments
Ecosystem	Terrestrial	Appropriate for evaluating impact of air emissions
Media activity concentration	IAEA SRS-19 air model	Generic dispersion model within ERICA. Established internationally recognised methodology. Provides consistency, allowing comparison between different assessments. May be overly conservative.
Release height	81.626 m	Reactor Building Vent 74.926 m (Reference 1-1) + 6.7 m plume rise under neutral atmospheric conditions
Distance to receptor	200 m	Distance to generic site boundary
Wind speed	5.0 m/s	Average wind speed value assumed for generic site
Fraction of time (wind blowing towards the direction of receptor)	0.25	Default value – conservative for generic site
Dry deposition coefficient	500 m/d	ERICA Default Value. These values are based on a recommendation that a total deposition coefficient for wet and dry deposition 1000 m/d is used for screening purposes for deposition of aerosols and reactive gases (Reference 5-4)
Wet deposition coefficient	500 m/d	
Surface soil density	260 kg/m2	ERICA Default Value. Value typical for crops on non-peat soils with a rooting zone depth of 0-20cm (Reference 5-4). Actual values of surface soil density may vary depending on the origin, mineral content and classification of the soil, but uncertainties about soil density are relatively small.
Duration of discharge	60 years	Lifetime of Westinghouse PWR plant (Reference 1-1)
Buildings nearby	yes	Reactor Building (Reference 1-1)
Building Height	70m	Reactor Building (Reference 1-1)
Buildings Surface Area Wall	3000m^2	Reactor Building (Reference 1-1)

Table 5.3-4

TIER 1 RESULTS OF ERICA TOOL ASSESSMENT ON AIR EMISSIONS

Isotope	Risk Quotient (unit less)	Limiting Reference Organism
H-3	2.26E-04	Detritivorous invertebrate
C-14	2.38E-03	Mammal (deer)
Mn-54	4.09E-09	Detritivorous invertebrate
Co-58	7.17E-08	Mammal (rat)
Co-60	1.51E-06	Mammal (rat)
Sr-89	1.42E-07	Reptile
Sr-90	7.10E-06	Reptile
Zr-95	1.72E-09	Detritivorous invertebrate, soil invertebrate (worm)
Nb-95	2.53E-09	Mammal (rat)
I-133	6.49E-06	Bird egg
Cs-134	6.06E-07	Mammal (deer)
Cs-137	2.55E-06	Mammal (deer)
I-131	1.71E-05	Bird egg
∑ Risk Quotients	2.64E-03	

Table 5.3-5

SENSITIVITY OF THE TIER 1 RESULTS OF ERICA TOOL ASSESSMENT FOR AIR EMISSIONS

Scenario	Stack Height (m)	Wind Speed (ms^{-1})	Distance to Receptor (m)	Sum of Risk Quotient (unitless)
A	81.626	5	200	2.64E-03
B	39.8	5	200	2.99E-03
C	81.626	1	200	1.32E-02
D	81.626	2	200	6.60E-03
E	81.626	5	200	2.64E-03
F	81.626	10	200	1.32E-03
G	81.626	5	50	5.82E-02
H	81.626	5	100	5.82E-02
I	81.626	5	200	2.64E-03
J	81.626	5	300	1.56E-03
K	39.8	1	50	2.91E-01

Table 5.3-6

WATER DISCHARGE DATA USED IN THE ERICA TOOL

Isotope	Westinghouse Predicted Operating Data [Reference 8]	Value Used in ERICA Tier 1 Assessment	Value Used in ERICA Tier 2 Assessment
	Bqs^{-1}	Bqs^{-1}	Bqs^{-1}
Tritium	1.06E+06	1.06E+06	1.06E+06
Carbon14	1.05E+02	1.05E+02	1.05E+02
Sodium-24	1.20E+00	–	1.20E+00
Chromium-51	1.46E+00	–	1.46E+00
Manganese-54	1.01E+00	1.01E+00	1.01E+00
Cobalt-58	1.30E+01	1.30E+01	1.30E+01
Iron-55	1.55E+01	–	1.55E+01
Iron-59	1.59E-01	–	1.59E-01
Cobalt-60	7.29E+00	7.29E+00	7.29E+00
Nickel-63	1.71E+01	1.71E+01	1.71E+01
Zinc-65	3.17E-01	–	3.17E-01
Rubidium-88	1.24E-02	–	1.24E-02
Strontium-89	7.61E-02	7.61E-02	7.61E-02
Strontium-90	7.93E-03	7.93E-03	7.93E-03
Yttrium-91	2.89E-03	–	2.89E-03
Zirconium-95	2.19E-01	2.19E-01	2.19E-01
Niobium-95	1.93E-01	1.93E-01	1.93E-01
Molybdenum-99	6.02E-01	–	6.02E-01
Technetium-99m	5.71E-01	–	5.71E-01
Ruthenium-103	3.81E+00	3.81E+00	3.81E+00
Silver-110m	8.24E-01	8.24E-01	8.24E-01
Iodine-131	4.76E-01	4.76E-01	4.76E-01
Iodine-132	6.34E-01	6.34E-01	6.34E-01
Iodine-133	9.20E-01	9.20E-01	9.20E-01

Table 5.3-6 (cont.)

WATER DISCHARGE DATA USED IN THE ERICA TOOL

Isotope	Westinghouse Predicted Operating Data [Reference 8]	Value Used in ERICA Tier 1 Assessment	Value Used in ERICA Tier 2 Assessment
	Bqs^{-1}	Bqs^{-1}	Bqs^{-1}
Iodine-134	1.87E-01	–	1.87E-01
Iodine-135	7.61E-01	–	7.61E-01
Caesium-134	2.41E-01	2.41E-01	2.41E-01
Caesium-136	2.95E-01	2.95E-01	2.95E-01
Caesium-137	7.29E-01	7.29E-01	7.29E-01
Barium-140	4.44E-01	–	4.44E-01
Lanthanum-140	5.71E-01	–	5.71E-01
Cerium-144	2.54E+00	2.54E+00	2.54E+00
Praseodymium-144	2.54E+00	–	2.54E+00
Plutonium-241	2.54E-03	2.54E-03	2.54E-03

Note:

Cl-36, Nb-94, As-76, Br-82, Rb-86, Tc-99, Ru-106, Sn-117m, Sb-122, Sb-124, Sb-125, I-129 U-234, U-235, U-238, Np-237, Pu-238, Pu-239, Pu-240, Pu-242, Am-241, Am-243, Cm-242, Cm-244 and all others < 1.17E-03 Bqs^{-1} each.

Table 5.3-7

INPUT DATA IN THE ERICA TIER 1 AND TIER 2 ASSESSMENT OF WATER EMISSIONS

Parameter	Input	Comments
Ecosystem	Marine	Appropriate for evaluating impact of water discharges to sea
Dose rate screening value	10 μGyh^{-1}	ERICA dose rate screening value
Uncertainty factor (UF)	3	The method using UFs is based on the assumption that the estimated doses, and RQs, are exponentially distributed, which is supported by the principle of maximum entropy. Under these circumstances a UF of 3 establishes a 5% probability of exceeding the dose screening value.
Media activity concentration	IAEA SRS-19 coastal model	Generic dispersion model within ERICA. Established internationally recognised methodology. Provides consistency, allowing comparison between different assessments. May be overly conservative.
Concentration Ratios	ERICA Default (Check boxes 1,2,3,5 and 7 activated	The activity concentrations of radionuclides in biota within the ERICA tool are predicted from media activity concentrations using equilibrium concentration ratios (CRs). For the marine environment, these are based on the ratio of whole body to filtered water activity concentrations. The default radioecology databases within the ERICA tool were used to provide CR values for all reference organisms. The CRs used in the ERICA default database are comprehensive, drawing on an extensive review of published literature and characterised by statistical information. Where there is no published data for particular reference organism-radionuclide combinations, the ERICA tool adopts a standard procedure to derive the best CR values. This procedure requires information to be selected from various other sources prioritised in the following order: 1) Similar taxonomy, 2) Similar reference organism, 3) From published reviews, 4) Specific activity models, 5) Similar biogeochemistry, 6) Similar biogeochemistry and taxonomy, 7) Similar biogeochemistry and reference organism, 8) Allometric or other modelling approaches, 9) Highest available value, 10) Reference organism in a different ecosystem, and 11) Combination of approaches. In order to completely populate the CR database for Tier 2 analysis with default ERICA values, it was necessary to check boxes that incorporated information from the following sources above: 1), 2), 3), 5), and 7).

Table 5.3-7 (cont.)

INPUT DATA IN THE ERICA TIER 1 AND TIER 2 ASSESSMENT OF WATER EMISSIONS

Parameter	Input	Comments
Distribution Coefficients	ERICA Default	For aquatic environments, the distribution coefficient (K_d) is used to relate equilibrium activity concentrations in sediments with those in water. However, it is recognised that the ERICA default K_d are mostly poorly defined statistically, but they are used in the absence of a specific site selection and site-specific Kd values.
Occupancy Factor	ERICA Default	The occupancy factor, the fraction of time that the aquatic organism spends at a specified location in its habitat (e.g., on water surface/in water/at water sediment interface/in sediment). The ERICA default occupancy factors have been selected to maximise the dose, such as those selected for the location in the habitat where highest doses might be expected. This may lead to an overestimation of the dose rate in some cases.
Radiation Weighting Factor Alpha	10.0	For a given unweighted absorbed dose rate, α-radiation may result in a more significant effect than β- or γ-radiation. Radiation weighting factors are introduced to account for the relative biological effectiveness of these different types of radiation. The radiation weighting factors used in ERICA have been adopted from FASSET. They have always been considered provisional values, applied for demonstration purposes only, and therefore, their application is arguably unsubstantiated. They might be considered conservative values.
Radiation Weighting Factor Beta/Gamma	1.0	
Radiation Weighting Factor Low beta	3.0	
Water depth (Chart Datum)	5 m	Generic site information [Reference 5-1]
Distance between release point and shore	150 m	Assumed value
Distance between release point and receptor	100 m	Assumed distance to nearest population of marine organisms of concern
Coastal current	0.1 m/s	Default value recommended for use when site-specific information is not available (Reference 5-4)

Table 5.3-8

TIER 1 RESULTS OF ERICA TOOL ASSESSMENT ON WATER DISCHARGES

Isotopes	Risk Quotient (unitless)	Limiting reference organism
H-3	1.78E-03	Phytoplankton
C-14	9.61E-03	Wading bird, reptile
Mn-54	1.70E-01	Polychaete worm
Co-58	3.86E-01	Polychaete worm
Co-60	5.50E-01	Polychaete worm
Ni-63	2.31E-04	Benthic mollusc
Sr-89	1.57E-06	Sea anemones or true corals – colony
Sr-90	3.06E-07	Sea anemones or true corals – colony
Zr-95	3.32E-02	Polychaete worm
Nb-95	1.18E-02	Polychaete worm
Ru-103	8.15E-03	Phytoplankton
Ag-110m	2.83E-03	Reptile
I-131	4.44E-05	Macroalgae
I-132	1.33E-04	Vascular plant
I-133	1.61E-04	Macroalgae
Cs-134	1.50E-04	Polychaete worm
Cs-136	2.51E-04	Polychaete worm
Cs-137	1.71E-04	Polychaete worm
Ce-144	2.69E-01	Polychaete worm
Pu-241	4.35E-07	Phytoplankton
∑ Risk Quotients	1.44+00	

Table 5.3-9

TIER 2 RESULTS OF WATER DISCHARGES USING THE ERICA DOSE RATE SCREENING VALUE

Organism	Total Dose Rate per organism (μGyh^{-1})	Dominant Sources of Dose Rate		RQ (expected value) (unitless)	RQ (conservative value) (unitless)
		Isotope	% Dose		
Polychaete worm	2.52E+01	Fe-59 Co-60 Co-58	76% 7% 5%	2.52E+00	7.57E+00
Macroalgae	1.34E+01	Fe-59 Co-60 Co-58	72% 7% 5%	1.34E+00	4.02E+00
Sea anemones or true corals – polyp	1.31E+01	Fe-59 Co-60 Co-58	73% 7% 5%	1.31E+00	3.92E+00
Benthic mollusc	1.23E+01	Fe-59 Co-60 Co-58	77% 7% 5%	1.23E+00	3.70E+00
Vascular plant	1.22E+01	Fe-59 Co-60 Co-58	77% 7% 5%	1.22E+00	3.65E+00
Benthic fish	1.15E+01	Fe-59 Co-60 Co-58	78% 8% 5%	1.15E+00	3.44E+00
Sea anemones or true corals – colony	1.13E+01	Fe-59 Co-60 Co-58	77% 8% 5%	1.13E+00	3.39E+00
Crustacean	1.10E+01	Fe-59 Co-60 Co-58	79% 8% 5%	1.10E+00	3.30E+00
Mammal	3.51E+00	Fe-55 Fe-59 C-14	65% 34% 1%	3.51E-01	1.05E+00
Reptile	3.50E+00	Fe-55 Fe-59 C-14	65% 33% 1%	3.50E-01	1.05E+00
(Wading) bird	2.76E+00	Fe-55 Fe-59 C-14	83% 16% 1%	2.76E-01	8.28E-01

Table 5.3-9 (cont.)

TIER 2 RESULTS OF WATER DISCHARGES USING THE ERICA DOSE RATE SCREENING VALUE

Organism	Total Dose Rate per organism (μGyh^{-1})	Dominant Sources of Dose Rate		RQ (expected value) (unitless)	RQ (conservative value) (unitless)
		Isotope	% Dose		
Zooplankton	9.09E-02	Fe-55 C-14 Ru-103	56% 20% 6%	9.09E-03	2.73E-02
Pelagic fish	3.87E-02	Fe-55 H-3 C-14	57% 13% 12%	3.87E-03	1.16E-02
Phytoplankton	3.23E-02	Fe-55 H-3 Fe-59	88% 7% 4%	3.23E-03	9.69E-03

Table 5.3-10

SENSITIVITY OF THE TIER 2 RESULTS OF ERICA TOOL ASSESSMENT FOR WATER DISCHARGES TO SEA

Scenario	Water Depth (m)	Distance between Release Point and Shore (m)	Distance between Release Point and Receptor (m)	Coastal Current (ms^{-1})	Total Dose Rate (μGyh^{-1})	No. Organisms greater with than ERICA Screening Value	No. Organisms "insufficient confidence"	No. Organisms "negligible risk"
A	5	150	100	0.1	25.2	8 (PW, Ma, SAp, BM, VP, BF, SAc, Cr)	2 (Mm, Re)	4 (WB, Zo, PF, Ph)
B	2	150	100	0.1	63.1	8 (PW, Ma, SAp, BM, VP, BF, SAc, Cr)	3 (Mm, Re, WB)	3 (Zo, PF, Ph)
C	7	150	100	0.1	18.0	1 (PW)	7 (Ma, SAp, BM, VP, BF, SAc, Cr)	6 (WB, Zo, PF, Ph, Mm, Re)
D	13	150	100	0.1	9.71	0	8 (PW, Ma, SAp, BM, VP, BF, SAc, Cr)	6 (WB, Zo, PF, Ph, Mm, Re)
E	5	50	100	0.1	25.2	8 (PW, Ma, SAp, BM, VP, BF, SAc, Cr)	2 (Mm, Re)	4 (WB, Zo, PF, Ph)
F	5	100	100	0.1				
G	5	200	100	0.1				
H	5	150	50	0.1	57.5	8 (PW, Ma, SAp, BM, VP, BF, SAc, Cr)	3 (Mm, Re, WB)	3 (Zo, PF, Ph)
I	5	150	80	0.1	32.9			

Table 5.3-10 (cont.)

SENSITIVITY OF THE TIER 2 RESULTS OF ERICA TOOL ASSESSMENT FOR WATER DISCHARGES TO SEA

Scenario	Water Depth (m)	Distance between Release Point and Shore (m)	Distance between Release Point and Receptor (m)	Coastal Current (ms^{-1})	Total Dose Rate (μGyh^{-1})	No. Organisms greater with than .ERICA Screening Value	No. Organisms "insufficient confidence"	No. Organisms "negligible risk"
J	5	150	220	0.1	9.87	0	8 (PW, Ma, SAp, BM, VP, BF, SAc, Cr)	6 (WB, Zo, PF, Ph, Mm, Re)
K	5	150	560	0.1	3.27	0	0	All
L	5	150	100	0.05	22.1	6 (PW, Ma, SAp, BM, VP, BF, SAc, Cr)	4 (Mm, Re, SAc, Cr)	6 (WB, Zo, PF, Ph, Mm, Re)
M	5	150	100	0.2	28.7	8 (PW, Ma, SAp, BM, VP, BF, SAc, Cr)	2 (Mm, Re)	4 (Zo, PF, Ph, WB)
N	5	150	100	0.5	33.9	8 (PW, Ma, SAp, BM, VP, BF, SAc, Cr)	3 (Mm, Re, WB)	3 (Zo, PF, Ph)
O	2	150	50	0.5	191	11 (PW, Ma, SAp, BM, VP, BF, SAc, Cr, Mm, Re, WB)	0	3 (Zo, PF, Ph)

Table 5.3-10 (cont.)

SENSITIVITY OF THE TIER 2 RESULTS OF ERICA TOOL ASSESSMENT FOR WATER DISCHARGES TO SEA

Scenario	Water Depth (m)	Distance between Release Point and Shore (m)	Distance between Release Point and Receptor (m)	Coastal Current (ms^{-1})	Total Dose Rate (μGyh^{-1})	No. Organisms greater with than .ERICA Screening Value	No. Organisms "insufficient confidence"	No. Organisms "negligible risk"
P	13	150	560	0.05	1.7	0	0	All

WB Wading Bird; BF Benthic Fish; BM Benthic Mollusc; Cr Crustacean; Ma Macroalgae
Mm Mammal; PF Pelagic Fish; Ph Phytoplankton; PW Polychaete Worm; Re Reptile
VP Vascular Plant; Zo Zooplankton; Sac Sea Anemones or true corals; Sac Sea Anemones or true corals – polyp

Table 5.3-11

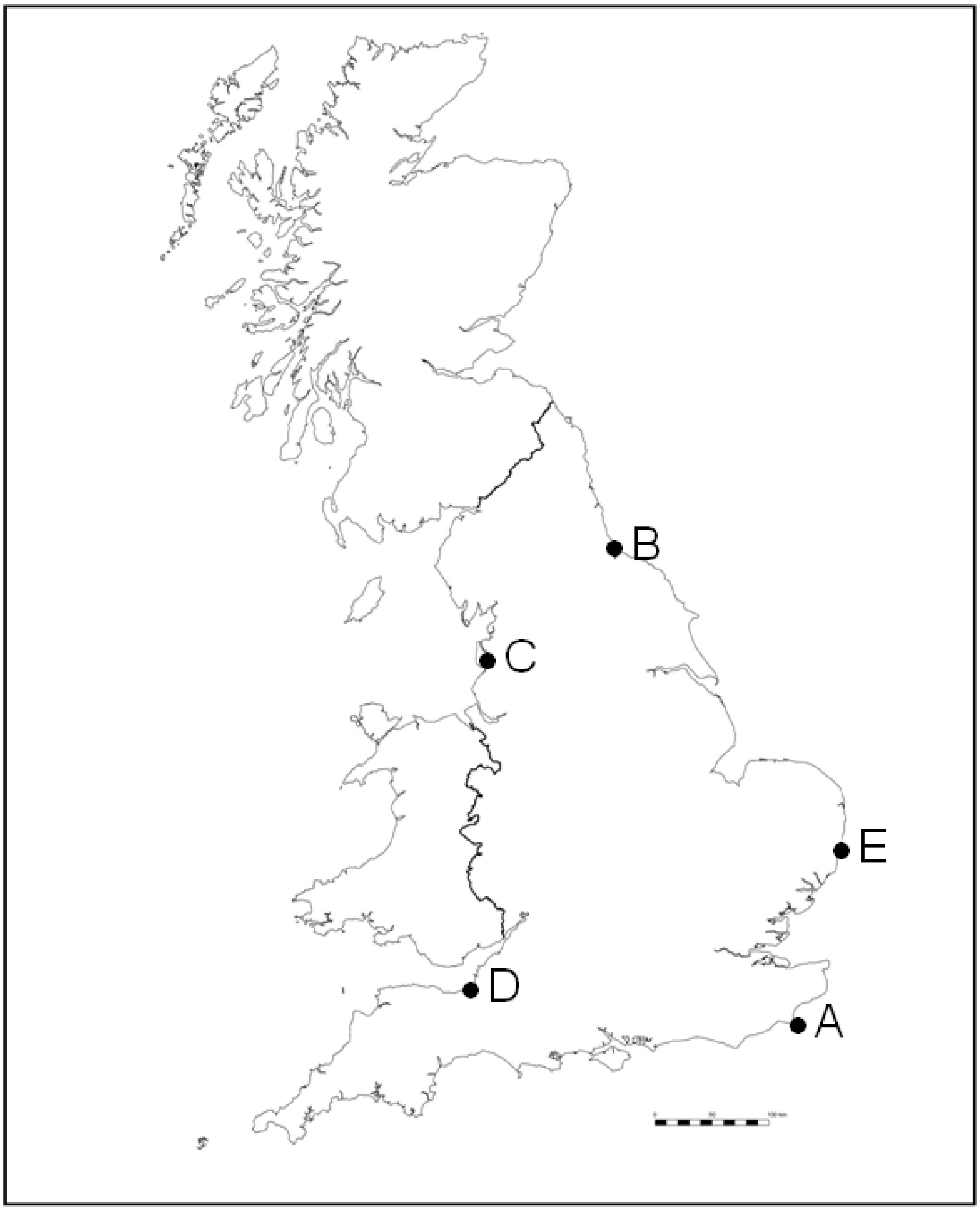
INPUT DATA IN THE WILDLIFE DOSE ASSESSMENT SPREADSHEET

Parameter	Input	Comments
Ecosystem	Terrestrial	
Concentration Rations	Spreadsheet Default	Spreadsheet Default Value
Occupancy Factor	Spreadsheet Default	Spreadsheet Default Value
Radiation Weighting Factor Beta/Gamma	1.0	Spreadsheet Default Value
Emission flow rate	38.13 m^3s^{-1}	Westinghouse Design (Reference 1-1)
Argon-41 concentration at receptor	0.360 Bqm^{-3}	Calculated in Reference 5-8 using Equations 4 to 6 from Reference 5-4
Krypton-85 concentration at receptor	1.911 Bqm^{-3}	Calculated in Reference 5-8 using Equations 4 to 6 from Reference 5-4
Distance to receptor	200 m	Distance to generic site boundary
Wind speed	5.0 m/s	Default value – conservative for generic site
Fraction of time (wind blowing towards the direction of receptor)	0.25	ERICA Default Value
Building Height	70m	Reactor Building (Reference 1-1)
Buildings Surface Area Wall	$3000m^2$	Reactor Building (Reference 1-1)

Table 5.3-12

OUTPUT DATA IN THE WILDLIFE DOSE ASSESSMENT SPREADSHEET

Organism	**Sum of Ar-41 and Kr-85 Dose Rate per Organism ($\mu Gy\ h^{-1}$)**
Ant	1.1E-04
Bacteria	6.4E-08
Bee	2.3E-04
Bird	1.5E-04
Bird Egg	1.1E-04
Caterpillar	2.6E-04
Earthworm	3.1E-08
Fungi	2.7E-04
Herb	1.7E-04
Lichen	1.4E-04
Mammal (carnivore)	5.7E-05
Mammal (herbivore)	4.9E-05
Reptile	6.3E-05
Rodent	4.6E-05
Seed	2.0E-04
Shrub	1.7E-04
Tree	1.7E-04
Woodlouse	1.4E-04



A – Dungeness, B – Hartlepool, C – Heysham, D – Hinkley, E – Sizewell

Figure 5.1-1. Location of Nuclear Power Stations Used to Establish the Generic Design Case

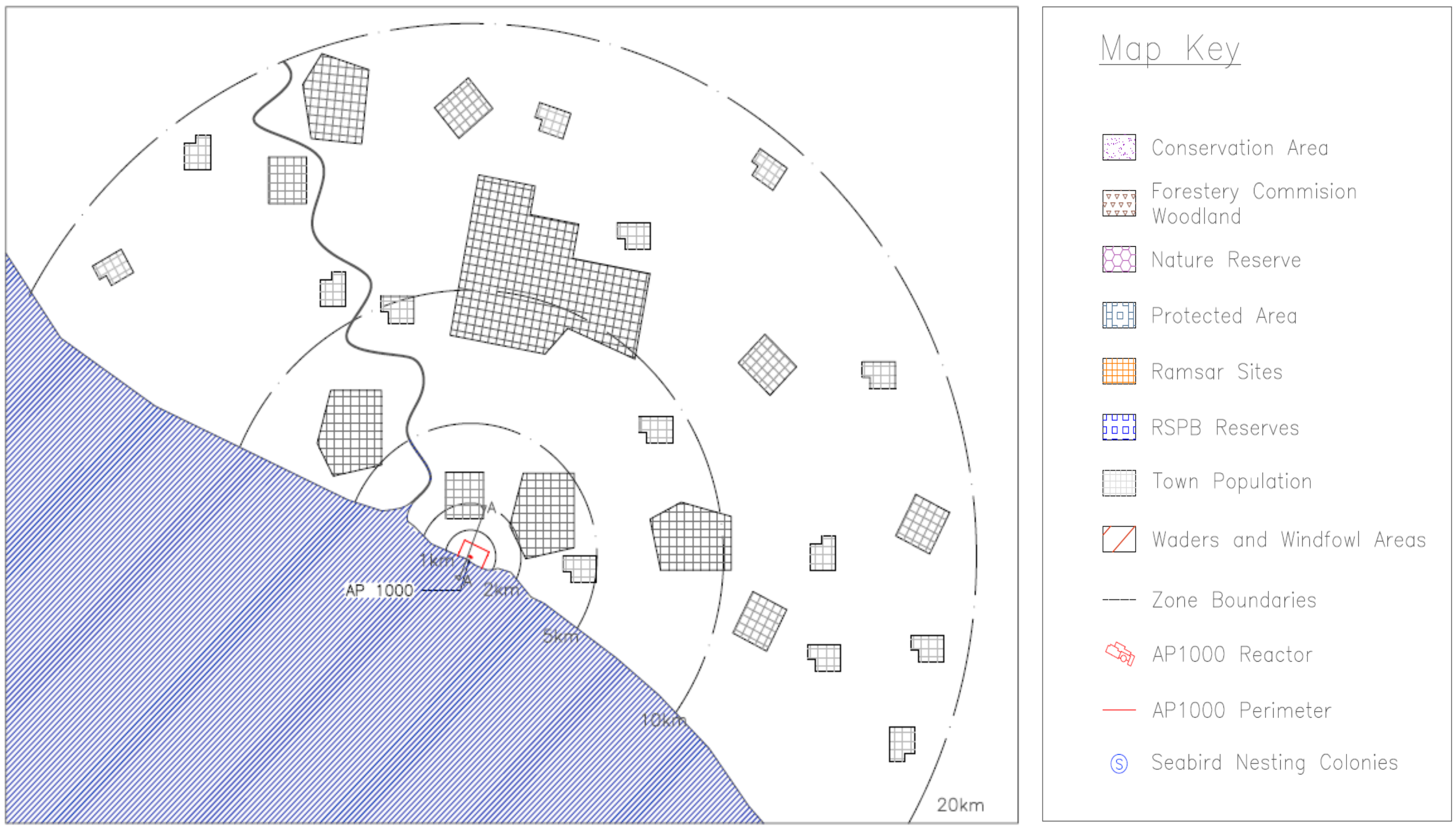


Figure 5.1-2. Population Centres for the Generic Design Case

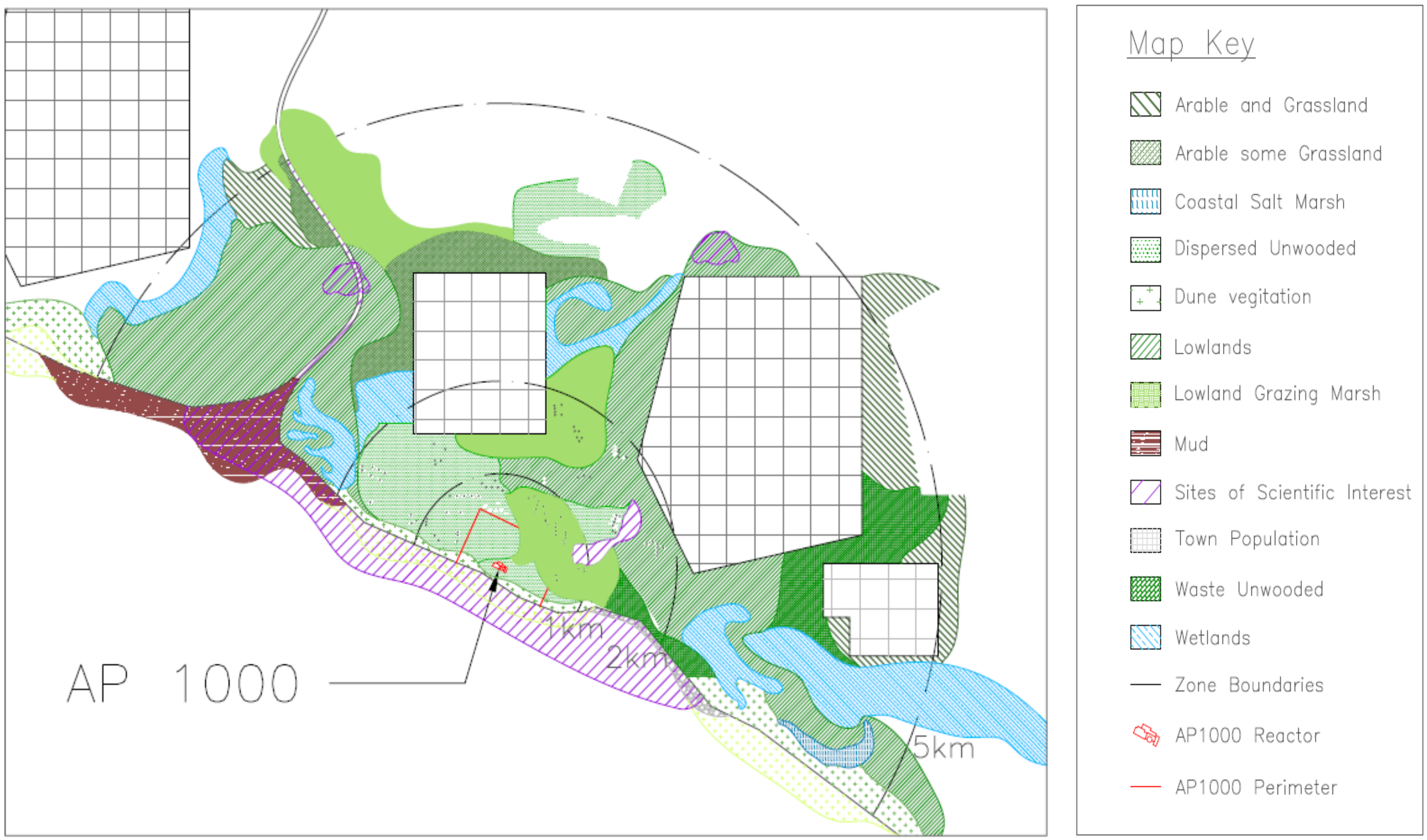


Figure 5.1-3. Land Use and Habitat Data for the Generic Design Case

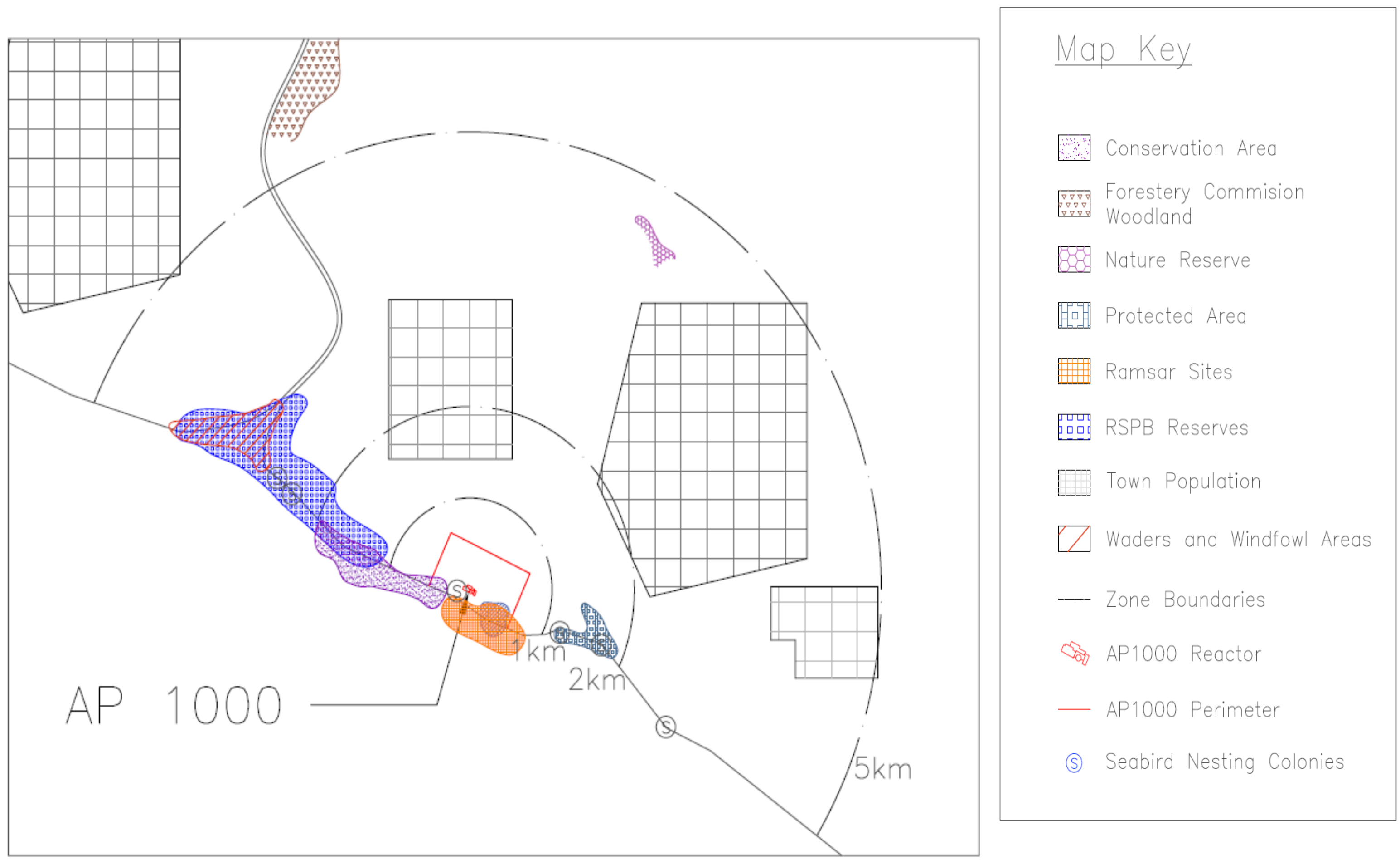


Figure 5.1-4. Sites of Special Interest for the Generic Design Case

6.0 ENVIRONMENTAL MONITORING

6.1 Proposed Regulatory Limits

The EA has produced guidance to assist nuclear regulators in setting consistent radioactive discharge limits for nuclear-licensed sites (Reference 6-1). This guidance has been used to calculate annual discharge limits for radioactive atmospheric emissions and radioactive liquid effluents from the **AP1000** NPP (Reference 6-2); this report is summarised below.

6.1.1 Selection of Isotopes

In selecting which isotopes warrant discharge limit calculations, consideration was given to those radionuclides 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.

The criteria for selection are shown in Table 6.1-1 for gaseous discharges and Table 6.1-2 for liquid discharges.

6.1.2 Limits Setting Procedure

The EA has produced guidance to assist nuclear regulators in setting consistent radioactive discharge limits for nuclear-licensed sites (Reference 6-1). This guidance suggests the following steps:

Step 1: Review all plants on the site that contribute to discharges, and identifying those where limits should be set.

As no specific site has been selected for the GDA process, no assumptions have been made regarding emissions and discharges from other plants located on the generic site. The emission and discharge data used to calculate limits are for the **AP1000** NPP alone.

Step 2: Derivation of the worst case annual plant discharges (WCPD) using the following formula:

$$WCPD = (1.5 \times D \times T \times A \times B) + C + L + N - I \qquad [1]$$

where:

1.5 = an EA-established factor which relates “worst case” to average discharges.

D = the representative average 12-month plant discharge, excluding discharges due to faulty plant operation.

T = a factor that allows for any future increases in throughput, power output, etc., relative to the review period.

A = a factor that allows for plant ageing. This allows for increases in discharges which result from changes within the plant as it ages that cannot be remedied or controlled by the operator.

B = a factor that allows for other future changes that are beyond the control of the operator. For example, in a reprocessing plant, these would include the need to deal with higher burn-up or shorter-cooled fuel; at a dockyard, the need to deal with wastes from a new class of submarines.

C = an allowance for decommissioning work beyond that carried out in the review period (and included in D).

L = an allowance for dealing with legacy wastes, beyond those dealt with in the review period (and included in D).

N = an allowance for new plant.

I = the reduction in discharges expected as a result of introducing improvement schemes before the new authorisation comes into force.

Since the Westinghouse **AP1000** NPP site is a new plant, allowances are not made for decommissioning work, legacy wastes, further new plant, or introduction of improvement schemes. Thus, factors C, L, N, and I become zero, and the worst case plant annual discharge becomes:

$$WCPD = 1.5 \times D \times T \times A \times B \qquad [2]$$

6.1.3 Calculated Limits

To estimate the worst case annual plant discharge, the parameters in equation [2] must be defined. The selection of parameters is discussed below:

- Representative 12-month plant discharge (D)

The predicted annual average plant discharges for an **AP1000** NPP site are given in Tables 3.3-6, 3.3-7, 3.3-8, and 3.4-6. These averages are broken down into monthly releases over the 18-month fuel cycle for air emissions and liquid discharges in Tables 6.1-3 and 6.1-4, respectively. The monthly breakdown accounts for periods of start-up, shutdown, and maintenance. To allow for periods when the discharge is likely to be higher than the predicted annual average, the 12 months at the end of each cycle (i.e., months 7-18) is used as the worst case representative 12-month plant discharge.

- Future Increase in Throughput (T)

There are no plans to increase the **AP1000** NPP throughput, power output, etc., within the foreseeable future. This parameter has been set to 1.

- Plant Ageing (A)

A margin of 10 percent is considered suitable to allow for increases in discharge as the plant ages. Therefore, this parameter is set to 1.1.

- Future Changes (B)

There are no foreseen future changes beyond the control of the operator. This parameter has been set to 1.

Therefore, the worst case annual plant discharge may be calculated for each air and liquid effluent to propose an annual limit as follows:

$$WCPD = 1.5 \times D \times T \times A \times B \qquad [2]$$

$$WCPD = 1.5 \times D \times 1 \times 1.1 \times 1 \qquad [3]$$

The results are summarised in Table 6.1-5 for air emissions and Table 6.1-6 for liquid discharges. The calculated annual limits are rounded to one significant figure as per the guidance in Reference 6-1.

6.1.4 Proposed Limits

It is appropriate that the **AP1000** NPP is regulated in a similar way to the existing NPPs in the UK. In this respect, it is proposed that annual limits for the **AP1000** NPP are restricted to those isotopes that already have established limits at UK NPPs (Reference 6-3).

The proposed annual limits for the **AP1000** NPP are compared directly against the limits for the Sizewell B PWR site air emissions in Table 6.1-7 and for liquid discharges in Table 6.1-8. The proposed limits for the **AP1000** NPP are compared directly against the limits for the UK AGRs in Table 6.1-9 and for liquid discharges in Table 6.1-10.

Figures 6.1-1 to 6.1-7 show how the predicted rolling annual average emissions for gaseous radioactive emissions compare with the proposed limits. Figures 6.1-8 to 6.1-10 show how the predicted rolling annual average emissions for liquid radioactive discharges compare with the proposed limits.

6.2 Monitoring Programmes

There are six types of environmental monitoring programmes that are typically used in the **AP1000** NPP system, as shown in Table 6.2-1.

6.2.1 Radiological Monitoring

This section provides a summary of the report on the discharge monitoring arrangements included in the **AP1000** NPP design (Reference 6-4). This report compared the **AP1000** NPP monitoring techniques with relevant UK regulatory guidance (References 6-5 to 6-7). It concluded that both the aerial and liquid effluent streams monitoring systems are in good agreement with the guidance.

The proposed **AP1000** NPP monitoring systems are broadly equivalent to monitoring systems currently used in operating UK NPPs. This provides confidence that the **AP1000** NPP monitoring techniques are in line with current best practice in the UK.

During the detailed **AP1000** NPP design, consideration will be given to any future Environment Agency Monitoring Certification Scheme (MCERTS) requirements for the monitoring of radioactive discharges. This may include sampling and monitoring equipment, qualifications of personnel involved in monitoring, and laboratory accreditation. None of the monitoring requirements require new technology or measurement techniques. This provides confidence that the **AP1000** NPP monitoring techniques can be designed and engineered to demonstrate BAT and meet applicable UK requirements.

6.2.1.1 Aerial Emissions

Gaseous radioactive effluents arise from a number of processes and in a number of different areas of the **AP1000** NPP. Discharges to the atmosphere from the plant are through two main vents:

- The main plant vent provides the release path for containment venting releases, auxiliary building ventilation releases, annex building releases, radwaste building releases and gaseous radwaste. The **AP1000** NPP WGS receives, processes, and discharges radwaste gases received during normal modes of plant operation including power generation, shutdown, and refuelling. About 90 percent of total annual aerial discharges (in Bq/y (Ci/y)) are through the main plant vent.

- The turbine building vent provides the release path for the condenser air removal system, land seal condenser exhaust, and the turbine building ventilation releases. Under ideal normal operation conditions, no radwaste is discharged through the turbine building vent. However, it is possible that a number of small leaks from the primary coolant to the secondary coolant cycle occur during the plant's operations, resulting in the possibility of a small aerial release from the turbine building vent.

Figure 6.2-1 provides a schematic overview of the aerial release points. The design of the stack monitoring system is still being developed; the requirements of M1 and M11 will be considered during this development. The instrument that will be used for flow rate measurement has not been specified. When the instrument is specified, the register of MCERTS-certified equipment will be reviewed to determine if a suitable instrument is available on the register. The exact locations of the monitoring point (MP), flow measurement point, upstream and downstream disturbances, and location of filtration have not been determined. A sample point will be chosen where the flow is well mixed and the velocity profile is relatively constant across the cross-section. The sample point will be sufficiently remote from the last disturbance in the stack. The rising stack is a straight run with no bends or disturbances in the line. Sample points will be chosen to ensure that the gas velocities across the ducts are approximately equal and that the gases are homogenous.

Tables 6.2-2 and 6.2-3 list the monitors that are associated with the main plant vent and turbine building vent, respectively. The tables include information on the type of monitor, radionuclides monitored, and detection ranges. The MP numbers relate to Figure 6.2-1.

The main plant vent radiation monitor (MP 8) is an off-line monitor for particulates, iodine and noble gases. All main plant aerial discharge streams have converged at this point and the main plant vent monitor provides the data needed for discharge reporting. Sampling is continuous under routine operating conditions by normal range noble gas, particulate, and iodine detectors. The particulate monitor uses a beta-sensitive scintillation detector that views a fixed filter. The iodine detector is a gamma-sensitive, thallium-activated, sodium iodide, scintillation detector that views a fixed charcoal filter. Krypton-85 and xenon-133 are monitored by an in-line beta-sensitive scintillation detector. In addition to the normal range krypton-85 and xenon-133 monitor, MP 8 also contains 2 accident range noble gas detectors, one for mid-range and one for high-range activity concentrations. These are beta/gamma-sensitive detectors with smaller sensitive volumes. Accident range particulate and iodine activity concentrations are monitored at MP 8 using particulate and iodine filters which would be taken to the on-site laboratory for analysis. In the event that elevated activity concentrations are detected by the normal range detectors, the monitored air flow would be redirected from the normal range detectors to the accident range detectors. In this way, the normal range detectors would not be damaged and/or contaminated by higher activity concentrations and remain available to resume monitoring once activity concentrations fall subsequently. The main plant vent monitor uses a sampling nozzle assembly that has flow and temperature sensors. Plant vent flow measurements are also provided to allow for calculation of total flow. The monitor at MP 8 also includes the capability to collect grab samples for further analysis in the on-site laboratory.

The turbine building vent discharge monitor (MP 10) is an in-line monitor for krypton-85 and xenon-133, using two beta/gamma-sensitive Geiger-Mueller tubes with overlapping detector ranges. The main purpose of this monitor is to provide data for discharge reporting in the event of a primary-to-secondary coolant leak. In addition, MP 9 provides nitrogen-16 detectors that are sensitive for detecting primary-to-secondary coolant leakage and are located near the steam generator main steam outlet and upstream of the turbine. The facility to collect manual grab samples is provided at two points before the turbine vent effluents converge and are released to the atmosphere; at the condenser air removal system and at the gland seal system. The grab samples can be analysed in the on-site laboratory.

The monitoring equipment and sample point will be located in the upper-most floors of the auxiliary building near the base of the main stack structure. They will be located on a purpose built skid, the design of which is still being developed. The design criteria are for the monitor installations and sample line connections to be located such that sufficient access is provided to easily service the monitor or sample line connection. The design of the skid will contain additional space around the front and rear of the equipment to allow access for maintenance activities.

There are additional MPs (MPs 1-7) upstream of the main plant vent that provide information that allow operators to locate the origin of any abnormal releases prior to discharge to the main plant vent. In addition, these internal vent monitors function as backup monitors for the main plant vent. They are discussed in the following paragraphs.

The radwaste building exhaust monitor (MP 1) and the health physics and hot machine shop exhaust monitor (MP 3) are off-line particulate monitors for measuring caesium-137 and strontium-90. Beta-sensitive scintillation detectors viewing fixed particulate filters are used in each. The monitors are located downstream of the exhaust fans. The primary purpose of these monitors is to alert the control room should activity concentrations exceed a predetermined set point.

The annex building (MP 2), containment air filtration and exhaust (MP 4), auxiliary building (MP 5), and fuel handling area (MP 6) exhaust radiation monitors are in-line monitors for krypton-85 and xenon-133, using beta-sensitive scintillation detectors. The monitors are located upstream of isolation valves which, when triggered, can reroute the waste streams through filtration exhaust units. The primary purpose of these monitors is to alert the control room should activity concentrations exceed a predetermined set-point.

The gaseous radwaste exhaust radiation monitor (MP 7) is an in-line monitoring detector for krypton-85 and xenon-133, using a beta-sensitive scintillation detector. The monitor is situated before the discharge reaches the plant vent or is diluted by any other flows. The detection range is higher than that of MP 2, 5, and 6. The primary purpose of this monitor is to alert the control room should activity concentrations exceed a predetermined set-point.

The monitoring of tritium and carbon-14 is currently not included in the design of the continuous monitoring systems. Typically, tritium and carbon-14 activity concentrations in the plant vent will be below minimum detectable levels of detectors. In order to determine their activity levels in aerial discharges from NPPs, off-line bubbler systems will be used to concentrate levels over several days or weeks to obtain measurable activity concentrations. The bubbler system, required for sampling of tritium and carbon-14, can be incorporated into the design of the main stack sampling system, depending on utility preference. The laboratory equipment used to analyse these samples is utility-operator-specific.

6.2.1.2 Water Discharges

There are two discharge streams for liquid effluents at the **AP1000** NPP, all of which are released into the environment via the seawater cooling return outfall.

1. Liquid Radwaste System (WLS)

   Treated liquid radwaste is collected in one of six monitor tanks before discharge to the seawater cooling return (see subsection 3.4.3.6). Prior to discharge to the sea, a grab sample is taken from the filled monitoring tank to be discharged. In order to obtain a representative sample, each tank is equipped with a recirculation line which has a sampling point (W1, W2, W3, W4, W5, or W6). The pump suction is taken from the bottom of the tank and returned to the top as quickly as possible to minimise the time needed for mixing. This will ensure the contents of the tank are fully mixed prior to sampling and will ensure a representative sample is obtained. The sample is dispatched for laboratory for analysis to confirm that radionuclide concentrations and activity levels are within acceptable limits.

   Effluent which does not meet the discharge limit can be returned to a waste holdup tank or recirculated directly through the filters and ion exchangers for further treatment.

   Effluent meeting discharge limits for radioactivity is pumped (design flow rate of ~22.7m^3/h) from the monitor tanks, batchwise, to the sea water cooling return sump through a common discharge line. It is mixed with the cooling water return of the CWS. In addition, the discharge line contains a radiation monitor (W7) with diverse methods of stopping the discharge. The first method closes an isolation valve in the discharge line, which prevents any further discharge from the WLS. The valve automatically closes and an alarm is actuated if the activity in the discharge stream reaches the monitor set point. The second method stops the monitor tank pumps.

   It should also be noted that the **AP1000** NPP design will be able to accommodate various systems such as administrative procedures and/or interlocks to avoid (a) an inadvertent discharge from the filled monitoring tank prior to sampling and confirmatory analysis and (b) discharge of the tank simultaneous to filling of the tank or other operations such as transfers between tanks or retreatment. The system chosen is site/utility operator-specific.

2. Waste Water System (WWS)

   The non-radwaste water, during normal plant operation and during plant outages, is handled by the WWS (see Section 4.2.6). Water is collected in the turbine building sumps, with a temporary storage capacity and a controlled source of fluid flow to the oil separator and WWRB.

   A radiation monitor (W9) located on the common discharge piping of the sump pumps provides an alarm upon detection of radioactivity in the waste water. In the event radioactivity is present in the turbine building sumps, the waste water is diverted from the sumps to the WLS for processing and disposal. The radiation monitor also trips the sump pumps on detection of radioactivity to isolate the contaminated waste water. Provisions are included for sampling the sumps (W8).

The waste water collected in the WWRB is sampled (W10) to confirm that radionuclide concentrations and activity levels are within acceptable limits. Waste water that complies with discharge limits will be released intermittently via the seawater cooling return sump for final discharge via the plant outfall to the sea.

A further MP (W11) for the final monitoring of the liquid wastes from minor discharge routes will be part of the Waste Water Basin design which is site-specific.

In addition to the waste water effluent streams, the sea water used to provide once-through non-contact sea water cooling for the SWS and the CWS will be sampled. Both cooling systems will allow grab samples (W12, W14) to be taken from the turbine building to enable chemical analysis of the sea discharge flows. The secondary side service water blowdown will be non-radioactive under normal operating conditions and will be discharged to the WWS or to the CWS. However, under a fault condition, the service water blowdown may become radioactive. If the radiation levels sensed by radiation monitor (W13) are above acceptable limits for sea discharge, it will require treatment (e.g., by the WLS). It should be noted that the design of the SWS and the CWS, including the location of the sampling and MPs, is site-specific.

The radiation monitors at W7, W9, and W13 are expected to be gamma sensitive, thallium activated, silver iodide scintillation counters. These will be set to detect Cs-137 in a concentration range of 3.7E04 to 3.7E08 Bqm^{-3} (2.8E-08 to 2.8E-04 $Cift^{-3}$). Specifications for the precision, bias, and availability/reliability will be developed at a later stage in the design. Commercially available RMS equipment will be used in the design. The **AP1000** NPP design will allow the data from the radiation monitors to be collected by the instrument and fed to the control system. The method by which the data is collated and reported to the EA is utility-operator-specific. The method for determining how the alarm threshold will be set has not been determined.

Each discharge line will be fitted with a flow meter. The instrument that will be used for flow rate measurement has not been specified. When the instrument is specified, the register of MCERTS certified equipment will be reviewed to determine if a suitable instrument is available on the register. The **AP1000** NPP design will allow the data from the flow meter to be collected by the instrument and fed to the control system. The method by which the data is collated and reported to the EA is utility-operator-specific.

The current **AP1000** NPP design will be able to accommodate both grab sampling as well as a proportional sampling in order to obtain a representative sample. Although the specification for the flow proportional sampler is utility-operator-specific, the equipment required to take a representative sample from each of the discharge lines will include a commercially available stationary waste water sampler equipped with an ancillary signal connection and a pressurised pipeline interface. These commercially available samplers have a range of programmable options for setting the frequency and size of sample volume. These will be capable of providing a representative sample from each of the discharge lines over a suitable time period.

The **AP1000** NPP reactor design will be able to accommodate a separate proportional sampler for each discharge line, which can be secured to allow an independent sample to be taken for regulatory purposes. Proportional samplers with a lockable cabinet are commercially available. All sampling and monitoring equipment will be housed in weather-shielded buildings and will be located in areas where access is controlled.

The **AP1000** NPP will have an on-site laboratory with the capability to be UKAS accredited to ISO17025. The utility operator will specify the equipment for the laboratory and implement the processes necessary to achieve ISO17025 accreditation. Tritium activity concentrations in the liquid effluent are likely to be below minimum detectable levels of on-line continuous detectors. The assessment of tritium can be carried out through analysis of grab samples.

Figure 6.2-2 provides a schematic overview of the liquid effluent processes. The exact location and total number of sample and MPs are site-specific.

Table 6.2-4 lists the monitors that are associated with each of the liquid discharge streams. The tables include information on the type of monitor, radionuclides monitored, and detection ranges. The MP (W) numbers relate to Figure 6.2-2.

6.2.1.3 Solid Waste

LLW packages will be put through an LRGS in order to "fingerprint" each waste package. Fingerprint analysis is the common name used for the practice of determining the range of activities and isotopes present in a waste stream. By building up a "fingerprint" of the isotopes from measurements in a consistent waste stream, it is possible to measure one isotope (e.g., Co-60 or Cs-137) and infer the presence and relative proportion of a range of other isotopes. So, when an LRGS is used, although some active isotopes may have no or very low gamma emissions and the LRGS can not 'see' them, their presence can be inferred.

Every ILW waste package will be "finger printed" using an HRGS within the ILW store to monitor its activity level before it is transferred to the ILW store. Waste package inventory records will be completed according to the required regulations to maintain an inventory record of each waste package and its location within the ILW store. The ILW store will be fitted with alpha and beta/gamma monitors to detect any leaks in activity.

The facility will exist to visually observe the waste packages within the ILW store via CCTV. If a waste package shows evidence of deviating from the specification during storage, i.e., via corrosion or damage, it will be placed in an allotted area within the ILW store and potentially put into a SCV.

There will be alpha and beta/gamma monitors within the auxiliary building to monitor the radioactivity during spent fuel handling operations and over the course of its storage within the cooling pond.

6.2.1.4 Dosimetry

Authorised discharges from NPPs can result in a small increase in public radiation exposure because of external exposure, inhalation, and the incorporation of radionuclides into food. A monitoring programme will be established to monitor on-site worker and off-site populations on a regular basis to ensure that exposures to radioactivity are within limits.

The programme for off-site monitoring will be established according to the site-specific conditions and the pathways that may result in people becoming exposed to radioactivity released into the environment. These pathways include:

In gaseous form

- inhalation of airborne radioactivity
- ingestion of food containing deposited radioactive materials
- external exposure from airborne radioactivity
- external exposure from radioactive material deposited on the ground

In liquid form

- ingestion of radioactivity incorporated into seafood
- external exposure from radioactivity deposited on inter-tidal areas
- external exposure from handling contaminated items

Examples of the type of environmental monitoring that may be carried out include analysis of milk, meat, grass, vegetables, soil, beach material, silt, sediment, fish, shellfish, and seaweed. Laboratory analysis of these samples will be dictated by the major releases

The monitoring will include determination of the contribution to the annual total dose of the maximum exposed individual from radioactive emissions and other nearby radioactive sources. Comparisons will be made with theoretical dose calculations.

6.2.2 Hydrological Monitoring

6.2.2.1 Groundwater

The groundwater monitoring scheme will be developed on the basis to the site-specific design and an understanding to the site-specific groundwater flow regime.

The groundwater monitoring scheme would typically include the following features:

- Installation of groundwater monitoring boreholes encircling the power plant site a distance of 100 m – 200 m (330 ft – 660 ft) from the plant.

- Separation of boreholes by distances of 50 m – 100 m (160 ft to 330 ft) on the down hydraulic gradient side of the site and 100 m – 200 m (330 ft – 660 ft) on the up hydraulic gradient of the site.

- Location of an up hydraulic gradient borehole on the site boundary as a source of reference.

- Groundwater monitoring wells with monitoring response zones in all of the identified groundwater bodies.

- Groundwater monitoring locations positioned in the proximity to any site activities thought to represent a higher than normal contamination risk to the subsurface (e.g., diesel storage).

- Sampling from any surface water feature that was located within the catchment area of the proposed development. Samples would be required from one upstream and one downstream location from the site.

- A quarterly groundwater monitoring programme with the aim of detecting any emissions entering or leaving the site.

Areas of the site to be specifically considered for the groundwater monitoring programme are:

- West of the auxiliary building in the area of the fuel transfer canal (which includes an outside wall)
- West and south of the radwaste building (which incorporates a curbed basemat, but does not have the monolithic basemat / wall nature of the auxiliary building)
- East of the auxiliary building rail bay and the radwaste building truck doors
- Close proximity to the diesel storage tanks

All assumptions relating to potential groundwater flow and subsurface permeability will be as a direct result of a conceptual site model. A site-specific conceptual site model will be developed prior to any detailed design of the groundwater monitoring programme.

6.2.2.2 Surface Water

There are no surface water features identified for the generic site; hence, no monitoring programme is proposed. A monitoring programme for adjacent surface water bodies may be developed if there is a potential impact at a specific site.

6.2.3 Ecological Monitoring

The impact of aerial and liquid effluent discharges on sensitive ecological receptors will be monitored on a regular basis. This monitoring will be site-specific and take into account the presence of sensitive and indicator species present at the site. The types of organisms that may be monitored include those shown in Table 5.1-4.

For example, polychaete worms inhabit marine sediments and provide the basis of the food chain for many higher marine species. As sessile species, they are predicted to receive the highest marine organism dose rate from the **AP1000** NPP discharges (see Section 5.3.3). They are also valuable biomonitors, quickly showing sensitivity to contaminants such as heavy metals through bioaccumulation. Environmental impacts can therefore be monitored through changes in the species richness, abundance, and community composition of the polychaete worms.

6.2.4 Thermal Monitoring

The temperature of cooling water discharges will be regularly monitored. The main cooling water will be returned to sea at a temperature differential of ~8°C (14°F) (see subsection 4.2.3.3). The heat will be dissipated as rapidly as possible by suitable design and location of the discharge point at each site. The dispersion of heat and the thermal plume will be characterised to identify the area where significant impacts may occur on benthic and pelagic species.

6.2.5 Chemical Monitoring

A programme of laboratory testing of the waste water will be established. Typical parameters that would be monitored include: pH, Eh, conductivity, temperature, oil and grease, total organic carbon, trace metals (e.g., copper, nickel, zinc, chromium, iron, cadmium, and mercury), phosphate, and ammonia.

Chemical monitoring and controlled (automatic) dosing of sodium hypochlorite will be carried out to ensure the minimum required dose is applied to control biofouling in the cooling water systems. Since the applied hypochlorite concentration will decrease through the CWS, chemical monitors are needed to ensure effective concentrations at critical points in the system. The chlorine residual in the cooling water discharge will be monitored.

Laboratory testing of by-products of chlorination will be carried out (e.g., trihalomethanes and halogenated acetic acids) to determine their concentrations in the seawater cooling outfall.

Chemical monitoring will be used in conjunction with the MCERTS flow monitoring (see subsection 6.2.1.2) to determine mass releases of non-radioactive pollutants from the **AP1000** NPP.

6.2.6 Meteorological Monitoring

Meteorological monitoring carried out on site will typically involve continuous instrumental monitoring of wind speed and direction, dry-bulb temperature, dew point temperature and rainfall.

6.3 Ground Information Required Before Construction

A variety of ground information will be gathered for the specific site before construction. This includes the following regional and site-specific geological, seismological, and geophysical information as well as conditions caused by human activities:

- Structural geology of the site
- Seismicity of the site
- Geological history
- Evidence of paleoseismicity
- Site stratigraphy and lithology
- Engineering significance of geological features
- Site groundwater conditions
- Dynamic behavior during prior earthquakes
- Zones of alteration, irregular weathering, or structural weakness
- Unrelieved residual stresses in bedrock
- Materials that could be unstable because of mineralogy or unstable physical properties
- Effect of human activities in the area

In addition to establishing the geotechnical requirements of the site, the nature of the site will need to be characterised to identify the likely behaviour of past or potential future contamination. This will take the form of a desk study which will determine historical land use (greenfield or brownfield) and to develop a conceptual site model which identifies contaminant-path-receptor relationships that may exist. The Phase 1 desk study would include:

- Historical land use of the site and surrounding area
- Geology and surface deposits
- Hydrogeology and hydrology of area (including flood risk)
- Characterisation of natural, historic, current, and potential future sources of ground contamination
- Pathways by which contaminants might migrate into, out of, and through the site
- Potential receptors that could be harmed by ground contamination (e.g., people, controlled waters, ecological, buildings, and building materials)
- Definition of the limitations and uncertainties relating to site information

The conceptual model developed at Phase 1 would identify potentially significant contaminant-pathway-receptor linkages and these may require further investigation for existing issues. For potential future releases, the results of the Phase 1 work would be used to determine appropriate management and control strategies to prevent and minimise impacts.

The desk study may need to be supplemented by a Phase 2 intrusive investigation to better define any existing site contamination and to improve understanding of the groundwater regime. This may include, but is not limited to:

- Investigation of ground by appropriate techniques e.g. trial pits, boreholes
- Collection and analysis of soil samples
- Determination particle size distribution, porosity, and permeability of soils and underlying bedrock
- Installation of standpipes for monitoring of gas and groundwater
- Baseline monitoring, sampling, and analysis of groundwater over appropriate time periods (typically 3 to 12 months)
- Use of groundwater level sensors to determine influences of groundwater level (e.g. rainfall and tidal effects)
- Baseline monitoring of gas over appropriate time periods (typically 6 to 24 months)
- Sampling and analysis of surface waters over appropriate time periods (typically 3 to 12 months)

The results would identify the need for any site remediation and/or further management and control strategies for the proposed development.

6.4 References

6-1 Science Report SC010034/SR, "Developing Guidance for Setting Limits on Radioactive Discharges to the Environment from Nuclear-Licensed Sites," Environment Agency, December 2005.

6-2 UKP-GW-GL-028, Rev. 2, "Proposed Annual Limits for Radioactive Discharge," Westinghouse Electric Company LLC, March 2011.

6-3 "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, Sizewell B Power Station," Environment Agency, December 2006.

6-4 UKP-GW-GL-029, Rev. 0, "**AP1000** Generic Design Measurement and Assessment of Discharges," Westinghouse Electric Company LLC, February 2009.

6-5 2004/2/Euratom, "Commission Recommendation 2004/2/Euratom of 18 December 2003 on standardised information on radioactive airborne and liquid discharges into the environment from nuclear power reactors and reprocessing plants in normal operation," Official Journal of the European Communities, pp. 36-46, January 2004.

6-6 Technical Guidance Note M11, "Monitoring of Radioactive Releases to Atmosphere from Nuclear Facilities," Environment Agency, 1999.

6-7 Technical Guidance Note M12, "Monitoring of Radioactive Releases to Water from Nuclear Facilities," Environment Agency, 1999.

Table 6.1-1

ISOTOPIC SELECTION CRITERIA FOR AIR EMISSION LIMIT CALCULATION

Selection Criteria	Isotopes Selected
Significant in terms of their radiological impact	>1% contribution to fisherman family dose (μSv / 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(1) 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 6.1-2

ISOTOPIC SELECTION CRITERIA FOR LIQUID DISCHARGE LIMIT CALCULATIONS

Selection Criteria	Isotopes Selected
Significant in terms of their radiological impact	>1% contribution to fisherman family dose (µSv / 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

Table 6.1-3

PREDICTED MONTHLY AIR RADIATION EMISSIONS DURING 18-MONTH CYCLE

Month	Predicted Monthly Air Radiation Discharges (TBq)													
	Radio Iodines	Noble Gases	Tritium	C-14	Ar-41	Co-60	Kr-85	Sr-90	I-131	Xe-131m	Xe-133	Cs-137	Other Particulate	Total
1	4.96E-05	0.298	0.132	0.045	0.093	2.69E-07	0.081	3.70E-08	1.73E-05	0.0513	0.090	1.11E-07	1.02E-06	0.568
2	4.96E-05	0.305	0.132	0.045	0.094	2.69E-07	0.085	3.70E-08	1.73E-05	0.0528	0.091	1.11E-07	1.02E-06	0.575
3	4.96E-05	0.312	0.132	0.045	0.094	2.69E-07	0.090	3.70E-08	1.73E-05	0.0546	0.091	1.11E-07	1.02E-06	0.583
4	4.96E-05	0.320	0.133	0.046	0.094	2.69E-07	0.095	3.70E-08	1.73E-05	0.0566	0.091	1.11E-07	1.02E-06	0.592
5	4.96E-05	0.329	0.134	0.046	0.095	2.69E-07	0.101	3.70E-08	1.73E-05	0.0589	0.092	1.11E-07	1.02E-06	0.602
6	4.96E-05	0.339	0.134	0.046	0.095	2.69E-07	0.108	3.70E-08	1.73E-05	0.0616	0.093	1.11E-07	1.02E-06	0.614
7	4.96E-05	0.351	0.135	0.046	0.096	2.69E-07	0.117	3.70E-08	1.73E-05	0.0647	0.093	1.11E-07	1.02E-06	0.628
8	4.96E-05	0.366	0.136	0.046	0.096	2.69E-07	0.127	3.70E-08	1.73E-05	0.0683	0.094	1.11E-07	1.02E-06	0.644
9	4.96E-05	0.383	0.137	0.047	0.097	2.69E-07	0.138	3.70E-08	1.73E-05	0.0727	0.095	1.11E-07	1.02E-06	0.664
10	4.96E-05	0.404	0.138	0.047	0.098	2.69E-07	0.153	3.70E-08	1.73E-05	0.0780	0.096	1.11E-07	1.02E-06	0.687
11	4.96E-05	0.430	0.140	0.048	0.099	2.69E-07	0.171	3.70E-08	1.73E-05	0.0847	0.098	1.11E-07	1.02E-06	0.717
12	4.96E-05	0.463	0.142	0.048	0.101	2.69E-07	0.194	3.70E-08	1.73E-05	0.0932	0.100	1.11E-07	1.02E-06	0.755
13	4.96E-05	0.508	0.145	0.050	0.103	2.69E-07	0.224	3.70E-08	1.73E-05	0.105	0.102	1.11E-07	1.02E-06	0.805
14	4.96E-05	0.570	0.149	0.051	0.105	2.69E-07	0.267	3.70E-08	1.73E-05	0.120	0.105	1.11E-07	1.02E-06	0.875
15	4.96E-05	0.662	0.155	0.053	0.110	2.69E-07	0.330	3.70E-08	1.73E-05	0.144	0.111	1.11E-07	1.02E-06	0.980
16	4.96E-05	0.815	0.165	0.056	0.117	2.69E-07	0.437	3.70E-08	1.73E-05	0.183	0.119	1.11E-07	1.02E-06	1.152
17	4.96E-05	1.117	0.184	0.063	0.130	2.69E-07	0.644	3.70E-08	1.73E-05	0.259	0.136	1.11E-07	1.02E-06	1.494
18	4.96E-05	2.031	0.242	0.083	0.171	2.69E-07	1.269	3.70E-08	1.73E-05	0.492	0.187	1.11E-07	1.02E-06	2.527
Total	8.93E-04	10.001	2.664	0.910	1.887	4.85E-06	4.662	6.66E-07	3.11E-04	2.054	1.887	2.00E-06	1.83E-05	15.463

Table 6.1-4

PREDICTED MONTHLY LIQUID DISCHARGES OF RADIOISOTOPES DURING 18 MONTH FUEL CYCLE

	Predicted Monthly Liquid Radiation Discharges (TBq)											
Month	**Tritium**	**Non-Tritium**	**C-14**	**Fe-55**	**Co-58**	**Co-60**	**Ni-63**	**Sr-90**	**Cs-137**	**Pu-241**	**Other Isotopes**	**Total**
1	2.473	1.43E-04	8.14E-05	1.20E-05	1.02E-05	5.62E-06	1.30E-05	5.96E-09	5.66E-07	1.99E-09	1.63E-05	2.473
2	2.481	1.52E-04	8.62E-05	1.27E-05	1.08E-05	5.96E-06	1.37E-05	6.33E-09	5.96E-07	2.11E-09	1.71E-05	2.481
3	2.489	1.61E-04	9.14E-05	1.35E-05	1.14E-05	6.33E-06	1.45E-05	6.70E-09	6.33E-07	2.23E-09	1.81E-05	2.489
4	2.498	1.71E-04	9.73E-05	1.44E-05	1.22E-05	6.73E-06	1.54E-05	7.14E-09	6.73E-07	2.38E-09	1.92E-05	2.499
5	2.509	1.83E-04	1.04E-04	1.54E-05	1.30E-05	7.18E-06	1.65E-05	7.66E-09	7.22E-07	2.55E-09	2.05E-05	2.509
6	2.522	1.97E-04	1.12E-04	1.65E-05	1.40E-05	7.73E-06	1.78E-05	8.25E-09	7.73E-07	2.75E-09	2.20E-05	2.522
7	2.536	2.13E-04	1.22E-04	1.78E-05	1.51E-05	8.36E-06	1.92E-05	8.92E-09	8.36E-07	2.97E-09	2.37E-05	2.537
8	2.554	2.32E-04	1.32E-04	1.94E-05	1.65E-05	9.10E-06	2.09E-05	9.73E-09	9.10E-07	3.24E-09	2.57E-05	2.554
9	2.574	2.55E-04	1.46E-04	2.13E-05	1.81E-05	9.99E-06	2.29E-05	1.07E-08	9.99E-07	3.56E-09	2.81E-05	2.574
10	2.599	2.83E-04	1.62E-04	2.36E-05	2.00E-05	1.11E-05	2.54E-05	1.18E-08	1.11E-06	3.96E-09	3.11E-05	2.600
11	2.631	3.17E-04	1.82E-04	2.65E-05	2.25E-05	1.24E-05	2.85E-05	1.33E-08	1.24E-06	4.44E-09	3.47E-05	2.631
12	2.671	3.61E-04	2.07E-04	3.02E-05	2.56E-05	1.42E-05	3.25E-05	1.52E-08	1.42E-06	5.07E-09	3.96E-05	2.671
13	2.724	4.22E-04	2.41E-04	3.51E-05	2.97E-05	1.65E-05	3.77E-05	1.77E-08	1.65E-06	5.88E-09	4.59E-05	2.724
14	2.798	5.03E-04	2.88E-04	4.18E-05	3.55E-05	1.96E-05	4.51E-05	2.12E-08	1.97E-06	7.07E-09	5.44E-05	2.799
15	2.909	6.25E-04	3.59E-04	5.22E-05	4.40E-05	2.45E-05	5.62E-05	2.63E-08	2.45E-06	8.77E-09	6.73E-05	2.909
16	3.092	8.25E-04	4.74E-04	6.88E-05	5.85E-05	3.23E-05	7.44E-05	3.49E-08	3.23E-06	1.16E-08	8.88E-05	3.092
17	3.453	1.22E-03	7.07E-04	1.02E-04	8.66E-05	4.81E-05	1.10E-04	5.18E-08	4.81E-06	1.72E-08	1.31E-04	3.455
18	4.548	2.43E-03	1.40E-03	2.03E-04	1.72E-04	9.51E-05	2.18E-04	1.03E-07	9.51E-06	3.43E-08	2.60E-04	4.550
Total	50.061	8.70E-03	5.00E-03	7.25E-04	6.14E-04	3.41E-04	8.33E-04	3.70E-07	3.42E-05	1.22E-07	9.44E-04	50.070

Table 6.1-5

CALCULATED ANNUAL LIMITS FOR AIR EMISSIONS

Air Effluent Input	Representative 12-Month Plant Discharge (D) (TBq/y)	Worst Case Plant Annual Discharges (WCPD) (TBq/y)	Calculated Annual Limit (TBq/y)
Radioiodines[(1)]	5.95E-04	9.82E-04	1E-03
Noble Gases[(2)]	8.099	13.363	13
Tritium	1.867	3.081	3
Carbon-14	0.638	1.053	1
Argon-41	1.323	2.182	2
Cobalt-60	3.22E-06	5.32E-06	5E-06
Krypton-85	4.070	6.716	7
Strontium-90	4.44E-07	7.33E-07	7E-07
Iodine-131	2.07E-04	3.42E-04	3E-04
Xenon-131m	1.76	2.91	3
Xenon-133	1.335	2.203	2
Caesium-137	1.33E-06	2.20E-06	2E-06
Other particulates	1.22E-05	2.01E-05	2E-05
Total Beta particulate[(3)]	1.72E-05	2.84E-05	3E-05
Total	11.928	19.681	20

Notes:
1. Radioiodines include I-131 and I-133.
2. Noble gases include Kr-85m, Kr-85, Kr-87, Kr-88, Kr-85, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, Xe-137, Xe-138.
3. Total beta particulate include Co-60 + Sr-90 + Cs-137 + other particulates.

Table 6.1-6

CALCULATED ANNUAL LIMITS FOR LIQUID DISCHARGES

Liquid Effluent Input	Representative 12-Month Plant Discharge (D) (TBq/y)	WCPD (TBq/y)	Calculated Annual Limit (TBq/y)
Tritium	35.09	57.90	60
Non-tritium	7.70E-03	1.27E-02	1E-02
Carbon-14	4.42E-03	7.30E-03	7E-03
Iron-55	6.42E-04	1.06E-03	1E-03
Cobalt-58	5.44E-04	8.97E-04	9E-04
Cobalt-60	3.01E-04	4.97E-04	5E-04
Nickel-63	6.91E-04	1.14E-03	1E-03
Strontium-90	3.24E-07	5.35E-07	5E-07
Caesium-137	3.01E-05	4.97E-05	5E-05
Plutonium-241	1.08E-07	1.78E-07	2E-07
Other isotopes(1)	1.07E-03	1.76E-03	2E-03
Total	35.104	57.922	60

Note:
1. Other isotopes = Non-tritium isotopes – (C-14+ Fe-55+Co-58+Co-60+Ni-63+Sr-90+Cs-137+Pu-241).

Table 6.1-7

COMPARISON OF PROPOSED AIR EMISSION LIMITS WITH SIZEWELL B PWR

Air Emission	AP1000 NPP Calculated Annual Limit (TBq/y)	AP1000 NPP Proposed Annual Limit (TBq/y)	Sizewell B Environment Agency New Limit [Reference 6-3] (TBq/y)
Radioiodines(1)	1E-03	1E-03	–
Noble Gases(2)	13	13	30
Tritium	3	3	3
Carbon-14	1	1	0.5
Argon-41	2	2	–
Cobalt-60	5E-06	–	–
Krypton-85	7	–	–
Strontium-90	7E-07	–	–
Iodine-131	3E-04	3E-04	5.0E-04
Xenon-131m	3	–	–
Xenon-133	2	–	–
Caesium-137	2E-06	–	–
Other particulates	2E-05	–	–
Total Beta particulates(3)	3E-05	3E-05	1.0E-04

Notes:
1. Radioiodines include I-131 and I-133.
2. Noble gases include Kr-85m, Kr-85, Kr-87, Kr-88, Kr-85, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-135, Xe-137, Xe-138.
3. Total beta particulate include Co-60 + Sr-90 + Cs-137 + other particulates.

Table 6.1-8

COMPARISON OF PROPOSED LIQUID DISCHARGE LIMITS WITH SIZEWELL B PWR

Air Emission	AP1000 NPP Calculated Annual Limit (TBq/y)	AP1000 NPP Proposed Annual Limit (TBq/y)	Sizewell B Environment Agency New Limit [Reference 6-3] (TBq/y)
Tritium	60	60	80
Non-tritium	1E-02	–	–
Carbon-14	7E-03	7E-03	–
Iron-55	1E-03	–	–
Cobalt-58	9E-04	–	–
Cobalt-60	5E-04	–	–
Nickel-63	1E-03	–	–
Strontium-90	5E-07	–	–
Caesium-137	5E-05	–	0.02
Plutonium-241	2E-07	–	–
Other isotopes (1)	2E-03	–	–
All isotopes without other limits	5E-03(2)	5E-03(2)	0.13(3)

Notes:
1. Other isotopes = Non-tritium isotopes – (C-14+ Fe-55+Co-58+Co-60+Ni-63+Sr-90+Cs-137+Pu-241).
2. All isotopes without other limits = Non-tritium isotopes – C-14.
3. All isotopes without other limits = Non-tritium isotopes – Cs-137.

Table 6.1-9

COMPARISON OF AIR EMISSION LIMITS WITH UK AGR SITES

Air Emission	AP1000 NPP Calculated Limits	AP1000 NPP Proposed Limits	Dungeness B	Hartlepool	Heysham 1	Heysham 2	Hinkley Point B
	(TBq/y)	(TBq/y)	(TBq/y)	(TBq/y)	(TBq/y)	(TBq/y)	(TBq/y)
Radioiodines(1)	1E-03	1E-03	1.50E-03	1.50E-03	1.50E-03	1.50E-03	1.50E-03
Noble Gases(2)	13	13	–	–	–	–	–
Tritium	3	3	12	10	10	10	12
Carbon-14	1	1	3.7	4.5	4.5	3.7	3.7
Argon-41	2	2	75	150	150	75	100
Cobalt-60	5E-06	–	–	–	–	–	–
Krypton-85	7	–	–	–	–	–	–
Strontium-90	7E-07	–	–	–	–	–	–
Iodine-131	3E-04	3E-04	–	–	–	–	–
Xenon-131m	3	–	–	–	–	–	–
Xenon-133	2	–	–	–	–	–	–
Caesium-137	2E-06	–	–	–	–	–	–
Other particulates(3)	2E-05	–	–	–	–	–	–
Beta particulates	3E-05	3E-05	–	–	–	–	–

Notes:
1. Radiodine = I-131 + I-133.
2. Noble Gases = Kr-85m + Kr-85 + Kr-87 + Kr-88 + Xe-131m + Xe-133m + Xe-133 + Xe-135m + Xe-136 + Xe-137 + Xe-138.
3. Other particulate = Total beta particulate - Co-60 - Sr-90 - Cs-137.

Table 6.1-10

COMPARISON OF LIQUID DISCHARGE LIMITS WITH UK AGR SITES

Air Emission	AP1000 NPP Calculated Limits	AP1000 NPP Proposed Limits	Dungeness B	Hartlepool	Heysham 1	Heysham 2	Hinkley Point B
	(TBq/y)	(TBq/y)	(TBq/y)	(TBq/y)	(TBq/y)	(TBq/y)	(TBq/y)
Tritium	60	60	650	650	650	650	650
Non-tritium[(1)]	1E-02	–	–	–	–	–	–
Carbon-14	7E-03	–	–	–	–	–	–
Iron-55	1E-03	–	–	–	–	–	–
Cobalt-58	9E-04	–	–	–	–	–	–
Cobalt-60	5E-04	5E-04	0.01	0.01	0.01	0.01	0.01
Nickel-63	1E-03	–	–	–	–	–	–
Strontium-90	5E-07	–	–	–	–	–	–
Caesium-137	5E-05	5E-05	0.1	0.1	0.1	0.1	0.1
Plutonium-241	2E-07	–	–	–	–	–	–
Other Particulates[(2)]	1E-03	1E-03	0.08	0.08	0.08	0.08	0.08

Notes:
1. Non-tritium = All isotopes – tritium.
2. Other particulate = Total beta particulate - Fe-55 - Co-58 - Co-60 – Ni-63 - Sr-90 - Cs-137 - Pu-241.

Table 6.2-1

MONITORING PROGRAMMES

Monitoring Programme	Descriptions
Radiological Monitoring	Collection of environmental samples (from air, water, sediment, fish and food products, as well as direct radiation levels) to determine the concentrations of radioactive constituents in the samples. Monitoring of annual total dose contributions to the maximum exposed individual from radioactive emissions and other nearby radioactive sources. Monitoring of on-site worker and off-site populations on a regular basis to ensure that exposures to radioactive are within limits.
Hydrological Monitoring	Periodic monitoring and subsequent sediment removal for maintenance from the cooling water system intake channel to minimise any impact to the raw water system operation. Bathymetric survey of the intake channel is expected after first year of operation to measure sediment build up and also to determine future dredging intervals. Monitoring of surface water and groundwater parameters are expected quarterly for the first year of operation, then annually. Operational monitoring concentrates on parameters are below: Surface water flow Groundwater flow Impact of sanitary and chemical waste retention methods on water quality Sediment transport Floodplain and wetlands
Ecological Monitoring	Procedures to monitor terrestrial species and habitats that could be adversely affected. Sampling and monitoring procedures on fish and aquatic species, and habitats that could be adversely affected by the intake or discharge of cooling water or other operational impacts.
Thermal Monitoring	Routine thermal monitoring of waste water discharges (specifically outfall, blow down, and electric power generation discharges).
Chemical Monitoring	Monitoring of discharges made through outfall for consistency. Monitoring of physical, biological, and chemical attributes. Monitoring of tanks containing oil or hazardous substances during tank filling operations. Monitoring procedures of continuous leak detection systems. Inspections to verify that hazardous waste is treated, stored, and disposed of.
Meteorological Monitoring	Collection and monitoring of data on-site conditions which includes wind speed and direction, dry-bulb temperature, dew point temperature, and rainfall.

Table 6.2-2

AP1000 NPP AERIAL EFFLUENT MONITORS AND DETECTION RANGES – MAIN PLANT VENT

Monitor Point (Fig 6.2-1)	Description	Detector	Category	Type	Isotopes	Nominal Detection Range		Purpose
						Minimum (Bq/m^3)	Maximum (Bq/m^3)	
8	Main Plant Vent	VFS-JE-RE101	Continuous off-line	Beta-sensitive scintillation detectors viewing fixed particulate filters	Sr-90, Cs-137	3.7E-02	3.7E+03	Normal operation range
		N/A	Continuous off-line	Manual removal of filter cartridges for analysis	Sr-90, Cs-137	Laboratory(1)		Accident range
		VFS-JE-RE102	Continuous off-line	Gamma-sensitive thallium-activated, sodium iodide scintillation counter	I-131	3.7E-01	3.7E+04	Normal operation range
		N/A	Continuous off-line	Manual removal of filter cartridges for analysis	I-131	Laboratory(1)		Accident range
		N/A	Grab sample	Manual collection of effluent sample for laboratory analysis	noble gases iodine particulates tritium	Laboratory(1)		n/a
		VFS-JE-RE103	Continuous in-line	Beta-sensitive scintillation detector	Kr-85, Xe-133	3.7E+03	3.7E+08	Normal operation range
		VFS-JE-RE104A	Continuous in-line	Beta/gamma-sensitive detector	Kr-85, Xe-133	3.7E+06	3.7E+12	Accident (mid) range
		VFS-JE-RE104B	Continuous in-line	Beta/gamma-sensitive detector	Kr-85, Xe-133	3.7E+09	3.7E+15	Accident (mid) range

Note:
1. Samples will be taken to the on-site chemical laboratory for analysis. Laboratory equipment has not been specified at this stage of the design, as it will be determined by the future plant operator.

Table 6.2-3

AP1000 NPP AERIAL EFFLUENT MONITORS AND DETECTION RANGES – TURBINE BUILDING VENT

Monitor Point (Fig 6.2-1)	Description	Detector	Category	Type	Isotopes	Nominal Detection Range		Purpose
						Minimum (Bq/m^3)	Maximum (Bq/m^3)	
10	Turbine Building Vent	TDS-JE-RE001	Continuous in-line	Beta/gamma-sensitive Geiger-Muller tubes	Kr-85, Xe-133	3.7E+04	3.7E+15	Accident Range
	Condenser Air Removal System	N/A	Grab sample	Manual collection of effluent sample for laboratory analysis	noble gases iodine tritium	Laboratory[1]		N/A
	Gland Seal System	N/A	Grab sample	Manual collection of effluent sample for laboratory analysis	noble gases iodine tritium	Laboratory[1]		N/A

Note:
1. Samples will be taken to the on-site chemical laboratory for analysis. Laboratory equipment has not been specified at this stage of the design, as it will be determined by the future plant operator.

Table 6.2-4

AP1000 NPP LIQUID EFFLUENT MONITORS AND DETECTION RANGES

Monitor Point (Fig 6.2-2)	Description	Detector	Category	Type	Isotopes	Nominal Detection Range		Purpose
						Minimum (Bq/m^3)	Maximum (Bq/m^3)	
W7	Radwaste building plant outfall	WLS-JS-RE229	Continuous in-line	Gamma-sensitive thallium-activated, sodium iodide scintillation counter	Cs-137	3.7E+04	3.7E+08	Normal and accident operation range
	Holding tanks	N/A	Grab sample	Manual collection of effluent sample for laboratory analysis	H-3 Cs-137	Laboratory(1)		n/a
W9	Turbine building waste water plant outfall	WLS-JS-RE021	Continuous off-line	Gamma-sensitive thallium-activated, sodium iodide scintillation counter	Cs-137	3.7E+03	3.7E+09	Normal and accident operation range
		N/A	Grab sample	Manual collection of effluent sample for laboratory analysis	H-3 Cs-137	Laboratory(1)		N/A
W13	Turbine building service water system	SWS-JE-RE008	Continuous off-line	Gamma-sensitive thallium-activated, sodium iodide scintillation counter	Cs-137	3.7E+03	3.7E+09	Normal and accident operation range
		N/A	Grab sample	Manual collection of effluent sample for laboratory analysis	H-3 Cs-137	Laboratory(1)		N/A

Note:
1. Samples will be taken to the on-site chemical laboratory for analysis. Laboratory equipment has not been specified at this stage of the design, as it will be determined by the future plant operator.

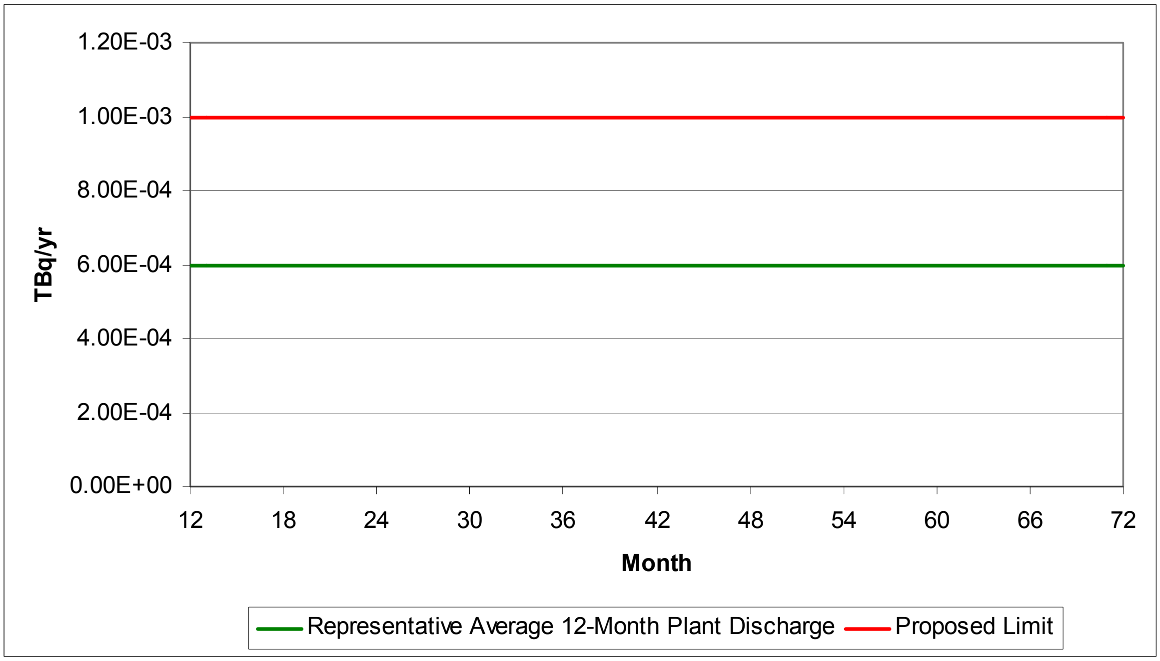


Figure 6.1-1. Comparison of Predicted Air Radioiodine Air Emission with Proposed Limits

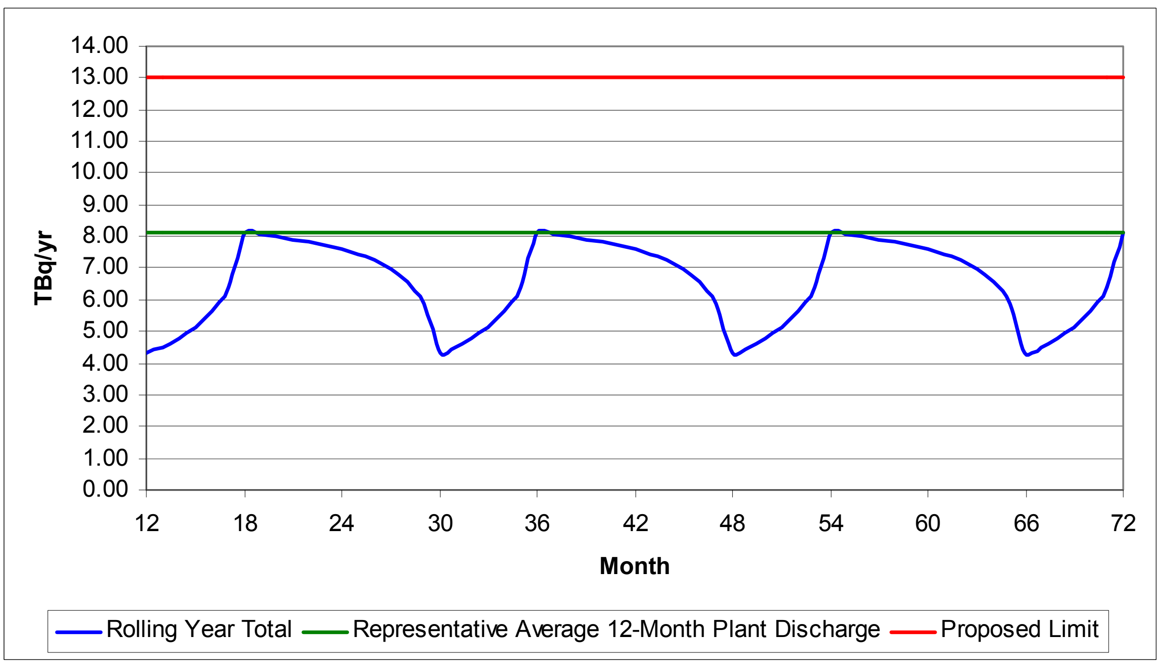


Figure 6.1-2. Comparison of Predicted Noble Gas Air Emissions with Proposed Limits

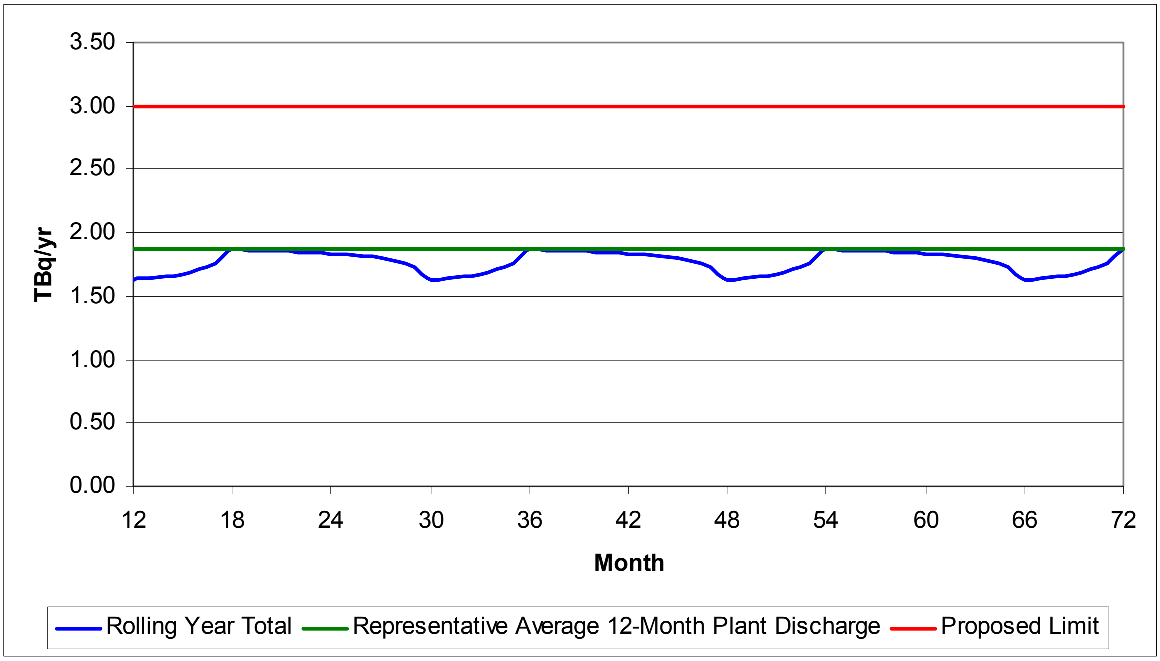


Figure 6.1-3. Comparison of Predicted Tritium Air Emissions with Proposed Limits

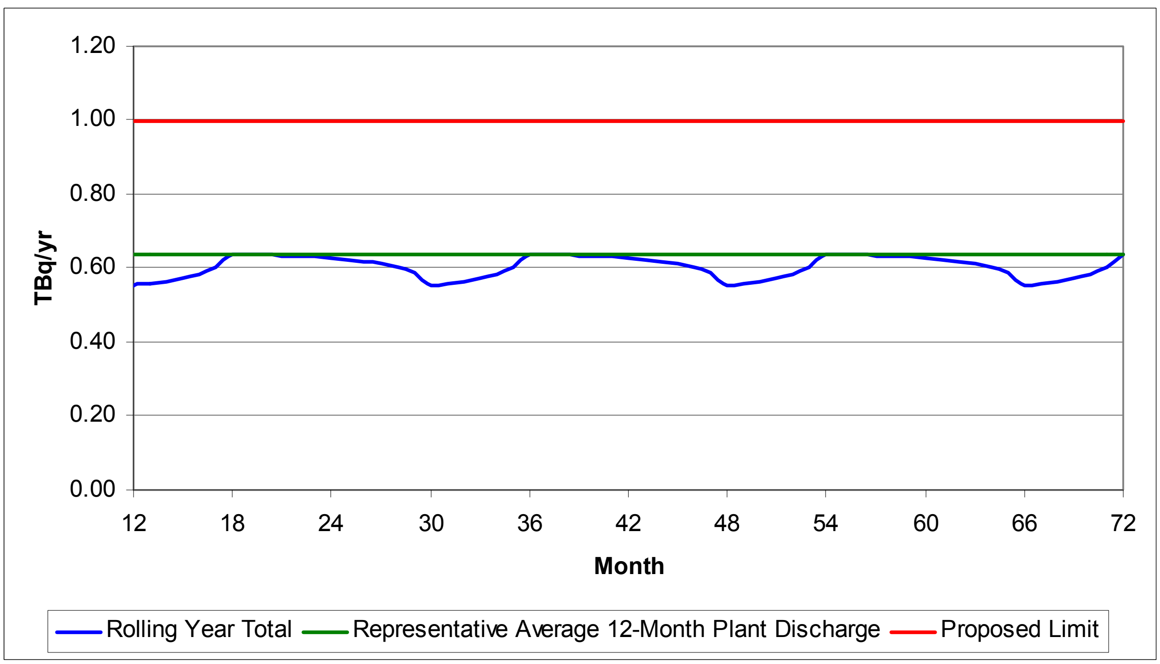


Figure 6.1-4. Comparison of Predicted Carbon-14 Air Emissions with Proposed Limits

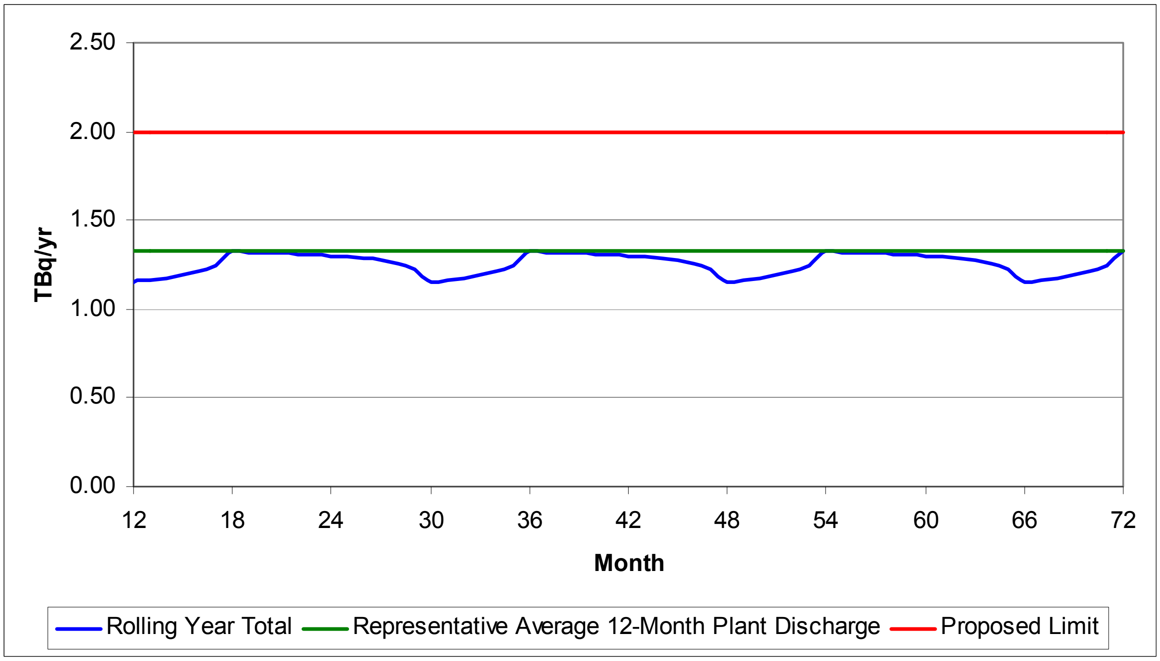


Figure 6.1-5. Comparison of Predicted Argon-41 Air Emissions with Proposed Limits

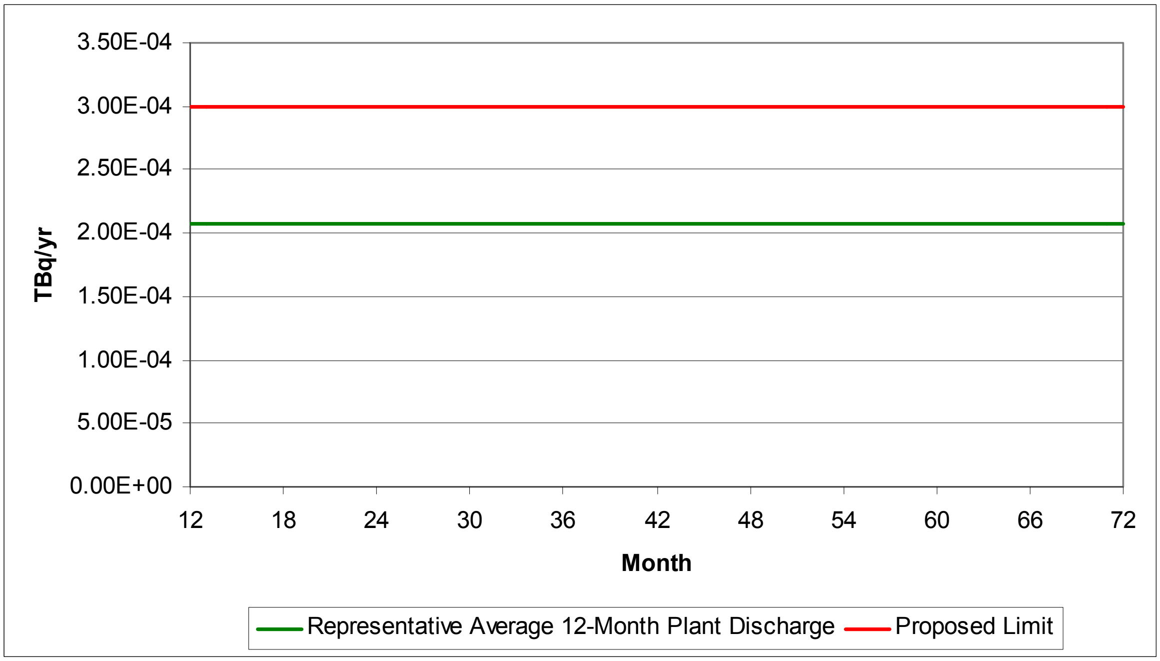


Figure 6.1-6. Comparison of Predicted Iodine-131 Air Emission with Proposed Limits

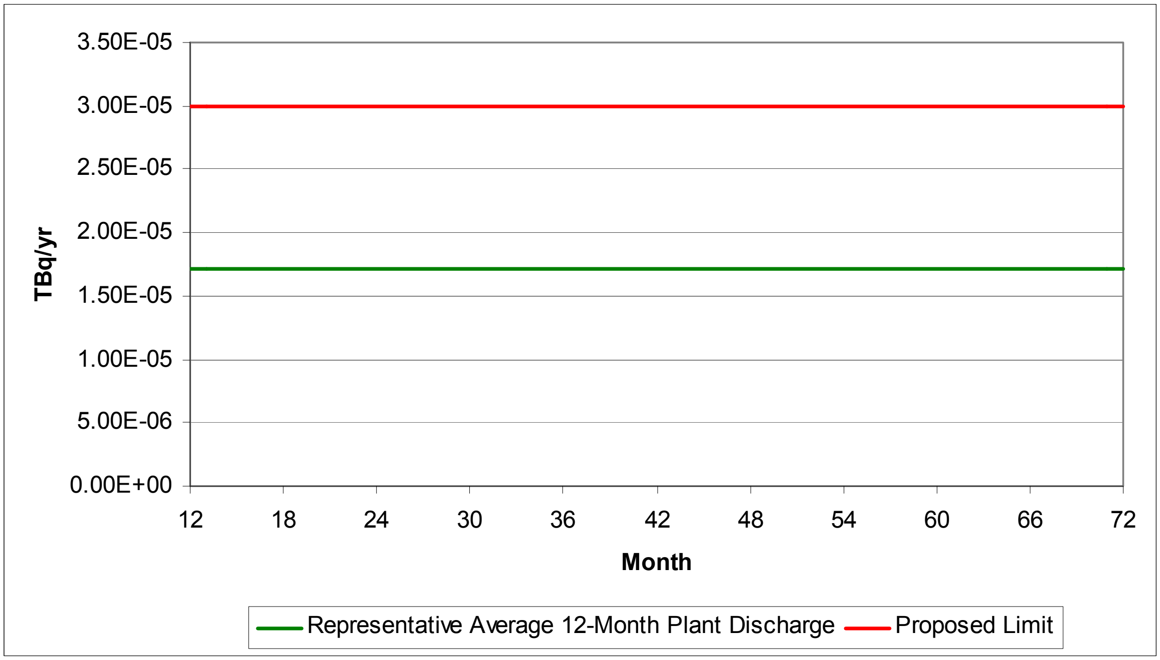


Figure 6.1-7. Comparison of Predicted Beta Particulate Air Emission with Proposed Limits

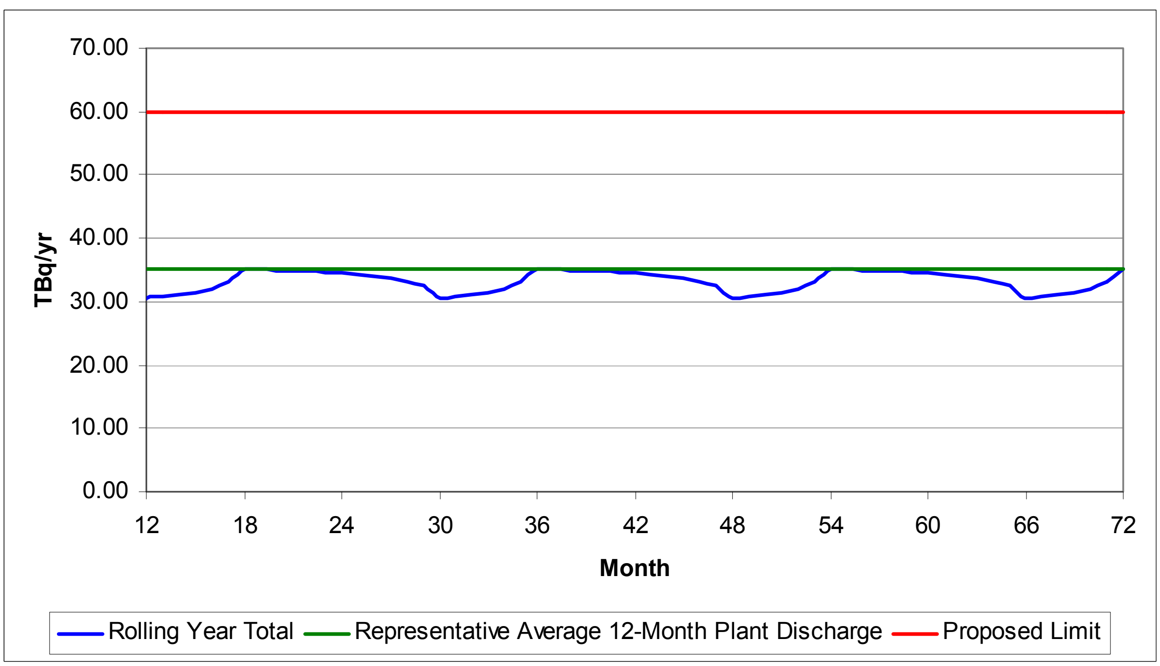


Figure 6.1-8. Comparison of Predicted Tritium Liquid Discharge with Proposed Limits

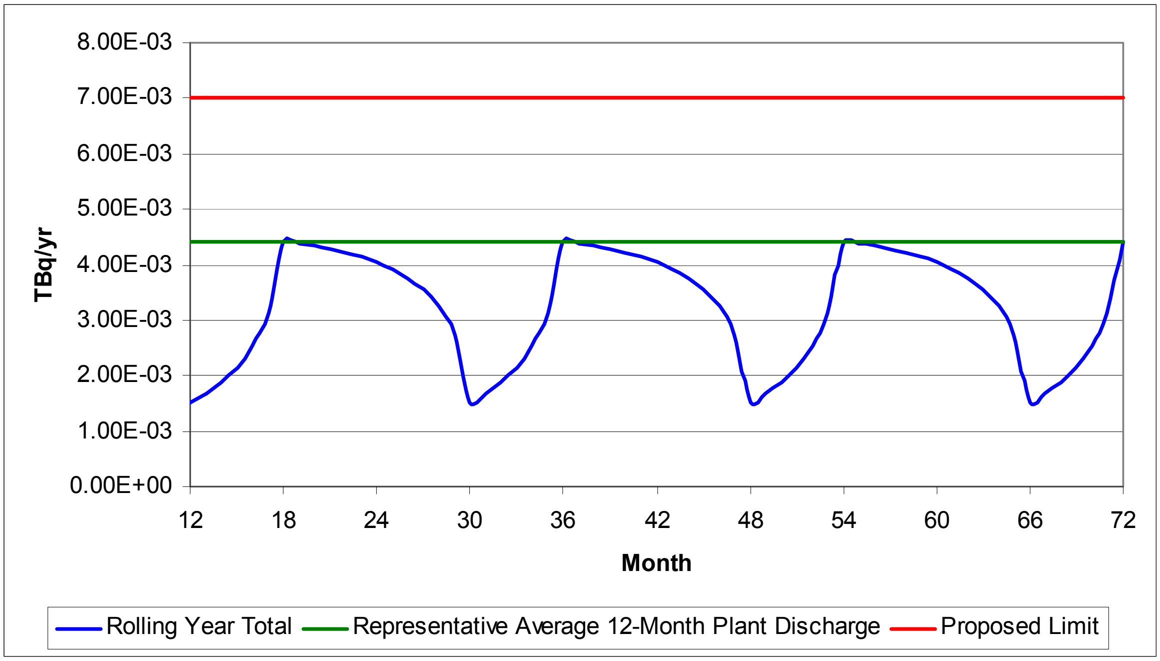


Figure 6.1-9. Comparison of Predicted C-14 Liquid Discharge with Proposed Limits

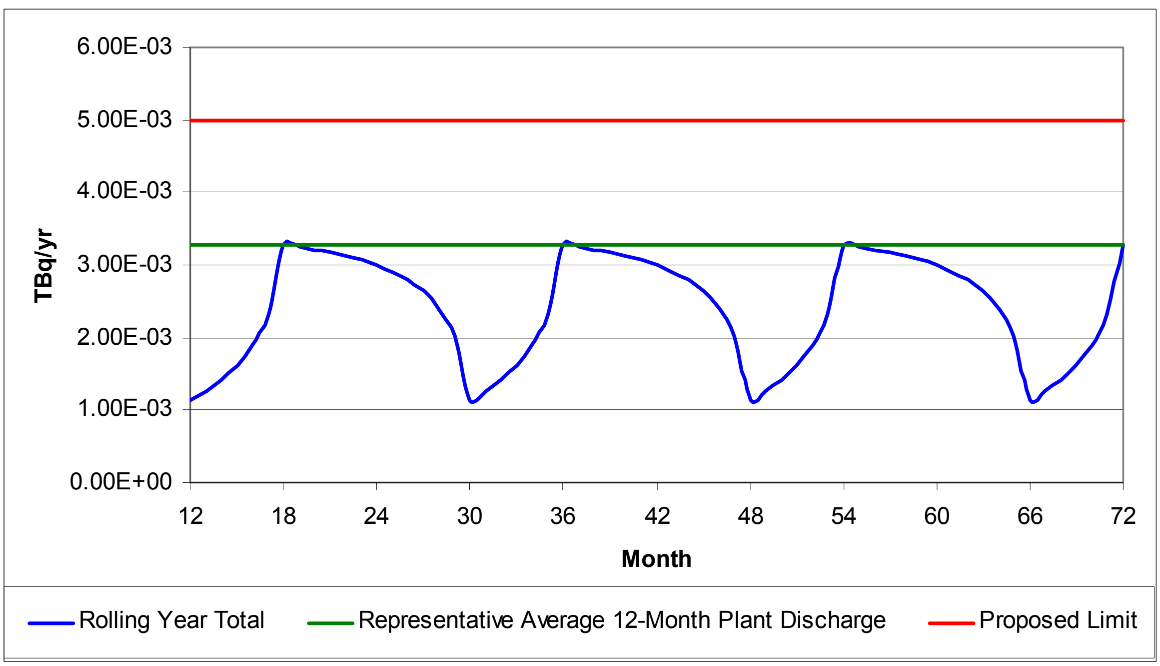


Figure 6.1-10. Comparison of Predicted Liquid Discharges of All Isotopes without Other Limits

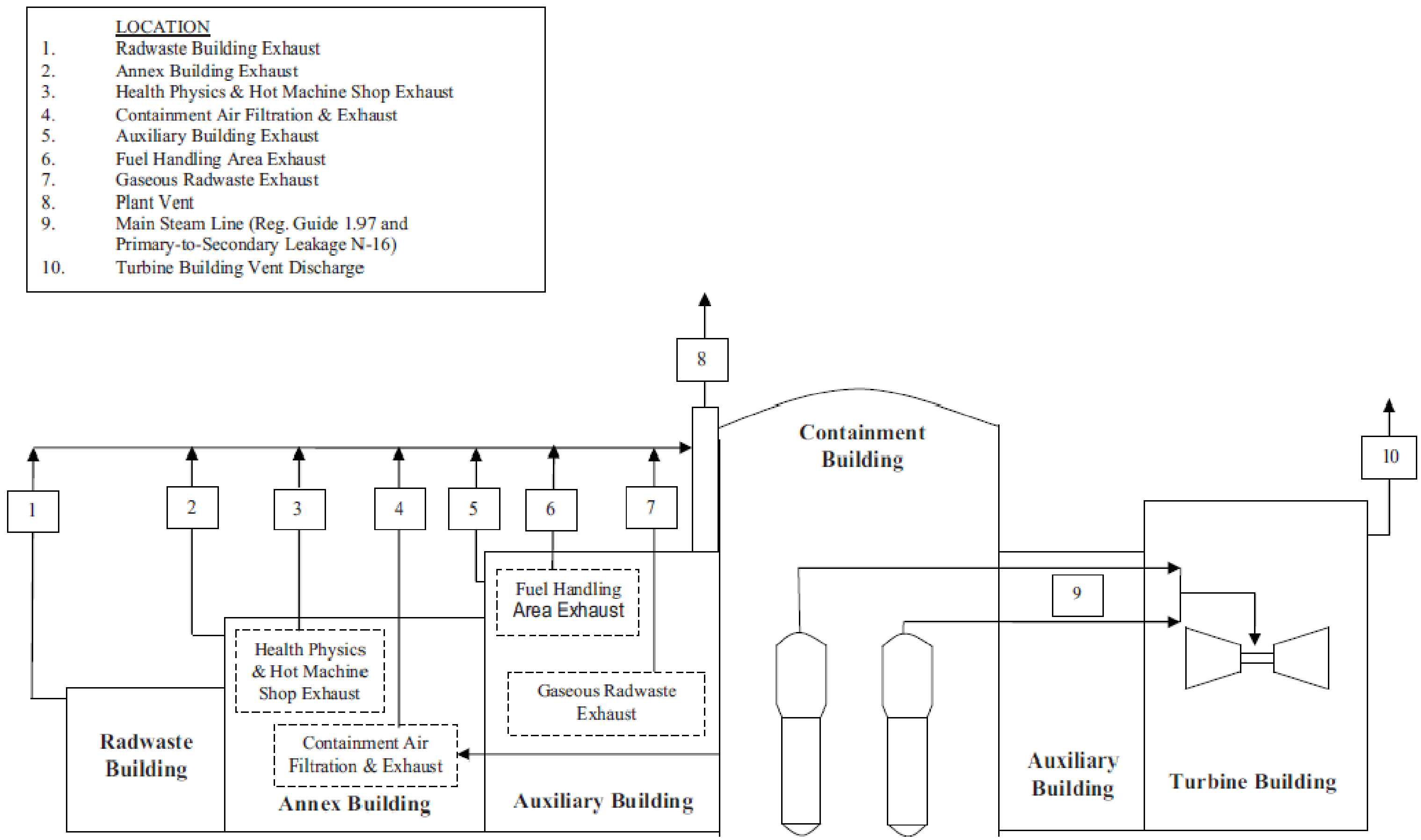


Figure 6.2-1. Schematic Illustrating Aerial Release Vents of the AP1000 NPPand Associated Monitors

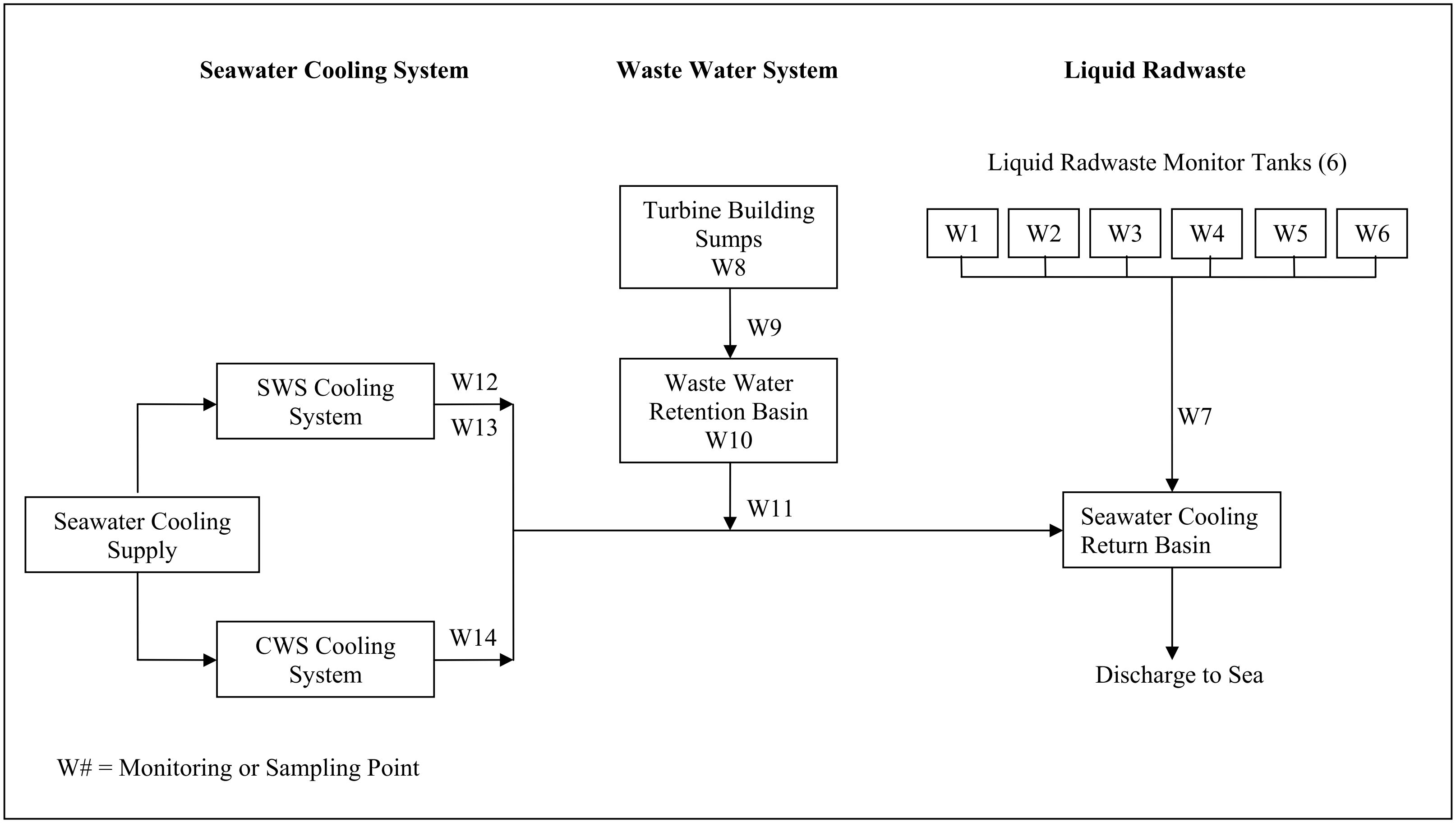


Figure 6.2-2. Schematic Illustrating the Liquid Discharge Monitoring Points of the AP1000 NPP and Associated Monitors

7.0 SELECTED CONSIDERATIONS FOR SPECIFIC SITES

7.1 Commentary on Sites with Multpile AP1000 NPP Units

For the purpose of GDA, it is assumed that the generic site will be occupied by one **AP1000** NPP and the information in this Environment Report reflects this single-unit case.

Systems related to the gaseous radwaste and liquid radwaste treatment will not be shared at a multi-unit site. A rough estimate of the impact and arrangements of a site with two, three, or more plants on the site could be made by multiplying the characteristic of interest by the number of units being evaluated. Note that the utilities may opt to use common circulating water abstraction and discharge points on a multiple-unit site.

Systems related to solid radwaste management, for example for treatment, storage, and transportation, could be shared at the utilities option. This is already the case for several existing PWRs. Better utilisation of space could be implemented by dedicating each of the multiple radwaste buildings to treating specific types of waste generated across the site. For example, one radwaste building could include equipment to treat site compactable waste, another to package site metallic waste, etc. Alternatively, a separate building could be constructed for treating site solid radwaste. Also, sharing of facilities would allow for operating experience to be shared across the **AP1000** NPPs on a site.

Locating multiple plants on the site does not affect the logistics of decommissioning the nuclear island, because each nuclear island on a site will operate independently and can be decommissioned independently. However, if the waste treatment facilities are shared, the decommissioning plan must ensure that the decommissioning waste does not affect units still operating and that the decommissioning waste does not exceed the capacity of the waste facility.

Openings and accessways in the nuclear island are not affected by the number of **AP1000** NPPs on a site and will not change when a single **AP1000** NPP or multiple **AP1000** NPPs are built on a site.

Once the site is selected, the utility will specify the number of units that will be built and operated on the site and where they will be located. This will define the distance between release points and influence how waste handling and radiation protection will be managed and integrated. The existing GDA submissions and generic site descriptions are relevant to site permitting because they provide the basic inputs to the site analyses.

7.2 Use of Cooling Towers for the Service Water System

The generic site is a coastal site (see Section 5.1) which abstracts seawater for once-through cooling of the CWS and SWS (see Section 4.2.3). This is acknowledged as a BAT solution for cooling at coastal nuclear power plants (Refences 7-1 and 7-2).

However, the PCSR (Reference 7-4) also retain the option for using cooling towers for the SWS. This arrangement may be used in circumstances where the use of seawater is not practical and there is a suitable supply of fresh water (e.g., river, lake, groundwater or mains supply) for the Raw Water System (RWS). The RWS would be required to provide the makeup flow to the cooling tower basin at a rate sufficient to compensate for losses due to evaporation, drift, and blowdown (typically 48 – 182 m^3h^{-1} (12,700 – 48,100 gallons/h)).

Major elements of the SWS cooling tower arrangement include two 100% capacity service water pumps, automatic backwash strainers, a two-cell cooling tower with a divided basin and associated piping, valves, controls and instrumentation. The service water system is arranged into two trains of components and piping. Each train includes one service water pump, one strainer, and one cooling tower cell. Each train provides cooling to one CCS component cooling water heat exchanger.

To maintain water quality in the SWS cooling towers, a range of chemicals may need to be injected into the SWS from the CFS. These chemicals may include a biocide, algaecide, pH adjustor, corrosion inhibitor, scale inhibitor, and silt dispersant. The selection of chemicals and their dose rate is dependent upon the RWS water supply and is a site-specific design.

A small blowdown flow is normally discharged from the SWS to the CWS or the WWS to control the level of solids concentration in the SWS inventory. Typically, this blowdown represents 0.2% – 0.75% of the main SWS flow (4700 m^3h^{-1}), which represents between 2 and 4 cycles of concentration.

Any decision to use cooling towers would be taken at the site-specific design stage when the environmental implications of the design would be assessed.

7.3 References

7-1 "Cooling Water Options for the New Generation of Nuclear Power Stations in the UK," SC070015/SR3, Environment Agency, June 2010.

7-2 UKP-GW-GL-034, Rev. 1, "Generic Assessment of the Impacts of Cooling Options for the Candidate Nuclear Power Plant **AP1000**," Westinghouse Electric Company LLC, February 2010.

7-3 Not used.

7-4 UKP-GW-GL-793, Rev. 1, "**AP1000** Pre-Construction Safety Report," Westinghouse Electric Company LLC, January 2017.

APPENDIX A
WASTE ARISINGS

A1 Identification of Waste Arisings from Solid, Liquid, and Gaseous Radioactive Sources

A2 Identification of Waste Arisings from Primary System Components

A3 Estimated Radwaste Arising from Large-Volume Components at Decommissioning

A4 Estimated Radwaste Arising from Small-Volume Components at Decommissioning

A5 Key for Preconditioning and Disposal Methods

A6 Steel and Concrete Rubble from Demolishing Various Modules

Appendix A1

IDENTIFICATION OF WASTE ARISINGS FROM SOLID, LIQUID, AND GASEOUS RADIOACTIVE SOURCES

System(1)	Waste Description	Rad/ Non-rad	LLW/ ILW/ HLW/ Mixed(2)	Physical/Chemical Description	Waste Form	Estimated Quantity						Disposability	
						normal/yr		maximum/yr		total quantity per life of plant(3)		Pre-Conditioning(4)	Disposal Route(4)
						cubic feet	cubic metres	cubic feet	cubic metres	cubic feet	cubic metres		
CVS	CVS Mixed Bed Resin	R	ILW	Spherical bead/resin compound	Solid	33.3	0.94	66.7	1.89	2,399	68	1	A
CVS	CVS Cation Bed	R	ILW	Spherical bead/resin compound	Solid	16.7	0.47	33.3	0.94	1,201	34	1	A
SFS	SFS Demineraliser	R	ILW	Spherical bead/resin compound	Solid	50	1.42	100	2.83	3,600	102	1	A
WLS	WLS unit 1 charcoal	R	ILW	Wet granular carbon	Solid	20	0.57	40	1.13	1,440	41	1	A
WLS	WLS unit 1 resin	R	ILW	Spherical bead/resin compound	Solid	40	1.13	80	2.27	2,880	82	1	A
WLS	WLS units 2,3,4	R	ILW	Spherical bead/resin compound	Solid	135	3.82	270	7.65	9,720	275	1	A
CPS	Condensate polisher spent resin	R	LLW	Spherical bead/resin compound	Solid	136	3.85	272	7.7	2,448	69	1	B
CVS	CVS RC filter cartridge	R	ILW	Metallic cylinder	Solid	1.12	0.03	2.24	0.06	81	2	2 or 4	A
SFS	SFS filter cartridge	R	ILW	Metallic cylinder	Solid	1.12	0.03	2.24	0.06	81	2	2 or 4	A
WLS	WLS inlet filter cartridge	R	ILW	Metallic cylinder	Solid	2.24	0.06	4.48	0.13	161	5	2 or 4	A
WLS	W LS outlet filter cartridge	R	ILW	Metallic cylinder	Solid	1.12	0.03	2.24	0.06	81	2	2 or 4	A
WSS	WSS resin fines filter cartridge	R	ILW	Metallic cylinder	Solid	1.12	0.03	2.24	0.06	81	2	2 or 4	A
WGS	WGS delay bed charcoal	R	LLW	Dry granular carbon	Solid	5.3	0.15	106.7	3.02	1,535	43	3 or 13	B or F
WGS	WGS guard bed charcoal	R	LLW	Dry granular carbon	Solid	5.3	0.15	10.7	0.3	383	11	3 or 13	B or F
CVS	makeup filter	N		Metallic cylinder	Solid	1.12	0.03	2.24	0.06	81	2	9	C
WSS	DAW	R	LLW	Compactable paper, tape, clothing, plastic, elastomers	Solid	4,750	134.52	7,260	205.61	315,120	8,924	3	B
WSS	DAW	R/N	LLW	Other non-compactable, metallic items, glass, wood	Solid	234	6.63	373	10.56	15,708	445	If N: 9, If R: 8 & 3	If N: C. If R: B

Appendix A1 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM SOLID, LIQUID, AND GASEOUS RADIOACTIVE SOURCES

System[1]	Waste Description	Rad/ Non-rad	LLW/ ILW/ HLW/ Mixed[2]	Physical/Chemical Description	Waste Form	Estimated Quantity						Disposability	
						normal/yr		maximum/yr		total quantity per life of plant[2]		Pre-Conditioning	Disposal Route[6]
						cubic feet	cubic metres	cubic feet	cubic	cubic feet	cubic metres		
WSS	DAW	R/N	Mixed	Small batteries/corrosive	Solid	5	0.14	10	0.28	360	10	If N: 9, If R: 8 & 3	If N: C. If R: B
WSS	Strippable coatings	R	LLW	Latex paint peelings	Solid							3	B
SGS	Sludge	R	LLW	Wet granular particles	Solid	1	0.028	3	0.085	84	2.38	15	J
						normal/gpd	**normal m3/d**	**maximum/gpd**	**maximum m3/d**	**gallons/lop**	**m3/lop**		
WLS	CVS shim bleed (liquid)	R	LLW	Diverted reactor coolant/dilute boric acid	Liquid	435	1.65	776	2.94	11,020,080	41,711	10	G
WLS	Equipment leaks	R	LLW	Dilute boric acid	Liquid	90	0.34	14400	54.5	2,835,000	10,731	10	G
WLS	Floor drains (dirty wastes)	R	LLW	Dilute boric acid	Liquid	1,200	4.54	5760	21.8	26,626,000	100,780	10	G
WLS	Sampling system drains	R	LLW	Dilute boric acid	Liquid	200	0.76	1000	3.79	4,560,000	17,260	10	G
WLS	Hot shower	R	LLW	Grey water	Liquid	0	0	0	0	0	0	14	I
WLS	Hand wash	R	LLW	Grey water	Liquid	200	0.76	2000	7.57	12,264,000	46,419	14	I
WLS	Equip and area decon	R	LLW	Detergent waste	Liquid	40	0.15	400	1.51	2,452,800	9,284	5 or 14	G or I
WLS	Chemical waste	R	LLW	Spent samples containing analytical chemicals	Liquid	7.14	0.03	14.28	0.05	187,639	710	14	I
WLS	Laundry (processed offsite)	N	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
WLS	Decon fluids	R	LLW	Liquid/w decon chemicals	Liquid	0.62	0	1.24	0	16,294	62	10	G
						normal cc/min	**normal m^3/hr**	**max cc/min**	**max m^3/hr**	**cft/lop**	**m^3/lop**		
WGS	RCDT drains	R	LLW	Gases containing hydrogen, nitrogen and fission gases	Gas	0	0	1.05	0.000063	1,170	0.0702	6	E
WGS	CVS shim bleed (gas)	R	LLW	Gases containing hydrogen, nitrogen and fission gasses	Gas	45.7	0.002742	81.6	0.004896	50,904	3.05424	6	E
						normal gpd	**normal m^3/d**	**max gpd**	**max m^3/day**	**gal/lop**	**m^3/lop**		
CVS	RCS heatup	R	LLW	Borated reactor coolant	Liquid	22,440	85	44,880	170	1,077,120	4,077	10	G

Appendix A1 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM SOLID, LIQUID, AND GASEOUS RADIOACTIVE SOURCES

System[(1)]	Waste Description	Rad/ Non-rad	LLW/ ILW/ HLW/ Mixed[(2)]	Physical/Chemical Description	Waste Form	Estimated Quantity						Disposability	
						normal/yr		maximum/yr		total quantity per life of plant[2]		Pre-Conditioning	Disposal Route[(6)]
						cubic feet	cubic metres	cubic feet	cubic	cubic feet	cubic metres		
CVS	Boron dilution near end-of–life (EOL)	R	LLW	Borated reactor coolant	Liquid	1,663	6	6,980	26	66,520	252	10	G
						normal gpm	**normal m^3/hr**	**max gpm**	**max m^3/hr**	**gal/lop**	**m^3/lop**		
BDS	Steam generator blow down	N/R	LLW if R	Secondary side coolant	Liquid	18.6	4.22	186	42.24	586,569,600	2,220,173	If N; 10. If R; 5	If N; D. If R; Gradioactive
WWS	Condensate demin rinses and backwashes	N		Demin water with minor solids	Liquid	0.052	0.01	2050	465.56	1,639,872	6,207	10	D
BDS	Condensate demin start-up bypass flow	N		Off specification demin water	Liquid	116.65	26.49	360	81.76	5,213,531,520	19,733,276	10	D
DTS	Reverse osmosis and electrodeionisation	N		Off specification demin water	Liquid	58.32	13.24	180	40.88	2,606,639,616	9,866,161	10	D
BDS	Fire testing drains	N		Demin water with minor solids	Liquid	0.38	0.09	750	170.33	11,983,680	45,358	10	D
Multiple	Turbine island waste water	N		Demin water with minor solids	Liquid	80.04	18.18	325	73.81	4,069,153,152	15,401,791	10	D
CDS	Condenser water box drain	N		Demin water with minor solids	Liquid	0	0	1,102	250.26	44,080	167	10	D
SWS	Strainer backwash	N		Demin water with minor solids	Liquid	4.2	0.95	3,000	681.3	132,451,200	501,329	10	D
CWS	Strainer backwash	N		Demin water with minor solids	Liquid	9.48	2.15	1,820	413.32	298,961,280	1,131,572	10	D
CPS	Condensate polisher rinse	N		Demin water with minor solids	Liquid	0.052	0.01	2,050	465.56	1,639,872	6,207	10	D
CWS	Circulating water system blow down	N		Heat sink water body	Liquid	4,719	1071.69	14,159	3215.52	148,818,384,000	563,279,273	10	D
WWS	Waste oil from oil separator (overflow)	N		Oily water mostly oil	Liquid	0	0	0.3	0.07	432	1.64	13[(5)]	C
WWS	Waste water from oil separator (underflow)	N		Oily water mostly water	Liquid	0	0	199.7	45.35	287568	1088.45	10	D

Notes:
1. See Glossary for definition of terms.
2. Hazardous and radioactive wastes mixed together
3. Includes the maximum quantity column once per five years for most entries.
4. See Appendix A5 for key of pre-conditioning and disposal routes.
5. Non-radioactive oil and radioactive oil will be kept separate, i.e. they will not be collected in the same vessel.

Appendix A2

IDENTIFICATION OF WASTE ARISINGS FROM PRIMARY SYSTEM COMPONENTS

System	Component	Type	Rad/ Non-rad	Room/ Location	Waste Source	Waste Description	Frequency(1)	Waste Quantity(2)				Radio-activity	Waste Form	Disposability	
								cubic feet each change	cubic metres each change	cubic feet plant life	cubic metres plant life			Pre-Conditioning(3)	Disposal Route(3)
CVS	Regenerative hx	Heat Exchanger	R	11209	Insulation replace	Insulation	1/lop	58.67	1.662	58.67	1.662	LLW	Solid	3	B
CVS	Makeup miniflo hx A & B	Heat Exchanger	R	12255	Insulation replace	Insulation	1/lop	21.47	0.608	21.47	0.608	LLW	Solid	3	B
CVS	Makeup miniflo hx A & B	Heat Exchanger	R	12255	Gasket replace	Compressible rigid plastic	1/lop	0.13	0.004	0.13	0.004	LLW	Solid	3	B
CVS	Letdown hx	Heat Exchanger	R	12255	Gasket replace	Compressible rigid plastic	1/lop	0.11	0.003	0.11	0.003	LLW	Solid	3	B
CVS	Letdown hx	Heat Exchanger	R	11209	Insulation replace	Compressible rigid plastic	1/lop	57.36	1.624	57.36	1.624	LLW	Solid	3	B
WLS	Reactor coolant drain tank hx	Heat Exchanger	R	11104	Gasket replace	Compressible rigid plastic	1/lop	0.04	0.001	0.04	0.001	LLW	Solid	3	B
WLS	Reactor coolant drain tank hx	Heat Exchanger	R	11104	Insulation replace	Insulation	1/lop	14.31	0.405	14.31	0.405	LLW	Solid	3	B
SFS	Spent fuel pool hx A	Heat Exchanger	R	12273	Minimal waste									9	B
SFS	Spent fuel pool hx B	Heat Exchanger	R	12275	Minimal waste									9	B
RNS	Residual heat removal hx	Heat Exchanger	R	12362	Insulation replace	Insulation	1/lop	201.12	5.696	201.12	5.696	LLW	Solid	3	B
RNS	Residual heat removal hx	Heat Exchanger	R	12362	Gasket replace	Compressible rigid plastic	1/lop	0.03	0.001	0.03	0.001	LLW	Solid	3	B
CCS	Component cooling hx A	Heat Exchanger	N	2000	Gasket replace	Neoprene	1/10 yrs	1.08	0.031	6.48	0.184		Solid	9	C
CCS	Component cooling hx B	Heat Exchanger	N	2000	Gasket replace	Neoprene	1/10 yrs	1.08	0.031	6.48	0.184		Solid	9	C
BDS	Steam gen blow down hx A&B	Heat Exchanger		20306	Minimal waste									9	C or B
various	hxs (other non-radioactive)	Heat Exchanger	N	various	Minimal waste									11	C
WLS	Reactor coolant drain tank	Tank	R	11104	Gasket replace	Compressible rigid plastic	3/lop	0.08	0.002	0.24	0.007	LLW	Solid	3	B
WLS	Waste hold-up tank A	Tank	R	12166	Gasket replace	Compressible rigid plastic	3/lop	0.08	0.002	0.24	0.007	LLW	Solid	3	B
WLS	Waste hold-up tank B	Tank	R	12167	Gasket replace	Compressible rigid plastic	3/lop	0.08	0.002	0.24	0.007	LLW	Solid	3	B
WLS	Waste monitor tank A	Tank	R	12363	Gasket replace	Compressible rigid plastic	3/lop	0.08	0.002	0.24	0.007	LLW	Solid	3	B
WLS	Waste monitor tank B	Tank	R	12365	Gasket replace	Compressible rigid plastic	3/lop	0.08	0.002	0.24	0.007	LLW	Solid	3	B
WLS	Waste monitor tank C	Tank	R	12265	Gasket replace	Compressible rigid plastic	3/lop	0.08	0.002	0.24	0.007	LLW	Solid	3	B
WLS	Waste monitor tank D	Tank	R	50355	Gasket replace	Compressible rigid plastic	3/lop	0.08	0.002	0.24	0.007	LLW	Solid	3	B
WLS	Waste monitor tank E	Tank	R	50355	Gasket replace	Compressible rigid plastic	3/lop	0.08	0.002	0.24	0.007	LLW	Solid	3	B
WLS	Waste monitor tank F	Tank	R	50355	Gasket replace	Compressible rigid plastic	3/lop	0.08	0.002	0.24	0.007	LLW	Solid	3	B
WLS	Effluent hold-up tank A	Tank	R	12171	Gasket replace	Compressible rigid plastic	3/lop	0.08	0.002	0.24	0.007	LLW	Solid	3	B
WLS	Effluent hold-up tank B	Tank	R	12172	Gasket replace	Compressible rigid plastic	3/lop	0.08	0.002	0.24	0.007	LLW	Solid	3	B

Appendix A2 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM PRIMARY SYSTEM COMPONENTS

System	Component	Type	Rad/ Non-rad	Room/ Location	Waste Source	Waste Description	Frequency(1)	Waste Quantity(2)				Radio-activity	Waste Form	Disposability	
								cubic feet each change	cubic metres each change	cubic feet plant life	cubic metres plant life			Pre-Conditioning(3)	Disposal Route(3)
WLS	Effluent hold-up tank C	Tank	R	12172	Gasket replace	Compressible rigid plastic	3/lop	0.08	0.002	0.24	0.007	LLW	Solid	3	B
CVS	Boric acid batching tank	Tank	N	40442	Minimal waste									9	C
CVS	Boric acid tank	Tank	N	381	Minimal waste									9	C
CVS	Chemical mixing tank	Tank	N	12255	Minimal waste									9	C
WLS	Zinc addition tank	Tank	N	2033	Minimal waste									9	C
WLS	Chemical waste tank	Tank	R	12264	Gasket replace	Compressible rigid plastic	once/18 mo	0.02	0.001	0.8	0.023	LLW	Solid	3	B
CCS	Component cooling water surge tank	Tank	N	20600	Minimal waste									9	C
CCS	Chemical addition tank	Tank	N	2000	Minimal waste									9	C
LOS	Lube oil reservoir	Tank	N	20407	Oil replace	Depleted oil	once/25 yrs(4)	2807	79.496	5614	158.992		Liquid	13(5)	C
various	Tanks (other non-radioactive)	Tank (160)	N	various										9	C
All Systems	Valves (potentially radioactive)	Valve (1461)	R	various	Packing replace	Compressible rigid plastic	once/5yrs	15.94	0.451	191.28	5.417	LLW	Solid	3	B
All Systems	Valves (non-radioactive)	Valve (3699)	N	various	Packing replace	Compressible rigid plastic	once/5yrs	40.3	1.141	483.6	13.696	LLW	Solid	9	C
CVS	Reactor coolant filter A	Filter	R	11209	Gasket replace	Compressible rigid plastic	once/yr	0.03	0.001	1.8	0.051	LLW	Solid	3	B
CVS	Reactor coolant filter B	Filter	R	11209	Gasket replace	Compressible rigid plastic	once/yr	0.03	0.001	1.8	0.051	LLW	Solid	3	B
CVS	Makeup filter	Filter	R	12156	Gasket replace	Compressible rigid plastic	once/yr	0.03	0.001	1.8	0.051	LLW	Solid	3	B
WLS	Waste pre-filter	Filter	R	12151	Gasket replace	Compressible rigid plastic	once/yr	0.03	0.001	1.8	0.051	LLW	Solid	3	B
WLS	Waste after filter	Filter	R	12151	Gasket replace	Compressible rigid plastic	once/yr	0.03	0.001	1.8	0.051	LLW	Solid	3	B
WLS	Resin fines filter	Filter	R	12471	Gasket replace	Compressible rigid plastic	once/yr	0.03	0.001	1.8	0.051	LLW	Solid	3	B
SFS	Spent fuel system filter A	Filter	R	12151	Gasket replace	Compressible rigid plastic	once/yr	0.03	0.001	1.8	0.051	LLW	Solid	3	B
CCS	Spent fuel system filter B	Filter	R	12151	Gasket replace	Compressible rigid plastic	once/yr	0.03	0.001	1.8	0.051	LLW	Solid	3	B
WLS	Charcoal deep bed filter	Adsorber	R	12151	Minimal waste									9	B
WGS	Guard bed	Adsorber	R	12155	Minimal waste									9	B
WGS	Delay bed A	Adsorber	R	12155	Minimal waste									9	B
WGS	Delay bed B	Adsorber	R	12155	Minimal waste									9	B
WGS	Moisture separator	Small vessel	R	12155	Minimal waste									9	B
CVS	Mixed bed demineraliser A	Ion exchanger	R	11209	Minimal waste									9	B

Appendix A2 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM PRIMARY SYSTEM COMPONENTS

System	Component	Type	Rad/ Non-rad	Room/ Location	Waste Source	Waste Description	Frequency(1)	Waste Quantity(2)				Radio-activity	Waste Form	Disposability	
								cubic feet each change	cubic metres each change	cubic feet plant life	cubic metres plant life			Pre-Conditioning(3)	Disposal Route(3)
CVS	Mixed bed demineraliser B	Ion exchanger	R	11209	Minimal waste									9	B
CVS	Cation bed demineraliser	Ion exchanger	R	11209	Minimal waste									9	B
CVS	Waste ion exchanger A	Ion exchanger	R	11209	Minimal waste									9	B
CVS	Waste ion exchanger B	Ion exchanger	R	11209	Minimal waste									9	B
CVS	Waste ion exchanger C	Ion exchanger	R	11209	Minimal waste									9	B
SFS	Spent fuel sys demin A	Ion exchanger	R	12151	Minimal waste									9	B
SFS	Spent fuel sys demin B	Ion exchanger	R	12151	Minimal waste									9	B
CVS	Makeup pump A	Pump	R	12255	Mech seals 2/pmp	Carbon/SiC	once/10 yr	0.1	0.003	1.2	0.034	LLW	Solid	3	B
CVS	Makeup pump/motor A	Pump	R	12255	Oil replace	Waste Oil	once/5yrs	1.34	0.038	16.08	0.455	LLW	Liquid	7(5)	F
CVS	Makeup pump B	Pump	R	12255	Mech seals 2/pmp	Carbon/SiC	once/10 yr	0.1	0.003	1.2	0.034	LLW	Solid	3	B
CVS	Makeup pump/motor B	Pump	R	12255	Oil replace	Waste oil	once/5yrs	1.34	0.038	16.08	0.455	LLW	Liquid	7(5)	F
WLS	Zinc injection pump	Pump	N	2033	Minimal waste									9	C
RNS	Residual heat removal pump A	Pump	R	12162	Mech seal 2/pmp	Carbon/SiC	once/5yrs	0.1	0.003	2.4	0.068	LLW	Solid	3	B
RNS	Residual heat removal pump B	Pump	R	12163	Mech seal 2/pmp	Carbon/SiC	once/5yrs	0.1	0.003	2.4	0.068	LLW	Solid	3	B
CCS	CCW pump A	Pump	R	20300	Mech seal 2/pmp	Carbon/SiC	once/5yrs	0.1	0.003	2.4	0.068	LLW	Solid	3	B
CCS	CCW pump B	Pump	R	20300	Mech seal 2/pmp	Carbon/SiC	once/5yrs	0.1	0.003	2.4	0.068	LLW	Solid	3	B
SFS	Spent fuel cooling pump A	Pump	R	12272	Mech seal 2/pmp	Carbon/SiC	once/5yrs	0.1	0.003	2.4	0.068	LLW	Solid	3	B
SFS	Spent fuel cooling pump/motor A	Pump	R	12272	Oil replace	Waste Oil	once/5yrs	1.34	0.038	16.08	0.455	LLW	Liquid	7(5)	F
SFS	Spent fuel cooling pump B	Pump	R	12274	Mech seal 2/pmp	Carbon/SiC	once/5yrs	0.1	0.003	2.4	0.068	LLW	Solid	3	B
SFS	Spent fuel cooling pump/motor B	Pump	R	12274	Oil replace	Waste Oil	once/5yrs	1.34	0.038	16.08	0.455	LLW	Liquid	7(5)	F
WLS	Degasifier separator pump A	Pump	R	12156	Replace pump	Canned Pump	once/lop	1	0.028	1	0.028	LLW	Solid	9	C or B
WLS	Degasifier separator pump B	Pump	R	12156	Replace pump	Canned Pump	once/lop	1	0.028	1	0.028	LLW	Solid	9	C or B
WLS	Degasifier discharge pump A	Pump	R	12158	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	Degasifier discharge pump B	Pump	R	12158	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	Effluent hold-up pump A	Pump	R	12271	Replace	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B

Appendix A2 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM PRIMARY SYSTEM COMPONENTS

System	Component	Type	Rad/ Non-rad	Room/ Location	Waste Source	Waste Description	Frequency(1)	Waste Quantity(2)				Radio-activity	Waste Form	Disposability	
								cubic feet each change	cubic metres each change	cubic feet plant life	cubic metres plant life			Pre-Conditioning(3)	Disposal Route(3)
					diaphragms										
WLS	Effluent hold-up pump B	Pump	R	12268	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	Effluent hold-up pump C	Pump	R	12268	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	Waste hold-up pump A	Pump	R	12268	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	Waste hold-up pump B	Pump	R	12268	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	Monitor pump A	Pump	R	12363	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	Monitor pump B	Pump	R	12365	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	Monitor pump C	Pump	R	12265	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	Monitor pump D	Pump	R	50355	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	Monitor pump E	Pump	R	50355	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	Monitor pump F	Pump	R	50355	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	Chemical waste pump	Pump	R	12264	Replace diaphragms	Buna n	once/5yrs	0.1	0.003	1.2	0.034	LLW	Plastic	3	B
WLS	RCDT pump A	Pump	R	11104	Mech seal 1/pmp	Carbon/SiC	3/lop	0.3	0.008	0.9	0.025	LLW	Solid	3	B
WLS	RCDT pump B	Pump	R	11104	Mech seal 1/pmp	Carbon/SiC	3/lop	0.3	0.008	0.9	0.025	LLW	Solid	3	B
WLS	Degasifier vacuum pump A	Pump	R	12156	Mech seal 2/pmp	Carbon/SiC	3/lop	0.1	0.003	0.6	0.017	LLW	Solid	3	B
WLS	Degasifier vacuum pump B	Pump	R	12156	Mech seal 2/pmp	Carbon/SiC	3/lop	0.1	0.003	0.6	0.017	LLW	Solid	3	B
WSS	Resin transfer pump	Pump	R	12372	Replace pump	Screw Pump	once/10 yr	0.1	0.003	0.6	0.017	LLW	Solid	9	C or B
WLS	Containment sump pump A	Pump	R	11104	Mech seal 1/pmp	Carbon/SiC	2/lop	0.1	0.003	0.2	0.006	LLW	Solid	3	B
WLS	Containment sump pump B	Pump	R	11104	Mech seal 1/pmp	Carbon/SiC	2/lop	0.1	0.003	0.2	0.006	LLW	Solid	3	B
FWS	Main Feedwater Pump A	Pump	N	20300	Mech seal 4/pmp	Carbon/SiC	once/5yrs	1	0.028	12	0.34		Solid	3	B
FWS	Main Feedwater Pump B	Pump	N	20300	Mech seal 4/pmp	carbon/SiC	once/5yrs	1	0.028	12	0.34		Solid	3	B
FWS	Main Feedwater Pump A	Pump	N	20300	Replace oil/100	Lube Oil	once/5yrs	13.37	0.379	160.44	4.544		Liquid	13(5)	C

Appendix A2 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM PRIMARY SYSTEM COMPONENTS

System	Component	Type	Rad/ Non-rad	Room/ Location	Waste Source	Waste Description	Frequency(1)	Waste Quantity(2)				Radio-activity	Waste Form	Disposability	
								cubic feet each change	cubic metres each change	cubic feet plant life	cubic metres plant life			Pre-Conditioning(3)	Disposal Route(3)
					gal										
FWS	Main Feedwater Pump B	Pump	N	20300	Replace oil/100 gal	Lube Oil	once/5yrs	13.37	0.379	160.44	4.544		Liquid	13(5)	C
various	Other non-rad pumps	Pump (35)	N	various	Replace parts									9	C
various	Compressor and vac pumps	Pump (18)	N	various	Replace parts	Metal/ Plastic								9	C
various	Shop and maint equipment	Tools	N	40358	Replace tools	Metal/ Plastic								9	C
								mass in pounds each change	**mass in tonne each change**	**mass in pounds plant life**	**mass in tonne plant life**				
EDS1	NNS1 125V 60 Cell 8 Hour Bat 1	Battery(6)	N	40307	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
EDS2	NNS2 125V 60 Cell 8 Hour Bat 1	Battery(6)	N	40309	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
EDS3	NNS1 125V 60 Cell 8 Hour Bat 2	Battery(6)	N	40307	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
EDS4	NNS2 125V 60 Cell 8 Hour Bat 2	Battery(6)	N	40309	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
APS	EDSS-DB-1A (60 cells)	Battery(6)	N	TBD	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
APS	EDSS-DB-1B (60 cells)	Battery(6)	N	TBD	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSA	Div A 125V 24 HR Battery 1A	Battery(6)	N	12101	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSA	Div A 125V 24 HR Battery 1B	Battery(6)	N	12101	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSB	Div B 125V 24 HR Battery 1A	Battery(6)	N	12104	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSB	Div B 125V 24 HR Battery 1B	Battery(6)	N	12104	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSB	Div B 125V 72 HR Battery 2A	Battery(6)	N	12204	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSB	Div B 125V 72 HR Battery 2B	Battery(6)	N	12204	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSC	Div C 125V 24 HR Battery 1A	Battery(6)	N	12102	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C

Appendix A2 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM PRIMARY SYSTEM COMPONENTS

System	Component	Type	Rad/ Non-rad	Room/ Location	Waste Source	Waste Description	Frequency(1)	Waste Quantity(2)				Radio-activity	Waste Form	Disposability	
								cubic feet each change	cubic metres each change	cubic feet plant life	cubic metres plant life			Pre-Conditioning(3)	Disposal Route(3)
IDSC	Div C 125V 24 HR Battery 1B	Battery(6)	N	12102	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSC	Div C 125V 72 HR Battery 2A	Battery(6)	N	12202	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSC	Div C 125V 72 HR Battery 2B	Battery(6)	N	12202	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSD	Div D 125V 24 HR Battery 1A	Battery(6)	N	12105	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSD	Div D 125V 24 HR Battery 1B	Battery(6)	N	12105	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSS	Spare 125V 60 Cell Battery A	Battery(6)	N	12103	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
IDSS	Spare 125V 60 Cell Battery B	Battery(6)	N	12103	Replace battery	Lead/Acid/ Plastic	once/20yrs(7)	39,000	18	78,000	35		Solid/ Liquid	9	C
								cubic feet each change	**cubic metres each change**	**cubic feet life of plant**	**cubic metres life of plant**				
VAS	Auxiliary/Annex Building Supply AHU A	HVAC filter	N	40601	Replace filters 12 pre-filters and 12 high efficiency	Fibreglass/ Metal	2/ year	96	2.72	11520	326.25		Solid	9	C
VAS	Auxiliary/Annex Building Supply AHU B	HVAC filter	N	40601	Replace filters 6 pre-filters and 6 high efficiency	Fibreglass/ Metal	2/ year	48	1.36	5760	163.13		Solid	9	C
VAS	Fuel Handling Area Supply AHU A	HVAC filter	N	40503	Replace filter 3 pre-filters and 3 high efficiency	Fibreglass/ Metal	2/ year	24	0.68	2880	81.56		Solid	9	C
VAS	Fuel Handling Area Supply AHU B	HVAC filter	N	40503	Replace filter 3 pre-filters and 3 high efficiency	Fibreglass/ Metal	2/ year	24	0.68	2880	81.56		Solid	9	C
VAS	CVS Pump Room Unit Cooler A	HVAC filter	N	12255	Replace filters 2 filter	Fibreglass/ Metal	2/ year	8	0.23	960	27.19		Solid	9	C
VAS	CVS Pump Room Unit Cooler B	HVAC filter	N	12255	Replace filters 2 filter	Fibreglass/ Metal	2/ year	8	0.23	960	27.19		Solid	9	C
VAS	RNS Pump Room Unit Cooler A	HVAC filter	N	12162	Replace filters 2 filter	Fibreglass/ Metal	2/ year	8	0.23	960	27.19		Solid	9	C
VAS	RNS Pump Room Unit Cooler B	HVAC filter	N	12163	Replace filters 2 filter	Fibreglass/ Metal	2/ year	8	0.23	960	27.19		Solid	9	C

Appendix A2 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM PRIMARY SYSTEM COMPONENTS

System	Component	Type	Rad/ Non-rad	Room/ Location	Waste Source	Waste Description	Frequency(1)	Waste Quantity(2)				Radio-activity	Waste Form	Disposability	
								cubic feet each change	cubic metres each change	cubic feet plant life	cubic metres plant life			Pre-Conditioning(3)	Disposal Route(3)
VAS	Active SPS Exhaust	HVAC Filter	R	Roof of Aux Bldg	Replace 12 Filters	Fiberglass / Metal	Once / 2yrs	48	1.36	1440	40.08	LLW	Solid	3	B
VAS	Active SPS Exhaust	HVAC Filter	R	Roof of Aux Bldg	Replace 12 Filters	Fiberglass / Metal	3/ year	48	1.36	8640	244.68	LLW	Solid	3	B
VAS	Passive SPS Exhaust	HVAC Filter	R	12562	Replace Filters	Fiberglass/ Metal	once/30 yrs	200 – 400	6 – 11	400 – 800	11 – 23	LLW	Solid	3	B
VAS	Passive Fuel Handling Area Exhaust	HVAC Filter	R	12562	Replace Filters	Metal	once/30yrs	200 – 400	6 – 121	400 – 800	11 – 23	LLW	Solid	3	B
VBS	Supplemental Air Filtration Unit A High Efficiency Filters	HVAC filter	N	12501	Replace filters 4 filter	Fibreglass/ Metal	once/5yrs	16	0.45	192	5.44		Solid	9	C
VBS	Supplemental Air Filtration Unit B High Efficiency Filters	HVAC filter	N	12501	Replace filters 4 filter	Fibreglass/ Metal	once/5yrs	16	0.45	192	5.44		Solid	9	C
VBS	Supplemental Air Filtration Unit A HEPA Filters	HVAC filter	N	12501	Replace filters 2 filter	Fiberglas/ Metal	once/5yrs	8	0.23	96	2.72		Solid	9	C
VBS	Supplemental Air Filtration Unit B HEPA Filters	HVAC filter	N	12501	Replace filters 2 filter	Fibreglass/Metal	once/5 yrs	8	0.23	96	2.72		Solid	9	C
VBS	Supplemental Air Filtration Unit A Charcoal Filter	HVAC filter	N	12501	Replace charcoal	Granulated charcoal	once/10 yrs	85.7	2.43	514.2	14.56		Solid	9	C
VBS	Supplemental Air Filtration Unit B Charcoal Filters	HVAC filter	N	12501	Replace charcoal	Granulated charcoal	once/10 yrs	85.7	2.43	514.2	14.56		Solid	9	C
VBS	MCR/TSC Supply AHU A	HVAC filter	N	12501	Replace filters 12 pre-filters and 12 high efficiency	Uncompacted fibreglass/metal	2/year	96	2.72	11520	326.25		Solid	9	C
VBS	MCR/TSC Supply AHU B	HVAC filter	N	40500	Replace filters 12 pre-filters and 12 high efficiency	Uncompacted fibreglass/metal	2/year	96	2.72	11520	326.25		Solid	9	C
VBS	A/C 1E Elect Room Supply AHU A	HVAC filter	N	12501	Replace filters 12 pre-filters and 12 high efficiency	Uncompacted fibreglass/metal	2/year	96	2.72	11520	326.25		Solid	9	C
VBS	B/D 1E Elect Room Supply AHU B	HVAC filter	N	12505	Replace filter 6 pre-filters and 6 high efficiency	Uncompacted fibreglass/metal	2/year	48	1.36	5760	163.13		Solid	9	C
VBS	A/C 1E Elect Room Supply AHU C	HVAC filter	N	12501	Replace filters 12 pre-filters and 12 high efficiency	Uncompacted fibreglass/metal	2/year	96	2.72	11520	326.25		Solid	9	C
VBS	B/D 1E Elect Room Supply AHU D	HVAC filter	N	12405	Replace filter 6 pre-filters and	Uncompacted fibreglass/metal	2/year	48	1.36	5760	163.13		Solid	9	C

Appendix A2 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM PRIMARY SYSTEM COMPONENTS

System	Component	Type	Rad/ Non-rad	Room/ Location	Waste Source	Waste Description	Frequency(1)	Waste Quantity(2)				Radio-activity	Waste Form	Disposability	
								cubic feet each change	cubic metres each change	cubic feet plant life	cubic metres plant life			Pre-Conditioning(3)	Disposal Route(3)
					6 high efficiency										
VFS	Containment Supply AHU Low Efficiency Filter A	HVAC filter	N	40503	Replace filter 2 pre-filters and 2 high efficiency	Uncompacted fibreglass/metal	2/year	16	0.45	1920	54.38		Solid	9	C
VFS	Containment Supply AHU Low Efficiency Filter B	HVAC filter	N	40503	Replace filter 2 pre-filters and 2 high efficiency	Uncompacted fibreglass/metal	2/year	16	0.45	1920	54.38		Solid	9	C
VFS	Containment Supply High Efficiency Filter A	HVAC filter	N	40503	Replace filter 2 pre-filters and 2 high efficiency	Uncompacted fibreglass/metal	2/year	16	0.45	1920	54.38		Solid	9	C
VFS	Containment Supply High Efficiency Filter B	HVAC filter	N	40503	Replace filter 2 pre-filters and 2 high efficiency	Uncompacted fibreglass/metal	2/year	16	0.45	1920	54.38		Solid	9	C
VFS	Containment Exh Upstream High Efficiency Filter A	HVAC filter	R	40551	Replace filter 4 filters	Uncompacted fibreglass/metal	once/3 yrs	16	0.45	320	9.06	LLW less than 5 mr/hr (.05 mSv/hr)	Solid	3	B
VFS	Containment Exh Upstream High Efficiency Filter B	HVAC filter	R	40552	Replace filter 4 filters	Uncompacted fibreglass/metal	once/3 yrs	16	0.45	320	9.06	LLW less than 5 mr/hr (.05 mSv/hr)	Solid	3	B
VFS	Containment exh HEPA Filter A	HVAC filter	R	40551	Replace filter 4 filters	Uncompacted fibreglass/metal	once/5 yrs	16	0.45	192	5.44	LLW less than 5 mr/hr (.05 mSv/hr)	Solid	3	B
VFS	Containment exh HEPA Filter B	HVAC filter	R	40552	Replace filter 4 filters	Uncompacted fibreglass/metal	once/5 yrs	16	0.45	192	5.44	LLW less than 5 mr/hr (.05 mSv/hr)	Solid	3	B
VFS	Cont exh charcoal filter A	HVAC filter	R	40551	Replace charcoal	Granulated charcoal	once/10 yrs	85.7	2.43	514.2	14.56	LLW less than 5 mr/hr (.05 mSv/hr)	Solid	3	B
VFS	Cont exh charcoal filter B	HVAC filter	R	40552	Replace charcoal	Granulated charcoal	once/10 yrs	85.7	2.43	514.2	14.56	LLW less than 5 mr/hr (.05 mSv/hr)	Solid	3	B
VFS	Cont exh downstream High Efficiency Filter A	HVAC filter	R	40551	Replace filter 4 filters	Uncompacted fibreglass/metal	once/3 yrs	16	0.45	320	9.06	LLW less than 5 mr/hr (.05 mSv/hr)	Solid	3	B
VFS	Cont exh downstream High Efficiency Filter B	HVAC filter	R	40552	Replace filter 4 filters	Uncompacted fibreglass/metal	once/3 yrs	16	0.45	320	9.06	LLW less than 5 mr/hr (.05 mSv/hr)	Solid	3	B
VHS	Health Physics & Hot Machine Shop AHU A	HVAC filter	N	40503	Replace filter 8 pre-filters and	Uncompacted fibreglass/metal	2/year	64	1.81	7680	217.5		Solid	9	C

Appendix A2 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM PRIMARY SYSTEM COMPONENTS

System	Component	Type	Rad/ Non-rad	Room/ Location	Waste Source	Waste Description	Frequency(1)	Waste Quantity(2)				Radio-activity	Waste Form	Disposability	
								cubic feet each change	cubic metres each change	cubic feet plant life	cubic metres plant life			Pre-Conditioning(3)	Disposal Route(3)
					8 high efficiency										
VHS	Health Physics & Hot Machine Shop AHU B	HVAC filter	N	40503	Replace filter 8 pre-filters and 8 high efficiency	Uncompacted fibreglass/metal	2/year	64	1.81	7680	217.5		Solid	9	C
VHS	Machine Tool Exhaust Fan Filter	HVAC filter	R	40503	Replace Filter	Uncompacted fibreglass/metal	2/year	8	0.23	960	27.19	LLW	Solid	3	B
VRS	Radwaste Bldg Supply AHU A	HVAC filter	N	50300	Replace filter 6 pre-filters and 6 high efficiency	Uncompacted fibreglass/metal	2/year	48	1.36	5760	163.13		Solid	9	C
VRS	Radwaste Bldg Supply AHU B	HVAC filter	N	50300	Replace filter 6 pre-filters and 6 high efficiency	Uncompacted fibreglass/metal	2/year	48	1.36	5760	163.13		Solid	9	C
VRS	Radwaste Blgd Exhaust	HVAC filter	R	50353	Repllace 18 HEPA	Uncompacted fibreglass/metal	Once / 2yrs	72	2.04	12960	61.17	LLW	Solid	3	B
VRS	Radwaste Blgd Exhaust	HVAC pre1filter	R	50353	Replace filter 18 pre-filters	Uncompacted fibreglass/metal	3/ year	72	2.04	12960	367.03	LLW	Solid	3	B
VTS	Turbine Bldg Personnel Area AHU A	HVAC filter	N	20510	Replace filter 4 pre-filters and 4 high efficiency	Uncompacted fibreglass/metal	2/year	32	0.91	3840	108.75		Solid	9	C
VTS	Turbine Bldg Personnel Area AHU B	HVAC filter	N	20510	Replace filter 4 pre-filters and 4 high efficiency	Uncompacted fibreglass/metal	2/year	32	0.91	3840	108.75		Solid	9	C
VTS	Turbine Bldg Electrical Eqp Room AHU A	HVAC filter	N	20510	Replace filter 6 pre-filters and 6 high efficiency	Uncompacted fibreglass/metal	2/year	48	1.36	5760	163.13		Solid	9	C
VTS	Turbine Bldg Electrical Eqp Room AHU B	HVAC filter	N	20510	Replace filter 6 pre-filters and 6 high efficiency	Uncompacted fibreglass/metal	2/year	48	1.36	5760	163.13		Solid	9	C
VXS	Annex Bldg General Area AHU A	HVAC filter	N	40499	Replace filter 4 pre-filters and 4 high efficiency	Uncompacted fibreglass/metal	2/year	32	0.91	3840	108.75		Solid	9	C
VXS	Annex Bldg General Area AHU B	HVAC filter	N	40499	Replace filter 4 pre-filters and 4 high efficiency	Uncompacted fibreglass/metal	2/year	32	0.91	3840	108.75		Solid	9	C
VXS	Annex Bldg Equipment Room AHU A	HVAC filter	N	40500	Replace filter 5 pre-filters and 5 high efficiency	Uncompacted fibreglass/metal	2/year	40	1.13	4800	135.94		Solid	9	C
VXS	Annex Bldg Equipment	HVAC filter	N	40500	Replace filter 5 pre-filters and	Uncompacted fibreglass/metal	2/year	40	1.13	4800	135.94		Solid	9	C

Appendix A2 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM PRIMARY SYSTEM COMPONENTS

System	Component	Type	Rad/ Non-rad	Room/ Location	Waste Source	Waste Description	Frequency(1)	Waste Quantity(2)				Radio-activity	Waste Form	Disposability	
								cubic feet each change	cubic metres each change	cubic feet plant life	cubic metres plant life			Pre-Conditioning(3)	Disposal Route(3)
	Room AHU B				5 high efficiency										
VXS	MSIV Compartment A AHU-A	HVAC filter	N	12506	Replace filter 2 filters	Uncompacted fibreglass/metal	2/year	8	0.23	960	27.19		Solid	9	C
VXS	MSIV Compartment B AHU-B	HVAC filter	N	12504	Replace filter 2 filters	Uncompacted fibreglass/metal	2/year	8	0.23	960	27.19		Solid	9	C
VXS	MSIV Compartment B AHU-C	HVAC filter	N	12504	Replace filter 2 filters	Uncompacted fibreglass/metal	2/year	8	0.23	960	27.19		Solid	9	C
VXS	MSIV Compartment A AHU-D	HVAC filter	N	12506	Replace filter 2 filters	Uncompacted fibreglass/metal	2/year	8	0.23	960	27.19		Solid	9	C
VXS	Switchgear Room AHU A	HVAC filter	N	40500	Replace filter 5 pre-filters and 5 high efficiency	Uncompacted fibreglass/metal	2/year	40	1.13	4800	135.94		Solid	9	C
VXS	Switchgear Room AHU B	HVAC filter	N	40500	Replace filter 5 pre-filters and 5 high efficiency	Uncompacted fibreglass/metal	2/year	40	1.13	4800	135.94		Solid	9	C
VXS	Mechanical Equipment Area AHU A	HVAC filter	N	40503	Replace filter 2 pre-filters and 2 high efficiency	Uncompacted fibreglass/metal	2/year	16	0.45	1920	54.38		Solid	9	C
VXS	Mechanical Equipment Area AHU B	HVAC filter	N	40503	Replace filter 2 pre-filters and 2 high efficiency	Uncompacted fibreglass/metal	2/year	16	0.45	1920	54.38		Solid	9	C
VXS	Valve/Piping Penetration Room AHU A	HVAC filter	N	12306	Replace filter 1 filter	Uncompacted fibreglass/metal	2/year	4	0.11	480	13.59		Solid	9	C
VXS	Valve/Piping Penetration Room AHU B	HVAC filter	N	12306	Replace filter 1 filter	Uncompacted fibreglass/metal	2/year	4	0.11	480	13.59		Solid	9	C
VZS	Service Module AHU A	HVAC filter	N	60310	Replace filter 2 pre-filters and 2 high efficiency	Uncompacted fibreglass/metal	2/year	16	0.45	1920	54.38		Solid	9	C
VZS	Service Module AHU B	HVAC filter	N	60310	Replace filter 2 pre-filters and 2 high efficiency	Uncompacted fibreglass/metal	2/year	16	0.45	1920	54.38		Solid	9	C
VZS	Engine Room AHU A	HVAC filter	N	60310	Replace filter 3 pre-filters and 3 high efficiency	Uncompacted fibreglass/metal	2/year	24	0.68	2880	81.56		Solid	9	C
VZS	Engine Room AHU B	HVAC filter	N	60310	Replace filter 3 pre-filters and 3 high efficiency	Uncompacted fibreglass/metal	2/year	24	0.68	2880	81.56		Solid	9	C

Appendix A2 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM PRIMARY SYSTEM COMPONENTS

System	Component	Type	Rad/ Non-rad	Room/ Location	Waste Source	Waste Description	Frequency(1)	Waste Quantity(2)				Radio-activity	Waste Form	Disposability	
								cubic feet each change	cubic metres each change	cubic feet plant life	cubic metres plant life			Pre-Conditioning(3)	Disposal Route(3)
various	Doors	Doors (266)	N	various	Replace gaskets	Fibreglass cloth	once/lop	22.17	0.63	22.17	0.63		Solid	9	C
various	Fire doors	Doors (157)	N	various	Replace gaskets	Fibreglass cloth	once/lop	13.08	0.37	13.08	0.37		Solid	9	C
various	Hatches	Hatches (33)	N	various	Replace gaskets	Fibreglass cloth	once/lop	5.51	0.16	5.51	0.16		Solid	9	C
FHS	Refuelling pool under water filtration system	Filter	R	1100	Replace cartridge	Pleated polyester	once/yr	2.95	0.08	177	5.01	LLW	Solid	9	B
FHS	Underwater cameras	Camera (4)	R	1100	Minimal waste							LLW	Solid	9	C or B
FHS	FHS underwater camera sys	Camera	R	1100	Minimal waste							LLW	Solid	9	C or B
FHS	Fixed underwater light	Light	R	1100	Minimal waste							LLW	Solid	9	C or B
FHS	Portable underwater light	Light	R	1100	Minimal waste							LLW	Solid	9	C or B
WSS	Resin slurry inlet camera	Camera	R	12471	Minimal waste							LLW	Solid	9	C or B
WSS	Resin slurry recirc camera	Camera	R	12372	Minimal waste							LLW	Solid	9	C or B
WSS	Resin slurry conditions monitor	Camera	R	12253	Minimal waste							LLW	Solid	9	C or B
4033	Hot machine shop decon system portable	Skid	R	40358								LLW	Solid	9	C or B
4033	Hot machine shop decon glove box	Skid	R	40358								LLW	Solid	9	C or B
all sys	Fans blowers and drives	Fans (122)	R/N										Solid	If N; 9. If R; 11	If N; C. If R; B
all sys	Instrumentation elements	Instruments (3337)	R/N	various	Replace elements							LLW	Solid	If N; 9. If R; 11	If N; C. If R; B
RXS	Control rod cluster	RX control (53)	R	none			once/20 yrs	198.75	5.63	596.25	16.9	ILW	Solid	12	H
RXS	Gray rod cluster	RX control (16)	R	none			once/20 yrs	60	1.7	180	5.1	ILW	Solid	12	H
RXS	Fuel assembly	RX control (157)	R	none	Burn-up	Spent fuel rods	40%/18 mos	485	13.74	19400	549.42	HLW		12	H
all sys	Wire and cable ac	ac circuits (2498)	R/N											If N; 9. If R; 3	If N; C. If R; B
all sys	Wire and cable dc	dc circuits (328)	R/N											If N; 9.	If N; C.

Appendix A2 (cont.)

IDENTIFICATION OF WASTE ARISINGS FROM PRIMARY SYSTEM COMPONENTS

System	Component	Type	Rad/ Non-rad	Room/ Location	Waste Source	Waste Description	Frequency[1]	Waste Quantity[2]				Radio-activity	Waste Form	Disposability	
								cubic feet each change	cubic metres each change	cubic feet plant life	cubic metres plant life			Pre-Conditioning[3]	Disposal Route[3]
														If R; 3	If R; B
various	Power transformers	Transformers (50)	R/N											If N; 9. If R; 3	If N; C. If R; B
RCS	Pressuriser heaters	Heater (31)	R	11403										9	C or B
CVS	Boric acid tank immersion heater	Heater (2)	N	381										9	C
CVS	Boric acid batching tank immersion heater	Heater	N	40442										9	C
VFS	Containment exhaust elec ht	Heater (2)	R	40601										9	C or B
BDS	Electrodeionisation filters A&B	Filter (4)	N	2000	Replace cartridge		once/6 mos	6.72	0.19	806	22.83			9	C
BDS	Electrodeionisation units A&B	Ion migration	R	2000	Replace stack	Resin/membrane module	once/12 yrs	27	0.76	135	3.82	LLW	Solid	1, possibly 13	B, Possibly F
BDS	Electrodeionisation units A&B (alternate supplier)	Ion migration	R	2000	Replace stack	Resin/membrane module	once/5 yrs	31.67	0.9	380	10.76	LLW	Solid	1, possibly 13	B, Possibly F
DTS	Electrodeionisation filters A&B	Filter (2)	N	20300	Replace cartridge		once/6mos	3.36	0.1	403	11.41			9	C
DTS	Electrodeionisation unit	Ion-migration	N	20300	Replace stack	Resin/membrane module	once/12 yrs	47.25	1.34	236	6.68			9	C
DTS	Reverse osmosis filters A&B	Filter (2)	N	20300	Replace cartridge		once/6 mos	3.36	0.1	403	11.41			9	C
DTS	Reverse osmosis units 1&2	Ion-migration	N	20300	Replace modules		once/7 yrs	557	15.77	4774	135.2			9	C

Notes:
1. Frequency based on design engineer experience volume is actual size.
2. Volume is actual size
3. See Appendix A5 for key of pre-conditioning and disposal routes
4. Based on a 79 m^3 (21,000 gallon) tank.
5. Radioactive and non-radioactive oil will not be stored in the same vessel.
6. Each cell is 295 kg (650 lb); of this weight, approximately 90% is lead. It is highly likely that the lead will be recycled and reused.
7. Two battery exchanges are used. The third will be included in determining decommissioning wastes.

Appendix A3

ESTIMATED RADWASTE ARISING FROM LARGE-VOLUME COMPONENTS AT DECOMMISSIONING

System	Waste Description		Waste Level	Volume[1]		Mass		Disposability		Notes
	Component	Type		cubic feet	cubic metres	pounds	tonnes	Pre-Conditioning	Disposal Route[5]	
CCS	Spent fuel system filter B	Filter	ILW	29	0.82	2,600	1.18	2	A	(2)
CVS	Letdown HX	Heat exchanger	ILW	116	3.28	14,040	6.38	9 & 4	A	
CVS	Reactor coolant filter A	Filter	ILW	29	0.82	2,600	1.18	2	A	(2)
CVS	Reactor coolant filter B	Filter	ILW	29	0.82	2,600	1.18	2	A	(2)
CVS	Regenerative HX	Heat exchanger	ILW	216	6.12	7,240	3.29	9 & 4	A	
RCS	Pressuriser	Tank	ILW	2,100	59.47	336,680	153.04	9 & 4	A	
RCS	Pressuriser heaters (31)	Heater	ILW	2	0.05	358	0.16	9 & 4	A	
RCS	SG 1 RX coolant pump	Pump	ILW	885	25.07	179,862	81.76	9 & 4	A	
RCS	SG 1 RX coolant pump	Pump	ILW	885	25.07	179,862	81.76	9 & 4	A	
RCS	SG 2 RX coolant pump	Pump	ILW	885	25.07	179,862	81.76	9 & 4	A	
RCS	SG 2 RX coolant Ppump	Pump	ILW	885	25.07	179,862	81.76	9 & 4	A	
RXS	Core barrel	Internals	ILW	3,627	102.72	132,900	60.41	9 & 4	A	
RXS	Core barrel hold down spring	Internals	ILW	4.65	0.13	2,277	1.0	9 & 4	A	

Appendix A3 (cont.)

ESTIMATED RADWASTE ARISING FROM LARGE-VOLUME COMPONENTS AT DECOMMISSIONING

System	Waste Description		Waste Level	Volume(1)		Mass		Disposability		Notes
	Component	Type		cubic feet	cubic metres	pounds	tonnes	Pre-Conditioning	Disposal Route(5)	
RXS	Core barrel nozzle	Internals	ILW	15	0.43	4,729	2.15	9 & 4	A	
RXS	Core shroud assembly	Internals	ILW	1,542	43.68	44,357	20.16	9 & 4	A	
RXS	Direct vessel injection A deflector	Internals	ILW	0.5	0.01	250	0.1	9 & 4	A	
RXS	Direct vessel injection B deflector	Internals	ILW	0.5	0.01	250	0.1	9 & 4	A	
RXS	Guide tube assemblies (69)	Internals	ILW	671	18.99	49,404	22.5	9 & 4	A	
RXS	Head and vessel pins	Internals	ILW	1	0.02	330	0.15	9 & 4	A	
RXS	Head cooling nozzles	Internals	ILW	–	–	–	–	9 & 4	A	
RXS	Irradiation specimen guide tubes	Internals	ILW	–	–	1,975	0.9	9 & 4	A	
RXS	lower core support Plate	Internals	ILW	144	4.07	46,342	21.06	9 & 4	A	
RXS	Lower support plate fuel alignment pins	Internals	ILW	0.37	0.01	94.2	0.04	9 & 4	A	
RXS	Non-threaded fasteners	Internals	ILW	–	–	–	–	9 & 4	A	
RXS	Radial supports (4)	Internals	ILW	2.28	0.06	1,142	0.52	9 & 4	A	
RXS	Reactor cavity neutron shield lower (4)	Internals	ILW	12.5	0.354	5,572	2.58	9 & 4	A	
RXS	Reactor cavityneutron shield middle (4)	Internals	ILW	12.5	0.354	5,052	2.3	9 & 4	A	
RXS	Reactor lower internals	Internals	ILW	(see individual pieces)		205,310	93.32	9 & 4	A	
RXS	Reactor upper internals	Internals	ILW			116,659	53.03	9 & 4	A	

Appendix A3 (cont.)

ESTIMATED RADWASTE ARISING FROM LARGE-VOLUME COMPONENTS AT DECOMMISSIONING

System	Waste Description		Waste Level	Volume(1)		Mass		Disposability		Notes
	Component	Type		cubic feet	cubic metres	pounds	tonnes	Pre-Conditioning	Disposal Route(5)	
RXS	Reactor vessel	Tank	ILW	11,603	328.61	600,166	272.8	9 & 4	A	
RXS	Secondary core support	Internals	ILW	70	1.99	4,464	2.03	9 & 4	A	
RXS	Support columns (42)	Internals	ILW	361	10.23	9,257	4.2	9 & 4	A	
RXS	Threaded structural Fasteners	Internals	ILW	–	–	–	–	9 & 4	A	
RXS	Upper core plate	Internals	ILW	65	1.85	7,105	3.2	9 & 4	A	
RXS	Upper core plate inserts (8)	Internals	ILW	0.25	0.01	116	0.1	9 & 4	A	
RXS	Upper support plate fuel alignment pins (314)	Internals	ILW	1.27	0.04	314	0.1	9 & 4	A	
RXS	Vortex suppression plate	Internals	ILW	11	0.3	3,195	1.45	9 & 4	A	
SFS	Spent fuel system filter A	Filter	ILW	29	0.82	2,600	1.18	2	A	(2)
WLS	Resin fines filter	Filter	ILW	29	0.82	2,600	1.18	2	A	(2)
WLS	Waste after-filter	Filter	ILW	29	0.82	2,600	1.18	2	A	(2)
WLS	Waste prefilter	Filter	ILW	29	0.82	2,600	1.18	2	A	(2)
All Sys	Valves (potentially radioactive) (1461)	Valve	LLW	–	–	–	–	9	C or B	
CCS	Component cooling water pump A	Pump	LLW	–	–	5961.30	2.70	9	C or B	
CCS	Component cooling water pump B	Pump	LLW	–	–	5961.30	2.70	9	C or B	
CVS	Cation bed demineraliser	Ion exchanger	LLW	74	2.1	17,200	7.82	9	C or B	

Appendix A3 (cont.)

ESTIMATED RADWASTE ARISING FROM LARGE-VOLUME COMPONENTS AT DECOMMISSIONING

System	Waste Description		Waste Level	Volume(1)		Mass		Disposability		Notes
	Component	Type		cubic feet	cubic metres	pounds	tonnes	Pre-Conditioning	Disposal Route(5)	
CVS	Makeup miniflow HX A	Heat exchanger	LLW	5	0.15	538	0.24	9	C or B	
CVS	Makeup miniflow HX B	Heat exchanger	LLW	5	0.15	538	0.24	9	C or B	
CVS	Makeup pump A	Pump	LLW	–	–	–	–	9	C or B	
CVS	Makeup pump B	Pump	LLW	–	–	–	–	9	C or B	
CVS	Makeup pump/motor A	Pump	LLW	224.60	6.36	–	–	9	C or B	
CVS	Makeup pump/motor B	Pump	LLW	224.60	6.36	–	–	9	C or B	
CVS	Mixed bed demineraliser A	Ion exchanger	LLW	74	2.1	17,200	7.82	9	C or B	
CVS	Mixed bed demineraliser B	Ion exchanger	LLW	74	2.1	17,200	7.82	9	C or B	
RCS	Steam generator 1	Heat exchanger	LLW	25,700	727.84	1,378,329	626.51	9	C or B	
RCS	Steam generator 2	Heat exchanger	LLW	25,700	727.84	1,378,329	626.51	9	C or B	
RNS	Residual heat removal HX	Heat exchanger	LLW	1,095	31.01	49,948	22.7	9	C or B	
RNS	Residual heat removal pump A	Pump	LLW	155.00	4.39	–	–	9	C or B	
RNS	Residual heat removal pump B	Pump	LLW	155.00	4.39	–	–	9	C or B	

Appendix A3 (cont.)

ESTIMATED RADWASTE ARISING FROM LARGE-VOLUME COMPONENTS AT DECOMMISSIONING

System	Waste Description		Waste Level	Volume[1]		Mass		Disposability		Notes
	Component	Type		cubic feet	cubic metres	pounds	tonnes	Pre-Conditioning	Disposal Route[5]	
RXS	Integrated head package dome insulation	Insulation	LLW	140	3.96	3,400	1.55	9	B	
RXS	Reactor integrated head package CRDM Cooling Fans	Integrated head	LLW	26.9	0.762	13,200	6	9	C or B	
RXS	Reactor integrated head package including CRDM latch housing (closure head assembly)	Head	LLW	7,685	217.64	145,000	65.91	11	C or B	
RXS	Reactor integrated head package radial arm hoist	Integrated head	LLW	20.1	0.569	9,858	4.48	9	C or B	
RXS	Reactor integrated head package remaining items	Integrated head	LLW	58.2	1.648	28,505	12.96	9	C or B	
RXS	Reactor integrated head package shroud	Integrated head	LLW	90	2.552	44,139	20.06	9	C or B	
RXS	Reactor integrated head package Top Plate	Integrated head	LLW	14.2	0.402	6,957	3.16	9	C or B	
RXS	Reactor integrated head package tripod	Integrated head	LLW	22.5	0.637	11,017	5.01	9	C or B	
RXS	Reactor vessel cavity reflective insulation	Insulation	LLW	625	17.7	15,000	6.82	3	B	
RXS	Studs, nuts, and washers (45 sets)	Fasteners	LLW	133	3.765	33,900	15.41	3	B	
SFS	Spent fuel cooling pump A	Pump	LLW	38.30	1.08	1495.00	0.68	9	C or B	

Appendix A3 (cont.)

ESTIMATED RADWASTE ARISING FROM LARGE-VOLUME COMPONENTS AT DECOMMISSIONING

System	Waste Description		Waste Level	Volume(1)		Mass		Disposability		Notes
	Component	Type		cubic feet	cubic metres	pounds	tonnes	Pre-Conditioning	Disposal Route(5)	
SFS	Spent fuel cooling pump B	Pump	LLW	38.30	1.08	1495.00	0.68	9	C or B	
SFS	Spent fuel cooling pump/motor A	Pump	LLW	84.70	2.40	3732.00	1.69	9	C or B	
SFS	Spent fuel cooling pump/motor B	Pump	LLW	84.70	2.40	3732.00	1.69	9	C or B	
SFS	Spent fuel pool HX A	Heat exchanger	LLW	67	1.9	3,327	1.51	9	C or B	
SFS	Spent fuel pool HX B	Heat exchanger	LLW	67	1.9	3,327	1.51	9	C or B	
SFS	Spent fuel sys. demin A	Ion exchanger	LLW	95	2.69	2,455	1.12	9	C or B	
SFS	Spent fuel sys. demin B	Ion exchanger	LLW	95	2.69	2,455	1.12	9	C or B	
VFS	Containment exhaust upstream high-efficiency filter A and B, HEPA filter A and B, charcoal filter A and B, and downstream high-efficiency filter A and B	HVAC filters	LLW	1,224	34.66	17,900	8.14	3	B	(3, 4)
WGS	Delay bed A	Adsorber	LLW	88	2.49	3,423	1.56	9	C or B	
WGS	Delay bed B	Adsorber	LLW	88	2.49	3,423	1.56	9	C or B	
WGS	Guard bed	Adsorber	LLW	8	0.23	511	0.23	9	C or B	
WGS	Moisture separator	Small vessel	LLW	0.53	0.02	51	0.02	9	C or B	

Appendix A3 (cont.)

ESTIMATED RADWASTE ARISING FROM LARGE-VOLUME COMPONENTS AT DECOMMISSIONING

System	Waste Description		Waste Level	Volume[1]		Mass		Disposability		Notes
	Component	Type		cubic feet	cubic metres	pounds	tonnes	Pre-Conditioning	Disposal Route[5]	
WGS	Waste gas system gas cooler	Heat exchanger	LLW	4	0.11	194	0.09	9	C or B	
WLS	Charcoal deep bed filter	Adsorber	LLW	68	1.92	1,888	0.86	9	C or B	
WLS	Chemical waste pump	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	Chemical waste tank	Tank	LLW	268	7.58	4,234	1.92	9	C or B	
WLS	Containment sump pump A	Pump	LLW	4.60	0.13	1300.00	0.59	9	C or B	
WLS	Containment sump pump B	Pump	LLW	4.60	0.13	1300.00	0.59	9	C or B	
WLS	Degasifier discharge pump A	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	Degasifier discharge pump B	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	Degasifier vacuum pump A	Pump	LLW	-	-	1400.00	0.64	9	C or B	
WLS	Degasifier vacuum pump B	Pump	LLW	-	-	1400.00	0.64	9	C or B	
WLS	Effluent holdup pump A	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	Effluent holdup pump B	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	Effluent holdup pump C	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	Effluent holdup tank A	Tank	LLW	3,846	108.93	25,520	11.6	9	C or B	
WLS	Effluent holdup tank B	Tank	LLW	3,846	108.93	25,520	11.6	9	C or B	
WLS	Effluent holdup tank C	Tank	LLW	3,846	108.93	25,520	11.6	9	C or B	
WLS	Monitor pump A	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	

Appendix A3 (cont.)

ESTIMATED RADWASTE ARISING FROM LARGE-VOLUME COMPONENTS AT DECOMMISSIONING

System	Waste Description		Waste Level	Volume(1)		Mass		Disposability		Notes
	Component	Type		cubic feet	cubic metres	pounds	tonnes	Pre-Conditioning	Disposal Route(5)	
WLS	Monitor pump B	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	Monitor pump C	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	Monitor pump D	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	Monitor pump E	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	Monitor pump F	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	RCDT pump A	Pump	LLW	13.24	0.37	850.00	0.39	9	C or B	
WLS	RCDT pump B	Pump	LLW	13.24	0.37	850.00	0.39	9	C or B	
WLS	Reactor coolant drain tank	Tank	LLW	128	3.62	2,569	1.17	9	C or B	
WLS	Reactor coolant drain tank HX	Heat exchanger	LLW	27	0.75	855	0.39	9	C or B	
WLS	Vapor condenser	Heat exchanger	LLW	62	1.75	3,175	1.44	9	C or B	
WLS	Waste holdup pump A	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	Waste holdup pump B	Pump	LLW	6.70	0.19	248.00	0.11	9	C or B	
WLS	Waste holdup tank A	Tank	LLW	2,072	58.68	15,317	6.96	9	C or B	
WLS	Waste holdup tank B	Tank	LLW	2,072	58.68	15,317	6.96	9	C or B	
WLS	Waste ion exchanger A	Ion exchanger	LLW	40	1.14	1,496	0.68	9	C or B	
WLS	Waste ion exchanger B	Ion	LLW	40	1.14	1,496	0.68	9	C or B	

Appendix A3 (cont.)

ESTIMATED RADWASTE ARISING FROM LARGE-VOLUME COMPONENTS AT DECOMMISSIONING

System	Waste Description		Waste Level	Volume(1)		Mass		Disposability		Notes
	Component	Type		cubic feet	cubic metres	pounds	tonnes	Pre-Conditioning	Disposal Route(5)	
		exchanger								
WLS	Waste ion exchanger C	Ion exchanger	LLW	40	1.14	1,496	0.68	9	C or B	
WLS	Waste monitor tank A	Tank	LLW	2,072	58.68	15,317	6.96	9	C or B	
WLS	Waste monitor tank B	Tank	LLW	2,072	58.68	15,317	6.96	9	C or B	
WLS	Waste monitor tank C	Tank	LLW	2,072	58.68	15,317	6.96	9	C or B	
WLS	Waste monitor tank D	Tank	LLW	2,072	58.68	15,317	6.96	9	C or B	
WLS	Waste monitor tank E	Tank	LLW	2,072	58.68	15,317	6.96	9	C or B	
WLS	Waste monitor tank F	Tank	LLW	2,072	58.68	15,317	6.96	9	C or B	
WSS	Spent resin tank A	Tank	LLW	304	8.6	3,909	1.77	9	C or B	
WSS	Spent resin tank B	Tank	LLW	304	8.6	3,909	1.77	9	C or B	
WSS	Resin transfer pump	Pump	LLW	-	-	2540.00	1.15	9	C or B	

Notes:

1. Volume basis may include protruding appendages such as nozzles and brackets allowing the component to fit into an overpack for transport without modification.
2. These filters are assumed to be drained, crushed, and crud-contaminated.
3. Radiation from these filters is <5 mr/hr (50 µSv/hr).
4. This mass and volume is for each of two containment air-handling units (A and B), each having overall dimensions of approximately 6' x 6' x 34' and consisting of a prefilter, a HEPA filter, a charcoal filter, and a post-filter with associated fans and enclosures.
5. See Appendix A5 for key of disposal routes.

Appendix A4

ESTIMATED RADWASTE ARISING FROM SMALL-VOLUME COMPONENTS AT DECOMMISSIONING

System	Waste Description		Waste Level	Disposability	
	Component	Type		Pre-Conditioning	Disposal Route(1)
All Systems	Fans, blowers, and drives (122)	Fans	LLW	9	C or B
All Systems	Instrumentation elements (3337)	Instruments	LLW	9	C or B
All Systems	Wire and cable ac (2498)	ac circuits	LLW	9	C or B
All Systems	Wire and cable dc (328)	dc circuits	LLW	9	C or B
BDS	Electrodeionisation units A and B	Ion-migration equipment	LLW	9	C or B
FHS	FHS underwater camera system	Camera	LLW	9	C or B
FHS	Fixed underwater light	Light	LLW	9	C or B
FHS	Portable underwater light	Light	LLW	9	C or B
FHS	Refueling pool underwater filtration system	Filter	LLW	2 or 15	B or J
FHS	Underwater cameras (4)	Cameras	LLW	9	C or B
VFS	Containment exhaust electric heater (2)	Heaters	LLW	9	C or B
VHS	Health Physics & Hot Machine Shop AHU A	HVAC filter	LLW	2 or 3	B
VHS	Health Physics & Hot Machine Shop AHU B	HVAC filter	LLW	2 or 3	B
WSS	Resin slurry conditions monitor	Camera	LLW	9	C or B
WSS	Resin slurry inlet camera	Camera	LLW	9	C or B
WSS	Resin slurry recirculation camera	Camera	LLW	9	C or B
–	Hot machine shop decon. glove box	Skid	LLW	9	C or B

Appendix A4 (cont.)

ESTIMATED RADWASTE ARISING FROM SMALL-VOLUME COMPONENTS AT DECOMMISSIONING

System	Waste Description		Waste Level	Disposability	
	Component	Type		Pre-Conditioning	Disposal Route[1]
–	Hot machine shop portable decon. system	Skid	LLW	9	C or B
–	Power transformers (50)	Transformers	LLW	9	C or B
–	Shield below MS59	Plate (tungsten)	LLW	9	C or B

Note:
1. See Appendix A5 for key of disposal routes.

Appendix A5

KEY FOR PRECONDITIONING AND DISPOSAL METHODS

Pre-Conditioning Method	
1.	Immobilisation in a cementitious grout within a 3 m^3 (100 ft^3) RWM approved drum
2.	Immobilisation in a cementitious grout within a 3 m^3 (100 ft^3) RWM approved box
3.	Compacted (Possible super compaction) into a 200 L (55 galion) RWM approved drum and placed into HHISO container
4.	Placed in "baskets" in the RWM approved box (possibly grouted), e.g., 4 m box
5.	Collection and passed to monitoring and sampling tanks with filtration/IX
6.	Passed to WGS delay beds
7.	Collection and storage in oil tanks
8.	Sorted dependent on size/type
9.	Monitoring and swabbing (over period of time) with potential cleaning/decontamination or size reduction
10.	Potential quenching, filtration, chemical treatment, and ion exchange, as necessary
11.	Size reduction and placed in HHISO
12.	Placed in HOLTEC flask
13.	Collect and store
14.	Collect in a chemical waste tank and de-water (cross flow filtration)
15.	Collect in 200L (55 gallon) Drum
Disposal Method	
A.	Site ILW store until UK repository becomes available
B.	Sent to LLW repository for storage
C.	Recycle or free issue
D.	Discharge to site drain
E.	Discharge to atmosphere
F.	Incineration
G.	Discharge via site effluent treatment
H.	Underground HLW storage facility
I.	Off-site contractor (e.g., NSG Environmental Ltd)
J.	Send to Inutec for drying and disposal at LLWR

Appendix A6

STEEL AND CONCRETE RUBBLE FROM DEMOLISHING VARIOUS MODULES

Module	Waste Description	LLW/ILW/ HLW/Mixed	Physical/Chemical Description	Waste Form	Steel Mass		Steel Volume(1)		Concrete Mass		Concrete Volume(1)	
					Ton	Tonne	ft^3	m^3	Ton	Tonne	ft^3	m^3
CA01	Concrete/Steel	LLW	Steel framework filled with concrete	Solid	1000	907	4167	118	4757	4315	63423	1796
CA02	Concrete/Steel	LLW	Steel framework filled with concrete	Solid	33	30	138	4	157	142	2093	59
CA03	Stainless Steel	LLW	Reinforced curved tank wall section	Solid	210	191	875	25	N/A	N/A	N/A	N/A
CA04	Concrete/Steel	LLW	Barrel shaped octagonal structure houses reactor vessel	Solid	31	28	129	4	526	478	7028(2)	199(2)
	Lead/Titanium	LLW	Lead encased titanium tubes	Solid	4	4	11	0	N/A	N/A	N/A	N/A
CA05	Carbon Steel	LLW	Steel framework filled with concrete	Solid	62	56	258	7	295	268	3932	111

Note:

1. Demolished volume is based on density of 7700 kg/m^3 (480 $pounds/ft^3$) for steel and 2400 kg/m^3 (150 $pounds/ft^3$) for concrete and 11,000 kg/m^3) (700 $pounds/ft^3$)for lead filled titanium tubes. Provisions should be included in storage facilities to accommodate packaged waste that may require two to three times demolished volume. With proper decontamination prior to demolition, most of this waste will be essentially all VLLW.
2. Concrete volume based on the assumption that a concentric section around the reactor vessel cavity in the vicinity of the core will contain enough activation products to be treated as an LLW.