| REV. | DATE | AUTHOR | CHECKED BY | STATUS | MODIFICATIONS / OBSERVATIONS | APPROVED BY |
|---|---|---|---|---|---|---|
| A | 2008-11-26 | Sarah Greenwood<br>Gary Craig<br>Stefan Rosenberger<br>BONNET Claude<br>THOUVENOT Rémi | BLASE Michael | FIN | First Issue | CHAUMIN Philippe<br>BRODT Alfred |

UK EPR

AREVA

| Project Code UKEPR | UK EPR | WORK PACKAGE |
|---|---|---|
| **AREVA NP** | | 1-3-11 |
| | | |
| | | TE |

**TITLE:**

# SOLID RADIOACTIVE WASTE STRATEGY REPORT (SRWSR)

| DOCUMENT KIND : | DCC : | CONFIDENTIALITY : |
|---|---|---|
| GENERAL DESCRIPTION | BC | RESTRICTED |

| AREVA NP | FGF | NESH-G/2008/en/0123 |
|---|---|---|
| Name of the issuing company | COMPANY CODE | DOCUMENT NUMBER |

AREVA

| Document type | | |
|---|---|---|
| GENERAL DESCRIPTION | | |
| Class: RESTRICTED | Number of pages | 257 |
| | Number of appendices | 0 |

Document Title

# SOLID RADIOACTIVE WASTE STRATEGY REPORT (SRWSR)

Short Title

| Rev | Date | Author | Checked by | Modifications / Observations | Status | Approved by |
|---|---|---|---|---|---|---|
| A | 2008-11-26 | Sarah Greenwood<br>Gary Craig<br>Stefan Rosenberger<br>BONNET Claude<br>THOUVENOT Rémi | BLASE Michael | First Issue | FIN | CHAUMIN Philippe<br>BRODT Alfred |
| | | | | | | |
| | | | | | | |
| | | | | | | |
| | | | | | | |
| | | | | | | |
| | | | | | | |

| Contract : | Project | UKEPR | NESH-G/2008/en/0123 |
|---|---|---|---|
| No EOTP: | File code | TE | |
| | Subdivision | | **INTERNAL IDENTIFICATION NUMBER** |

# List of revisions

| Rev. | Date | Scope of revision | Section/Page |
|---|---|---|---|
| A | 2008-11-26 | First issue | All<br>I |

## List of ABBREVIATIONS

| Acronym | Definition |
|---|---|
| ADR | European agreement concerning international carriage of dangerous goods by road |
| ALARA | As Low As Reasonably Achievable |
| BAT | Best Available Technique |
| BERR | Business, Enterprise and Regulatory Reform (Department of) |
| BNI | Balance of Nuclear Island |
| BPEO | Best Practicable Environmental Option |
| BPM | Best Practicable Means |
| CA | Controlled Area |
| CoRWM | Committee on Radioactive Waste Management |
| COSHH | Control Of Substances Hazardous to Health (regulations) |
| CSTS | Coolant Storage and Treatment System |
| CVCS | Chemical and Volume Control System |
| DEFRA | Department for Environment, Food and Rural Affairs |
| DBE | Design Basis Earthquake |
| DfT | Department for Transport |
| DSC | Dry Shielded Container |
| DWMP | Decommissioning Waste Management Plan |
| EMIT | Examination, Maintenance, Inspection and Testing |
| FPPS/FPCS | Fuel Pond Purification System/Fuel Pond Cooling System |
| GDA | Generic Design Assessment |
| GDF | Geological Disposal Facility |
| HEPA | High Efficiency Particulate in Air |
| HHISO | Half Height ISO (International Standards Organisation) |
| HLW | High Level Waste |
| HSE | Health and Safety Executive |
| HSM | Horizontal Storage Module |
| HVAC | Heating Ventilation and Air Conditioning |
| IAEA | International Atomic Energy Agency |
| IER | Ion Exchange Resin |
| ILW | Intermediate Level Waste |

| | |
|---|---|
| IRR 99 | Ionising Radiation Regulations 1999 |
| ISF | Intermediate Storage Facility |
| ISFSI | Interim Spent Fuel Storage Installation |
| LLW | Low Level Waste |
| LLWR | Low Level Waste Repository |
| LoC | Letter of Compliance |
| MCP | Main Coolant Pipe |
| MOX | Mixed Oxide Fuel |
| MPS | Main Primary System |
| MRWS | Managing Radioactive Waste Safely |
| NCA | Non Controlled Area |
| NDA | Nuclear Decommissioning Authority |
| NII | Nuclear Installations Inspectorate |
| NPP | Nuclear Power Plant |
| NSSS | Nuclear Steam Supply System |
| NUHOMS® | NUTECH Horizontal Modular Storage |
| OND | Office for Nuclear Development |
| OSPAR | Convention for the Protection of the Marine Environment of the North East Atlantic (Oslo-Paris) |
| PCER | Pre-Construction Environmental Report |
| PRS | Periodic Review of Safety |
| PZR | Pressurizer |
| QA | Quality Assurance |
| RB | Reactor Building |
| RCCA | Rod Cluster Control Assembly |
| RCP | Reactor Coolant Pump |
| RSA 60 | Radioactive Substances Act 1960 |
| RSA 93 | Radioactive Substances Act 1993 |
| RID | European agreement concerning international carriage of dangerous goods by rail |
| RWMD | Radioactive Waste Management Directorate |
| SAPs | Safety Assessment Principles |
| SEPA | Scottish Environmental Protection Agency |
| SG | Steam Generator |
| SoLA | Substances of Low Activity (exemption order) |

| | |
|---|---|
| SRWS | Solid Radioactive Waste Strategy |
| SQEP | Suitably Qualified and Experienced Personnel |
| SSER | Safety, Security and Environmental Report |
| SWTC | Shielded Waste Transport Container |
| TAG | Technical Assessment Guide |
| TC | Transport Container |
| UK EPR | United Kingdom EPR (i.e. the plant) |
| UV | Ultra Violet Light |
| VLLW | Very Low Level Waste |
| WTB | Waste Treatment Building |

## Glossary of Terms

| Term | Explanation |
|---|---|
| Activity | A measure of the radioactivity of a substance, measured in Becquerel (Bq). |
| Aforesaid | A previously stated item. |
| Azimuth Distortion | The circumferential variation of neutron flux within the reactor core. |
| Base Case | The fundamental status of the proposal and issues, which are explored in more detail in this document. |
| Baseline | This identifies activities that could impact on nuclear safety and demonstrates that the licensee has adequate organisational structures, resources and competences to manage safety effectively. |
| Balance of Nuclear Island | The areas within the nuclear island that are not the primary circuit. |
| Best Available Technique | Best Available Techniques (BAT) are required to be considered (under EC Directive 96/61) in order to avoid or reduce emissions resulting from certain installations and to reduce the impact on the environment as a whole. Use of BAT is required by the Environment Agency when licensing the major potentially polluting industries under the IPPC legislative regime. BAT takes into account the balance between the costs and environmental benefits. |
| Best Practicable Environmental Option | A decision-making procedure that takes into account the protection and conservation of the environment. It justifies the option that provides the most benefit or the least damage to the environment as a whole, at acceptable cost, in the long term as well as the short term. |
| Commissioning | The testing of a process facility prior to routine or normal operation that ensures that all equipment works as intended prior to full active operation. |
| Conditioning | The process by which waste is physically or chemically processed to make it disposable. |
| Containment | Layers of defence intended to prevent radioactive materials reaching the environment if the confinement is breached. |
| Contingency | A way of continuing an operation or group of operations after an unwanted event has occurred. A safety control measure that forms part of a series of safety control measures designed to prevent the occurrence of an accident. |
| Controlled Area | A defined area in which specific protection measures and safety provisions are required for controlling exposures to or preventing the spread of contamination during normal working conditions, and preventing or limiting the extent of potential exposures. |
| Controlled Waste | Forms of waste from a nuclear facility that are not radioactive. The Controlled Waste Regulations 1992 defines Controlled Waste as household, industrial and commercial waste. |
| Convection | The transfer of heat and moisture by the movement of a fluid. |

| Term | Explanation |
|---|---|
| Conventional Island | The area of a nuclear facility site that contains the non-nuclear support buildings and services. |
| Criticality | A situation where a sufficient quantity of fissile material is assembled in the right geometry and concentration for a self sustaining nuclear chain reaction to take place. |
| Decommissioning | The administrative and technical actions taken to allow a nuclear site to be deconstructed and de-licensed after it has been shut down from a state of commercial operation. |
| Decontamination | Removal of unwanted radioactive contamination by a chemical or mechanical process. |
| Decontamination Factor | The ratio of the quantity of contamination before clean up treatment to that after treatment. A measure of the efficiency of the clean up process. |
| Design Basis Earthquake | A postulated earthquake that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to assure public health and safety. |
| Discharge | The release of a waste into the environment without the intention of retrieval. |
| Disposal | The placing of waste in a predefined location for long-term storage without the intention to retrieve at a later date. |
| Effluent | Untreated waste-water or other liquid. |
| Encapsulation | The process of sealing radioactive materials to prevent leakage of contamination. |
| Epoxy Resins | Thermosetting epoxide polymer that cures when mixed with a catalyst. |
| Failure | Where a system or operation does not complete its specified task or function. |
| Fission | The splitting of a heavy nucleus of uranium or plutonium into at least two other nuclei and the release of a relatively large amount of energy. Two or three neutrons are usually released during this type of transformation. |
| Fuel Assembly | A matrix design module containing nuclear fuel that is designed to facilitate the loading and unloading of fuel from the reactor core. |
| Geological Disposal Facility | This is the planed radioactive waste disposal facility. It will be built deep underground and will offer shielding and long-term protection due to its geological features. |
| Heterogeneous | A diverse mixture of components varying in size and nature. |
| High Level Waste (HLW) | Waste for which account must be taken of the radioactive decay heat in its management. HLW requires shielding and cooling. |
| Homogenisation | The process by which a material is made uniform in composition and concentration. |
| Interim Storage | The storage of radioactive materials for an defined amount of time prior to final disposal. |

| Term | Explanation |
|---|---|
| Intermediate Level Waste (ILW) | Waste for which the radioactivity content exceeds 4 GBq per tonne of alpha emitting radionuclides or 12 GBq per tonne of beta/gamma emitting radionuclides. |
| Ion Exchange Resin | An insoluble matrix substance, usually in the form of small individual beads, containing chemicals that is used to remove unwanted soluble radioactive materials from the EPR water auxiliary circuits. |
| $K_{eff}$ | $K_{eff}$ (effective) represents the rate of growth of the nuclear fission reaction. It is the rate of change of the neutron population within a system i.e. the ratio of production to loss (leakage and absorption). |
| Leachability | A measure of the propensity and solubility of a species in water (predominantly ground water). |
| Legacy | Those wastes that have been long term (surface) stored pending a long-term disposal solution. |
| Lifecycle | It is the different stages within the existence of either the nuclear facility or the waste, ranging from creation through to either decommissioning or disposal. |
| Low Level Waste (LLW) | Waste that has radioactivity content less than the threshold for ILW. LLW forms that contain sufficient radioactive material to require action for the protection of people, but not so much that it requires shielding in handling or storage as Low Level Wastes are not heat generating. |
| Neutron Flux | The number of neutrons passing through 1 square metre of area in 1 second. |
| Neutron Shield | Materials with a high neutron capture cross section. |
| Nuclear Island | The area of land on which the facilities that contain radioactive material are situated on a nuclear facility, typically containing the nuclear reactor core, steam generators and associated systems. It is shielded and separated from the outside world. |
| Osmosis | Movement of a solvent (e.g. water) through a semi-permeable membrane from a solution of high solute concentration to one of low solute concentration through a concentration gradient. |
| Passive Safety | A form of safety precaution that does not require control systems or human intervention. |
| Periodically | After a certain point in time and at intervals thereto. For example annual reviews etc. |
| Polymer | A large (macro) molecule comprising of repeat units typically connected by covalent chemical bonds. |
| Pool | An area that is filled with water that acts to absorb and moderate the radioactivity of radioactive material (such as spent fuel). |
| Primary Circuit | This is the reactor coolant system. It involves the circulation of primary coolant stream moving from the reactor vessel, where heat is exchanged to the coolant, on to the steam generator where the heat is transferred to create steam, and back to the reactor vessel. |

| Term | Explanation |
|---|---|
| Probability | The ratio of the number times the event is predicted to occur out of the total number of possible outcomes. |
| Radioactive Waste | Solid, liquid, and gaseous materials from nuclear operations that are radioactive or become radioactive (contaminated) and for which there is no further use. They will require appropriate management such as effective treatment to make safe and may require disposal. |
| Radionuclide | An unstable nuclide that emits ionising radiation. |
| Raw Waste | Waste that has not undergone any treatment or processing. |
| Redundant | A part or operation that does not serve any purpose at this point in time but is a means to maintain safety and operability over time. |
| Reprocessing | The extraction and separation of usable materials and waste from spent nuclear fuel. |
| Retrieve | The act of removal of material from its existing location, such as the retrieval of fuel from within the reactor core or the retrieval of radioactive waste from a long–term storage facility. |
| Secondary Waste | Waste that arises as a by-product from the treatment of waste from the nuclear facility. |
| Segregated | Waste that has been removed as a waste stream and organised into groups that are defined either by size, activity, usability or origin. |
| Seismic Event | The abrupt release of energy in the earth's crust causing an earth vibration or earthquake. |
| Site Footprint | The totality of ground area occupied by the nuclear facility including the nuclear and conventional islands. |
| Spent Fuel | Fuel that has been used within the nuclear reactor to an extent that it can no longer sustain a chain reaction. It is, however, still radioactive and will require appropriate treatment and disposal. |
| Supercompaction | The compaction of a material with a very high force of up to 2000 te. |
| Swarf | Metal turnings and scrap, usually from the separation of spent nuclear fuel, from the canning and cladding material prior to reprocessing. |
| Tritium | A radioactive isotope of hydrogen (one proton, two neutrons). Because it is chemically identical to natural hydrogen, tritium can easily be taken into the body by any ingestion path. It decays by beta emission and has a radioactive half-life of about 12.3 years. |
| Very Low Level Waste | A sub-category of Low Level Waste that is excluded from the requirement for regulatory control appropriate to radioactive wastes, as the radiological hazards are judged to be sufficiently low. The regulatory controls relevant to controlled wastes do apply to this form of waste. |
| Waste | A material that has no further useful commercial, industrial or manufacturing purpose and is either discarded, planned to be discarded or required to be discarded. |

| Term | Explanation |
|---|---|
| Waste Hierarchy | A principle of categorisation for the sentencing of wastes. The hierarchy includes the following waste sentencing principles, in order of most desirable first, prevent, re-use, recycle, recover, dispose. |
| Waste Streams | Groupings of wastes based on common waste origin, character or waste management route. |
| White Paper | Theses are issued by the UK government and lay out the policy or intentions of the UK government on a particular issue. |

## Table of Contents

# 1 EXECUTIVE SUMMARY

The Solid Radioactive Waste Strategy Report (SRWSR) presents waste and spent fuel management options to the Flamanville 3 (FA3) reference case for the UK EPR. It has been produced as a supporting document in the UK EPR Step 3 Submission and is referenced within the relevant chapters of the SSER 2008.

This Solid Radioactive Waste Strategy Report (SRWSR) describes how solid radioactive waste and spent fuel generated by the UK EPR over its complete lifecycle can be managed within the constraints of the current UK Government Policy and regulatory requirements. It has been produced to satisfy Health and Safety Executive and Environment Agency requirements.

This SRWSR describes and characterises the predicted arisings of solid radioactive waste and spent fuel from the UK EPR. This SRWSR sets options for waste and spent fuel treatment, conditioning, packaging, record keeping, storing, transporting and disposal. The options set out are based on nationally or internationally proven technologies and the experience gained from EPR and AREVA projects. They have been set out in recognition that some potential UK EPR operators may wish to adopt alternative waste management and spent fuel options to those of the Reference Case.

Although the Reference Case is supported by a Best Available Technique (BAT) analysis and an impact assessment, the options presented in this SRWSR provide a high degree of confidence that a BAT case can be made by utilities.

This SRWSR is in accordance with the UK Government Base Case Assumptions of the Decommissioning and Waste Management Plan (DWMP) for new build reactors. It identifies the facilities required to manage waste and spent fuel arisings of a single EPR for up to 100 years from the start of its 60 year operational life. It also considers the impact of multi-reactor units as a variant to the base case configuration.

A description of the Waste Treatment Building (WTB) for operational waste and Interim Storage Facilities (ISF) for Intermediate Level Waste (ILW) and for spent fuel for the UK EPR are provided.

The schematic below (Figure 1) illustrates the waste arisings and flows within the scope of the SRWSR.

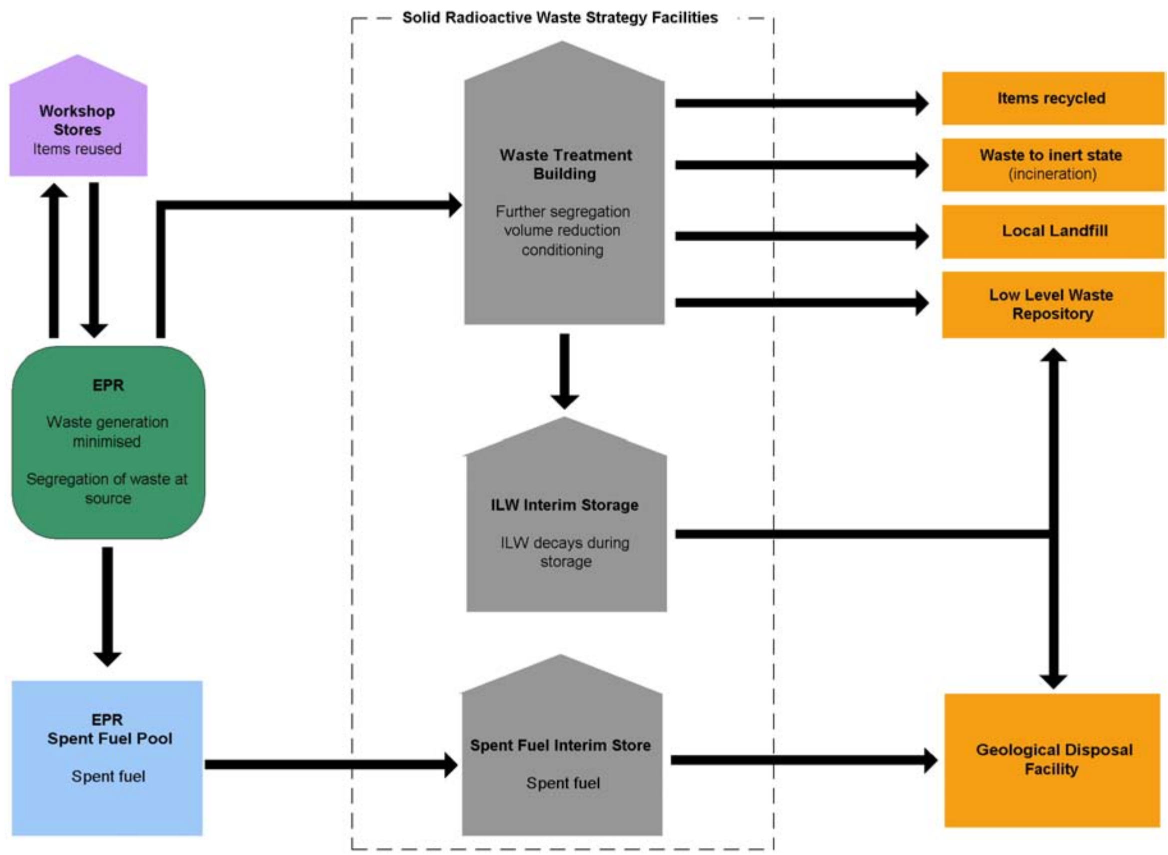



**FIGURE 1: RADIOACTIVE WASTE STREAM FLOW DIAGRAM**

Central to this SRWSR are four themes:

1. Avoidance of waste or waste minimisation;
2. Use of current UK practices or internationally proven technology;
3. Confidence that UK regulatory requirements for waste management and decommissioning are understood and can be implemented;
4. Flexibility within possible management options to address potential changes in UK waste legislation and Government Policy.

The SRWSR describes how the generation of waste will be avoided or minimised where reasonably practicable. It takes into account the waste minimisation techniques that have developed through evolutionary design features of the EPR and demonstrates that the nature of the UK EPR operational waste is very similar to those radioactive waste streams produced by other operating PWRs.

All of the UK EPR reactor wastes can be readily classified into the currently recognised UK waste categories. Further, the wastes generated by the UK EPR are fully enveloped within current UK waste management experience. The presence of relatively short-lived Intermediate Level Waste (ILW) provides an opportunity to significantly reduce (by on-site decay storage) the volumes of waste for disposal.

The requirement for quality assurance and record keeping over the UK EPR lifecycle is addressed in outline.

The options presented provide a high degree of confidence that the waste and spent fuel can be treated and conditioned into a safe state pending final disposal (or an alternative final solution). Furthermore, the options provide confidence that waste and spent fuel can be safely stored, retrieved, handled and transported prior to disposal as practices evolve.

## 1.1 Exemptions, Assumptions and Uncertainties

This SRWSR takes into account the liquid waste streams produced by a UK EPR which are delivered to the Waste Treatment Building for treatment and solidification. It does not address liquid and gaseous waste discharges, non-radioactive waste generated at the plant or wastes generated from the operation of the interim storage facilities.

This SRWSR assumes that the UK EPR will commence decommissioning immediately after shutdown and de-fuelling, and describes the radioactive waste generated from the dismantling of the nuclear island. It does not take account of the non-radioactive waste generated from the decommissioning of the conventional island and administration buildings. It does not take account of the radioactive and non-radioactive waste generated from the decommissioning of the interim waste and spent fuel storage facilities.

This SRWSR assumes that the options presented can be used by utilities to demonstrate that the design makes use of Best Available Techniques. The intention is to preserve as much flexibility as possible whilst providing confidence that BAT solutions can be identified by utilities in the future. This is judged to be sensible given the range of uncertainties at this stage. In particular, no specific sites for the UK EPR have been agreed and issues relating to the future treatment and transport of certain wastes need to be resolved with the UK Government and the Nuclear Decommissioning Authority (NDA).

This SRWSR assumes that waste service providers will continue to be available to provide waste treatment and disposal serivices.

It is anticipated that this SRWSR will be revised and updated and used as a basis to develop more specific strategies in the future.

## 1.2 Conclusions

This SRWSR sets out management arrangements for processing and interim storage of waste and spent fuel generated by the UK EPR in accordance with the UK Government policy and regulatory constraints. The SRWSR provides a high degree of confidence that the challenges associated with the management of solid waste and spent fuel from the UK EPR are fully understood and that solutions are available with the envelope of current UK and international experience. Specifically it demonstrates that:

- The production of radioactive waste will be avoided and where this is not reasonably practicable, the quantity of waste produced will be minimised;
- The radioactive wastes generated by the UK EPR are similar of those wastes generated by operating PWRs and all waste streams have process routes to interim storage or final disposal / solution;
- A number of waste and spent fuel options to the FA3 Reference Case which are based on nationally or internationally proven technologies and the experience from EPR and AREVA projects are available for the UK EPR;
- The dominance of short-lived radionuclides in some ILW will enable it to be declassified to LLW within the on-site interim storage period.

The base case waste treatment facilities for each UK EPR site include:

- Waste Treatment Building for the receipt, segregation, treatment and conditioning of solid radioactive wastes;
- Interim Storage Facility for solid ILW packages;
- Spent fuel Interim Storage Facility for receipt, packaging and safe interim storage of spent fuel.

The facilities are modular in design and may be adapted to cope with future process changes and to maintain an appropriate capacity storage volume.

## 2 SCOPE OF THIS SOLID RADIOACTIVE WASTE STRATEGY

This report presents possible solid radioactive waste strategies for the UK EPR programme. The SRWS identifies and describes the solid radioactive waste and spent fuel characteristics arising from the operating and decommissioning cycles or lifecycle of the UK EPR with respect to the current UK regulatory requirements. The operating cycle of the UK EPR is 60 years.

It sets out management arrangements for processing and interim storage of the waste and spent fuel prior to ultimate disposal in accordance with UK regulatory requirements and the Government's DWMP base case.

For each waste stream generated by the UK EPR this strategy describes the "cradle to grave" waste route demonstrating the following:

- No new or orphan wastes are generated by the EPR;
- All waste streams are classified according to the current UK classification;
- Waste is avoided or segregated at source;
- Waste is minimised;
- Waste is processed using proven and internationally recognised technologies and placed in standard packages for safe handling;
- ILW from operations and decommissioning and spent fuel is maintained in safe and secure state until disposal;
- Lifetime records are managed throughout the lifecycle.

This strategy examines the processes and describes the facilities to process the wastes and provide interim storage of ILW and spent fuel. The main facilities and processes are listed below:

A Waste Treatment Building to:

- House processing and packaging systems for all operational waste streams produced by the reactor;
- Enable processes which are installed to be removed and replaced in the future for more modern techniques;
- Provide adequate buffer storage for all waste streams prior to processing and disposal;
- Provide adequate space and handling for safe operation and adaptation for decommissioning activities;
- Provide building modularity to cater for extension (although it would be also an option to build a separate store for decomissioning wastes).

An ILW Interim Store to:

- Store all ILW generated by reactor operations for up to 100 years after first fuel loading;
- Provide building modularity to cater for extension and cater for decommissioning ILW (although it would be also an option to build a separate store for decomissioning wastes);

- Account for the impact of radioactive decay on operational ILW quantities.

A Spent Fuel Interim Store to:

- Store all spent fuel assemblies and activated components generated by the reactor for up to 100 years and final disposal. This strategy describes different wet and dry spent fuel interim storage technologies;
- Cater of the EPR's high burnup spent fuel;
- Cater for the safe import and export of fuel and which is compatible with the UK EPR Spent Fuel Pool;
- Cater for extension and the interim storage of fuel assemblies from other EPR reactors;
- Provide adequate space and handling for safe operation and monitoring.

This strategy also addresses decommissioning of the EPR and notably:

- Immediate decommissioning scenario and programme following 60 years of reactor operation;
- The quantities of solid radioactive waste generated from the decommissioning of the Nuclear Island. The waste inventory considers all Nuclear Island buildings are demolished to one metre below ground level. Building structures below -1m will be cleaned but are left in place;
- The dominance of short-lived radionuclides in some decommissioning ILW will enable it to be declassified to LLW within the on site interim storage period;
- Features facilitating decommissioning of the EPR;
- It considers that the end state of the Nuclear Island decommissioning is one where residual radioactivity has been reduced to acceptable levels enabling the licence conditions to be removed and the land reused for other purposes.

The intention is that this strategy will remain open and flexible. This will ensure that options are not foreclosed prematurely in terms of looking for and taking advantage of new developments. This might for example include the potential for centralised waste processing (incineration and melting/recycling) and storage facility service providers and spent fuel reprocessing.

A number of industrial processes, recognised to represent international good practice, are used by different plant operators to treat radioactive wastes and store spent fuel. These processes have evolved over the years and performance has improved through the integration of operational feedback experience by the plant operators and plant designers.

## 2.1 Base Case Assumptions

In developing this strategy we have aligned with existing Government Policy and guidance concerning the Decommissioning and Waste Management Plan (DWMP) Base Case [Refs.1,2 and 3] for new nuclear power stations in the UK.

The Base Case makes the following assumptions:

**Reactor**

The Base Case will be based on a single EPR station operating for 60 years.

**Radioactive waste LLW and ILW**

- LLW will be disposed of promptly after it has been generated to a suitable disposal facility;
- LLW arising during operation and decommissioning will be packaged on site by the operator and dispatched to a disposal facility promptly after they have been generated. Operators will be required to ensure that any facilities needed for packaging are available on site although it is assumed that LLW will not be conditioned on site;
- ILW from operations and decommissioning will be stored in safe and secure interim stores assumed to be on the site of the nuclear power station until decommissioning has been completed and a geological disposal facility is available to take the waste;
- Operational ILW will be conditioned and packaged as soon as reasonably feasible after it is produced and before storage on-site;
- LLW and ILW will be segregated and managed to ensure that volumes are minimised.

**Spent fuel**

- Spent fuel will be disposed of in a geological disposal facility;
- New nuclear power stations will use uranium or uranium oxide fuel;
- Spent fuel from new nuclear power stations will not be reprocessed;
- Spent fuel will be stored in cooling ponds for a period of time, followed by storage in safe and secure interim stores on the site of the power station until decommissioning has been completed and disposal facilities are available to accommodate it.
- The interim spent fuel storage facility is technically capable of being maintained or refurbished to last 100 years from the time spent fuel is firsat placed in it.

**Interim Storage**

- Utilities will be obliged to provide safe and secure interim storage facilities that are technically capable of being maintained or replaced to last for at least 100 years from the time when the waste or spent fuel is first emplaced in them;
- Utilities will be obliged to provide the stores as they are needed subject to agreement with regulators and Secretary of State.

**Decommissioning**

- The Base Case assumes that decommissioning begins when the station is shut down and ceases generating electricity for the last time;
- The Base Case assumes that the final site end state will be such that all station buildings and facilities have been removed, the site returned to a state agreed with the regulators and the planning authority and released from the control of the nuclear site licence;
- Early decommissioning of the reactor and maintenance of the interim storage facilities for 100 years (from generation of the first waste package).

**Waste Minimisation**

Operators are expected to set down the steps they will take throughout all of the stages of the station’s life to ensure that waste activity and volumes are minimised throughout reactor life; for example by careful segregation of waste arisings and by minimisation of secondary wastes.

## 2.2 EPR Waste Strategy Lifecycle

One of a number of possible EPR waste strategy lifecycles is summarised below (Table 1). This is an indicative decommissioning programme only. Decomissioning programmes will be defined by the utilities.

Table 1: Possible EPR Waste Strategy Lifecycle

| Activity | Year |
|---|---|
| **Reactor Operation (60 years)** | |
| Start of EPR | 0 |
| Shutdown of EPR | +60 (assuming no life extension) |
| First operational waste processed | +1 |
| Last operational waste processed | +61 |
| First fuel discharge to pool | +1.5 |
| Last fuel discharge to pool | +60 |
| | |
| **Reactor Decommissioning (12 years)** | |
| Decontamination of primary circuit | +61 |
| Dismantling of Nuclear Island | +66 to +72 |
| Dismantling of primary circuits | +66 to +70 |
| End of decommissioning | +72 |
| | |
| **ILW Interim Storage Facility (100 years)** | |
| Start of operation | +1 |
| End of operation | +100 |
| First ILW package to facility | +1 |
| Last ILW package to facility | +70 |
| Last ILW package to leave facility for disposal | +100 |
| | |
| **Spent fuel Interim Storage Facility (100 years)** | |
| Start of operation | +10 |
| End of operation | +100 |
| First spent fuel assembly to facility | +10 |
| Last spent fuel assembly to facility | +70 |
| Last spent fuel assemblies to leave ISF for disposal | +100 |
| | |
| **Geological Disposal Facility available** | +100 |

# 3 REGULATORY BASELINE

The purpose of this section is to describe the regulatory context of the management of solid radioactive waste in the UK and to define the main categories of solid radioactive waste.

The last comprehensive statement of government policy on radioactive waste was set out in Cm2919 'Review of Radioactive Waste Management Policy' [Ref.4]. This policy has been subsequently amended or clarified by a number of policy statements, details of which can be found at reference 5. Most recently the government has issued Cm7386 'Managing Radioactive Wastes Safety' [Ref.1] which describes a framework for implementing the deep geological disposal of radioactive waste including High Level Waste (HLW) and legacy spent fuel. We have taken account of those aspects of the policy on legacy spent fuel that are also applicable to new power station spent fuel. A number of government departments set policy relating to radioactive waste management. The key departments are described below.

## 3.1 Department for Environment, Food and Rural Affairs

The Department for Environment, Food and Rural Affairs (DEFRA) is a UK Government Policy Department. DEFRA sets environmental policy across a number of areas including radioactive substances management.

## 3.2 Department for Business, Enterprise and Regulatory Reform

The government department responsible for formulating policy on nuclear energy is the Department for Business, Enterprise and Regulatory Reform (BERR). BERR was created on the 28th June 2007. The Energy Group of BERR deals with energy related matters from production to supply and is responsible for delivering the UK government's policy goals of safe, secure and sustainable energy supplies and ultimately a low-carbon economy. The predecessor of BERR, the Department for Trade and Industry produced a consultation document on the future of nuclear power [Ref.2]. This was followed by a White Paper in January 2008 on the future of nuclear power [Ref.3]. The White Paper concluded that, faced with the challenges of addressing climate change and ensuring security of supply, nuclear power should play a role alongside other low carbon technologies in providing the UK with energy. It stated that the electricity industry should be allowed to build and operate New Nuclear Power Stations, subject to meeting planning and regulatory requirements. The White Paper also stated that it will be the responsibility of the energy companies to fund, develop and build New Nuclear Power Stations in the UK. Provision should also be made to meet the full financial burden of decommissioning and waste management costs. The legislative framework for this is outlined in The Energy Bill (2008) [Ref.6], which requires any operator of a New Nuclear Power Station to have a Funded Decommissioning Programme, approved by the Secretary of State, in place before construction of a New Nuclear Power Station begins. BERR has recently issued a consultation document on the guidance for the Funded Decommissioning Programme for New Nuclear Power Stations [Ref.7 and Ref.8].

The Office for Nuclear Development (OND) was formed by BERR on 18th September 2008. The OND has policy ownership for the Government's responsibilities on nuclear safety, security and safeguards.

## 3.3 Department for Transport

The Radioactive Materials Transport Division of the Department for Transport (DfT) is the competent body in the UK for approving packages for the transport of radioactive material. The regulation of transport packages is based on international standards and recommendations from the International Atomic Energy Agency (IAEA) in particular the IAEA Transport Regulations. The transport of nuclear materials is regulated under The Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations (2007) [Ref.9]. These regulations transpose the Commission Directive 2006/89/EC on transport of dangerous goods by road and 2006/90/EC on the transport of dangerous goods by rail. These directives require that the ADR (the European Agreement concerning the international carriage of dangerous goods by road) and the RID (the european agreement concerning international carriage of dangerous goods by rail) are implemented in member states.

## 3.4 Health and Safety Executive Safe Management of Radioactive Waste on UK Nuclear Sites

The Health and Safety Executive (HSE) regulate the operators of nuclear establishments through a nuclear site licence granted under the Nuclear Installations Act 1965 (as amended). A nuclear site licence has 36 Licence Conditions, with which the licensee (the operator) is required to demonstrate compliance. These Licence Conditions cover the safety aspects of design, manufacture, construction, commissioning, operation, maintenance and decommissioning of the installation and the management of radioactive waste on site.

Whilst virtually all of the Licence Conditions are relevant in some respect to operations involving radioactive waste, a smaller number have been framed specifically to regulate the aspects of the management of radioactive waste that fall within the remit of HSE. Those conditions that specifically relate to the management of waste are listed and described below. These have been taken into account in the design and planned operation of the EPR for when a proposed operator applies for a nuclear site licence.

### 3.4.1 Licence Condition 32 Accumulation of Radioactive Waste

Licence Condition 32 states that:

1. The licensee shall make and implement adequate arrangements for minimising so far as is reasonably practicable the rate of production and total quantity of radioactive waste accumulated on the site at any time and for recording the waste so accumulated.
2. The licensee shall submit to the Executive for approval such part or parts of the aforesaid arrangements as the Executive may specify.
3. The licensee shall ensure that once approved no alteration or amendment is made to the approved arrangements unless the Executive has approved such alteration or amendment.
4. Without prejudice to paragraph (1) of this condition the licensee shall ensure that radioactive waste accumulated or stored on the site complies with such limitations as to quantity, type and form as may be specified by the Executive.

5. The licensee shall, if so specified by the Executive, not accumulate radioactive waste except in a place and in a manner approved by the Executive.

This Licence Condition requires the licensee to make and implement adequate arrangements for minimising so far as is reasonably practicable the rate of production and total quantity of radioactive waste accumulated on the site at any time and for maintaining records of accumulated waste. The underlying purpose of the Licence Condition is to ensure that licensees have regard to the total quantities of waste that accumulate on a site and that where required the form of the waste is such that it is passively safe[1].

### 3.4.2 Licence Condition 33 Disposal of Radioactive Waste

Licence Condition 33 requires that:

The licensee shall, if so directed by the Executive, ensure that radioactive waste accumulated or stored on the site is disposed of as the Executive may specify and in accordance with an authorisation granted under the Radioactive Substances Act 1960 (RSA 60) or, as the case may be, the Radioactive Substances Act 1993 (RSA 93; Ref. 10).

The majority of Certificates of Authorisation are now granted under RSA93. Any New Nuclear Power Station will apply for an authorisation under RSA93.

This Licence Condition provides the HSE with power to direct a licensee to dispose of waste. In practice this power is rarely used.

### 3.4.3 Licence Condition 34 Leakage and Escape of Radioactive Material and Radioactive Waste

Licence Condition 34 states that:

1. The licensee shall ensure, so far as is reasonably practicable, that radioactive material and radioactive waste on the site is at all times adequately controlled or contained so that it cannot leak or otherwise escape from such control or containment.
2. Notwithstanding paragraph (1) of this condition the licensee shall ensure, so far as is reasonably practicable, that no such leak or escape of radioactive material or radioactive waste can occur without being detected, and that any such leak or escape is then notified, recorded, investigated and reported in accordance with arrangements made under condition 7.
3. Nothing in this condition shall apply to discharges or releases of radioactive waste in accordance with an approved operating rule or with disposal authorizations granted under RSA60 or, as the case may be, RSA93.

[1] In the context of decommissioning and the interim storage of nuclear matter, providing and maintaining a safety function by minimising the need for active safety systems, monitoring or prompt human intervention.

This Licence Condition requires that the licensee shall ensure, so far as is reasonably practicable, that radioactive material and radioactive waste on the site is at all times adequately controlled or contained so that it cannot leak or otherwise escape from the containment and that in the event of any fault or accident which results in a leak or escape, there are adequate means for detecting the leak and reporting the incident to the HSE.

### 3.4.4 Safety Assessment Principles

The NII provides guidance to its inspectors on the assessment of safety cases in their Safety Assessment Principles (SAPs) [Ref.11]. The SAPs relating to radioactive waste are as follows:

- A strategy should be produced and implemented for the management of radioactive waste on a site (RW.1);
- The generation of radioactive waste should be prevented or, where this is not reasonably practicable, minimised in terms of quantity and activity (RW.2);
- The accumulation of radioactive waste should be minimised (RW.3);
- Radioactive waste should be characterised and segregated to facilitate subsequent safe and effective management (RW.4);
- Radioactive waste should be processed into a passively safe state as soon as is reasonably practicable (RW.6);
- Radioactive waste should be stored in accordance with good engineering practice and in a passively safe condition (RW.5);
- Information that might be required now and in the future for the safe management of radioactive waste should be recorded and preserved (RW.7).

The SAPs also contain decommissioning principles. These reflect the regulatory expectation that decommissioning should be taken into account throughout all stages of the design and operation of nuclear plant. The decommissioning principles are as follows:

- Facilities should be designed and operated so that they can be safely decommissioned (DC.1);
- A decommissioning strategy should be prepared and maintained for each site and should be integrated with other relevant strategies (DC.2);
- Decommissioning should be carried out as soon as is reasonably practicable taking relevant factors into account (DC.3);
- A decommissioning plan and programme should be prepared and maintained for each nuclear facility throughout its lifecycle to demonstrate that it can be safely decommissioned (DC.4);
- The facility should be made passively safe before entering a care and maintenance phase (DC.5)[2];

[2] BERR guidance indicates that immediate decommissioning is preferred. The EPR reactor can be decommissioned over a period of twelve years immediately following shutdown with an assumed 40-year safe interim storage of packaged wastes. Each utility will develop their own decommissioning programme.

- Throughout the whole life-cycle of a facility the documents and records that might be required for decommissioning purposes should be identified, prepared, updated and retained (DC.6);
- Organisational arrangements should be established and maintained to ensure safe and effective decommissioning of facilities (DC.7);
- The safety management system should be periodically reviewed and modified as necessary prior to and during decommissioning (DC.8).

The HSE and the Environment Agency regulate nuclear sites under a Memorandum of Understanding [Ref.12]. This ensures a complementary approach to the management of radioactive waste.

Updated regulatory guidance on the management of ILW (and HLW) was issued in December 2007 [Ref.13].

The HSE and the Environment Agency have produced guidance on the Generic Design Assessment (GDA) process for New Nuclear Power Stations [Ref.14]. This guidance outlines the steps in the GDA process and the information that the Requesting Parties must provide in order for the regulators to assess the adequacy of the UK EPR.

## 3.5 Environment Agency

The Environment Agency was established by the 1995 Environment Act and is a Non-Departmental Public Body of the DEFRA. The Environment Agency in England and Wales and Scottish Environment Protection Agency (SEPA) in Scotland regulate the discharge or disposal of radioactive waste. This encompasses discharges into the atmosphere, sea, controlled waters and transfer for disposal at another site (for example to the Low Level Waste Repository (LLWR)). Discharges of radioactive waste to the environment (by all media) require authorisation under the RSA93. The Environment Agency have developed interim principles for Radioactive Substances Regulation [Ref.15]. Central to these principles is the selection of the Best Practicable Environmental Option (BPEO) for radioactive waste management and application of the Best Available Technique (BAT) to minimise the volume and activity of waste produced.

### 3.5.1 Radioactive Substances Regulation Environmental Principles

The Environment Agency provides interim guidance to their regulators to underpin decisions that they make relating to radioactive substances regulations [Ref.16]. The Radioactive Substances Regulation Environmental Principles (REPs) relating to radioactive waste are as follows:

- A strategy should be produced for the management of all radioactive substances;
- Producers, owners and users of radioactive substances should be accountable for the costs of managing and disposing of their radioactive substances, for associated regulation and research and for rectifying environmental damage;
- Organisations should learn from their own and others’ experience so as to continually improve their ability to protect the environment;

- Radioactive substances should be managed to avoid placing a burden on future generations their environment such that it compromises their ability to meet their needs;
- The best available technique should be used to ensure that production of radioactive waste is prevented and where that is not practicable minimised with regard to activity and quantity;
- When making decisions about the management of radioactive substances, the best available techniques should be used to ensure that the resulting environmental risk and impact are minimised;
- Radioactive substances should be characterised using the best available techniques so as to facilitate their subsequent management, including waste disposal;
- Radioactive substances should be stored using the best available techniques so that their environmental risk and environmental impact are minimised and that subsequent management, including disposal is facilitated;
- Sufficient records relating to radioactive substances and associated facilities should be made and managed so as: to facilitate the subsequent management of those substances and facilities; to demonstrate whether compliance with requirements and standards has been achieved; and to provide continuing assurance about the environmental impact and risks of the operations undertaken, including waste disposal;
- All exposures of any member of the public and of the population as a whole to ionising radiation shall be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account;
- Radiation doses to individual people shall be below the relevant dose limits and in general should be below the relevant constraints;
- A facility should be designed as to allow for defence in depth against the occurrence of radiological impacts to people and the environment;
- External and internal hazards that could affect the delivery of an environment protection function should be identified and the best available techniques used to avoid or reduce any impact;
- Facilities should be designed, built and operated using the best available techniques to minimise the impacts on people and the environment of decommissioning operations and the management of decommissioning wastes.

## 3.6 Nuclear Decommissioning Authority Radioactive Waste Management Directorate

The NDA was established by The Energy Act in 2004. The NDA is responsible for the decommissioning and clean up of all civil public-sector nuclear sites in the UK including management of radioactive wastes arising from these activities. In October 2006, following publication of the recommendations made by the Committee on Radioactive Waste Management (CoRWM), the government announced that the NDA would also be responsible for the implementation of the UK's geological disposal programme [Ref.17].

The role of the NDA RWMD is to develop and implement a GDF for the UK's higher activity wastes. The RWMD is not a regulator. Prior to April 2007 the responsibility for geological disposal rested with NIREX. In April 2007 NIREX was subsumed into NDA as RWMD.

The GDF is, according to current plans, being designed to accommodate some types of LLW (that does not meet the acceptability criteria for disposal at LLWR), Intermediate Level Waste (ILW), High Level Waste (HLW) and certain legacy spent fuel [Ref.18]. RWMD has produced a Generic Waste Package Specification [Ref.19] which defines the criterion that waste packages will be required to meet for transport to and disposal in a GDF. In light of the government White Paper on the future of nuclear power [Ref. 3], the ability of the GDF to accommodate waste from new build stations such as the UK EPR is also being addressed under the Managing Radioactive Waste Safely (MRWS) programme.

The transport of waste packages from a waste producing site to the proposed Geological Disposal Facility (GDF) will be carried out by the Nuclear Decommissioning Authority (NDA) Radioactive Waste Management Directorate (RWMD) [Ref.20]. The risks arising from this operation are assessed in a Generic Transport Safety Assessment [Ref.21]. The RWMD's Letter of Compliance (LoC) process, provides assurance that a waste transport package meets the foreseen requirements for transport of packaged radioactive waste from its site of interim storage as well as disposal within the proposed GDF.

# 4 CATEGORIES OF SOLID RADIOACTIVE WASTE

Radioactive waste is categorised in the UK dependent on the level of radioactivity in the waste. These types are LLW, ILW and HLW or heat generating waste. In addition to the waste categories listed above spent fuel is also included in the scope of this report. Further information on the UK regulatory context for the management of spent fuel is provided below. spent fuel is not a waste.

Most of the waste produced by the UK EPR, especially during decommissioning, will not be radioactive. Each waste arising will be assessed in order to determine if it is exempt from regulatory control under RSA 93 [Ref.10].

## 4.1 Exempt Wastes

RSA93 contains a number of Exemption Orders for waste which due to its low activity levels is exempt from regulatory control under the Act. These exclusions or exemptions mostly require the radioactivity concentration or specific activity of the article or substance to be below specified values, which must be satisfied in addition to the other requirements. The Substances of Low Activity (SoLA) Exemption Order [Ref.10] specifies that wastes containing artificial radioactivity of less than 0.4 Bq/g, which are essentially insoluble in water, are exempt from regulatory control under RSA93. The UK nuclear industry has produced an Industry Code of Practice for clearance and exemption of materials and waste [Ref.22]. The Code of Practice details the principles, processes and practices that should be used when determining whether an article or material may be released from any further controls on the basis of radiological protection considerations.

## 4.2 Very Low Level Waste

LLW containing lower levels of activity is further sub-categorised VLLW. The definitions of this waste and disposal routes for the waste are described below.

VLLW falls into two categories these are:

- Low volume very low level solid waste (Low volume VLLW);
- High volume very low level solid waste (High volume VLLW).

Low volume VLLW is defined in reference 23 as:

> *'radioactive waste which can be safely disposed of to an unspecified destination with municipal, commercial or industrial waste. Each 0.1 m$^3$ of waste must contain less than 400 kilobecquerels (kBq) of total activity or single items containing less than 40 kBq of total activity'.*
>
> *For wastes containing carbon-14 or tritium:*
>
> - *'In each 0.1 m$^3$, the activity limit is 4000 kBq for carbon-14 and tritium taken together, and,*
> - *for any single item, the activity limit is 400 kBq for carbon-14 and tritium taken together.*
>
> *'*

High volume VLLW is defined as 'radioactive waste with maximum concentrations of four megabecquerels per tonne (MBq/te) of total activity which can be disposed of to specified landfill sites. For waste containing tritium the concentration limit for tritium is 40 MBq/te. Controls on the disposal of this material after removal from the premises will be necessary in a manner specified by the environmental regulators, for example, disposal to an authorised waste disposal site.

## 4.3 Low Level Waste

LLW is defined in reference 23 as radioactive waste having a radioactive content not exceeding 4GBq/te of alpha or 12 GBq/te of beta/gamma activity.

Disposal of the majority of LLW in the UK is currently via the LLWR near the village of Drigg in Cumbria. Any LLW streams arising from the lifecycle of the EPR must therefore meet the acceptance criteria for the LLWR. This report describes the proposed techniques to pretreat, treat and condition the LLW to reduce the volume and activity of consignments to the LLWR.

## 4.4 Intermediate Level Waste

ILW is defined as waste with radioactivity levels which exceed the upper boundary for LLW, but which does not generate significant amounts of heat. The majority of ILW requires conditioning into an acceptable (passively safe) form prior to interim storage. Options for the conditioning of ILW to arise from the EPR are described within this document. The intention is that the final disposal location of packaged ILW will be in a GDF.

## 4.5 High Level Waste

HLW is defined as waste in which the temperature may rise significantly as a result of decay heat from the radioactive component of the waste. Heat generation therefore has to be taken into account when designing interim storage and disposal facilities. It is currently planned that HLW will be disposed of in the GDF. Regulatory guidance on the management of HLW can be found at reference 13.

## 4.6 Spent Fuel

- In the UK spent fuel is not classified as waste. However, due to the long half-life of the nuclides contained within spent fuel and the associated high levels of radioactivity the management of spent fuel is a key issue for the design of any New Nuclear Power Station. The UK government's consultation document on new nuclear power [Refs. 7 and 8] and the White Paper [Ref. 3] stated that; 'The government has concluded that any nuclear power stations that might be built in the UK should proceed on the basis that spent fuel will not be *'reprocessed'*. A concept for the direct disposal of spent fuel has not yet been finalised, this is currently being examined by the RWMD. RWMD predict that the disposability of spent fuel will be assessed through a similar process to that currently in place for ILW. A strategy will therefore be required and provision made in the design of new reactors for interim storage of spent fuel for the lifetime of the plant. Possible options for this spent fuel interim store are discussed later in this report.

# 5 CURRENT POSITION OF THIS STRATEGY

This strategy will be submitted to the Environment Agency and HSE as a Supporting Document to the Step 3 GDA Pre-Construction Environmental Report (PCER) submission in November 2008.

This section describes the progress in dialogue that EdF and AREVA have made to date (November 2008) to agree the potential means of disposing of the UK EPR radioactive waste streams through establish routes and those future disposal routes being implemented by the UK Government (like the GDF).

## 5.1 Liaison With UK Regulators and Potential Waste Service Providers

During the GDA EdF and AREVA have engaged with regulators and potential waste service providers to assess the disposability of the possible waste streams. We are committed to ensuring that the quantities of waste produced as a by-product of generating electricity are kept as low as possible and in a form which is readily disposable. The UK EPR design has evolved over a number of years by learning from Europe-wide operational practice and implementing lessons learnt to eliminate the creation of radioactive wastes where possible. Where waste creation cannot be avoided, we will endeavour to work with utilities, regulators and waste service providers to find viable solutions to minimise the creation of waste, maximise reuse and recycle and finally as a last resort dispose of the waste.

This section describes the progress in dialogue that has made to date (November 2008) through applying innovative thinking, state-of-the-art designs and engaging close liaison with UK regulators and potential waste service providers.

### 5.1.1 Current Main Waste Service Providers

As well as the waste management services provided by onsite facilitates, there are other off-site waste service providers which the EPR may utilise to ensure a safe disposal of the waste and optimisation of waste minimisation.

#### 5.1.1.1 RWMD

The RWMD currently assesses the disposability of ILW and some LLW (which are not acceptable for disposal at LLWR) through the LoC assessment process [Ref.24]. The transport of ILW to the GDF will be managed and operated by the operator of the disposal facility [Ref.25 and 26]. The ILW will be transported in robust shielded transport containers that have been approved for use in the UK and developed specifically for the safe transport of ILW.

As part of the GDA process the RWMD has developed a conceptual LoC process. RWMD is conducting a ‘nature and quantities’ assessment of the conditioned ILW packages that will be produced during the operation and eventual decommissioning of the UK EPR. A similar conceptual process has also been adopted for spent fuel. These assessments are ongoing at present and the following paragraphs describe progress in this area.

AREVA and the utilities have proposed waste conditioning and packaging options to RWMD.

The independent RWMD assessment results will be published in a Disposability Report in spring 2009. This publically available Disposability Report will outline the viability and practicability of the waste streams and package types proposed.

#### 5.1.1.2 LLWR

The LLWR is now managed by a new site licence company (LLWR Management Ltd.) who have recently updated and re-issued the Conditions for Acceptance for wastes consigned for disposal of LLW. This new guidance highlights the need for increased segregation, recycling, declassification and use of the Very Low Level Waste (VLLW) disposal route to prolong the lifetime capacity of the LLWR. In the future the LLWR could organise for combustible and metallic LLW to be collected from the site and transported to the chosen treatment facility. It may be that alternative waste service providers are engaged for incineration and recycling services.

Under the management contract it is also the remit of LLWR Management Ltd. to formulate a long term strategy for the future management of the UK's LLW.

AREVA and EdF have submitted a waste characterisation documentation to enable LLWR Management Ltd. to verify the acceptability of UK EPR LLW to the LLWR. LLW will be transported to the LLWR using the ISO containers recommended by LLWR Management Ltd.

#### 5.1.1.3 Local Landfills

The acceptability of Very Low Level Waste (VLLW) to local landfill is a matter for site specific assessments which will need to be assessed once the national position regarding VLLW is finalised.

# 6 OPERATIONAL WASTE CHARACTERISTICS

## 6.1 Introduction

During the operation of the UK EPR, waste streams will be generated. Liquid effluent generated during the operation of the UK EPR will be treated to generate a clean discharge. The treatment of liquid effluent will generate solid waste as a by product. For completeness this waste is included below.

In this section these waste streams are documented in terms of estimated total quantities and annual arisings. The physical, chemical and radiological characterisation is presented.

## 6.2 Waste Characterisation Methodology

The waste streams that will be generated from operation of a UK EPR have been characterised by EdF by examining the waste arisings from several operating French reactors. These are expected values and variations in their absolute accuracy does not effect the content of this strategy.

## 6.3 Summary of Operational Waste Arisings

The UK EPR will generate operational wastes that will be classified as ILW or LLW based on beta-gamma activity only. Table 2 shows a summary of the expected waste arisings from the UK EPR. Figure 2 and Figure 3 illustrate the percentage make-up of the wastes when segregated into LLW and ILW arisings.

**TABLE 2: SUMMARY OF RAW WASTE ARISINGS FROM OPERATION OF A UK EPR**

| Waste Stream | Waste Classification at Time of Production | Annual Rate of Production $m^3$ / year | Total Volume Expected Over Operational Phase of EPR $m^3$ |
|---|---|---|---|
| Ion Exchange Resins | ILW | 3 | 180 |
| Spent Filters | ILW / LLW | 5 | 300 |
| Dry Active Waste | ILW/LLW | 51 | 3060 |
| Tank Sludges | ILW / LLW | 1 | 60 |
| Evaporator Concentrates | LLW | 3 | 180 |
| Low Activity Resins | VLLW | 7.5 | 450 |
| Air and Water Filters | LLW | 4 | 240 |
| Oils | LLW | 2 | 120 |
| Metal Maintenance Waste | LLW/VLLW | 6 | 360 |
| **Total ($m^3$)** | - | **82.5** | **4950** |

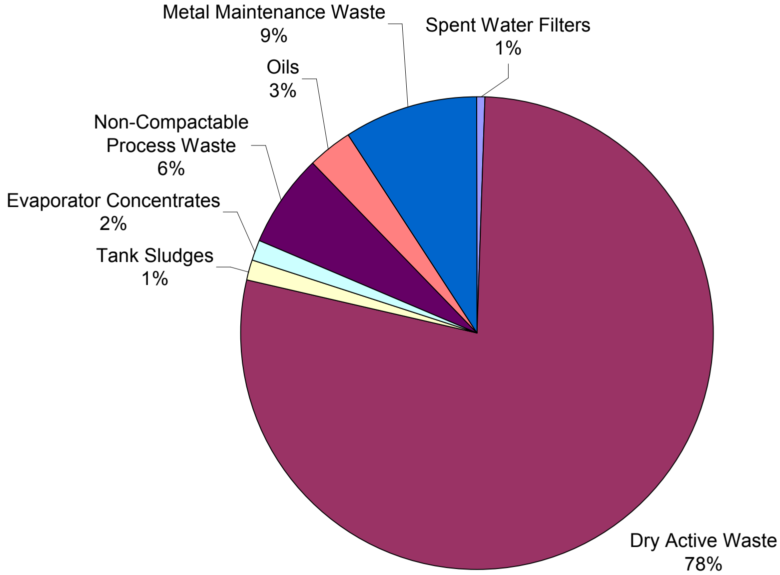



**FIGURE 2: UK EPR LLW AT TIME OF PRODUCTION**

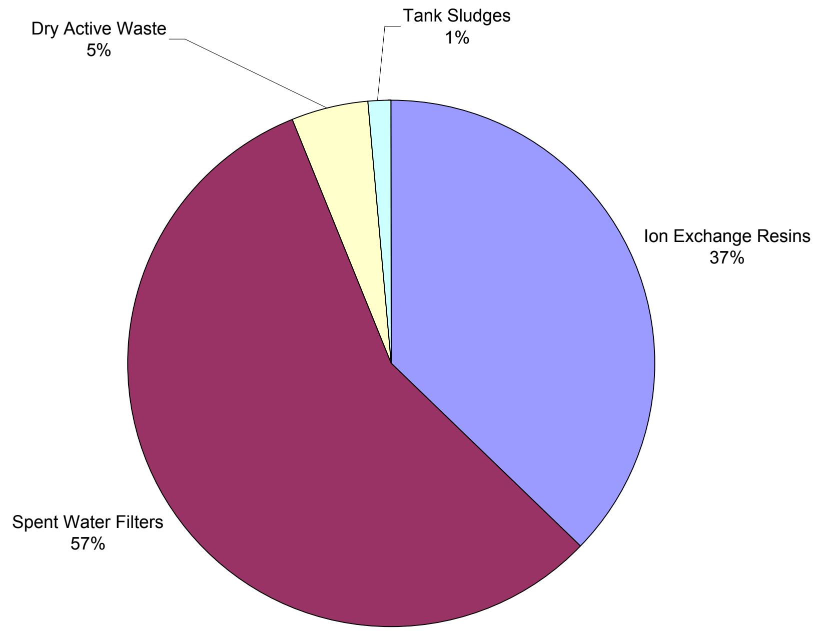



**FIGURE 3: UK EPR ILW AT TIME OF PRODUCTION**

### 6.3.1 Ion Exchange Resins

**TABLE 3: WASTE STREAM DATASHEET FOR ION EXCHANGE RESINS**

| **Waste Stream** | **Ion Exchange Resins** |
|---|---|
| **Waste Origin** | Ion exchange beds are used to minimise soluble radioactive material This material results from corrosion in the primary circuit (mainly in the steam generators) and activation of chemicals in the primary circuit) in the following UK EPR water auxiliary circuits:<br>• Chemical and Volumetric Control System;<br>• Coolant purification system;<br>Spent fuel Storage Compartment Treatment System. The ion exchange resins in the beds are periodically changed to optimise their performance. The spent resins are treated as waste. |
| **Waste Physical Description** | Small spheres (diameter range 0.3 - 1.2mm) of organic resins with polystyrenic, phenolic, acrylic or formophenolic skeleton<br>(cationic resins strongly acid, anionic resins strongly basic and mixed bed resins) |
| **Nature of Radioactive Material** | Activated corrosion products and fission products removed from auxiliary water circuits. |
| **Annual Arising** | 3 $m^3$ |
| **Total Arising** | 180 $m^3$ |
| **Waste Classification at Time of Generation** | ILW |
| **Main Radionuclides** | Nickel-63<br>Cobalt-60<br>Caesium-137<br>Cobalt-58<br>Silver-110m |
| **Hazardous Substances** | Boron: 9000 ppm<br>Mercury: 20 ppm |

### 6.3.2 Spent Filters

**TABLE 4: WASTE STREAM DATASHEET FOR SPENT FILTERS**

| **Waste Stream** | **Spent Filters** |
|---|---|
| **Waste Origin** | Corrosion occurring in the steam generators results in particulate corrosion products becoming mobile in the primary circuit [Ref.27]. Upon passing through the reactor core, these corrosion products can become neutron activated. Filters are used to capture and hence minimise such particulate material in the EPR water auxiliary circuits. Spent filter cartridges arise from the treatment lines of water auxiliary circuits: Chemical and Volumetric Control System, Boron Recycle System, Liquid Waste Treatment System, spent fuel Storage Compartment Treatment System. Water filters are withdrawn from operation on the basis of clogging and/or dose rate. |
| **Waste Physical Description** | Cartridges are composed principally of stainless steel supports with glass fibre filter and some organic materials. The amount of particulate radioactive material (metallic oxides) trapped on each filter may be variable. |
| **Nature of Radioactive Material** | Particulate activated corrosion products filtered from water auxiliary circuits. |
| **Annual Arising** | 5 $m^3$ |
| **Total Arising** | 300 $m^3$ |
| **Waste Classification at Time of Generation** | 275 $m^3$ ILW<br>25 $m^3$ LLW |
| **Main Radionuclides** | Cobalt-58<br>Iron-55<br>Cobalt-60<br>Silver-110m<br>Manganese-54 |
| **Hazardous Substances** | Boron: 6,000 ppm<br>Lead: 425 ppm<br>Nickel: 210 ppm<br>Chromium: 240 ppm<br>Cadmium: 11 ppm<br>Arsenic, antimony, mercury: 5 ppm<br>Beryllium, selenium: 0.2 ppm |

### 6.3.3 Dry Active Waste

**TABLE 5: WASTE STREAM DATASHEET FOR DRY ACTIVE WASTE**

| **Waste Stream** | **Dry Active Waste** |
|---|---|
| **Waste Origin** | Dry active waste is generated through routine and maintenance operations in the EPR nuclear island. The waste consists of contaminated personal protection equipment, monitoring swabs, plastic, clothing, contaminated tools and other process consumables. Mainly arising during outages these wastes are collected into plastic bags. The bags are put into a cast iron 200 l drum or container. |
| **Waste Physical Description** | Pre-compacted heterogeneous waste (technological waste or operational waste) segregated pieces of metal, plastic, clothes, glassware, rubble. |
| **Nature of Radioactive Material** | Contamination with fission products and activation products. |
| **Annual Arising** | 51 $m^3$ |
| **Total Arising** | 3060 $m^3$ |
| **Waste Classification at Time of Generation** | 1 $m^3$ ILW<br>50 $m^3$ LLW |
| **Main Radionuclides** | Iron-55<br>Cobalt-58<br>Cobalt-60<br>Silver-110m<br>Nickel-63 |
| **Hazardous Substances** | Boron: 100 ppm<br>Antimony: 1000 ppm |

### 6.3.4 Tank Sludges

**TABLE 6: WASTE STREAM DATASHEET FOR TANK SLUDGES**

| **Waste Steam** | **Tank Sludges** |
|---|---|
| **Waste Origin** | During operation of an EPR, particulates can settle as a sludge in the buffer and storage tanks associated with the water auxiliary circuits (e.g. Liquid Waste Treatment System, Liquid Effluents Release System etc.). These tanks are periodically cleaned out and the sludge accumulations removed for treatment as waste. |
| **Waste Physical Description** | A sludge consisting of settled metal oxide particulate. |
| **Nature of Radioactive Material** | Sludge is contaminated with activated corrosion products and fission products. Activation products and fission products, beta and beta/gamma emitters (hydrogen-3, cobalt-60, cobalt-58, manganese-54, zinc-65, silver-110m, antimony-125, caesium-134, caesium-137, beryllium-10, carbon-14, sodium-22, chlorine-36, calcium-41, iron-55, nickel-59, nickel-63, selenium-79, strontium-90, molybdenum-93, zirconium-93, niobium-94, technetium-99, palladium-107, silver-108m, tin-121m, tin-126, iodine-129, caesium-135, samarium-151) and a very low amount of alpha emitters. |
| **Annual Arising** | 1 $m^3$ |
| **Total Arising** | 60 $m^3$ |
| **Waste Classification at Time of Generation** | ILW / LLW |
| **Main Radionuclides** | Iron-55<br>Cobalt-58<br>Cobalt-60<br>Zinc-65<br>Nickel-63 |
| **Hazardous Substances** | Boron: 1.000 ppm<br>Mercury: 5 ppm<br>Lead: 335 ppm<br>Nickel: 165 ppm<br>Chromium: 190 ppm<br>Antimony: 5 ppm<br>Cadmium: 10 ppm |

### 6.3.5 Evaporator Concentrates

**TABLE 7: WASTE STREAM DATASHEET FOR EVAPORATOR CONCENTRATES**

| **Waste Steam** | **Evaporator Concentrates** |
|---|---|
| **Waste Origin** | The UK EPR proposes to make use of evaporation for the minimisation of radioactive liquid effluents arising from the non-recyclable Liquid Waste Treatment System. Evaporation will be used to minimise the discharge of active aqueous effluents. Evaporation of effluents results in the production of a sludge-like concentrate that will contain the bulk of the radioactivity initially present in aqueous effluent streams. These concentrates contain on average 40,000 ppm of boron and have a total salinity of 300 g/L and are liable to crystallise if concentrated further. |
| **Waste Physical Description** | Concentrated sludge containing some activated metal oxides. The concentrate contains a high level of boron (17,000 ppm) and can crystallise if the boron concentration exceeds 40,000 ppm. |
| **Nature of Radioactive Material** | Sludge is contaminated with activated corrosion products and fission products. |
| **Annual Arising** | 3 $m^3$ |
| **Total Arising** | 180 $m^3$ |
| **Waste Classification at Time of Generation** | LLW |
| **Main Radionuclides** | Iron-55<br>Cobalt-58<br>Cobalt-60<br>Silver-110m<br>Nickel-63 |
| **Hazardous Substances** | Boron: 1.000 ppm<br>Mercury: 5 ppm<br>Lead: 335 ppm<br>Nickel: 165 ppm<br>Chromium: 190 ppm<br>Antimony: 5 ppm<br>Cadmium: 10 ppm |

### 6.3.6 Low Activity Resins

**TABLE 8: WASTE STREAM DATASHEET FOR LOW ACTIVITY RESINS**

| **Waste Steam** | **Low Activity Resins** |
|---|---|
| **Waste Origin** | The Steam Generator Blowdown System is used to recycle blowdown water to the EPR secondary circuit [Ref. 28]. The steam generator blowdown is purified by the use of two parallel filters for the removal of suspended solids and two parallel demineralisation lines which use ion exchange resins to perform the demineralisation. |
| **Waste Physical Description** | Very low active Ion exchangers resins removed from the purification treatment lines of the Steam Generator Blowdown System. Made of balls or grains (diameter ranges 0.3 - 1.2mm) of organic resins with polystyrenic, phenolic, acrylic or formophenolic skeleton (cationic resins strongly acid, anionic resins strongly basic and mixed bed). Suppliers: PUROLITE, DOWEX, ROHM&HAAS. |
| **Nature of Radioactive Material** | Contamination with soluble activated corrosion products and fission products up to 10 Bq/g. |
| **Annual Arising** | 7.5 $m^3$ |
| **Total Arising** | 450 $m^3$ |
| **Waste Classification at Time of Generation** | LLW / VLLW |
| **Main Radionuclides** | Activation products and fission products, beta and beta/gamma emitters (Cobalt-60, caesium-134, caesium-137, beryllium-10, carbon-14, chlorine-36, calcium-41, iron-55, nickel-59, nickel-63, selenium-79, strontium-90, molybdenum-93, zironium-93, niobium-94, technetium-99, pallasium-107, silver-108m, tin-121m, tin-126, iodine-129, caesium-135, samarium-151). |
| **Hazardous Substances** | Lead: 22 ppm<br>Chromium: 18 ppm<br>Nickel: 4 ppm<br>Mercury: < 0.5 ppm |

### 6.3.7 Air and Water Filters

**TABLE 9: WASTE STREAM DATASHEET FOR AIR AND WATER FILTERS**

| **Waste Steam** | **Air and Water Filters** |
|---|---|
| **Waste Origin** | All radiation controlled areas of the nuclear auxiliary building, fuel building, safeguards buildings, reactor building, operational production centre, access building and waste treatment building are served by dedicated ventilation systems. The extract from these systems is subject to a number of airborne activity abatement techniques, including the use of HEPA filtration, before discharge to the environment. The abatement systems will therefore produce a number of spent HEPA filters over the course of reactor operations. These HEPA filters have the potential to be classed as radioactive waste should they have arrested any radioactive particulate or aerosol.<br><br>In addition, water filters may arise from filtering of the low active effluent filtration (Gaseous Treatment System, Liquid Waste Treatment System, Steam Generator Blowdown System) |
| **Waste Physical Description** | Cartridges are composed principally of stainless style supports with glass fibre filter and some organic materials. The amount of particulate radioactive material (metallic oxides) trapped on each filter may be variable. |
| **Nature of Radioactive Material** | Particulate (metal oxide) contamination up to a few Bq/g. |
| **Annual Arising** | 4 $m^3$ |
| **Total Arising** | 240 $m^3$ |
| **Waste Classification at Time of Generation** | LLW |
| **Main Radionuclides** | Cobalt-58<br>Iron-55<br>Cobalt-60<br>Silver-110m<br>Manganese-54 |
| **Hazardous Substances** | Boron: 8150 ppm<br>Lead: 904 ppm<br>Chromium: 333 ppm<br>Nickel: 291 ppm<br>Arsenic, antimony, mercury: 7 ppm<br>Beryllium: 0.3 ppm<br>Selenium: 3 ppm<br>Cadmium: 15 ppm |

### 6.3.8 Oils

**TABLE 10: WASTE STREAM DATAHEET FOR OILS**

| **Waste Steam** | **Oils** |
|---|---|
| **Waste Origin** | Oils are used in the lubrication of process pumps. Oils arising from circulators and pumps used to transfer radioactive liquids have the potential to become contaminated. |
| **Waste Physical Description** | Lubricating oil from various circulation pumps. |
| **Nature of Radioactive Material** | Lubricating oils have the potential to become slightly contaminated during service. |
| **Annual Arising** | 2 $m^3$ |
| **Total Arising** | 120 $m^3$ |
| **Waste Classification at Time of Generation** | LLW |
| **Main Radionuclides** | Cobalt-58<br>Iron-55<br>Cobalt-60<br>Silver-110m<br>Manganese-54 |
| **Hazardous Substances** | Waste oil is considered to be a hazardous substance. |

### 6.3.9 Metal Maintenance Waste

**TABLE 11: WASTE STREAM DATASHEET FOR METAL MAINTENANCE WASTE**

| **Waste Steam** | **Metal Maintenance Waste** |
|---|---|
| **Waste Origin** | During maintenance operations a variety of metals wastes can be generated. These arise from the replacement of engineering components. Redundant metal components / equipment arising from maintenance operations in the nuclear island may be contaminated. |
| **Waste Physical Description** | Metal wastes will consist of components of engineering equipment that have been replaced during maintenance operations. |
| **Nature of Radioactive Material** | Surface contaminated metal components. |
| **Annual Arising** | 6 $m^3$ |
| **Total Arising** | 360 $m^3$ |
| **Waste Classification at Time of Generation** | LLW / VLLW |
| **Main Radionuclides** | Cobalt-58<br>Iron-55<br>Cobalt-60<br>Silver-110m<br>Manganese-54 |
| **Hazardous Substances** | None identified. |

## 6.4 Storage of Radioactive Waste to Allow Reclassification

A major factor in the management of radioactive wastes is the time period for which they remain hazardous. It is uniquely inherent in the nature of radioactive materials that, through the passage of time, they become less radioactive. For radioactive wastes where the activity is dominated by relatively short lived isotopes, the waste can be reclassified to a lower hazard category after a suitable period of interim storage.

The radioactivity of operational ILW arising from the operation of the UK EPR is dominated at the time of arising by relatively short lived radionuclides. After a period of safe interim storage, the radioactivity of some of this waste will have decayed to such levels that the waste would no longer be classified as ILW. This would present the opportunity to manage the waste as LLW.

Decay storage is an important technique for the re-classification or exemption of radioactive wastes. Decay storage is often used for wastes arising from medical uses of radioactive material or for some sealed sources [Ref. 29] where the radioactive material is typically pure and has a short half-life (e.g. less than one hundred days).

Decay storage is a robust strategy for waste management as it does not require any mechanical or chemical waste treatment, and thus does not generate any secondary radioactive discharges. The site operator must simply ensure that the waste is securely and safely stored for the period needed for decay.

With regard to the UK EPR, some of the waste generated during operations is classified as ILW at the time of generation. The dominant radioisotopes at the time of generation in these wastes are cobalt-60 (half life of 5.27 years), caesium-137 (half life of 30.2 years) and iron-55 (half-life of 2.7 years). The design lifetime of the EPR interim waste store is 100 years, with operational wastes being interim stored for a maximum period of between 40 years (for wastes arising at the end of operations) and 100 years (for wastes arising near the beginning of operations). This storage period is greater than the half lives of the dominant radioisotopes within most operational wastes streams at the time of waste generation.

A portion of the waste that is categorised as ILW at the time of generation will have decayed sufficiently during the period of interim storage on site that it could be reclassified as LLW at the time of disposal. The paragraphs below illustrate this strategy with respect to EPR operational waste arisings.

### 6.4.1 Decay Storage of UK EPR Spent Water Filters

Figure 4 shows the radioactive decay characteristics for the spent water filters arising from operation of the UK EPR which, pessimistically, would be ILW at the time of generation. The graph is the total activity for all radionuclides predicted to be present in this waste stream. The primary contributors to the total initial radioactivity of this particular waste are cobalt-60, caesium-137 and iron-55. The levels of radionuclides with longer half-lives are present at levels below the LLWR threshold of 12 GBq/te beta-gamma therefore this waste decays to LLW within a few decades. Long lived radionuclides are present in quantities above the thresholds for both VLLW and Substances of Low Activity (SoLA) and so controlled decay storage to VLLW or SoLA is not feasible.

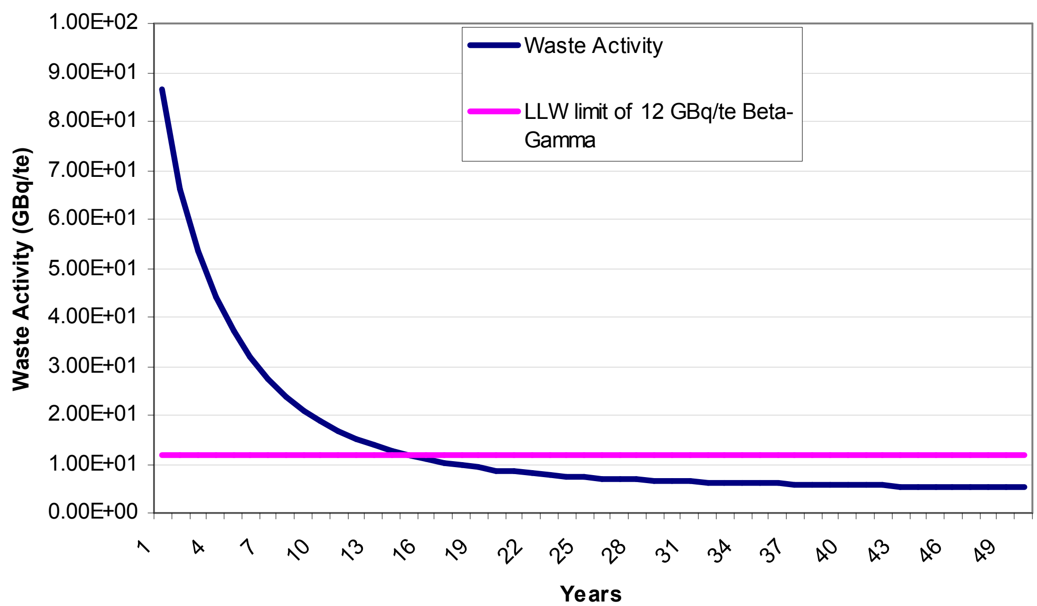



**FIGURE 4: DECAY STORAGE GRAPH FOR EPR SPENT WATER FILTERS**

### 6.4.2 Decay Storage of ILW Dry Active Waste

Figure 5 shows the decay characteristics of ILW Dry Active Waste. With this waste it is cobalt-60 and iron-55 that dominate the activity over the first few years. These decay significantly over the first decade after waste generation, allowing the waste to be reclassified as LLW after ten years.

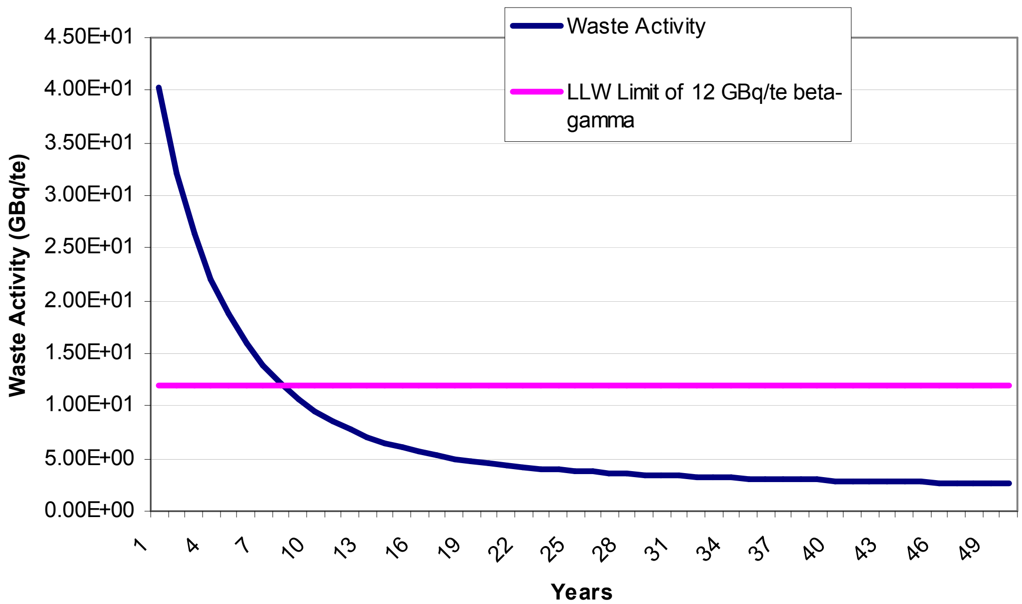



**FIGURE 5: DECAY STORAGE OF ILW DRY ACTIVE WASTE**

### 6.4.3 Decay Storage of EPR Spent Ion Exchange Resins

Figure 6 shows the radioactive decay characteristics for the spent ion exchange resins arising from operation of the UK EPR that, pessimistically, would be ILW at the time of generation. The graph shows the total activity for all radionuclides predicted to be present in this waste stream. As with the spent water filters, the primary contributors to the total radioactivity of this particular waste are cobalt-60, iron-55 and caesium-137. These decay to non-significant levels within a few decades, however, nickel-63 is present in sufficient quantities to preclude decay storage of this waste to LLW on the interim storage timescale of the UK EPR. However, the level of nickel-63 (with a half life of 100 years) decays sufficiently over three centuries for this waste to be classified as LLW after 300 years.

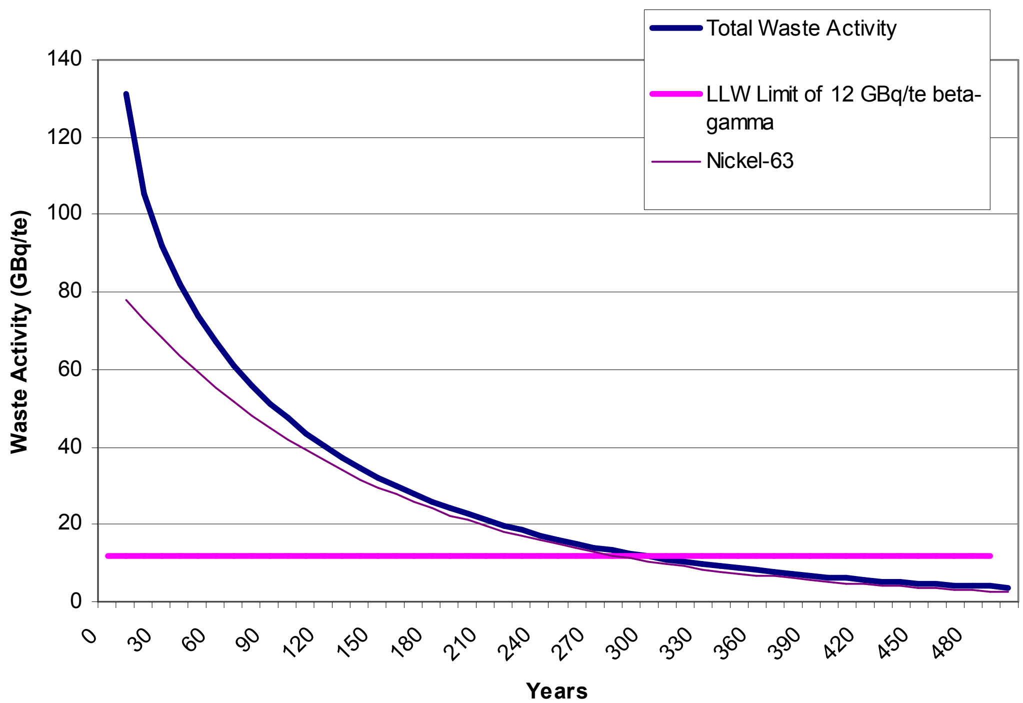



**FIGURE 6: DECAY STORAGE OF EPR SPENT ION EXCHANGE RESINS**

It should be noted that the radionuclide inventory data has been sourced from analysis of wastes arising from currently operating reactors. The UK EPR has been optimised to minimise the generation of waste arisings from the experience from operating these reactors and as such the decay graphs presented above are pessimistic.

# 7 UTILITY OPERATIONAL WASTE TREATMENT AND CONDITIONING PROCESSING OPTIONS

This section outlines the waste treatment and conditioning options that AREVA believe can be feasibly deployed for the treatment of the different types of solid radioactive wastes arising from operation of the UK EPR.

Within the constraints of the regulatory and licensing baseline for the UK EPR it is intended that there will be flexibility for individual utilities to select and optimise their own waste management strategies. This flexibility will permit changes to waste management techniques to reflect recent developments and national and international practices as new or improved options become available.

## 7.1 The Waste Hierarchy

Developing waste management options which are sustainable is an underling principle of this strategy. The waste hierarchy (see Figure 7) is a cornerstone of sustainable waste management [Refs. 30, 31 and 32]. This concept is best described by reference to waste minimisation, reuse, recycling and recovery, with disposal only undertaken where it is unavoidable.

The waste hierarchy sets out the order in which options for waste management should be considered based on environmental impact. Interpretation of the waste hierarchy finds solid waste preferable to liquid discharge and liquid discharge preferable to airborne discharge (the principle of concentrate and contain).

The aim of applying the waste hierarchy is to extract the maximum benefits from resources and to generate the minimum amount of waste for disposal.

A combination of UK EPR waste management options selected would ensure the application of the waste hierarchy, specific for each solid, liquid and gaseous waste is applied. The waste hierarchy principle will also be applied specifically to key radionuclides that are identified as having the greatest dose impact on discharge. This helps to ensure that the activity discharged to the environment is As Low As Reasonably Practicable (ALARP).

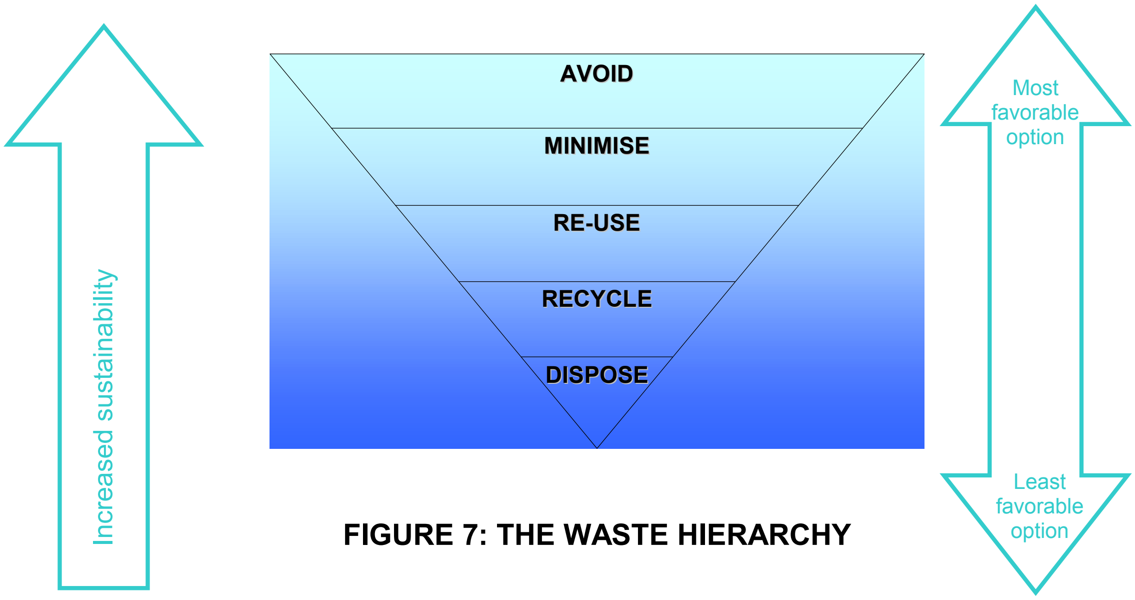



**FIGURE 7: THE WASTE HIERARCHY**

## 7.2 Best Available Technique

It is a legal requirement under the RSA93 that the Best Available Technique (BAT) must be applied both to minimise the production of relevant wastes and to minimise the discharge of radioactivity into the environment.

In regard to the Reference Case (Based on the Flamanville 3 EPR), Chapter 8 of the PCER provides an evaluation of environmental options considered and shows that the BATs have been used to minimise the production and discharge and disposal of waste.

As BAT is largely a relative, not absolute, criterion, it is essential that utilities understand the full range of factors that may influence the selection of a chosen waste management option against a range of possible alternatives. Combinations of the options presented within this SRWSR could be used by utilities to demonstrate that their chosen waste management option makes use of the BAT.

In the UK there are two particular challenges that need to be addressed by New Power Plant utilities:

- recognising the influence of site specific issues; and
- identifying what represents international ‘good practice’ through applying experience gained by the utilities in the development of specific plant and processes.

It is the vendors intention to provide confidence that BAT solutions can be identified whilst preserving as much flexibility as possible to allow the utilities to demonstrate BAT. This is judged to be sensible approach given the range of uncertainties at this stage.

## 7.3 Selected Techniques for the Treatment and Conditioning of Solid Waste Arisings

The practices selected for use in the EPR have evolved over recent decades in various plants around the world. The options are listed below (see Table 12) and described in more detail later in this report.

**TABLE 12: OVERVIEW OF FEASIBLE TECHNIQUES AVAILABLE FOR THE TREATMENT AND CONDITIONING OF EPR WASTE ARISINGS**

| Desired effect | Selected options |
|---|---|
| Concentrating and containing waste | Evaporation, filtration, ion exchange, separation, drying and dewatering. |
| Volume reduction of waste | Avoidance, sorting, decay storage, incineration, compaction, shredding, decontamination, biological treatment, melting. |
| Waste encapsulation | Cementation, polymer encapsulation. |

The options listed in Table 12 and the following figures (see Figure 8, Figure 9 and Figure 10) illustrate the waste management techniques applied to liquid and solid waste treatment that form the basis of this strategy.

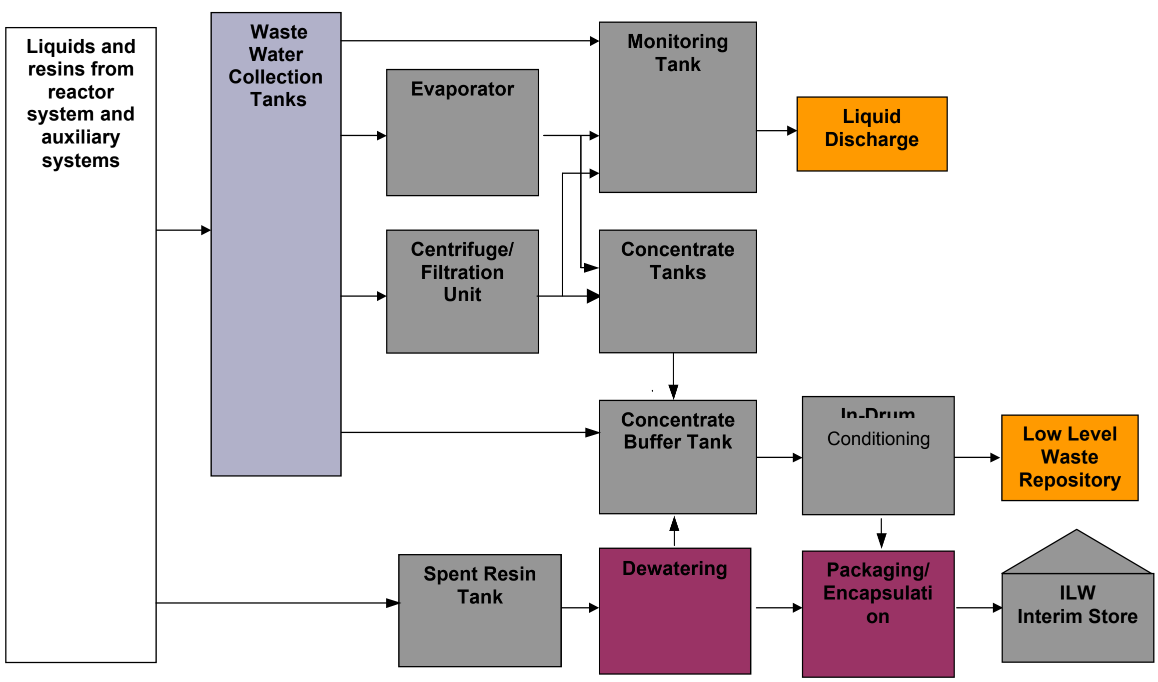



**FIGURE 8: LIQUID EFFLUENT TREATMENT SYSTEM**

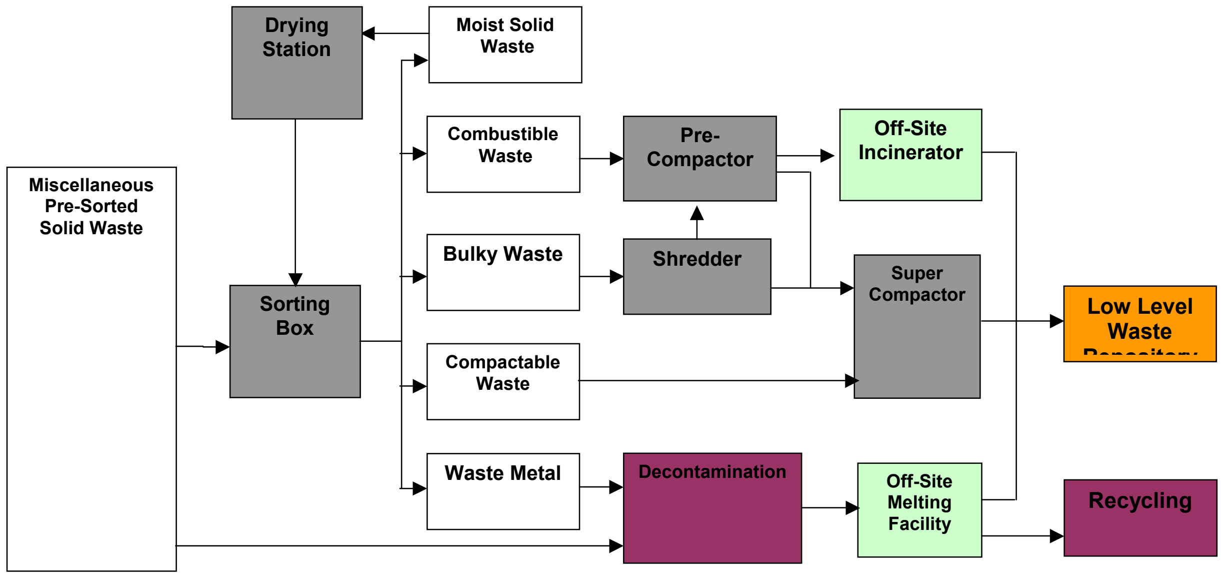



**FIGURE 9: STORAGE AND TREATMENT OF SOLID LLW**

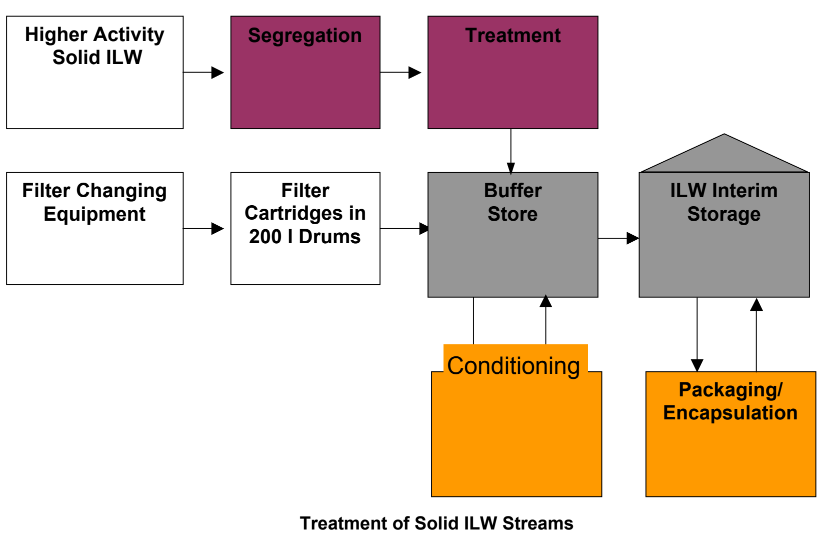



**FIGURE 10: STORAGE AND TREATMENT OF SOLID ILW STREAMS**

It is intended that this strategy will incorporate effective, tried and tested waste management options. The options above seek to take a broad view of feasible treatment and conditioning techniques as required to inform a robust waste management strategy.

In accordance with good waste minimisation practice, waste arising from the operation of the UK EPR will be segregated at source on the basis of its activity and its physical and chemical characteristics. Waste with a high liquid content, for example solid wastes arising from the treatment of liquid waste, will be dewatered. This process reduces the mass of the waste and the mobility of the radionuclides contained within the waste and facilitates downstream conditioning of the waste.

All nuclear material will be retained on-site until it has been appropriately conditioned and packaged or declared as exempt from regulatory control.

LLW will be conditioned in accordance with the conditions for acceptance for the selected disposal route [Ref. 33] and transferred off-site. Only ILW will be stored on site for extended periods until such a time as the GDF becomes available. Any ILW produced will be processed and packaged in accordance with the methods agreed in the Letter of Compliance issued by RWMD. This will ensure, so far as is practicable that the waste is suitable for eventual disposal in a GDF and can be stored in a passively safe state on-site in the interim time period. During the timescale for disposal of ILW to a disposal facility it is possible that some wastes may decay below the ILW threshold limits. Although initially stored as ILW these waste streams will be re-categorised, removed from the waste store and shipped as LLW.

Within the timeframe for final disposal of ILW there may be significant developments in treatment and conditioning processes. For example, advanced treatment and conditioning processes such as separation, transmutation and immobilisation of actinides and fission products which are currently at an early stage of development may become available within this time frame. Therefore, waste forms that provide the option for such advanced conditioning in the future whilst permitting safe interim storage may become the preferred option. Care is therefore required to ensure that the current selected treatment and conditioning options do not preclude future processing. An example of a situation where this could occur is in wastes arising from the treatment of EPR waste liquid (evaporator concentrates, resins). Early encapsulation of these materials would not only increase storage and disposal volumes but could also hinder future waste conditioning. Therefore de-watering of residues has been selected as the preferred approach (e.g. dewatering of resins, in-drum drying of evaporator concentrates) as this technique has been proven in other countries and minimises the volume of waste generated, whilst facilitating future waste conditioning [Ref. 34 and Ref. 35].

### 7.3.1 Liquid Effluent Treatment By Evaporation

Evaporation is the most effective method for chemical and radiological purification of liquid effluent from a Nuclear Power Station [Ref. 36]. The evaporator separates the dissolved and suspended liquid effluent constituents, concentrating the radioactivity in the resulting residue (see Figure 11).

Clean water leaving the evaporator can be discharged or recycled directly following monitoring to confirm compliance with discharge authorisations. The evaporator concentrate will be subject to further conditioning prior to disposal.

**Application**

Evaporation can be used for:

- Equipment drains and sump water;
- Decantation water;
- Liquid chemical waste (from decontamination operations, laboratory);
- Laundry water (detergents);
- Regenerants (chemicals).

**Design Data**

- System throughput $4m^3/h$ (typical);
- Decontamination factor (bottoms, distillate): up to $10^7$.

**Advantages**

- Reduces radioactive discharges by concentrating particulate in the evaporate;
- Minimises discharges of chemical and solid impurities;
- High decontamination factor;
- Proven system design/technology;
- Well documented operational experience;
- Customised application;
- Continuous and automatic operation.

- **Disadvantages**

- Unsuitable for liquids containing a high foam component;
- Evaporators can have higher operational costs than non-thermal separation technologies.

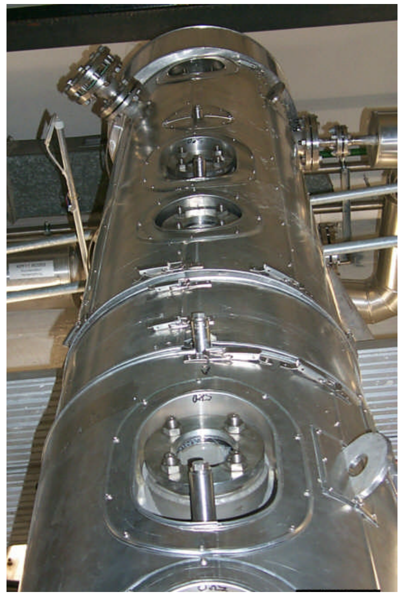


**FIGURE 11: EVAPORATOR USED FOR LIQUID EFFLUENT TREATMENT**

### 7.3.2 Liquid Effluent Treatment By Biological Treatment

Biological liquid effluent treatment can be used as a pre treatment step for liquid effluent containing a high concentration of detergents and/or organic materials; for example laundry effluent or shower arisings [Ref.37].

Biological liquid effluent treatment results in the removal of the majority of the organic radioactive and non-radioactive waste constituents. Typically biological treatment is performed under aerobic conditions. Bacterial action is used to decompose the biodegradable consitituents of the liquid effluent into carbon monoxide and carbon dioxide.

Following completion of the treatment the bacteria sludge generated is treated using a separator to extract the bacteria into a concentrated sludge. The bacteria are then treated using hydrogen peroxide prior to transfer of the residual material to other systems. There is no transfer of bacteria to other systems.

Following treatment the effluent can be directly discharged or routed for further treatment. The remaining concentrate contains primarily mineralised residuals that require further conditioning prior to final disposal.

**Application**

Treatment of all types of liquid effluent, primarily used for liquid effluent containing undissolved solids such as laundry water, drains from showers and sinks, sump water, regenerants, etc.

**Design Data**

- System throughput: 70 m³ (batchwise, typical);
- Decontamination factor of approximately 20.

**Advantages**

- Reduces chemical oxygen demand in discharges;
- Minimises volume of waste for disposal;
- Reduces organic content in residues;
- High decontamination factor for radionuclides bound to organic constituents of the waste;
- No secondary waste generated;
- Can be combined with a separator or separator units.

**Disadvantages**

- Growth of micro-organisms can be inhibited by the presence of inappropriate chemcials.

**FIGURE 12: BIOLOGICAL TREATMENT OF LIQUID EFFLUENT**

### 7.3.3 Liquid Effluent Treatment by Cross Flow Filtration

Cross flow filtration systems have been used in many Nuclear Power Stations worldwide for the treatment of radioactive liquid effluent.

There are several types of cross flow filtration systems available. These include: reverse osmosis, nano-filtration, ultrafiltration and microfiltration. The diversity of techniques available means that the techniques can be adapted to allow the treatment of specific liquid effluent types [Refs. 36, 37 and 38].

With careful design selection the cross flow filtration systems can be used to remove virtually all dissolved and suspended solids from radioactive liquid effluent.

Following monitoring, clean water leaving the cross flow filtration system can be directly discharged or sent for additional treatment. The concentrates resulting from the process require additional conditioning prior to disposal.

**Application**

- Treatment of all types of liquid effluent containing dissolved and undissolved solids.

**Design Data**

- System throughput: 10 $m^3/h$ (typical);
- Decontamination factor: 10 to1000 (system and nuclide specific, typical values).

**Advantages**

- Volume reduction;
- High decontamination factor;
- No secondary waste;
- Proven and compact design;
- Automatic operation.

**Disadvantages**

- Concentrated sludges/slurries have to be conditioned prior to disposal;
- Membrane can be prone to blocking or fouling.

### 7.3.4 Liquid Effluent Treatment By End of Pipe Filtration

End of pipe filtration systems are used to remove suspended solids from liquid effluent containing low concentrations of suspended solids [Refs. 36 and 37]. The use of this technique is not economical for liquid effluent containing high quantities of suspended solids as the filters are rapidly blinded under these circumstances.

The used filter cartridges are sent for conditioning and disposal. The volume of used filter cartridges generated from the use of this technique can be minimised by the use of self-cleaning filters and filter configurations that are amenable to back flushing.

The typical minimum practical pore size of end of pipe filters used for cleaning of the radioactive liquid effluent is approximately 1 to 5 µm. The smaller the filtration pore size the more difficult it is to accommodate the volumes and flow rates required during liquid effluent treatment.

Following monitoring, the liquid can be directly discharged or sent for additional treatment.

**Application**

- Treatment of all types of liquid effluent for the removal of undissolved suspended solids.

**Design Data**

- System throughput: 1-10 $m^3/h$ (typical);
- Decontamination factor: 2-10 (system and nuclide specific, typical values).

**Advantages**

- Cost effective;
- Proven and compact design;
- Automatic operation.

**Disadvantages**

- Process has to be carried out in batches;
- Secondary waste produced in the form of spent filter cartridges;
- Filter has to be appropriately sized to enable through flow of liquid to stop unnecessary binding and secondary waste generation.

### 7.3.5 Liquid Effluent Treatment Using Ion Exchangers

Ion exchangers can be used to remove dissolved material from radioactive liquid effluent [Ref.39]. The combination of selective anionic and cationic media allow the removal of non-radioactive and radioactive impurities. Ion selective media are added to the treatment system where there is a requirement to remove specific radionuclides.

Following treatment, the clean liquid effluent is continuously monitored prior to discharge. The depleted exchange media are then either regenerated for continued use or transferred to the appropriate waste conditioning system.

**Application**

- Treatment of all types of liquid effluent for the removal of dissolved material.

**Design Data**

- System throughput: 1-10 $m^3/h$ (typical);
- Decontamination factor: 2-10 (system and nuclide specific, typical values).

**Advantages**

- Cost effective;
- Proven and compact design;
- Automatic operation.

**Disadvantages**

- Only suitable for dissolved ionic species (e.g. not suitable for particulate cobalt);
- Feed stream needs pH conditioning and may need pre-filtering since particulates in the feed stream can foul (block) the ion-exchange beds or pass straight through the bed;
- A variation in pH or competing ions in the feed can cause the target ions to be displaced back into solution.

### 7.3.6 Solid Waste Treatment By Segregation

Segregation ideally starts at the point of origin with in-situ separation processes. For example, this includes sorting of the contaminated solid waste into categories that allows routing to further treatment processes.

**Solid Waste Segregation In The Controlled Area Workshop**

Large and heterogeneous waste (e.g. worn out items) is either segregated in-situ or sent to the controlled area workshop for segregation and size reduction using dedicated cutting equipment. Components are repaired where possible to minimise waste production. However, it is not always practicable to repair components in this way. For example, some pumps, valves and motors cannot be repaired in-situ. These waste components will be reduced in size and sent for treatment as appropriate [Refs.40 and 35]. Careful decontamination and size reduction can facilitate the removal of the most contaminated areas on a redundant item. This may facilitate only a small proportion of the waste being categorised as radioactive. The remainder of the waste item can sent for disposal as controlled waste.

**Solid Waste Segregation In A Sorting Box**

A sorting box will be used to verify the efficiency of the in-situ separation of waste placed directly into a disposal drum. In addition it will also serve as a primary segregation device to safely separate waste in accordance with the selected downstream waste treatment technique. The majority of the bulk solid waste material will be routed through the sorting box. The use of this technique will permit manual sorting of the waste into the different waste types.

**Application**

- Segregation can be used for contaminated solid waste.

**Design Data**

- Segregation is carried out by hand.

**Advantages**

- Simplifies downstream waste treatment.

**Disadvantages**

- Requires the worker to be in close proximity to the contaminated items resulting in potential operator doses.

### 7.3.7 Solid Waste Drum Compaction

An in-drum compactor or bailing press will be used to reduce the volume of the solid waste generated. The compactor can be integrated into the sorting box or be used as a standalone unit permitting pre-compaction of the waste prior to disposal and/or supercompaction. The typical pressing force applied is 200 kN.

**Application**

- Treatment of solid waste with high voidage.

**Design Data**

- Volume reduction factors for compaction vary according to the compaction force and the feed material but are typically between 3 and 10.

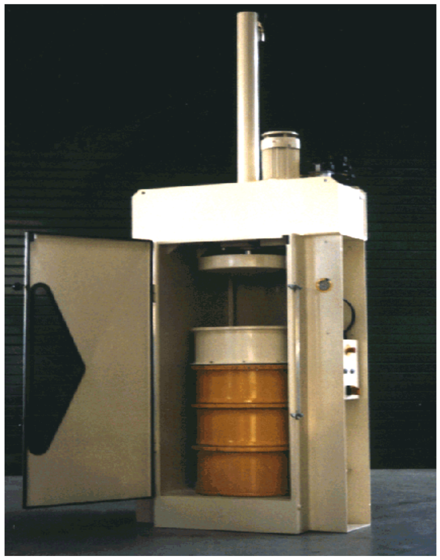
**Advantages**

- Significant waste volume reduction.

**Disadvantages**

- Not suitable for:
    - Non-compactable objects;
    - Free liquors;
    - High moisture content wastes.

**FIGURE 13: IN-DRUM COMPACTOR FOR REDUCTION OF SOLID WASTE**



### 7.3.8 Solid Waste Drying

Drying of the solid waste reduces the mobility of the waste and prevents degradation of the waste container through internal corrosion. Drying will be carried out using a drum drying device equipped with an electrical heater. This consists of a bottom heater and shell heaters which fully enclose the drum permitting heating of the outer surface of the drum. An isolated hood connects the drum to the ventilation system and allows the vapour to be vented to the ventilation system. All solid waste must be dried prior to sending it to a supercompactor and/or long term storage.

**Application**

- Wastes with high moisture content.

**Advantages**

- End product is dry waste which is suitable for disposal.

**Disadvantages**

- Drying by high heat application is not suitable for all wastes.

### 7.3.9 Solid Waste Shredding

A shredder will be used to reduce the volume of waste. This form of size reduction increases the mass of waste that can be placed into a container. Packaged shredded items can then be further volume reduced through supercompaction [Ref. 36].

**Application**

- Dry solids such as paper, clothes, plastics, wood and some light metal.

**Advantages**

- Cost effective;
- Easy to use;
- Facilitates compaction.

**FIGURE 14: USE OF A SHREDDER TO TREAT SOLID WASTE**

### 7.3.10 Solid Waste Decontamination

A stationary decontamination system consists of a decontamination booth for large components and an ultrasonic bath for the decontamination of smaller components.

Larger items of waste will be segregated in the decontamination booth and cleaned using high pressure water. Smaller parts will be placed directly into the ultrasonic bath. Any contaminated liquids arising from decontamination will be sent to the liquid treatment system for processing. Following treatment the decontaminated parts can be reused, classified as LLW, VLLW or exempt from regulatory control.

**Application**

- Solid waste items with surface contamination.

**Advantages**

- Suitable for hard to reach contamination;
- High decontamination factors for loose contamination.

**Disadvantages**

- Produces large volumes of waste water that must be treated.

### 7.3.11 Solid Waste Treatment By Melting

Metal, especially redundant steel components that are within the specified activity levels, or have reduced activity levels following up-stream decontamination, can be routed to an off-site melting facility [Refs.41 and 42]. In this technique the metal is melted and dependent on its final activity level is either released as exempt from regulatory control or recycled for future use within the nuclear industry, for example as shielding blocks or cast iron waste disposal containers.

**Application**

- Metal wastes (e.g. steel components).

**Advantages**

- Can be used for volume reduction;
- Removes areas of high contamination by forming a homogenised material;
- The metal product may be suitable for recycling;
- Concentrates radionuclides in the slag.

**Disadvantages**

- Energy and infrastructure requirements.

### 7.3.12 Controlled Clearance Of Solid Waste

Solid wastes that have been decontaminated will be sent to a monitoring system that forms an integral part of the solid waste management process. The system is typically housed within a transport container and scans the content of a package to monitor the content for the presence of residual contamination. Once the waste under scrutiny has been verified as exempt from regulatory control, it is shipped off-site as non-radioactive material [Ref.43].

### 7.3.13 Solid Waste Treatment Through Incineration

Combustible wastes will be segregated at source and can be sent in batches to an off-site provider for incineration. Incineration reduces the volume of waste requiring disposal and removes the organic content of the waste [Ref.44]. The resulting product primarily consists of dry inorganic materials (e.g. slag, ashes, solidified scrubber solutions) that can be safely stored, if required and disposed. Typically, incineration can result in a volume reduction factor of between 25 to 100.

**Application**

- Combustible wastes.

**Design Data**

- Volume reduction factor between 25 and 100.

**Advantages**

- Destroys hazardous organic compounds and flammable solvents;
- High volume reduction factors;
- Generates a stable inert ash.

**Disadvantages**

- Off-gas treatment generates secondary wastes.

### 7.3.14 Solid Waste Super Compaction

A super compactor can be used to compact specially designed compactable drums. This will result in a waste stream that is significantly reduced in volume and virtually void free.

The compaction factor is determined by the material composition and overall mix of materials. Typical compaction factors range from 3 to 8 with an average of approximately 5 [Ref.35].

The resulting pucks are sorted in accordance with their height. The pucks are individually selected and transferred to the final drum storage to maximise the packing density.

**Advantages**

- High volume reduction factors;
- Voidage elimination.

**Disadvantages**

- Not suitable for:
    - Non-compactable objects;
    - Pressurised containers.


**FIGURE 15: SUPER COMPACTOR USED FOR SPECIALLY DESIGNED COMPACTABLE DRUMS**

### 7.3.15 Conditioning Evaporator Concentrates By In Drum Drying

Evaporator concentrates and filter sludges are two significant radioactive wet wastes produced during the normal operation of nuclear power plants. Although the solids and salts content can vary significantly depending on the origin of the wastes, drying of these wastes results in a significant reduction in the volume of waste generated [Ref.45]. The final waste form is a dry, solid product that is easy to dispose of. If the waste contains a sufficiently high salt content, the dried waste will form a solid, monolithic block encapsulating sludge and other solids (e.g. resins). The negative effects of product ageing are virtually eliminated or minimised due to the minimial residual water content of the dried wastes. Residual water is typically bound as crystal water in the salt. Where resins are encapsulated the salt will fully saturate the resins during conditioning.

In this technique the disposal container (e.g. a cast iron container) is placed under vacuum and heated, resulting in evaporation of the liquid component of the waste. The resulting vapours condense outside the waste container and are collected and transferred to the liquid waste tanks for further treatment.

**Application**

Drying is a suitable technique for volume reduction of:

- Evaporator bottoms;
- Decontamination solutions;
- Sludges;
- Spent resins.

**Design Data**

- System throughput per filling station: up to 140 litres per day;
- The system can be designed with several filling stations;
- Residual moisture of product: Less than 10% by weight achievable with no freestanding water remaining;
- Volume reduction factor: Up to 10 (related to waste with solid content of 20% by weight).

**Advantages**

- High volume reduction factors;
- In-container processing, i.e. waste is processed in the final disposal container;
- Waste forms a solid product inside container (e.g. 200 litre drums);
- Mobile skid-mounted units are available;
- Proven design with extensive operational experience.

**FIGURE 16: CONDITIONING BY EVAPORATION**

### 7.3.16 Conditioning Through Dewatering and Drying

Dewatering of spent resins and sludge can be performed in a disposal container which is equipped with the appropriate dewatering internals and connections to a dewatering device. The dewatering device applies a negative pressure to the container internals extracting all free water [Ref. 46]. The efficiency of this technique can be increased by applying a heated jacket to the externals of the disposal container and condensing the resulting water vapours. The end product is a dry waste product.

Dewatered resins can either be dryed or transferred to an in drum drying facility to be embedded in evaporator concentrate. Another option that could be explored is the regerneration of the resins and discharge of the resins as LLW and the regenerant, following drying as ILW for storage.

**Application**

- Dewatering of spent resins.

**Design Data**

- System throughput up to 300 litres/day;
- The system can be designed with several filling stations;
- Residual moisture of product: no free standing water.

**Advantages**

- In-container processing, i.e. waste is processed in-situ in the final disposal container;
- Skid-mounted units available;

- Due to the minimal residual water content of the resins the negative effects of product aging are virtually eliminated.

**Disadvantages**

- None identified.

**FIGURE 17: CONDITIONING BY DEWATERING**

## 7.4 Packaging of operational radioactive wastes

The wastes that will arise from the operation of the UK EPR will require appropriate management to protect both human health and the environment. Some of these wastes will contain levels of radioactivity that are so low that they are exempt from regulatory control and can be managed and disposed of in accordance with standard waste regulations. All radioactive wastes will require packaging that is appropriate for long term storage or disposal.

The waste packaging to be used for any given waste stream is designed to ensure that the waste is safely contained. This containment may be a simple waste drum (e.g. for very low level wastes such as slightly contaminated protective clothing) or may be a sealed and shielded container (for higher activity wastes). Each radioactive waste stream will be placed in a specific waste package designed for the particular features of the waste and the long term management requirements.

### 7.4.1 Exempt Waste and VLLW Packaging

In most cases exempt waste and VLLW will not have any specific packaging requirements imposed because of radiological protection considerations. The packaging and disposal of these wastes will be determined by other factors such as the nature of the material and contamination with other pollutants. For example, unless it can be re-used elsewhere, concrete waste that is exempt or VLLW will have to be disposed of as controlled waste to a properly licensed landfill site. The regulatory controls applied to these wastes will ensure both public and environmental safety.

VLLW waste will be packaged in 200 litre carbon steel drums or other suitable containers without conditioning being performed. As such it will be shipped and disposed. Redundant metal components will be disposed of directly. No melting for recycling is assumed in the estimates presented below in the following table.

**TABLE 13: VLLW WASTE ARISINGS ESTIMATES**

| | Metal maintenance wastes | Low activity resins |
|---|---|---|
| Estimated raw waste volume ($m^3/yr$) | 2.4 | 7.5 |
| AREVA Processing Baseline | Sorting, packaging and disposal | Packaging and disposal |
| AREVA Packaging Baseline | 200 l drum | 200 l drum |
| Estimated number of unconditioned 200 l drums per year | 13.3 | 41.7 |
| Volume per package ($m^3$) | 0.2 | 0.2 |
| Disposal Volume ($m^3$) | 2.66 | 8.34 |
| Total Disposal Volume ($m^3$) | 11 | |

### 7.4.2 LLW Packaging

For the UK EPR the 200 litre carbon steel or stainless steel drum will be the primary package used for LLW waste forms. These drums will provide a passively safe waste form for interim storage.

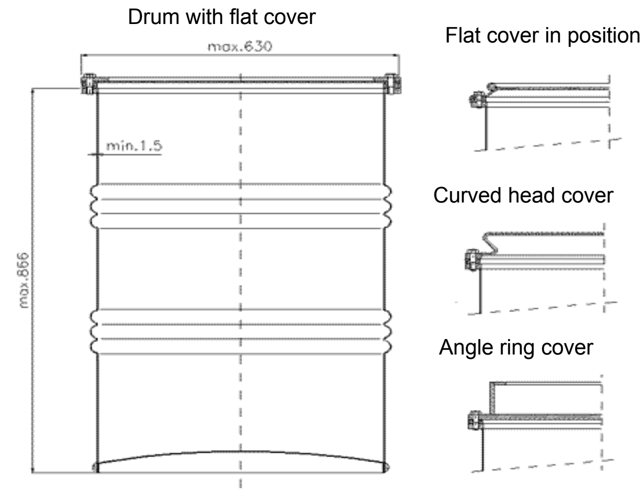



**FIGURE 18: 200 LITRE CARBON STEEL OR STAINLESS STEEL DRUM**

**TABLE 14: PREDICTED ANNUAL WASTE PACKAGING ARRANGEMENTS FOR SHIPMENT TO LLW DISPOSAL FACILITY**

| | Evaporator Concentrates | Air and Water Filters | Dried active waste | Dried active waste | Oils | Metal maintenance wastes |
|---|---|---|---|---|---|---|
| Estimated volume (m³/y) | 3 | 4 | 37.5 | 12.5 | 2 | 3.6 |
| Conditioning process | In-Drum Drying | Compaction/ shredding | Raw for Incineration | Super compaction | Incineration | Melting for recycling |
| Primary packaging | 200 l carbon steel drum | 200 l carbon steel drum | 200 l carbon steel drum | 200 l carbon steel drum | 200 l carbon steel drum | 200 l carbon steel drum |
| Number of primary packages | 27.8 | 7.4 | 6.0 | 30.2 | 0.1 | 2.7 |
| Volume primary packages (m³/y) | 5.6 | 1.5 | 1.2 | 6.0 | 0.0 | 1.3 |
| Total volume primary packages (m³/y) | 15.6 | | | | | |

The disposal volumes provided are based on the assumption that no additional packaging or overpacking is required to dispose of these waste containers.

The waste volumes for combustible waste, redundant metal components and oil are the estimated volumes of the residual waste that must be deposited in the disposal site. As these wastes are shipped for off-site treatment and conditioning, the volumes reported will be residual volumes generated at the processing site rather than at the NPP. Additional secondary waste volumes generated at the site of processing (e.g. protective clothing, residuals from off gas filters at the incinerator) are possible but these are not included in the estimates given in the table. Table 15 lists some of the waste package types that may be used.

ILW that will meet the LLW threshold limits during the timescales for decay storage on site prior to disposal off-site may be packaged in primary containers only.

### 7.4.3 ILW Packaging

The waste packages to be used for ILW will, as far as possible, be designed to be used with generic handling equipment and storage or disposal concepts. The majority of waste packages are based on a standard design of 500 litre drum, as defined by the NDA. Although the external dimensions, lifting features and mass limits are common to all designs the internal design can vary to accommodate different waste types. For example, a 500 l drum may have internal shielding, may be designed for direct encapsulation of waste or may be designed to receive a smaller drum that will then be grouted in place.

**TABLE 15: PACKAGES FOR OPERATIONAL WASTE**

| Waste Package | Waste Type | Comments |
|---|---|---|
| 500 l stainless steel drum | ILW | These can be shielded by adding an internal layer of cementitious grout. |
| Macroencapsulation of a 200 l drum in a 500 l drum | ILW | This option could be used for non-compactable ILW or for providing additional shielding. |
| Cast iron container | ILW | These containers are designed for use with a variety of waste types including dewatered resins and wastes with a high specific activity. |

#### 7.4.3.1 500 Litre NDA Annular Grouted Drum

The current NDA ILW disposal concept is based on the use of 500 l stainless steel drums (as specified by RWMD) as the main waste package type. Figure 19 shows how these could be grouped to assist in interim storage.

**FIGURE 19: 500 LITRE DRUMS**

#### 7.4.3.2 Conditioning Through Macro Encapsulation

ILW not suitable for placement in cast iron containers, will be placed into 200 litre drums [Ref.47]. These wastes will include solid waste and spent filters. If required, the wastes will then be grouted into the drums. The 200 litre drums will then be placed into the 500 litre annular NDA drums and the void space filled with cement grout.

**Application**

- Encapsulation of dried:
  - Evaporator bottoms;
  - Filter sludges / slurries;
  - Spent resin;
  - Decontamination solutions;
  - Solid radwaste (for example, spent filter elements, super compacted waste etc.).

**Design Data**

- System throughput: 3 drums/hour.

**Advantages**

- Proven system design;
- Customised application;
- The final properties of grout prepared with clean water can be tuned to the characteristics needed e.g. leachability, compressive strength and water resistance;
- Macroencapsulated waste forms do not affect chemistry or the aging characteristics of the cement.

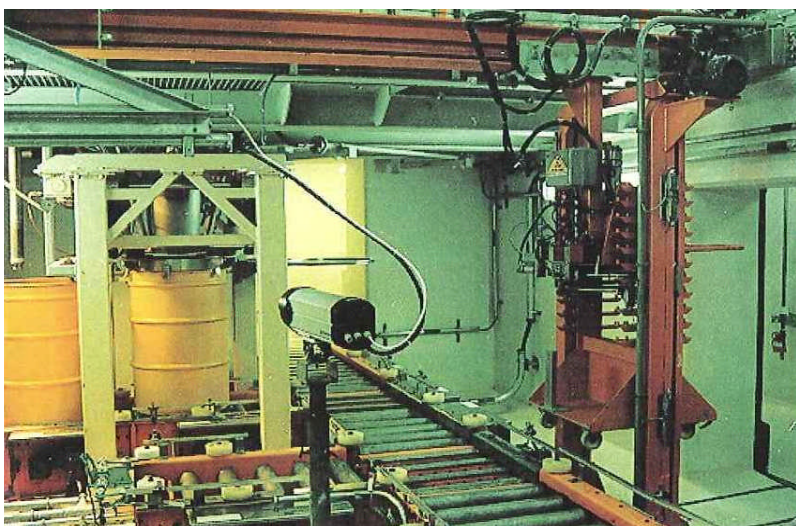


**FIGURE 20: CONDITIONING BY MACRO ENCAPSULATION**

#### 7.4.3.3 Cast Iron Container

The cast iron container is a multi-purpose cask mainly used for the processing, handling and storage of ILW. As this container provides a safe handling and storage enclosure it is suitable for dewatered resins, evaporator concentrates and other components of higher specific activities.

This container can be manufactured from metals contaminated with low levels of activity, allowing recycling of metals which would otherwise have to be disposed of as radioactive waste.

The cask outer dimensions of 1500 mm height, and 1060 mm diameter are fixed but the internal and lid configurations can be adapted to the needs of different waste streams. Typical empty weights can be between 4000 kg and 9400 kg providing volumes between 135 l and 680 l. (This description is based on the MOSAIK Type II container which has been subject to the concept stage LoC NDA assessment.)

A key advantage of the use of this container is the accessibility of the container during processing, handling and storage due to the shielding provided. This permits the use of mobile, modular conditioning systems.

**FIGURE 21: MOSAIK CAST IRON CONTAINER**

### 7.4.3.4 ILW Package Volumes

The AREVA waste packaging technology, including the use of cast iron containers, differs from the procedures generally adopted with the UK nuclear industry. Table 16 and Figure 22 compares some aspects of the two approaches and shows that the AREVA concept could expect to produce a much lower volume of waste. The AREVA concept design technology will be further developed and will be assessed by UK regulatory authorities. If the design meets the requirements of these regulators (in terms of safety and environmental performance) then the packaging technology will be proposed for the UK EPR waste treatment facilities.

**TABLE 16: COMPARISON OF THE STORAGE AND DISPOSAL VOLUMES FOR THREE PACKAGING CONCEPTS**

| | Operational waste type | Ion Exchange Resins | Sludge | Water filters | Technological operational waste | Total |
|---|---|---|---|---|---|---|
| | Estimated raw waste volume (m³/y) | 3 | 1 | 5 | 1 | |
| RWMD Concept | Conditioning process | Cementation | Cementation | Inactive Grouting | Inactive Grouting | |
| RWMD Concept | Primary packaging (m³) | 500 l drum | 500 l drum | 200 l drum | 500 l drum | |
| RWMD Concept | Interim storage volume | 40 | 6.7 | 5 | 1.8 | 53.5 |
| RWMD Concept | Secondary packaging | none | none | 500 l drum grouted | None | |
| RWMD Concept | Disposal volume (m³/y) | 40 | 6.7 | 15 | 1.8 | 63.5 |
| Cast Iron Concept | Conditioning process | Dewatering & drying | In-drum drying combined with evaporator concentrate | Inactive grouting | Inactive grouting | |
| Cast Iron Concept | Primary packaging | Cast Iron 15 cm PB 40 | Cast Iron 15 cm | 200 l drum | 500 l annular grouted drum | |
| Cast Iron Concept | Interim storage volume | 11.3 | 2.5 | 5 | 1.5 | 20.3 |
| Cast Iron Concept | Secondary packaging | None | None | 500 l drum grouted | None | |
| Cast Iron Concept | Disposal volume (m³/y) | 11.3 | 2.5 | 15 | 1.8 | 30.6 |
| 200 Litre Drum Concept | Conditioning process | Dewatering & drying | In-drum drying combined with evaporator concentrate | Inactive grouting | Inactive grouting | |
| 200 Litre Drum Concept | Primary packaging | 200 l drum | 200 l drum | 200 l drum | 200 l drum | |
| 200 Litre Drum Concept | Interim storage volume | 3.3 | 0.9 | 5.0 | 1.5 | 10.8 |
| 200 Litre Drum Concept | Secondary packaging | 500 l drum grouted | 500 l drum grouted | 500 l drum grouted | 500 l drum grouted | |
| 200 Litre Drum Concept | Final package volume (m³/y) | 10 | 2.8 | 15 | 4.5 | 32.3 |

The total quantity of the operational waste produced for interim storage and disposal highly depends on the conditioning and packaging processes selected.

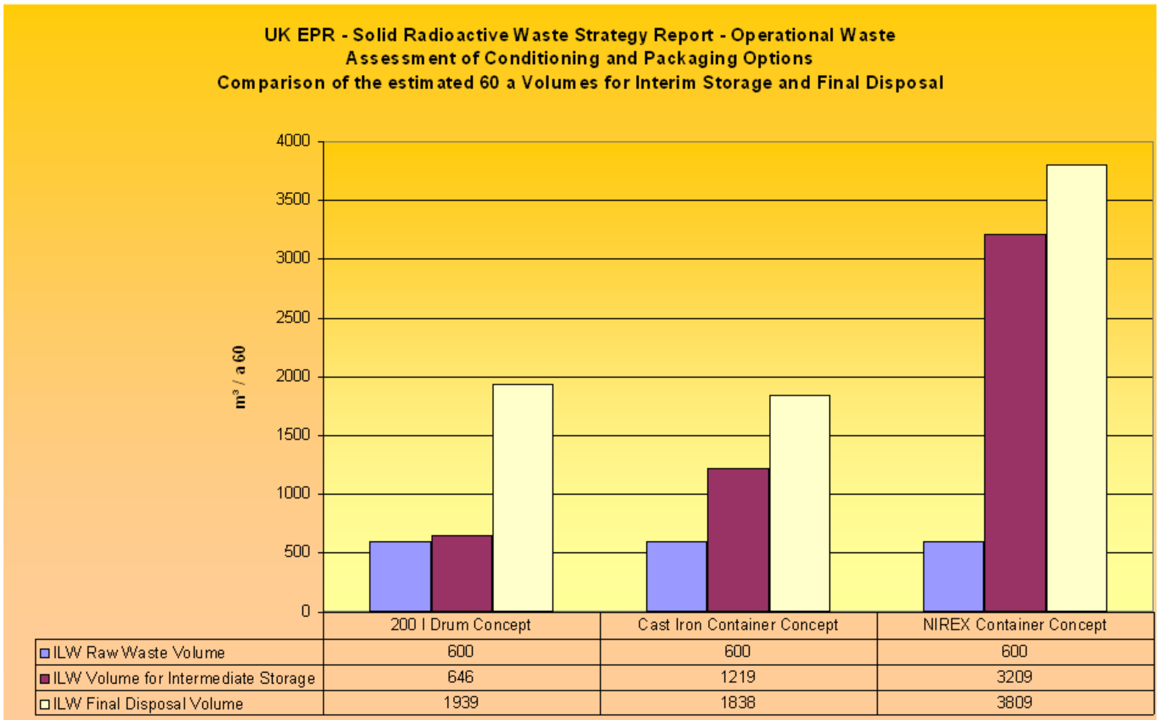



**FIGURE 22: COMPARISON OF ESTIMATED 60 YEAR VOLUMES FOR INTERIM STORAGE AND FINAL DISPOSAL**

The conditioned volumes were calculated using average values based on similar wastes conditioned. The waste container packaging efficiency is assumed at 90%.

It should be noted, that the interim storage volume highly depends on the handling strategy and the timing of final conditioning.

ILW that will meet the LLW threshold limits during the timescales for decay storage on site prior to disposal off-site may be packaged in primary containers only.

# 8 DECOMMISSIONING WASTE

This section examines the nature and quantity of radioactive waste to be generated through the decommissioning of the nuclear island of a single UK EPR unit. The decommissioning methodology is outlined. It is explained how the design of the UK EPR will facilitate its decommissioning. This study does not take into account the conventional waste or non-radioactive waste generated from the dismantling of the conventional island plant (principally the turbine house and electrical switchgear) or the administrative buildings and non-nuclear site infrastructure. This section does not take into account decommissioning of the ILW and Spent Fuel Interim Stores.

## 8.1 EPR Plant Inventory and Radiological Characteristics After 60 Years of Operation

### 8.1.1 EPR Design Features for Decommissioning

The UK EPR has been designed with maintenance and decommissioning in mind. The future decommissioning of the EPR has been planned and optimised at the design stage. Thus the design will enable decommissioning to be performed to minimise radiation doses to workers and minimise radioactive waste generation. In particular, the design incorporates the following features:

- Choice of materials of construction to minimise activation;
- Optimisation of neutron shielding;
- Optimisation of access routes to nuclear areas;
- Reactor systems design;
- Ease of removal of major process components;
- Submerged disassembly of reactor pressure vessel;
- Modular thermal insulation;
- Fuel cladding integrity;
- Design for decontamination;
- Prevention of contamination spread;
- Minimisation of hazardous materials;
- Summary of design principles.

These are discussed in more detail below.

#### 8.1.1.1 Choice Of Materials Of Construction To Minimise Neutron Activation

Materials with a minimum propensity to become radioactive through activation (directly or through their corrosion products) have been selected for equipment subjected to irradiation.

Cobalt-59, a stable isotope, is present in stainless steel and other alloys. Upon activation by neutrons it transmutes into cobalt-60 which is an intense gamma radiation emitter. Although cobalt-60 has a relatively short half life of 5.27 years, its presence is significant for decommissioning, if not for long term waste management. Therefore, elimination, wherever possible, of the use of alloys containing high cobalt levels in areas subject to neutron irradiation has been implemented. In particular the use of stellite for hard facing surfaces has been minimised.

Where complete elimination of cobalt containing materials cannot be achieved, low cobalt content alloys have been selected. Alloy 690 has been selected instead of alloy 600 for the steam generator tubes. This minimises the quantity of cobalt in the corrosion products circulating in the primary system. The amount of cobalt in some steel components has been limited to 6 ppm.

#### 8.1.1.2 Optimisation Of Neutron Shielding

Shielding and barriers have been optimised to minimise the activation and contamination of equipment. For example, the UK EPR design incorporates neutron shielding around the core. This shielding reduces neutron activation of material and equipment in the vicinity of the core and thereby simplifies the clean-up of the structure while minimising the volume of active waste. The neutron shield will become activated during service. The neutron shield is modular by conception and has been designed for ease of removal from the reactor core.

#### 8.1.1.3 Optimisation Of Access Routes To Nuclear Areas

The design of access points in nuclear areas, handling equipment and access routes has been optimised in order to reduce the expected duration of exposure of workers (provisions facilitating removal of components and structure, and facilitating personnel access). The design makes use of international experience from past reactor decommissioning projects and measures taken to facilitate maintenance (for example, feedback from the replacement of steam generators in currently operating PWR plants).

#### 8.1.1.4 Reactor Systems Design

The design of reactor systems (core instrumentation, steam generators, reactor coolant pumps, pressuriser, heat exchangers etc.) has been optimised to facilitate decommissioning.

Systems have been designed to minimise the creation, transportation and deposition of contamination.

#### 8.1.1.5 Ease Of Removal Of Major Process Components

The major process components within the reactor building (and reactor building access routes) have been designed to allow their removal from the reactor building each as a single item without the need for size reduction. This can allow their removal to other facilities for size reduction, waste conditioning and packaging.

#### 8.1.1.6 Submerged Disassembly Of Reactor Pressure Vessel

The design of the reactor compartment (fitted with a lost steel framework embedded in the concrete) makes it possible to fill the reactor compartment with water to allow the pressure vessel to be disassembled whilst submerged. The reactor compartment is lined with decontaminable steel to prevent contamination of its concrete structure.

#### 8.1.1.7 Modular Thermal Insulation

Modular assembly of the thermal insulation on the main primary circuit allows its easy removal. The lagging which clips put into place reduces time required for its removal and therefore reduce dose uptake by operators.

#### 8.1.1.8 Fuel Cladding Integrity

Improvement of fuel cladding to further reduce the low likelihood of fuel leakages thus minimising the probability of the release of alpha and beta emitters into the primary circuit, minimising the extent of primary circuit contamination.

#### 8.1.1.9 Design For Decontamination

Design arrangements facilitate the decontamination of room and equipment:

- Placing of injection seals, drainage lines and tanks, and sampling points to aid post operational clean-out and use of decontamination techniques;
- Provision of coating or lining of walls. For example, the presence of a metal liner on the internal wall of the reactor building protects the reinforced concrete against contamination and will aid the clean-up operations and the subsequent demolition of the reactor building.

#### 8.1.1.10 Prevention Of Contamination Spread

The spread of contamination is minimised by appropriate measures such as containment, zoning, ventilation and segregated drains.

#### 8.1.1.11 Minimisation Of Hazardous Materials

The use of materials constituting controlled industrial waste is minimised as far as possible.

#### 8.1.1.12 Summary Of Design Principles

In summary, the design of the EPR reactor includes measures which will:

- Minimise the volume of radioactive waste;
- Minimise the toxicity of the waste;
- Minimise the activity level of irradiated components;
- Minimise the spread of contamination;
- Permit easy decontamination;
- Facilitate the access of personnel and machines and the removal of waste from the reactor building;
- Reduce the operator intervention time.

These measures optimise the dismantling of the reactor, limit the dose uptake for the corresponding operations and limit the quantity and activity of the nuclear waste produced in comparison to earlier PWR designs, which did not plan or consider decommissioning at the design stage.

### 8.1.2 Physical Inventory

Particular focus has been directed to the design of the central reactor equipment within the reactor building (i.e. the primary circuit and its close environment) where components have the greatest potential to become activated or contaminated.

The primary circuit and its immediate surrounds (illustrated in Figure 23) consists of:

- The reactor pressure vessel (comprising internals and equipped cover head);
- The 4 primary loops which each contain a reactor coolant pump, a steam generator and the hot/cold/ U branch (main coolant pipes);
- The pressuriser;
- The reactor pit;
- Thermal insulation;
- Vessel supports and structures;
- Core instrumentation;
- The auxiliary circuit up to the first isolation device.

The characteristics of these components are presented in Table 17. Equipment surrounding the nuclear steam supply system and the rest of the nuclear island, shown in Figure 24 has been considered globally by type of equipment and for the purposes of estimating the classification of decommissioning waste. These are listed in Table 18 and Table 19. Concrete arising from the decontamination of on-site buildings is expected to amount to about 240 $m^3$.

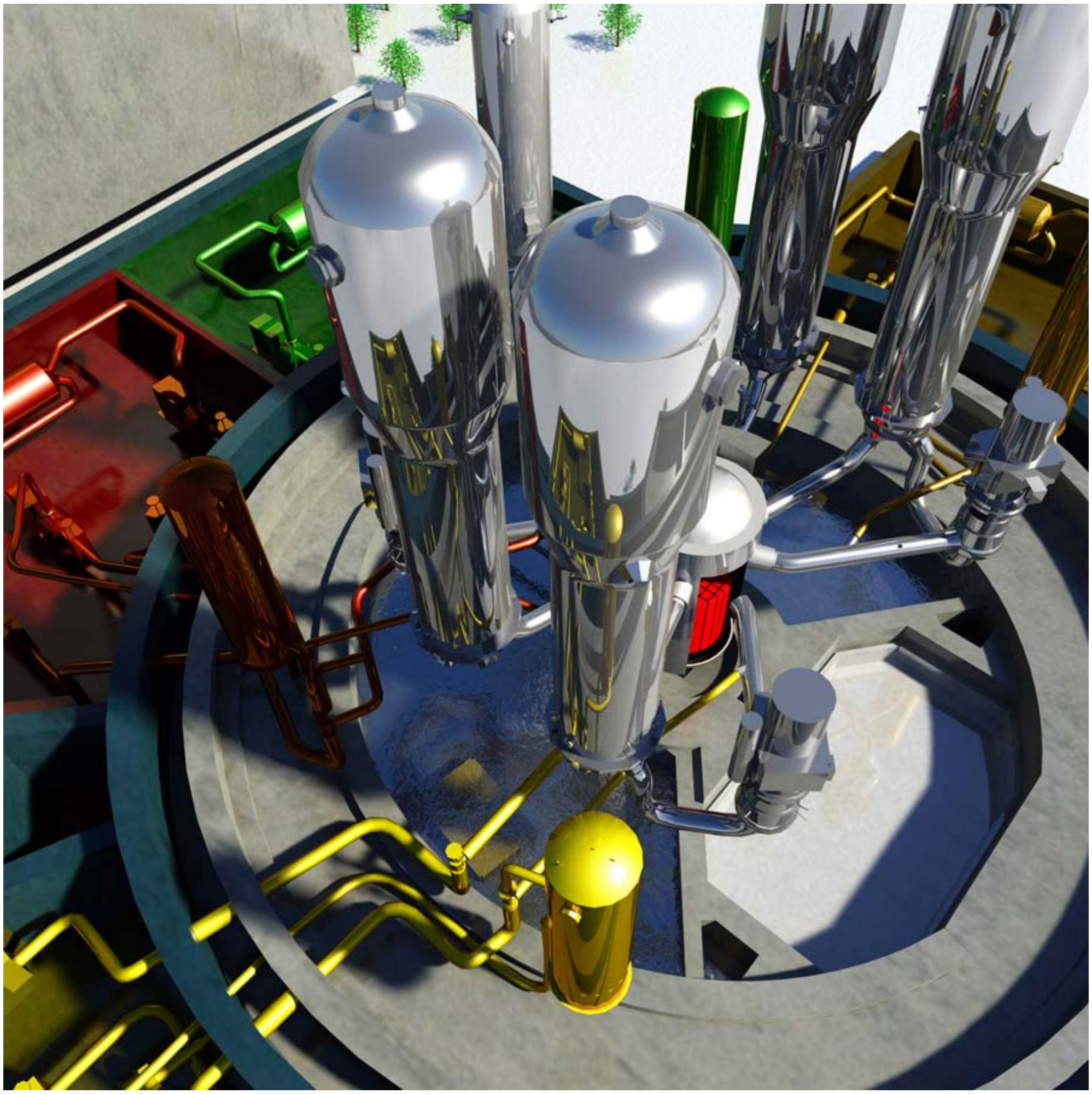


**FIGURE 23: EPR PRIMARY CIRCUIT**

## TABLE 17: PRIMARY CIRCUIT WASTE CHARACTERISTICS

| Component | Mass (te) | % Stainless steel | % Carbon steel | % Other materials | Waste classification (*) |
|---|---|---|---|---|---|
| Vessel head | 116.00 | 6% | 94% | - | LLW |
| Vessel head insulation | 5.20 | 73% | - | 27% Isover Ultimate© | VLLW |
| Pressure Vessel | 225.00 | 6% | 94% | - | ILW |
| | 185.00 | 6% | 94% | - | LLW |
| Vessel insulation | 7.00 | 100% | - | - | LLW |
| Lower pressure vessel internals | 198.05 | 100% | - | - | ILW |
| Upper pressure vessel internals | 19.99 | 100% | - | - | ILW |
| | 54.10 | 100% | - | - | LLW |
| Primary pipe circuit | 147.80 | 100% | - | - | LLW |
| Steam generators (x4) | 1474.00 | - | - | - | LLW |
| | 726.00 | - | - | - | VLLW |
| Reactor coolant pumps (x4) | 108.80 | 100% | - | - | LLW |
| | 339.20 | - | - | | VLLW |
| Pressuriser expansion line | 9.13 | 100% | - | - | LLW |
| Pressuriser | 150.00 | 6% | 94% | - | LLW |
| Reactor pit | 180.00 | - | 12% | 88% concrete | ILW |
| | 483.00 | - | 12% | 88% concrete | LLW |
| Primary Circuit Pipework Insulation | 345.20 | 73% | - | 27% Isover Ultimate© | VLLW |
| Primary Circuit Pipework Support | 482.84 | - | - | - | VLLW |
| TOTAL | 1898.44 te | | | | VLLW |
| | 2734.83 te | | | | LLW |
| | 623.04 te | | | | ILW |

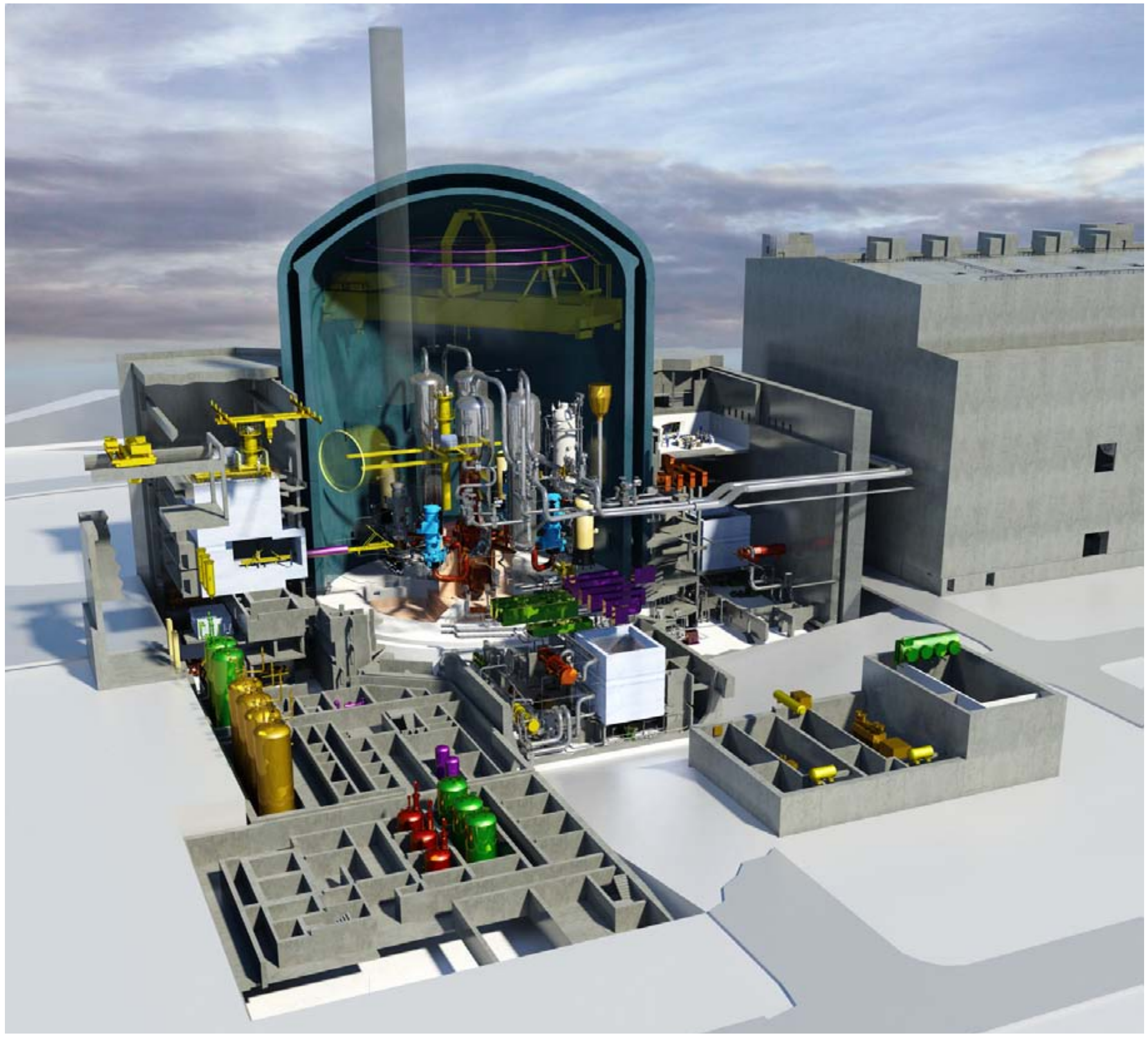


**FIGURE 24: EPR CUTAWAY**

**TABLE 18: BALANCE OF NUCLEAR STEAM SYSTEM SUPPLY INVENTORY**

| | Quantity | Mass (te) | Length (m) | % stainless steel mass | % carbon steel mass | Other | Mixture | Waste category |
|---|---|---|---|---|---|---|---|---|
| **Heat exchangers** | **10** | **117.60** | **-** | - | - | - | x | LLW |
| **Tanks** | **22** | **435.10** | **-** | - | - | - | x | LLW |
| | **20** | **-** | **-** | | - | - | x | LLW |
| **Valves** | **2080** | **290.40** | **-** | - | - | - | x | LLW |
| | **91** | **-** | **-** | - | - | - | x | LLW |
| **Sensors** | **1189** | **17.80** | **-** | - | - | - | x | LLW |
| **Pipes** | **-** | **516.10** | **16332.40** | 31.27% | 68.73% | - | - | LLW |
| **Pipe insulator** | - | **102.60** | - | - | - | 100% Isover Ultimate | - | LLW |
| **Pipe insulator support** | - | **13.40** | - | 77% | - | 23% Al | - | LLW |
| **Pumps** | **37** | **136.20** | **-** | - | - | - | x | LLW |
| **Filter** | **14** | **89.80** | **-** | - | - | - | x | LLW |
| | **20** | **-** | **-** | - | - | - | - | LLW |
| **Cabletray** | **-** | **3.11** | **1104.61** | - | - | 75% Cu | 25% plastic | LLW |
| **I & C** | **-** | **-** | **-** | - | - | - | X | LLW |
| **Handling** | **-** | **-** | **-** | - | - | - | X | LLW |
| **Fuel Rack** | **-** | **-** | **-** | - | - | - | X | LLW |
| **Others** | **-** | **536.80** | **-** | - | - | - | X | LLW |
| **Total** | - | **2259.00** | - | - | - | - | - | LLW |

**TABLE 19: BALANCE OF NUCLEAR ISLAND INVENTORY**

| | | Controlled Area (te) | Waste | Non Controlled Area (te) | Waste Type |
|---|---|---|---|---|---|
| **Mechanical Equipment** | Heat Exchanger | 69 | LLW | 172 | VLLW |
| | Tanks | 292 | LLW | 11 | VLLW |
| | Valves | 141 | LLW | 24 | VLLW |
| | Sensors | 48 | LLW | 12 | VLLW |
| | Storage racks | 185 | LLW | 0 | VLLW |
| | Pipes | 800 | LLW | 200 | VLLW |
| | Ventilation | 134 | LLW | 33 | VLLW |
| | Handling | 1050 | VLLW | 12 | Conventional |
| | Pumps (Hydraulic section) | 22 | LLW | 66 | VLLW |
| | Shielded doors | 400 | VLLW | 0 | Conventional |
| **Electrical Equipment** | Cabletray | 407 | VLLW | 413 | Conventional |
| | Cable | 161 | VLLW | 145 | Conventional |
| | Electrical Equipment rooms | 0 | VLLW | 408 | Conventional |
| Others | | 1133 | LLW | - | - |
| | | 69 | VLLW | - | - |
| | | LLW | 2824 te | | |
| | | VLLW | 2605 te | | |
| | | Conventional | 978 te | | |
| | | **Total** | 6407 te | | |

### 8.1.2.1 Radiological Characteristics After 60 Years Of Operation

### 8.1.2.2 Primary Circuit

**Contamination**

The expected radionuclide contamination in components arising during normal operation has been derived by the designer. This has been based on up to date operational experience from French and German reactors. The figures have been revised to reflect the advanced technical, geometrical and design features of the UK EPR.

During plant operation, corrosion products can deposit on the inner surface of pipes and components, and build up as loosely bound deposits and/or a solid contamination layer.

The forecasted EPR corrosion product specification is given in Table 20.

**TABLE 20: RADIOACTIVE DEPOSIT OF CORROSION PRODUCTS ON INNER SURFACE OF THE MAIN COOLANT LOOPS**

| Nuclide | Primary piping Hot/Cold legs (Bq/m$^2$) | Steam Generators (Bq/m$^2$) |
|---|---|---|
| Fe-59 | 7.0E7 to 2.0E8 | 5.0E7 to 1.2E8 |
| Mn-54 | 2.5E8 to 4.0E8 | 6.5E7 to 1.3E8 |
| Co-58 | 3.0E9 to 5.2E9 | 2.5E8 to 2.6E9 |
| Co-60 | 5.0E8 to 9.8E8 | 2.5E8 to 5.0E8 |

Prior to dismantling, the primary circuit will be chemically decontaminated. The contamination will be collected on ion exchange resins. The contamination data for the primary circuit has been used for the characterisation of the waste (ion exchange resins) produced during the full decontamination of the primary circuit.

The entire primary system to be decontaminated has an overall volume of about 395 m$^3$ and a total surface area of 15800 m$^2$. The steam generators make the greatest contribution to the overall surface area (14000 m$^2$).

Note: These figures exclude component replacement.

**Activation**

Activation only occurs to primary circuit components exposed to the neutron flux from the reactor core. These components are illustrated in Figure 25.

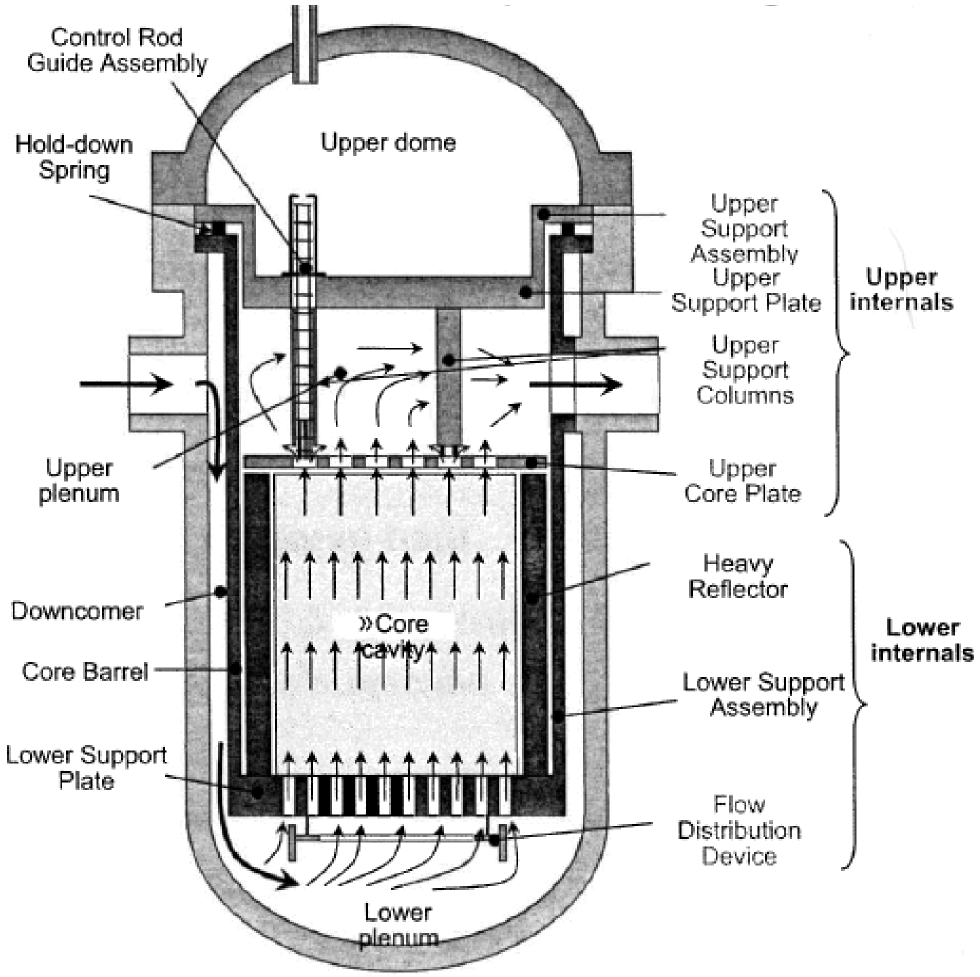



**FIGURE 25: EPR COMPONENTS DESIGN – REACTOR PRESSURE VESSEL INTERNALS**

**Purpose and Method**

The purpose of the assessment is to predict the extent of activation of main reactor pit components of the UK EPR. This assessment determines the waste classification ILW, LLW or VLLW for such components.

The activation modelling tool used was DARWIN-PEPIN 2.1.1, which resolves the Bateman equation. This calculation code required the following input data:

- Material type (concrete, steel);
- Level of neutron flux (energetic repartition into 3 groups);
- Power and operating history.

The output data is the activation in Bq/g for each radionuclide. From this data, a waste classification level is assigned. The global activity of each waste stream is calculated by considering the associated mass of materials and components.

Calculations were performed for several materials and flux levels.

Complementary dose rate assessments are made to define the packaging requirements for each raw waste stream using the MERCURE 5.3 code.

**Input Data**

**Composition of Material**

- *Steels*

The materials specified for fabrication of the reactor pressure vessel and associated internal components are as follows:

- Z2 CN19-10 + N2 stainless steel for all internal components near the core;
- Z3 CN18-10 + N2 stainless steel for others internals components;
- 16 MND 5 ferritic steel for the reactor pressure vessel (lined with stainless steel for corrosion protection).

The metallurgical compositions of each element have been maximised to the upper bound. Compositions used in calculations are given in Table 21. The density is normalised to 1.0; it means that calculations are made for 1 gram of material. Trace steel impurities have not at this stage been modelled.

**TABLE 21: COMPOSITION OF METALS USED IN CALCULATIONS**

| | **Z2 CN 19-10 + N2** | **Z3 CN 18-10 + N2** | **16 MND 5** |
|---|---|---|---|
| **Element** | **Mass ratio** | **Mass ratio** | **Mass ratio** |
| **DENSITY** | **1.0672000** | **1.0787680** | **1.0297300** |
| C | 0.00035 | 0.00040 | 0.00220 |
| Si | 0.0100 | 0.0100 | 0.0030 |
| Mn | 0.0200 | 0.0200 | 0.0160 |
| Ni | 0.1000 | 0.1100 | 0.0080 |
| Cr | 0.2000 | 0.2000 | 0.0025 |
| Mo | | | 0.0057 |
| S | 0.00015 | 0.0002 | 0.0001 |
| P | 0.00030 | 0.00035 | 0.00008 |
| Cu | 0.0100 | 0.0100 | 0.0008 |
| B | | 0.000018 | |
| N | 0.0008 | 0.0008 | |
| Ta | | | |
| Co | **0.0006** | **0.002** | **0.0003** |
| Fe | 0.7250 | 0.7250 | 0.9907 |
| Nb | | | |
| Ti | | | |
| Eu | | | |
| Ag | | | |
| Pb | | | |
| V | | | |
| Se | | | |
| Al | | | 0.0004 |
| W | | | |

*Concrete*

Concrete composition used in calculations is given in Table 22. The composition examined here was supplied by the civil works department for a particular EPR construction site. This composition is relatively rich in chemical elements (and therefore interesting for the activation calculation) but it would be different and specific for each EPR due to the differing sources of aggregate used.

**TABLE 22: COMPOSITION OF CONCRETE USED IN CALCULATIONS**

| Element | % by weight |
|---|---|
| Na | 1.7 |
| Mg | 2.3 |
| Al | 6.8 |
| Si | 25.1 |
| P | 0.1 |
| K | 1.9 |
| Ca | 7.1 |
| Ti | 0.5 |
| Mn | 0.05 |
| Fe | 4.5 |
| S | 0.2 |
| Ga | 0.002 |
| Ba | 0.04 |
| B | 0.001 |
| O | 47.9 |
| C | 0.07 |
| H | 0.67 |
| **Sum** | **99 %**<br>Rest: not significant trace elements |

For concrete, site specific evaluations would have to be performed, considering the actual composition for that particular site.

**Level of Neutron Flux**

For this assessment EPR design neutron flux data have been used: in particular, the assessment of the impact of the neutron propagation study outside of the core.

Note:

- No loading factor has been considered (100% nominal power has been assumed for all operations);
- The maximum flux level has been assumed to apply evenly to all areas of a component. In particular no flux axial distribution has been modelled, the axial mid-point flux has been conservatively assumed to apply to the entire length of each component;
- The worst radial propagation has been chosen: azimuth distortion of the core effect can lead to factor 3 or 4 between maximum and minimum level flux at one axial position.

Consequently, the activation calculations have produced highly conservative estimates for radionuclide inventories. However, it will be very difficult to derive a good average for some input parameters. This would require a more specific neutron propagation study to be carried out.

**Power and Operating History**

This EPR operating programme assumed for the activation calculations is as follows:

- 38 operating fuel cycles of 18 months (100% nominal power) followed by plant shutdown of 16 days;
- A longer plant shutdown is considered every 10 years (40 days);
- The total operating time is 60 years;
- The thermal power is 4500 MWth.

**Activation Results**

*Steels*
Based on the results of the activation calculations, steels have been grouped into three families of ILW dependent on activity levels:

- The most activated:
    - Neutron shield and lower support plate;
    - Specific activity associated with $^{60}Co$ ~ 2.1E+9 Bq/g for raw waste 5 years[3] after shutdown.
- The intermediate activated:
    - Core barrel, upper core plate, flow distribution device and upper support columns (which are located under the upper support plate);
    - Specific activity associated with $^{60}Co$ ~ 3.6E+8 Bq/g for raw waste 5 years after shutdown.
- The less activated:
    - Pressure vessel + cladding (in the active region of the fuel assemblies);
    - Specific activity with $^{60}Co$ ~ 2.8E+5 Bq/g for raw waste 5 years after shutdown.

All other internal structures, namely the upper support columns (above upper support plate), the upper support plate, the upper support assembly, the pressure vessel (upper and lower fuel assemblies) and vessel cladding (upper and lower fuel assemblies) will be classified as LLW at the time of production.

*Concrete*

It is predicted that the following categories of waste will be generated for the reactor pit:

- 5 years after shutdown:
    - 60 cm depth of concrete is ILW;
    - Then 90 cm depth is LLW;
    - The remainder is VLLW.

- 40 years after shutdown:
    - 80 cm depth is LLW;

[3] Based on 60 years of electricity production for an EPR followed by an assumption of immediate dismantling of a single reactor, the ILW resins (from primary circuit decontamination) are produced in year 61 and ILW solid dismantling wastes between years 67 and 70. Therefore, 5 years after shutdown has been chosen to evaluate conservative radiological characteristics.

- The remainder is VLLW.

These concrete thicknesses lead to the following volume and mass of waste (density 2.35 g/cm3):

**TABLE 23: VOLUME MASS OF WASTE**

| | ILW concrete | LLW concrete |
|---|---|---|
| **5 years after shutdown** | 77 $m^3$ / 180 te | 205 $m^3$ / 483 te |
| **40 years after shutdown** | none | 112 $m^3$ / 264 te |

No account has been taken in the modelling of the effect of steel reinforcement or concrete impurities.

### Dose Rate Calculations

### Calculation Assumptions

Dose rate calculations inform the selection of the type of packaging that can be used for each raw activated waste.

Calculations take into account:

- Geometry of each package type: The 3 $m^3$ and 4 m boxes are assessed to be suitable for packaging these wastes at year 100 based on transport dose rate limits;
- The waste packing density within the packages is assumed to be about 3 te/$m^3$ (based on 13% steel and 87% concrete). The bulk density of the raw waste is estimated to be 1 te/$m^3$.

Dose rates have been calculated for several source intensities, in order to define the specific activity limit for $^{60}$Co (The dominant contributor to external package dose rates) that would satisfy the transport dose rate criteria (contact dose rate < 2mSv/hr and 1m dose rate < 0.1mSv/hr).

### Results

*The most activated wastes*

These wastes are derived from dismantling of the neutron shield and the lower support plate.

**TABLE 24: SPECIFIC ACTIVITIES OF WASTE PACKAGES FOR MOST ACTIVATED WASTES**

| Packaging | Maximum specific activity ($^{60}Co$) allowable with respect to transport dose rate criteria | Specific activity ($^{60}Co$) of this packed waste |
|---|---|---|
| 3-cubic metre box within a SWTC 285 transport container | 1E+8 Bq/g of raw waste | ~2.1E+7 Bq/g of raw waste ***40 years after shutdown*** |
| 3-cubic metre box with an additional 100 mm internal steel shielding, and a SWTC 285 transport container | 5E+9 Bq/g of raw waste | ~2.1E+9 Bq/g of raw waste ***5 years after shutdown*** |

*The typical activated wastes*

These wastes are derived from the dismantling of the core barrel, the upper core plate, the flow distribution device and the upper support columns.

**TABLE 25: SPECIFIC ACTIVITIES OF WASTE PACKAGES FOR TYPICAL ACTIVATED WASTES**

| Packaging | Maximum specific activity ($^{60}Co$) allowable with respect to transport dose rate criteria | Specific activity ($^{60}Co$) of this packed waste |
|---|---|---|
| 3-cubic metre box with a SWTC 285 transport container | 1E+8 Bq/g of raw waste | ~1E+6 Bq/g of raw waste ***40 years after shutdown*** |
| 3-cubic metre box with an additional 100 mm internal steel shielding, and a SWTC 285 transport container | 5E+9 Bq/g of raw waste | ~3.6E+8 Bq/g of raw waste ***5 years after shutdown*** |

*The least activated wastes*

These wastes are derived from the reactor pressure vessel and the sections of the reactor pressure vessel cladding near the centre point of the fuel assemblies (The upper and lower sections will be LLW).

**TABLE 26: SPECIFIC ACTIVITIES OF WASTE PACKAGES FOR LEAST ACTIVATED WASTES**

| Packaging | Maximum specific activity ($^{60}Co$) allowable with respect to transport criteria | Specific activity ($^{60}Co$) of this packed waste |
|---|---|---|
| 4-metre box with an additional 100 mm internal concrete shielding | 7E+3 Bq/g of raw waste | ~2.8E+3 of raw waste ***40 years after shutdown*** |
| 4-metre box with an additional 400 mm internal concrete shielding | 3E+5 Bq/g of raw waste | ~2.8E+5 Bq/g of raw waste ***5 years after shutdown*** |

#### 8.1.2.3 Other Equipment (Nuclear Steam Supply System (Except Primary Circuit), Nuclear Island, Maintenance)

Classifications of waste of the Nuclear Steam Supply System and Balance of Nuclear Island inventories are indicated respectively in Table 18 and Table 19. Contaminated concrete (excluding the activated concrete of the reactor pit) is classified LLW for 75t and VLLW for 455t.

### 8.1.3 Dismantling and waste management options

Techniques for safely decommissioning and dismantling an EPR already exist and have been demonstrated on previous decommissioning projects.

Four main factors will have influence on the decommissioning strategy adopted by a utility:

- Acceptance for disposal of ILW to the proposed UK deep geological disposal facility;
- Acceptance for disposal of LLW to a Low Level Waste Repository;
- Legal and regulatory framework at the time of decommissioning;
- Radiological status of the plant after 60 years of operation.

Strategic options will be refined during the operational phase of the EPR through dialogue with relevant regulators. Experience gained from other decommissioning and dismantling projects will be considered.

At the present time, the baseline scenario for decommissioning and dismantling is:

Dismantling methodologies considered for the EPR is as follows:

- Dismantling of strongly and moderately activated components remotely under water. These components are located in the reactor pit which is designed to be flooded to enable work to be carried out on reactor core components. The reactor core components, especially the neutron shield, have been designed for ease of dismantling and removal;
- Dismantling of contaminated components and slightly activated components in contact with air;
- Dismantling will make maximum use of the EPRs static and dynamic containment. The access routes to the reactor containment building have been designed to allow import of dismantling equipment and export of large components;
- Use of auxiliary buildings erected for the dismantling. Once the reactor has been shut down and fuel removed from the nuclear island, redundant auxiliary buildings can be refurbished to support decommissioning and waste management;
- Cutting of components will be completed to separate and categorise waste with respect to radiological classification and to provide size-reduced pieces which are compatible with the designated packaging;
- Main Coolant Pipes and Reactor Coolant Pumps are removed from their location inside the Reactor Building to a workshop at the building floor service in order to be size reduced for packaging;
- Removal of Steam Generators as complete units from their respective shielded enclosures (“pillboxes”) to a treatment building outside of the reactor building;
- The polar crane within the reactor building has been designed for the handling of heavy equipment and reactor components during decommissioning. Lighter components can be handled by other means specific to the task.

- Various cutting techniques will be available to size reduce equipment. The most appropriate technique will be applied depending on specific physical characteristics of the piece (size, thickness, type of material) and intervention principles. The following can be used:
    - Steam generators: thermal (plasma, torch), mechanical (circular saw, band saw);
    - Reactor coolant pump: thermal (plasma, torch), mechanical;
    - Main Coolant Pipe: thermal (plasma), mechanical (tool used for SG replacement);
    - Pressuriser: thermal (torch);
    - Reactor pressure vessel internals: thermal (plasma), mechanical (circular or band saw), abrasive water jet;
    - Pressure vessel / pressure vessel head: thermal (plasma), mechanical (circular saw);
    - Reactor pit concrete: diamond wires and core boring tools, circular saw;
    - Reactor pressure vessel supporting ring: circular saw.

- Decontamination techniques:

    - Full decontamination process is scheduled for the primary circuit, using a process such as CORD-UV. The process will be performed on the intact primary circuit after defuelling. The CORD-UV process allows:

        - the oxidation to carbon dioxide by means of UV (ultraviolet light) of the organic chemicals that are used;
        - the completion of the entire decontamination with only one fill of water as the circulation water is cleaned by ion exchange;
        - the removal of the oxide layers present and a controlled dissolution of a layer of base metal material;
        - The cycle sequence of the decontamination (pre-oxidation, reduction, decontamination, chemical decontamination) may be repeated until the activity is removed and fixed on ion exchange resins;
        - The use of UV wet oxidation and ion exchange minimises the amount of secondary waste produced.

    - Item specific decontamination will be performed in a workshop to allow some reclassification and even recycling or controlled clearance for disposal (on a case by case basis).

This baseline can be adapted to include additional options such as:

1. Removal of components (reactor pressure vessel, steam generator) in one piece as demonstrated with various projects worldwide. This may:

    - Reduce operations and need of auxiliary installations;
    - Reduce duration of the decommissioning;
    - Allow recycling of the reactor component materials to fabricate waste containers to minimise volume for disposal.

2. The steam generators may be fully dismantled and packaged within the reactor containment building.

3. Where there are several reactors on the same site or several sites could be dismantled within a few years of each other, the implementation of a centralised facility for size reduction and packaging of reactors components may be feasible. This will:
   - Use the same installation to dismantle and pack components of several reactors, several sites;
   - Allow development of very specialised / heavy tools;
   - Important constraints to deal with the transport of such large components needs to be taken account of.

4. Final dismantling may be adapted to take advantage of radiological decay (as seen with the use of interim storage for short lived ILW). In this case, the decommissioning sequence would be as follows:
   - Removal of fissile materials (i.e. spent fuel) and radioactive liquids;
   - Survey of this “interim storage nuclear installation.”

**TABLE 27: ADVANTAGES AND DISADVANTAGES OF PROLONGING FINAL DISMANTLING**

| – Advantages | – Disadvantages |
|---|---|
| • Take advantage of future improvements in decommissioning technology.<br>• Allow decay of activated waste to limit quantity and volume of ILW (which is also the case with the interim storage requested) and in general allow some declassification of waste which will reduce the overall cost.<br>• Avoid having to complete a full decontamination of the primary circuit.<br>• Allow immediate transport of packaged ILW to the Geological Disposal Facility (if available).<br>• Reduce dose uptake of dismantling operations.<br>• Reduce the biological protection required (in-situ and for the packaging).<br>• Reduce the packed volume of waste (where radiological limits determine packaging factors) and then the associated cost.<br>– | • Increased long term monitoring costs.<br>• Potential loss of knowledge.<br>• Constraints on the future development of the site.<br>• Financial burden of extended care and maintenance of a disused facility.<br>• Need for new handling and process equipment.<br>• More stringent regulation.<br>– |

### 8.1.4 Baseline decommissioning scenario

The decommissioning of a nuclear facility comprises several technical operations and administrative processes whose end point is the site’s regulatory de-licensing.

The following decommissioning sequence applies and may proceed from the decision to the permanent shut down of the facility following regulatory consent to proceed:

- Removal of fissile materials and radioactive liquids; post shutdown but with the nuclear systems still operational, full in-situ decontamination of primary circuit using a process such as CORD-UV. The removal of fissile materials, radiological liquids and most of the contamination eliminates the largest part of the radiological hazard.
- Depending on the technical requirements, demolition or refurbishment of the conventional and non nuclear plant and construction of decommissioning specific service facilities;

- Dismantling of the activated and contaminated equipment and structures
- Demolishing and removal of what remains of the facility to a pre-defined end state which has been agreed with the Regulators and local planning officers, (partial or total de-licensing). In the reference case strategy the end state is assumed to include the radiological decontamination of all buildings and their demolition to one meter below ground level.

The wastes produced by these operations will be removed from the site, possibly after interim storage on the site. This will almost certainly be the case for ILW. LLW and VLLW should be removed from site without the need for interim storage.

The corresponding programme is outlined in Figure 26. This is an indicative decommissioning programme only. Decomissioning programmes will be defined by the utilities.

| No. | Task | Years from Decommissioning | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | -1 | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 | 12 | 13 | 14 |
| 1 | DISMANTLING OF 1 EPR UNIT | | | | | | | | | | | | | | | |
| 2 | Cessation of unit operation | | | | | | | | | | | | | | | |
| 3 | Permanent shutdown of the unit | | | | | | | | | | | | | | | |
| 4 | Dismantling of the unit | | | | | | | | | | | | | | | |
| 5 | Nuclear Auxiliary Building | | | | | | | | | | | | | | | |
| 6 | Fuel Building | | | | | | | | | | | | | | | |
| 7 | Reactor Building | | | | | | | | | | | | | | | |
| 8 | Auxiliary circuit dismantling prior to primary circuit | | | | | | | | | | | | | | | |
| 9 | Dismantling of primary circuit | | | | | | | | | | | | | | | |
| 10 | Dismantling of the 4 loops with SG evacuation | | | | | | | | | | | | | | | |
| 11 | Dismantling of the SG in a dedicated workshop | | | | | | | | | | | | | | | |
| 12 | Dismantling of RPV, intervals and reactor pit | | | | | | | | | | | | | | | |
| 13 | End of dismantling of auxiliary equipment | | | | | | | | | | | | | | | |
| 14 | RB decontamination and cleaning | | | | | | | | | | | | | | | |
| 15 | Other buildings (safeguard building, waste building…) | | | | | | | | | | | | | | | |
| 16 | Demolition of buildings | | | | | | | | | | | | | | | |

SG – Steam Generator
RPV – Reactor Pressure Vessel
RB – Reactor Building

**FIGURE 26: INDICATIVE DISMANTLING PROGRAMME FOR ONE EPR UNIT**

The dismantling scenario for the primary circuit is as follows:

- Preliminary decontamination of the primary circuit with CORD-UV type process in order to:
  - Reduce dose rates to allow contact ("hands-on") dismantling of many of the primary circuit components;
  - Minimise volume of packed waste (reduction of required biological shielding) even with production of secondary waste (Ion exchange resins).
- Preparation for primary circuit dismantling, dismantling of last auxiliary pipes;
- Export of the steam generators out of the reactor building for dismantling in a dedicated workshop:
  - Separation and export of the upper sections of the steam generators to a non-contamination area workshop (the upper sections will have only contacted fluids from the secondary reactor circuits and thus would be highly unlikely to be significantly contaminated);
  - Export of the lower steam generator section (containing the primary circuit tube bundle) to the contamination area workshop constructed to accommodate this size of component.
- Export and dismantling of the reactor coolant pumps;
- Dismantling and export of the main coolant pipes;
- Export and dismantling of the pressuriser;
- Preparation for internals dismantling with adjustment of the water level in the reactor compartment;
- Dismantling of upper core internals associated with the pressure vessel head under water on its storage stand in the reactor pool (the pressure vessel head is removed for refuelling the reactor and has a dedicated storage stand within the reactor pit;
- Dismantling of thermal shield in the reactor compartment pool;
- Dismantling of lower core internals under water, on its storage stand in the reactor compartment pool;
- Dismantling of reactor pressure vessel;
- Dismantling of reactor vessel head;
- Dismantling of activated part of the reactor compartment (the first 60 cm depth of concrete at the level of the reactor core);
- End of primary circuit dismantling - Radiological decontamination and demolition of the buildings.

This scenario for primary circuit dismantling and, by extrapolation, the complete reactor site dismantling, is based on:

- Use of currently practised, realistic and achievable operations and techniques guaranteeing the control of the dismantling; feasibility is demonstrated through the use of techniques validated worldwide.
- Technical choice and operation modes reducing dose uptake to operators and members of the public:

- Minimisation of operator dose uptake (limitation of required personnel, limitation of work at contact, minimisation of operation durations, use of mobile radiological shielding protection);
- Use as far as necessary of services like static and/or dynamic containment to avoid any contamination risk;
- Minimisation of dismantling cost;
- Limitation of investment cost; use as far as possible existing equipment present on site from normal operations;
- Limitation of technical risk by using existing and proven technologies;
- Management of the waste and minimisation of secondary and induced waste generated.

### 8.1.5 Raw waste characteristics

#### 8.1.5.1 ILW

##### 8.1.5.1.1 Contaminated ILW

Based on preliminary calculations, the contaminated ILW resulting from the immediate dismantling of the whole plant (considering classification at the dismantling date) will be limited to the waste resulting from decontamination of the primary circuit which results in the production of contaminated ion exchange resins.

This waste amounts to about 30 to 40 $m^3$ of resins. The β/γ activity on the resins will range between 1E5 Bq/g and 4.5E5 Bq/g.

Based on 60 years of electricity production for an EPR followed by the immediate dismantling of a single reactor, the ILW resins will be produced in year 61.

After interim storage up to the 100 years after the beginning of reactor operation (40 years after reactor shutdown), the activity of these wastes will be significantly reduced and a proportion of may be reclassified as LLW.

##### 8.1.5.1.2 Activated ILW

Based on preliminary calculations, the ILW resulting from the immediate dismantling of the whole plant (considering classification at the dismantling date) will be limited, depending of the decay time considered, to:

- The neutron shield;
- Most of the reactor pressure vessel internals (upper and lower internals, excepted the far upper sections of these components);
- The core shells of the reactor vessel (part of the reactor vessel located near the core cavity);
- Part of the concrete of the reactor compartment.

These wastes amount to approximately 450 te of raw solid metallic waste and 180 te of concrete.

According to the boundary calculations, the most activated components (Neutron shield and lower support plate) produce decay heats (10 and 5kW/$m^3$ respectively) close to the ILW limit of <2kW/$m^3$. These components will cool quickly in the reactor pool and can be treated as ILW within the timeframe associated with immediate decommissioning.

After a decay period of approximately 30 years some of the ILW (some metallic wastes and the whole concrete) can be re-classified as LLW.

Based on a 60 years of electricity production for an EPR followed by the immediate dismantling of a single reactor, the ILW solid metallic wastes are produced during years 67 and 68 and ILW solid concrete wastes during years 69 and 70.

The effects of the activity decay up to the year 100 for these 2 wasteforms are as follows:

- After the 32 to 33 years of decay, about 10% of the activated metallic components can be re classified as LLW, the other 90% remains classified as ILW;
- After 30 to 31 years of decay, the concrete of the reactor compartment is at worst classified LLW.

#### 8.1.5.2 LLW / VLLW

Radiological characteristics of the decommissioning LLW and VLLW have not been detailed in this report.

#### 8.1.5.3 Summary

The best estimate to date of the quantities of waste produced during one EPR unit decommissioning is summarised in Table 28.

**TABLE 28: QUANTITIES OF WASTE PRODUCED DURING DECOMMISSIONING**

| | Mass (te) | | |
|---|---|---|---|
| | ILW | LLW | VLLW |
| CPP | 673 | 2735 | 1898 |
| NSSS | 0 | 2259 | 0 |
| BNI | 0 | 2824 | 2605 |
| CONCRETE | 0 | 75 | 455 |
| Total | 673 | 7893 | 4958 |

The production of radioactive waste against time is illustrated at Figure 27.

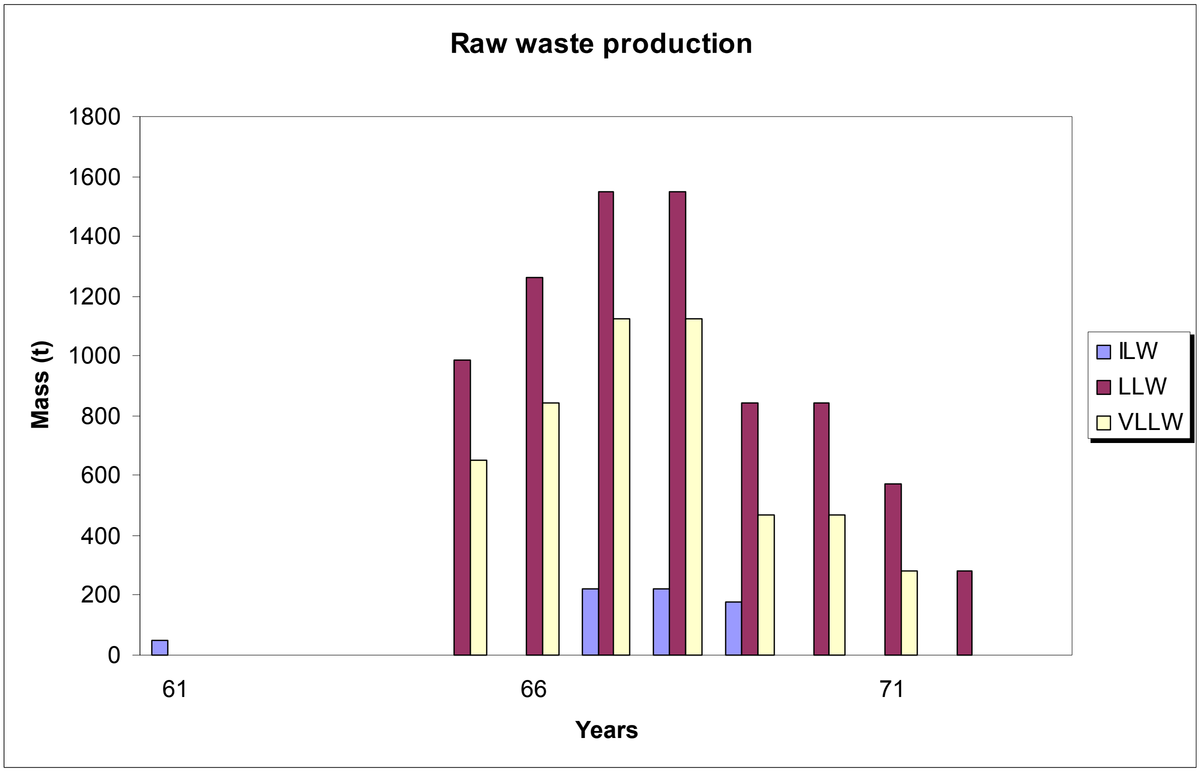



**FIGURE 27: RAW WASTE PRODUCTION AGAINST TIME**

### 8.1.6 Waste processing

#### 8.1.6.1 ILW

##### 8.1.6.1.1 Contaminated ILW

This consists of the Ion-Exchange Resins used during the full decontamination of the primary circuit.

Ion exchange resins will be encapsulated in a solid matrix which is in direct contact with the waste container. Based on the NIREX report N/104, the container proposed for this waste is the unshielded waste package “500 litre Solids Drum”.

A transport container providing an additional shielding not bigger than 285 mm of steel is then required for transport and waste handling in the proposed deep geological disposal facility.

Ion-exchange resins can be dewatered or immobilised in a polymer matrix (epoxy) inside this container.

Epoxy resins is usually mixed into a container pre-equipped with a stirring device. Epoxy matrix enables a greater quantity of ion exchange resin to be immobilised in the matrix than cement mortar. Note that if the waste containment can be assured by the container itself, no additives need be considered.

Data related to waste processing at production time have been summarized in Table 30.

The ion exchange resins are the main secondary ILW produced during the post operational clean-out operations. Some additional secondary ILW, (e.g. activated swarf) will also be produced, depending on the cutting technique used for dismantling the pressure vessel and associated internals but this is not expected to be a significant arising.

**FIGURE 28: 500 LITRE DRUMS**

The indicative radiological characteristics in Bq per drum at the production time are given in Table 29.

**TABLE 29: RADIOLOGICAL CHARACTERISTICS PER DRUM AT PRODUCTION TIME**

| | Min. | Max. |
|---|---|---|
| Fe59 | 4.76E+09 | 1.16E+10 |
| Mn54 | 7.12E+09 | 1.36E+10 |
| Co58 | 3.92E+10 | 2.57E+11 |
| Co60 | 2.47E+10 | 4.92E+10 |
| Total | 7.58E+10 | 3.31E+11 |

**TABLE 30: ION EXCHANGE RESINS PROCESSING**

| | |
|---|---|
| **Nature of waste stream:** | Ion-exchange resin used for decontamination of the primary circuit (full system decontamination) |
| **Raw waste volume:** | 30 to 40 $m^3$ |
| **Proposed matrix:** | Polymer matrix (epoxy) |
| **Package type:** | NDA specification 500L Solids Drum (V=0.5 $m^3$) |
| **Raw waste per package:** | About 150 kg |
| **Waste form density:** | 1500 $kg/m^3$ |
| **Waste package mass:** | 750 kg |
| **Number of packages:** | About 370 |
| **Physical/chemical composition:** | The waste in an average package is expected to comprise:<br>• 20% of resin<br>• 80% of polymer matrix<br>– |
| **Reference date:** | At reactor dismantling (year 61) |
| **External gamma dose rates:** | At 1m: < 0.1 mSv/h.<br>At contact: < 2 mSv/h.<br>With a 285 mm steel shielding transport container as indicated in NIREX report N/104. |

### 8.1.6.2 Activated ILW

Metallic and concrete wastes from the dismantling of the activated components near the reactor core are solid wastes on the basis of the guidance given in NDA report N/104. Assuming that these wastes are transported from the place of interim storage to the phased geological disposal facility in year 100 and considering their radiological characterisation at this date the waste packaging scenario is as follows:

- The shielded 4 metre boxes were developed for packaging wastes arising from the dismantling of nuclear facilities. This freight container type package incorporates concrete liner shielding that can have various thicknesses adapted to the activity level of the wastes. Based on the maximum gross mass of 64t for such a package, with the estimated activation of the components and the requirements in terms of dose rate at contact (2mSv/h) and at 1 metre (0,1mSv/h) from this transport container, this kind of container is not suitable for all of the ILW produced during the dismantling of an EPR. The shielding required for the containment of the core shells of the reactor vessel is of the order of 100mm thick. The other metallic ILW resulting from the dismantling of the reactor (Neutron shield, internals) are not compatible with this kind of package. For concrete waste use of 4 metre boxes without additional shielding is possible. The characteristics of this conditioned waste are given in Table 31.

**FIGURE 29: 4M$^3$ BOX**

- The 3 cubic metre boxes potentially incorporating additional shielding will be used for ILW (Neutron shield and most of the reactor vessel internals) that is not suitable for packaging in 4m boxes. In addition, this packaged waste will required the use of a transport container (of the type Standard Waste Package Container mentioned in the report N/104) providing an additional shielding of 285mm of steel. The characteristics of this conditioned waste are given in Table 32.

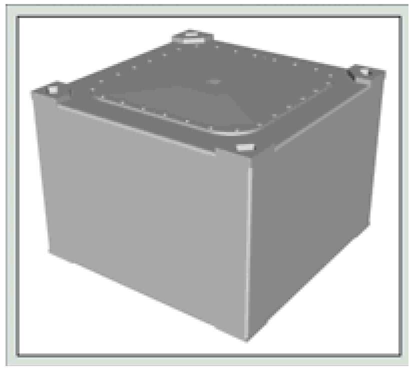


**FIGURE 30: $3M^3$ BOX CORNER LIFT VARIANT**

**FIGURE 31: $3M^3$ BOX MID-SIDE LIFTING VARIANT**

All these metallic wastes will be encapsulated in a cement grout.

Note: after interim storage, concrete wastes resulting from the dismantling of the reactor compartment are at worst classified LLW.

## TABLE 31: 4 M$^3$ WITH 100 MM CONCRETE SHIELDING CHARACTERISTICS

| | |
| --- | --- |
| **Nature of waste stream:** | **Reactor vessel:** Parts from the reactor vessel (cladding included) near the core |
| **Raw waste volume:** | About 23 m$^3$ |
| **Proposed matrix:** | Mortar matrix |
| **Package type:** | NDA specification 4 metre Box with 100 mm concrete shielding |
| **Raw waste per package:** | 18 to 22 tonnes per package (limited by filing ratio) |
| **Waste form density:** | 3140 to 3320 kg\m$^3$ ($\rho_{stainless\ steel}$=7850 kg/m$^3$ & $\rho_{grout}$=2300 kg/m$^3$) |
| **Waste package mass:** | 61 to 64 tonnes<br>$m_{package} = m_{4\ meter\ Box} + m_{immobilised\ grout} + m_{steel\ waste} + m_{concrete\ shielding}$ |
| **Number of packages:** | About 10 |
| **Physical/chemical composition:** | The waste in an average package by weight is expected to comprise:<br>– 40% of ferritic steel waste,<br>– 2% of stainless cladding steel waste,<br>– 58% of mortar. |
| **Derivation of case:** | DARWIN PEPIN calculation with representatives flux in the reactor vessel |
| **Reference date:** | After interim storage (year 100) |
| **Total package activity:** | **4 metre Box:** 1.36 TBq βγ |
| **Waste classification:**<br>**(based on mass of wasteform)** | >12 000 Bq/g - ILW<br>**4 metre Box:** 7.6 E 4 Bq/g βγ |
| **External gamma dose rates:** | At 1m: < 0.1 mSv/h<br>At contact: < 2 mSv/h<br>From the package. |

**TABLE 32: 3 M³ BOX CHARACTERISTICS**

| | |
|---|---|
| **Nature of waste stream:** | • **Upper Internals:** Upper support columns (under upper support plate) and Upper core plate<br>• **Lower internals:** Core barrel, Flow distribution device, Neutron shield, Lower support plate. |
| **Raw waste volume:** | About 27 $m^3$ |
| **Proposed matrix:** | Mortar matrix |
| **Package type:** | NDA specification 3 cubic metre Box<br>Additional transport shielding SWTC-285 described in the NDA generic waste package specification is needed for transport. |
| **Raw waste per package:** | 2.6 to 3.7 ton per package |
| **Waste form density:** | 3150 to 3520 kg/$m^3$ ($\rho_{stainless\ steel}$=7850 kg/m3 & $\rho_{grout}$=2300 kg/$m^3$) |
| **Waste package mass:** | 7.7 ton<br>$m_{package} = m_{3\ metre\ box} + m_{immobilised\ grout} + m_{steel\ waste}$ |
| **Number of packages:** | About 75 |
| **Physical/chemical composition:** | The waste in an average package by weight is expected to comprise:<br>– 17% of stainless steel waste,<br>– 83% of mortar. |
| **Derivation of case:** | DARWIN PEPIN calculation with representatives flux in the reactor vessel |
| **Reference date:** | After interim storage (years 100) |
| **Total package activity:** | **3 metre Box**: 2440 TBq $\beta\gamma$ |
| **Waste classification:**<br>**(based on mass of wasteform)** | >12 000 Bq/g – ILW<br>**3 metre Box**: 2.1E8 Bq/g < Activity $\beta\gamma$ < 1.25E9 Bq/g |
| **External gamma dose rates:** | At 1m: < 0.1 mSv/h<br>At contact: < 2 mSv/h<br>From the transport container SWTC 285. |

#### 8.1.6.3 LLW / VLLW

The LLW and VLLW are solid wastes produced during the dismantling of a reactor.
The packaging scenario considered in the present report is:

- Conditioning of the LLW in HHISO (Half Height ISO) container;
- No special packaging for the VLLW for which only the global volume is considered.

#### 8.1.6.4 Summary

In summary, after interim storage on site, the ILW resulting from decommissioning of an EPR unit is as follows: 390t of raw solid activated metallic waste corresponding to about 392m$^3$ of packed waste conditioned in approximately:

- 10 “4-metre boxes” with an additional 100mm concrete shielding;
- 75 “3 cubic metre boxes”.

For these 3 cubic metre boxes, a transport container not bigger than the SWTC 285 would provide adequate shielding for transport and waste handling at the repository site.

The corresponding summary of waste package is given in Table 31 and Table 32 (related respectively to the 4m boxes with an additional 100 mm concrete shielding and the 3m$^3$ boxes).

The other nuclear waste (LLW and VLLW) produced during the decommissioning of one EPR unit correspond respectively to about 480 HHISO-20’ containers or 5400 m$^3$.

For waste storage and transport purposes, the production of ILW during the decommissioning of an EPR unit will produce 310t of raw solid activated metallic waste corresponding to about 670m$^3$ of packed waste conditioned in approximately:

- 28 “4-metre boxes” with an additional 400mm concrete shielding;
- 62 “3-cubic metre boxes” with an additional 100mm steel shielding.

For these 3-cubic metre boxes, a transport container not bigger than the SWTC 285 would provide adequate shielding for transport to and waste handling at the repository site.

About 136t of high activity metallic wastes (corresponding to neutron shield and lower support plate) are classified as ILW and are pakced in the 3 cubic metre boxes with an additional steel shielding.

The characteristics of this conditioned waste are given in Table 35 for the 4 metre boxes and in Table 36 for the 3 cubic metre boxes.

- 180t of raw solid activated concrete waste corresponding to about 132m$^3$ of packed waste conditioned in approximately six “4-metre boxes”.

  The characteristics of this conditioned concrete waste are given in Table 37.

- 30 to 40 m$^3$ of ion-exchange resins packed in approximately 370 “500-litre drums” (see Table 31) and transportable to the repository using an SWTC.

The other nuclear waste (LLW and VLLW) produced during the decommissioning of an EPR unit correspond respectively to about 610 HHISO-20' container (LLW) or 5500 $m^3$ (VLLW).The overall production of waste packages is indicated in Table 33.

**TABLE 33: OVERALL PRODUCTION OF WASTE PACKAGES**

| | Waste package (nb) | |
|---|---|---|
| | ILW | LLW |
| ILW - 500 liter drum | 370 | |
| ILW - 4 metre box - 400 mm | 28 | |
| ILW - 3 cubic metre box - 100 mm | 167 | |
| ILW - 4 metre box | 6 | |
| LLW - HHISO 20' | | 610 |

The production of packed waste against time in terms of package number per type and packed volume are illustrated on the Figure 32 and Figure 33.

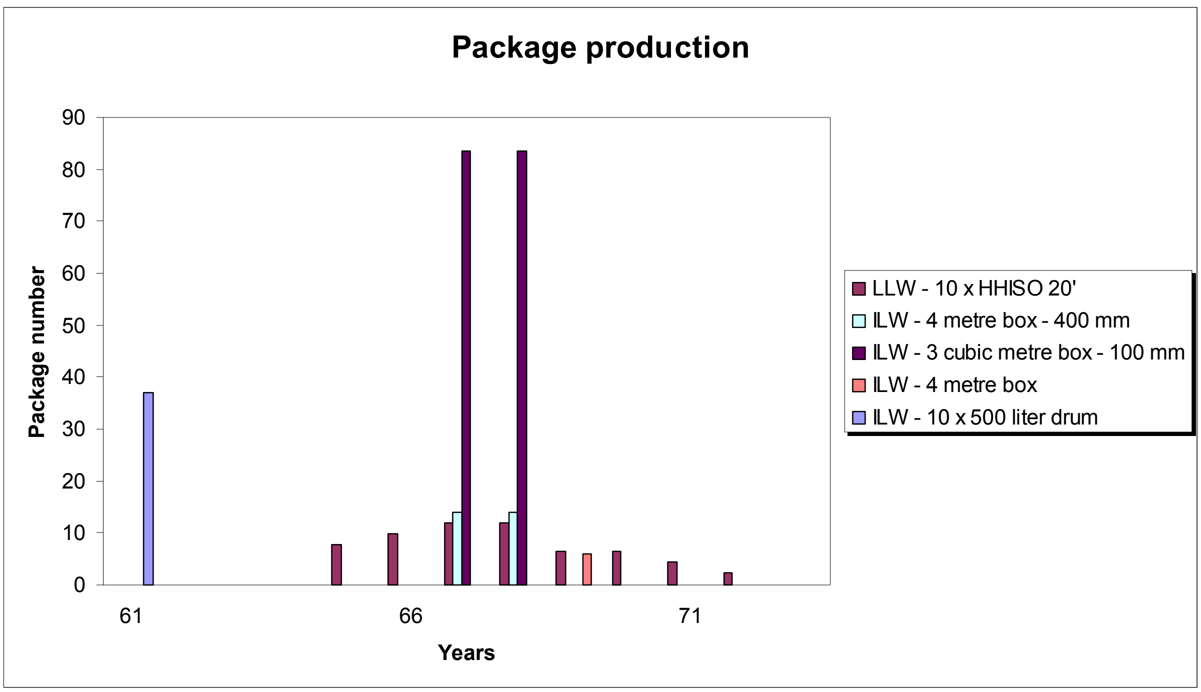



**FIGURE 32: NUMBER OF WASTE PACKAGES PRODUCED AGAINST TIME**

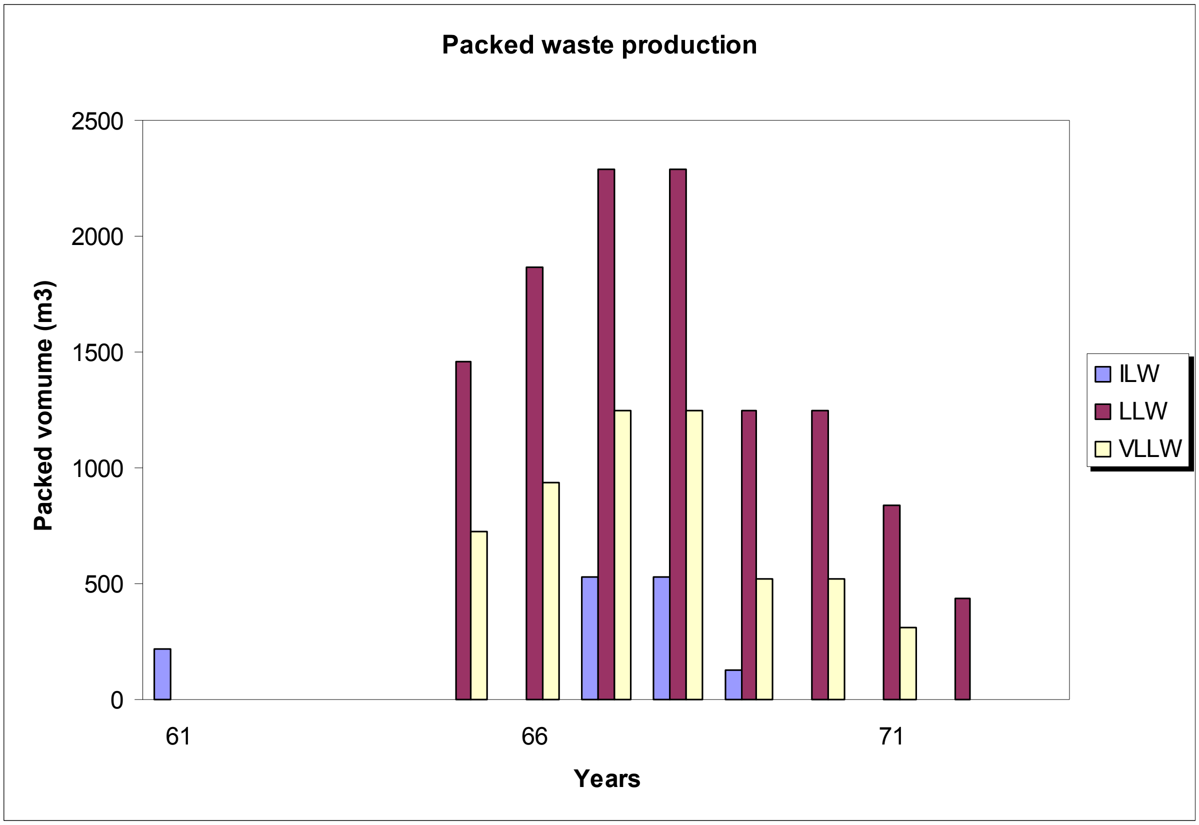



**FIGURE 33: PACKED VOLUME PRODUCED AGAINST TIME**

Further packaging optimisation assessments will be necessary to optimise the radiological inventories, raw waste loading per package and criteria for interim storage.

The impact of the interim storage up to 100 years is shown in Table 34.

**TABLE 34: IMPACT OF TIME ON ILW**

| ILW Waste | At production time | | | | After interim storage (+) | | | |
|---|---|---|---|---|---|---|---|---|
| | **Waste quantity** | **Type of package** | **Nb** | **Packed volume ($m^3$)** | **Waste quantity** | **Type of package** | **Nb** | **Packed volume ($m^3$)** |
| **IER from decontamination** | 30/40 $m^3$ | 500l drum | 370 | 220 | 0 | - | - | 0 |
| **Activated components** | 225 te | 4 metre box - 400mm | 28 | 600 | 180 te | 4 metre box - 100mm | 10 | 220 |
| | 221 te | 3 cubic metre box – 100 mm | 167 | 450 | 210 te | 3 cubic metre box | 75 | 200 |
| **Activated concrete** | 180 te | 4 metre box | 6 | 130 | 0 | - | - | 0 |
| **TOTAL** | **626 te 30/40 $m^3$** | **-** | **571** | **1400** | **390 te** | **-** | **85** | **420** |

(*) comprising the HLW after some years of decay
(+) same kind of evolution in case of deferred dismantling

**TABLE 35: 4 METRE BOX WITH 400 MM CONCRETE SHIELDING CHARACTERISTICS FOR METALLIC WASTE**

| | |
|---|---|
| **Nature of waste stream:** | **Reactor vessel:** Parts from the reactor vessel (cladding included) near the core |
| **Raw waste volume:** | About 29 $m^3$ |
| **Proposed matrix:** | Mortar matrix |
| **Package type:** | NDA specification 4 metre Box with 400 mm concrete shielding |
| **Raw waste per package:** | 7.4 to 10.6 per package (limited by filling ratio). |
| **Waste form density:** | 3140 to 3500 kg\m$^3$ ($\rho_{stainless\ steel}$=7850 kg/m$^3$ & $\rho_{grout}$=2300 kg/m$^3$) |
| **Waste package mass:** | 54 to 56 tonnes<br>$m_{package} = m_{4\ meter\ Box} + m_{immobilised\ grout} + m_{steel\ waste} + m_{concrete\ shielding}$ |
| **Number of packages:** | About 28 |
| **Physical/chemical composition:** | The waste in an average package by weight is expected to comprise:<br>– 38% of ferritic steel waste,<br>– 2% of stainless steel waste,<br>– 60% of mortar. |
| **Derivation of case:** | DARWIN PEPIN calculation with representatives flux in the reactor vessel |
| **Reference date:** | At shut down (years 65) |
| **Total package activity:** | **4 metre Box with 400mm internal concrete shielding:** 10.7 TBq βγ |
| **Waste classification:**<br>**(based on mass of wasteform)** | >12 000 Bq/g - ILW<br>**4 metre Box with 400mm internal concrete shielding:** 1.33 E6 Bq/g βγ |
| **External gamma dose rates:** | At 1m: < 0.1 mSv/h<br>At contact: < 2 mSv/h<br>From the package. |

## TABLE 36: 3 CUBIC METRE BOX WITH 100 MM STEEL SHIELDING CHARACTERISTICS FOR METALLIC WASTE

| | |
|---|---|
| **Nature of waste stream:** | **Lower & upper internals from EPR pressure vessel:** Core barrel, Upper core plate, Flow distribution device, Upper support column (under upper support plate)**.** |
| **Raw waste volume:** | About 11 $m^3$ |
| **Proposed matrix:** | Mortar matrix |
| **Package type:** | NDA specification 3 cubic metre Box with 100mm internal steel shielding Additional transport shielding SWTC-285, described in the NDA Generic Waste Package Specification, is needed. |
| **Raw waste per package:** | 1.3 ton per package (limited by package heat output). |
| **Waste form density:** | 2800 kg\m$^3$ ($\rho_{stainless\ steel}$=7850kg/m3 & $\rho_{grout}$=2300kg/m$^3$) |
| **Derivation of case:** | DARWIN PEPIN calculation with representatives flux in the reactor vessel |
| **Reference date:** | At shut down (years 65) |
| **Total package activity:** | **3 cubic metre Box with 100mm shielding:** 2000 TBq βγ |
| **Waste classification:**<br>**(based on mass of wasteform)** | >12 000 Bq/g - ILW<br>**3 cubic metre Box with 100mm shielding:**<br>9.2E8 Bq/g < Activity βγ < 1.6E9 Bq/g |
| **External gamma dose rates:** | At 1m: < 0.1 mSv/h<br>At contact: < 2 mSv/h<br>From the transport container SWTC 285. |

**TABLE 37: 4 METRE BOX CHARACTERISTICS FOR ACTIVATED CONCRETE**

| | |
|---|---|
| **Nature of waste stream:** | Concrete of reactor compartment |
| **Source of documentation:** | NIREX Report N/104-GWPS Volume 1 & 2 Issue 2,<br>Fundamentals of radioactive waste management |
| **Waste stream identifiers:** | 'to be allocated' |
| **Raw waste volume:** | About 77 $m^3$ |
| **Proposed matrix:** | Mortar matrix |
| **Package type:** | NDA specification 4 metre Box |
| **Raw waste per package:** | 30.7 tonnes max per package (limited by useful volume). |
| **Waste form density:** | 2340 kg\m$^3$ ($\rho_{pitgrout}$=2360kg/m$^3$ $\rho_{grout}$=2300kg/m$^3$) |
| **Waste package mass:** | 50 ton Max<br>$M_{package}=m_{container} + m_{immobilised\ grout} + m_{steel\ waste} + m_{steel\ shielding}$ |
| **Number of packages:** | Approximately 6 |
| **Physical/chemical composition:** | The waste in an average package by weight is expected to comprise:<br>68% of concrete waste,<br>32% of mortar matrix. |
| **Derivation of case:** | DARWIN PEPIN calculation with representatives flux in the reactor vessel |
| **Reference date:** | At shut down (years 65) |
| **Total package activity:** | 4 metre Box: 0.6 GBq βγ |
| **Waste classification:**<br>**(based on mass of wasteform)** | >12 000 Bq/g - ILW<br>4 metre Box:<br>Global Activity βγ 1.99E+04 Bq/g |
| **External gamma dose rates:** | At 1m: < 0.1 mSv/h<br>At contact: < 2 mSv/h<br>From the package. |

# 9 SPENT FUEL

## 9.1 EPR Spent fuel Characteristics

### 9.1.1 Summary Description of the Core and Fuel Assemblies

The EPR reactor core contains the nuclear fuel in which the fission reaction occurs. The remainder of the active core structure serves either to support the fuel, control the chain reaction or to channel the coolant.

The reactor core consists of fuel rods held in bundles by spacer grids and top and bottom fittings. The fuel rods consist of uranium oxide pellets stacked in a cladding tube, plugged and seal-welded to encapsulate the fuel (see Figure 34). The fuel rods are arranged in fuel assemblies (see Figure 35 and Table 38) which are formed by a 17 x 17 array of 265 fuel rods and 24 guide thimbles. The 24 guide thimbles are joined to the grids and to the top and bottom nozzles. The guide thimbles may also hold rod cluster control assemblies, neutron source rods or the in-core instrumentation. Guide thimbles that do not contain one of these components are fitted with plugs to limit the bypass flow. The grid assemblies consist of an arrangement of interlocked straps. The straps contain spring fingers and dimples for fuel rod support, as well as coolant mixing vanes. The reactor core consists of 241 assemblies.

The fuel assemblies are made from a variety of fuel formulations. These include different enrichments of uranium and some fuel assemblies also contain a neutron poison, gadolinium oxide ($Gd_2O_3$). The gadolinium oxide is mixed with the fuel and depletes slowly with burn up which helps to control the fission process. For core refuelling, the number and the characteristics of the fresh assemblies depend on reactor operating parameters and the fuel management strategies e.g. cycle length, type of loading, fuel management regime, fuel type, etc. Typically, around one third of the fuel assemblies are replaced at each refuelling.

**TABLE 38: FUEL ASSEMBLY DATA**

| | |
|---|---|
| **Assemblies** | |
| Arrangement of fuel rods | 17 by 17 square array |
| External maximum section (mm x mm) | 214 x 214 |
| Maximum length (mm) | 4859 |
| Active length (mm) (Average, at 20 °C) | 4200 |
| Overall weight (kg) | 780 |
| Uranium mass (kg) | 527.5 |
| **Rods** | |
| Number of fuel rods | 265 |
| Fuel rod outer diameter (mm) | 9.5 |
| Cladding thickness (mm) | 0.57 |
| Rod pitch (mm) | 12.6 |
| **Cladding** | |
| Material | M5 |

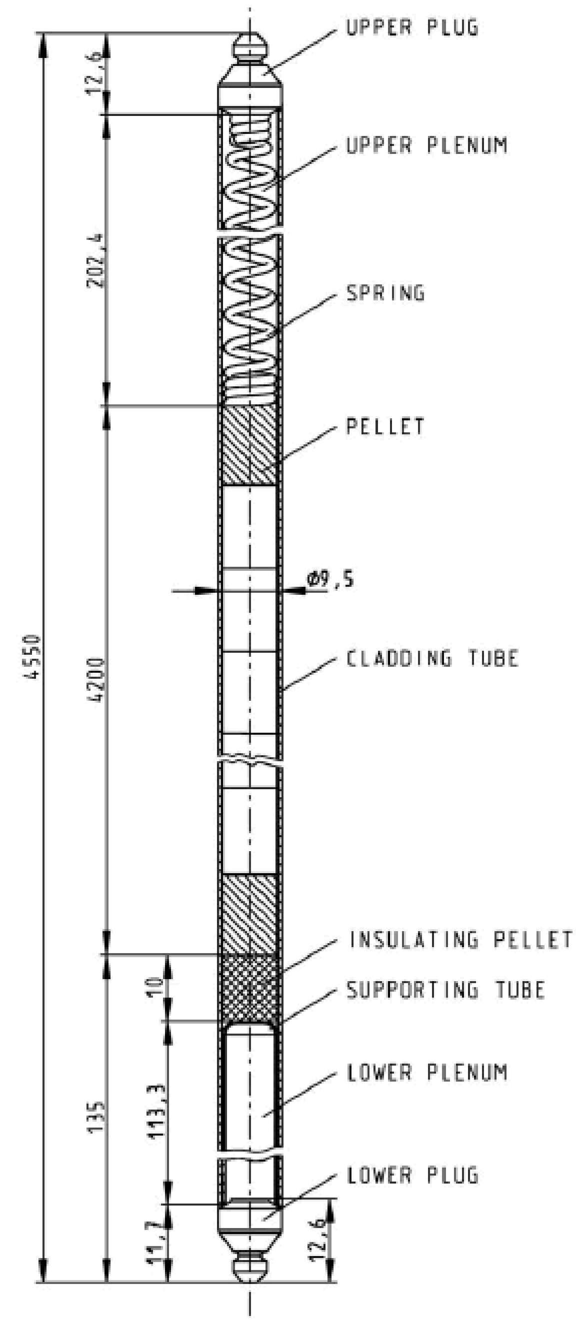



**FIGURE 34: FUEL ROD**

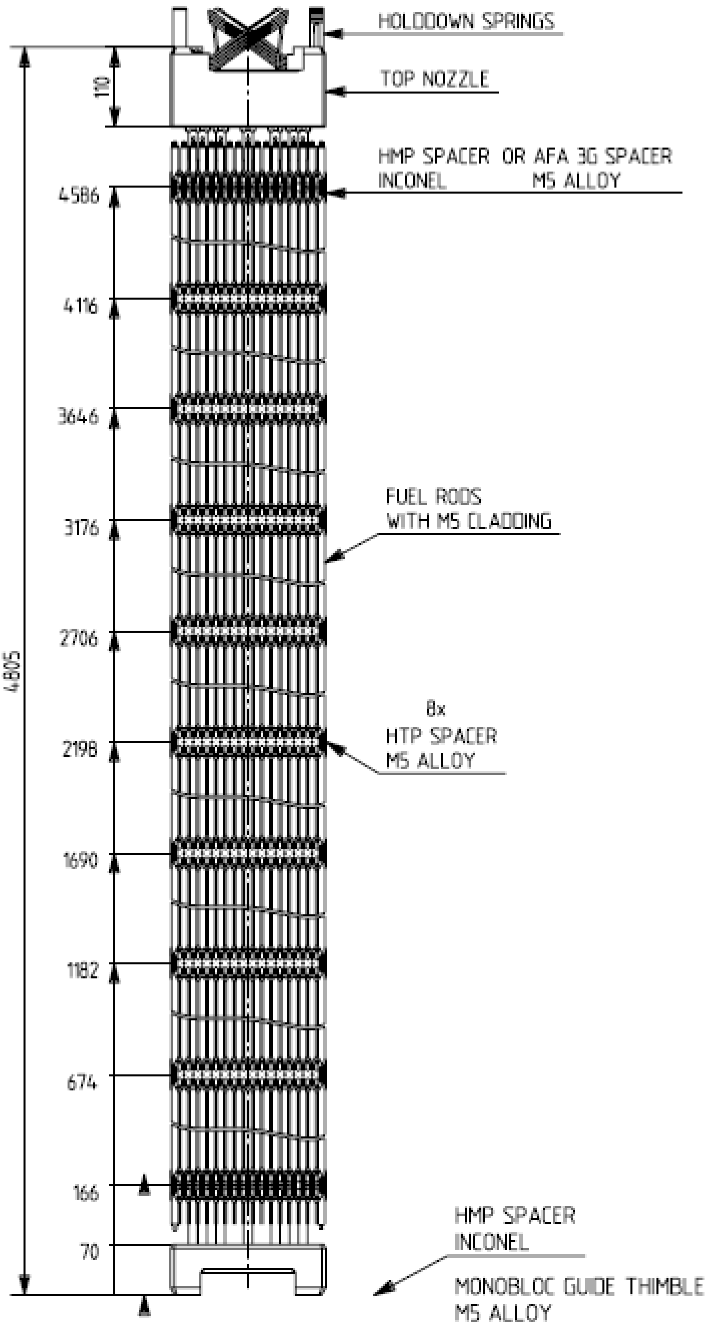



**FIGURE 35: FUEL ASSEMBLY**

### 9.1.2 Thermal Characteristics

The EPR reactor is designed to use both MOX and uranium oxide fuels. For the purposes of this report it is assumed that the reactor uses uranium oxide fuel only. The uranium oxide fuel is designed with a variety of enrichments, the maximum being 5%. The isotopic composition of the spent fuel depend upon the initial enrichment, whether the fuel has been produced from natural uranium only or whether reprocessed uranium has been used and the fuel management regime conditions to which it is subject in the reactor. The average core region fuel burnup is less than 65,000 MWd/tU.

Spent fuel assemblies are discharged from the reactor and placed into the spent fuel pool to cool and decay for a period of about 10 years before being moved to an interim storage facility. The decay heat generated by an EPR spent fuel assembly (fuel manufactured with natural uranium) which has undergone four 13 month reactor cycles and 10 years of cooling in the Spent Fuel Pool is approximately 1,400 Watts at the time of interim storage. A spent fuel assembly manufactured from reprocessed uranium, under the same conditions, would have a heat output of 2000 Watts.

### 9.1.3 Spent Fuel Quantities

The UK EPR is designed for an operational life of 60 years. At any given time the operational reactor will contain around 127 tonnes of enriched uranium fuel. Reactor refuelling will take place at the end of reactor cycles which can range between 12 and 22 months depending on the fuel management regime adopted. The quantities of spent fuel discharged from the reactor during refuelling can be up to 80 assemblies. The EPR produces between 40 and 60 spent fuel assemblies per year of operation. The number of fuel assemblies produced over the operation of the EPR is 3400 (this equates to 1794 te of uranium).

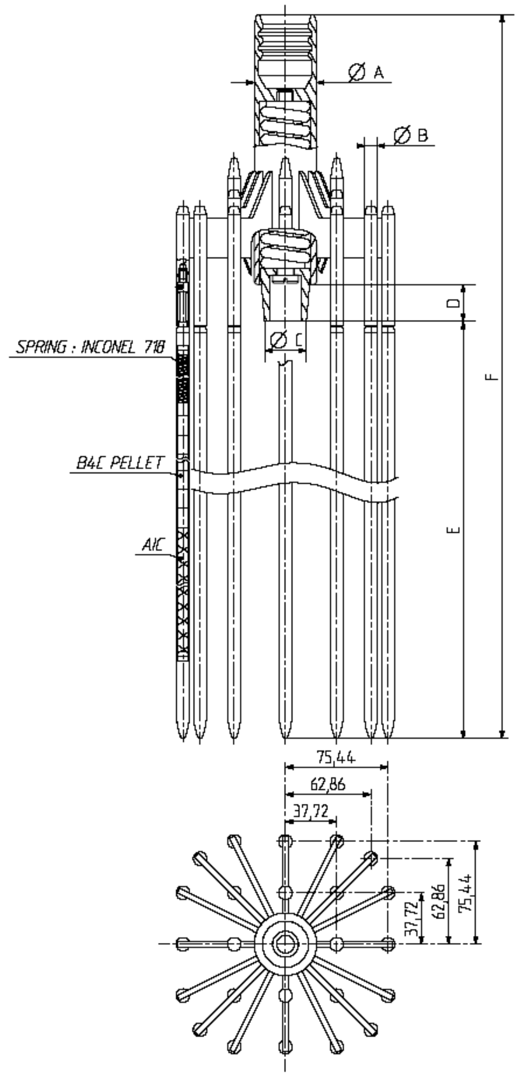
## 9.2 Summary of Core Activated Components

During the operation of the UK EPR, a number of core components used to control or measure neutron activity will need to be replaced during outages. These components when removed from the reactor are highly activated and transferred to the Spent Fuel Pool where they are left to radiologically decay.

**TABLE 39: WASTE STREAM DATASHEET FOR CORE ACTIVATED COMPONENTS**

| Waste Component | Waste Classification at Time of Production | Rate of Production | Total Quantity Expected over 60 year operational phase of EPR (components) |
|---|---|---|---|
| Rod Cluster Control Assemblies (Spider + 24 clad AIC-$B_4C$ absorbing rods) | HLW/ILW | 89 RCCAs replaced every 15 years | 356 |
| In core Instrumention (Aeroball finger tubes) | HLW/ILW | 12 fingers replaced every 15 years | 48 |

These components could be dismantled and packaged inside the Spent Fuel Pool and transferred by flask to the Interim Storage Facility for further decay.




| | |
|---|---|
| Dimension A | 46.74 maxi |
| Dimension B | 9.19 maxi |
| Dimension C | 32.07 maxi |
| Dimension D | 26.7 mini |
| Dimension E | 4492 |
| Dimension F | 4717.5 |

Figure 36: 17 x 17 EPR Rod Cluster Control Assembly (RCCA)

## 9.3 Spent Fuel Treatment and Storage Options

### 9.3.1 Overall Strategy and Assumptions

Fuel cycle back-end solutions consist of spent fuel reprocessing/recycling or direct disposal.

- Spent fuel reprocessing/recycling option is currently implemented for the treatment of spent fuel generated in a number of NPPs worldwide. In this case, spent fuel needs to be stored for a short duration prior to the treatment of spent fuel;
- No spent fuel direct disposal facility is currently available in the world. Research and development programmes are developed, including at laboratory sites, to assess the acceptability of the disposal facility. In this case, an interim storage facility is required (for several decades) before ultimate disposal can be performed.

Both strategies have been selected by various utilities and countries. Others have not made their choice yet and are in a “wait and see” position which also requires interim storage facilities to be available. Therefore, whatever spent fuel back-end option is selected, interim storage facilities are required to allow NPPs to continue to operate. Different storage facilities are operated in the world, each adapted to a specific case: wet or dry technology, short term or long term (up to 100 years), facility on each NPP site or a centralised facility. Moreover, the objectives of these facilities are to protect public, personnel and environment and to keep the fuel assemblies under safe conditions and to maintain their integrity to enable them to be handled and export them for their eventual treatment.

In the UK, the current BERR base case makes the assumption that spent fuel will not be reprocessed and that interim storage facilities for at least 100 years are required.

Various spent fuel storage technologies with different vendors/designers have been implemented worldwide:

- Wet storage:
  - in almost all NPPs (common practice for initial cooling);
  - in centralised facilities coupled with reprocessing (LaHague in France, Sellafield in UK, Mayak in Russia, Rokkashoin Japan);
  - in centralised long term interim storage facilities (Clab in Sweden, Gosgen in Switzerland, Kozloduy in Bulgaria, Krasnoyarsk in Russia, Tihange in Belgium).
- Dry storage:
  - In vaults (CASCAD and TOR in France, Paks in Hungary, Fort St Vrain in US, Habog at Borselle in the Netherlands);
  - In metallic casks: at several NPPs including ISAR and Gorleben in Germany, at Doel in Belgium, at Zwilag in Switzerland, at Surry in USA;
  - In NUHOMS systems on several US NPPs sites and at Metzamor in Armenia.

For the purpose of this report, interim storage technologies have been assessed to determine which options are adapted for the UK EPR spent fuel. AREVA and its subsidiary companies have extensive expertise in the field of spent fuel management enabling a range of different facilities and technologies to be considered for the UK EPR. The solutions described hereafter are AREVA ones.

#### 9.3.1.1 Spent Fuel Unloading Technologies

Spent fuel removed from a reactor must be cooled for an initial period before it can be placed into interim storage. For the UK EPR, fuel assemblies removed from the reactor are cooled underwater in the spent fuel pool for up to 10 years before they are prepared for interim storage. On arrival at the interim storage facility spent fuel can be removed underwater or within a dry unloading cell. Both technologies have been used successfully to receive spent fuel assemblies at AREVA’s La Hague Plant in France.

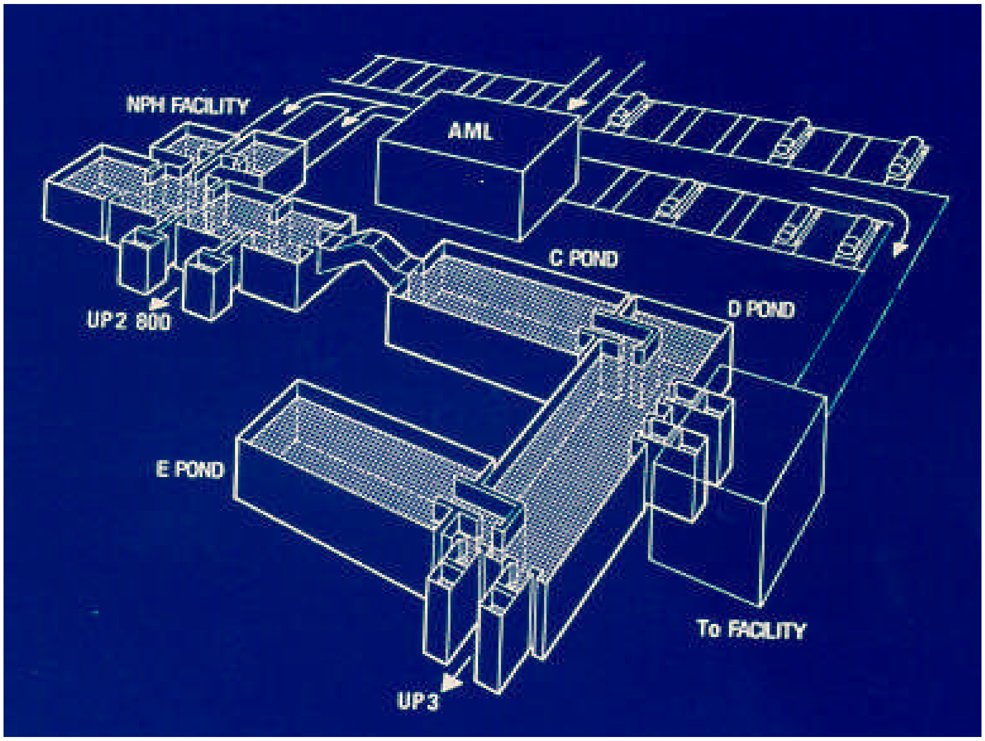



**FIGURE 37: LA HAGUE SPENT FUEL STORAGE COMPLEX**

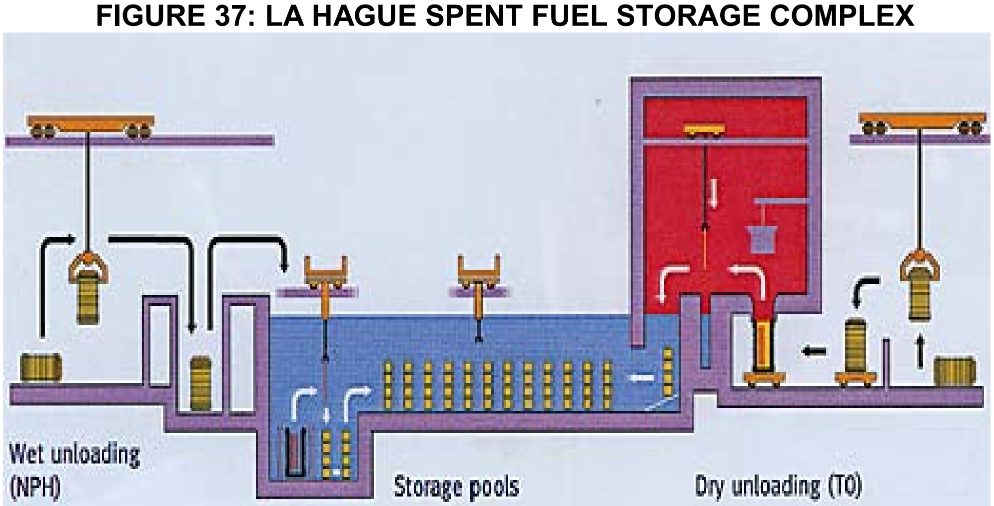



**FIGURE 38: GENERAL PRINCIPLES OF UNLOADING OPERATIONS AT LA HAGUE**

Incoming transport containers are monitored to assess the external dose rate and to detect possible surface contamination prior to transfer to the unloading facilities. Transport container maintenance and decontamination facilities are also provided.

**FIGURE 39: TRANSPORT CONTAINER INTERIM STORAGE**

#### 9.3.1.1.1 Dry Unloading Technology

This technology has been implemented at the completely automated T0 facility at the La Hague reprocessing plant. This facility places each incoming spent fuel transport container onto a self-driven trolley and the transport container is prepared for unloading. Once prepared, the transport container is transferred and docked underneath the fuel assembly unloading cell.

**FIGURE 40: TRANSPORT CONTAINER PREPARATION OPERATIONS**

**FIGURE 41: TRANSPORT CONTAINER DOCKING SYSTEM**

The T0 facility transport container docking system connects the transport container internal cavity to the unloading cell. The connection ensures that the fuel assemblies remained isolated from the external atmosphere and prevents the spread of any contamination. The fuel assemblies are automatically removed from the transport container by handling systems inside the unloading cell.

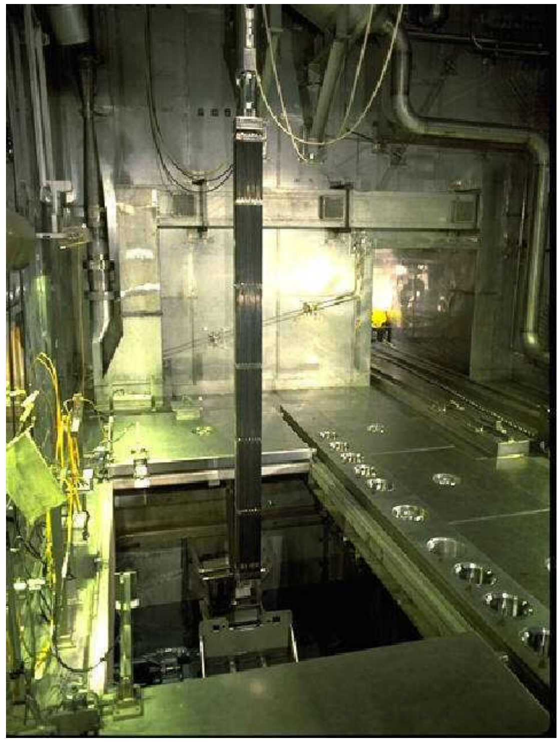


**FIGURE 42: FUEL ASSEMBLY REMOVED INTO UNLOADING CELL**

The fuel assemblies are then transferred and immersed in a cooling/rinsing enclosure, where a sipping test is performed to check the integrity of the fuel rods. The fuel assemblies are then transferred to the interim storage facility.

The empty transport container is monitored for residual radioactivity and closed for transfer to the preparation station where it is cleaned, rinsed and vacuum-dried and prepared for reuse.

### 9.3.1.1.2 Underwater Unloading Technology

An underwater unloading facility has been installed and successfully operated at AREVA's La Hague facility and similar facilities have been built elsewhere. Commissioned in 1981, this facility has a pool in which the transport container is fully immersed prior to unloading. It is important to note that all operations are completed under water. The transport container lid is opened and the transport container is prepared for unloading. The fuel assemblies are unloaded one at a time and placed into storage baskets. A submerged conveyor transfers the storage baskets from the unloading pool to one of the interim storage pools.

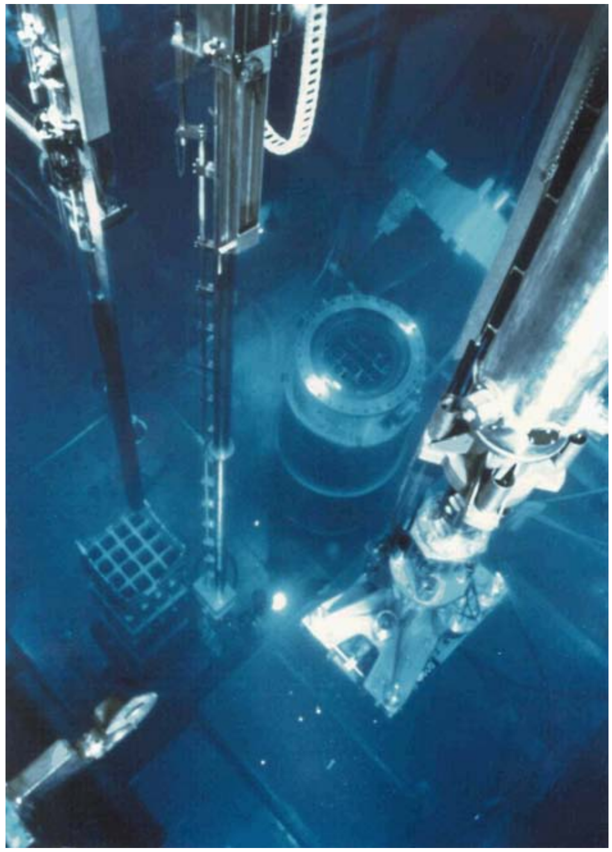


FIGURE 43: TRANSPORT CONTAINER IMMERSED IN NEW POOLS HAGUE FACILITY

FIGURE 44: BASKET TRANSFER BY SUBMERGED CONVEYOR

#### 9.3.1.1.3 Comparison of Dry and Underwater Unloading Technologies

The principal distinguishing features of the two options are compared in the following table.

**TABLE 40: MAIN ADVANTAGES AND DISADVANTAGES OF UNLOADING OPTIONS**

| | Dry unloading | Underwater unloading |
|---|---|---|
| **Safety and radiological issues** | Height differences in the unloading sequence are smaller, leading to a reduced risk from dropped loads. | |
| | Operator doses tend to be lower when operations are undertaken in an active cell. Dry unloading is a faster operation leading to a lower exposure time. | The need for additional decontamination and the generation of additional liquid waste will lead to increased operator doses. |
| **Environmental issues** | "Dry" unloading does not entirely remove the need for water as there is still a need for sipping tests and for cooling/rinsing of the transport container. | This option requires the use of large volumes of water leading to increased secondary waste management issues. |
| | There is a reduced need for transport container decontamination, leading to a reduction in associated wastes. | |
| **Performance issues** | Transport container unloading is much quicker as there are fewer handling and decontamination operations. | The existing underwater facilities are more flexible and can handle a range of transport container types. Although this should not be a significant issue as transport containers should be of a single design. |
| | | The need for an unloading pool and for effluent management facilities leads to increased financial burden for outlay and operations. |

Depending on specific requirements, either wet or dry unloading technology can be selected. For the purpose of this report, dry unloading is the selected option as it presents much smaller secondary waste arisings and the reduced exposure to operators. This solution also provides adequate flexibility and ease of handling of different spent fuel packages.

### 9.3.1.2 Overview of Spent Fuel Storage Technologies

After the initial cooling phase an interim storage period is required before spent fuel can be disposed. This interim storage facility may be located on site as defined by the base case or off the reactor site, serving several nuclear power plants.

The storage facility can involve either dry or wet technology, and will be designed to meet the following requirements:

- To ensure safe operations (e.g. by preventing a criticality incident and maintaining effective containment);
- To provide radiological protection to public, workers and the environment at all times in compliance with dose limits and ensuring that all doses are As Low As Reasonably Practicable;
- To ensure thermal cooling to maintain spent fuel integrity;
- To facilitate fuel assembly monitoring and retrievability;
- To maintain spent fuel in a condition appropriate for final disposal.

Five interim storage solutions (based on AREVA proven technologies) are discussed in detail below:

- Underwater storage;
- Dry storage in a metal flask;
- Dry storage in concrete storage modules (e.g.Nuhoms);
- Dry storage in TN NOVA (Nuhoms evolution);
- Dry storage in a vault.

Among these 5 options, one wet and two dry solutions have been identified and assessed for the UK EPR. They are described in chapter 13.

#### 9.3.1.2.1 Spent fuel Underwater Storage

**Underwater Spent Fuel Storage in GÖSGEN**

A further example of spent fuel storage in a pool is the GÖSGEN facility located on the GÖSGEN nuclear power plant in Switzerland.

This Wet Spent Fuel Storage Facility can accommodate both uranium and MOX Fuel assemblies and is an independent extension of the existing spent fuel storage reactor pool.

One of the main characteristics of this new storage facility is the almost exclusive use of passive cooling for the pool, based on natural air convection, which means that no active components are implemented (such as pumps).

In its final configuration, the wet storage facility accommodates up to 1000 irradiated fuel assemblies stored in racks.

The facility consists of the storage building, an adjacent service structure and two cooling towers. The storage building houses the fuel assemblies storage pool and the cask pool: it is a cubical, steel reinforced concrete structure of 17 m X 35 m X 25 m total height.

The service structure is limited since the proximity of the existing plant allows the use of existing systems, such as HVAC.

Structures are designed considering not only the operational loads and the weather conditions (e.g. the wind and snow loads), but also hazards of external origin (earthquake, flooding, aircraft crash, explosion).

All of the storage equipment is such that – in normal operating conditions as well as in degraded operation or in incident situations:

- subcriticality is ensured at all times;
- decay heat is removed;
- no unacceptable radiation exposure may occur for the public or the personnel.

The fuel elements placed in standard transport containers are transferred from the power plant to the interim storage facility on a heavy vehicle running on a rail-track. Transport containers are compatible with either wet or dry transport conditions and arrive in the horizontal position in the wet storage facility.

In the reception area, the transport container is upended and transferred first to the floor of the storage pool through a hatch in the ceiling and then into the container pool. The fuel elements are unloaded using a special handling device operated from a crane.

In the pool the fuel elements are placed in compact storage racks. They are cooled by natural convection of the pool and by water circulated from the bottom through the racks.

Criticality safety is always ensured by appropriately arranging the fuel elements in the racks, i.e. by suitably spacing them out and by installing neutron absorbers in the storage racks.

Pool water cleaning systems are built into the pool to ensure good visibility for the operators when handling the fuel.

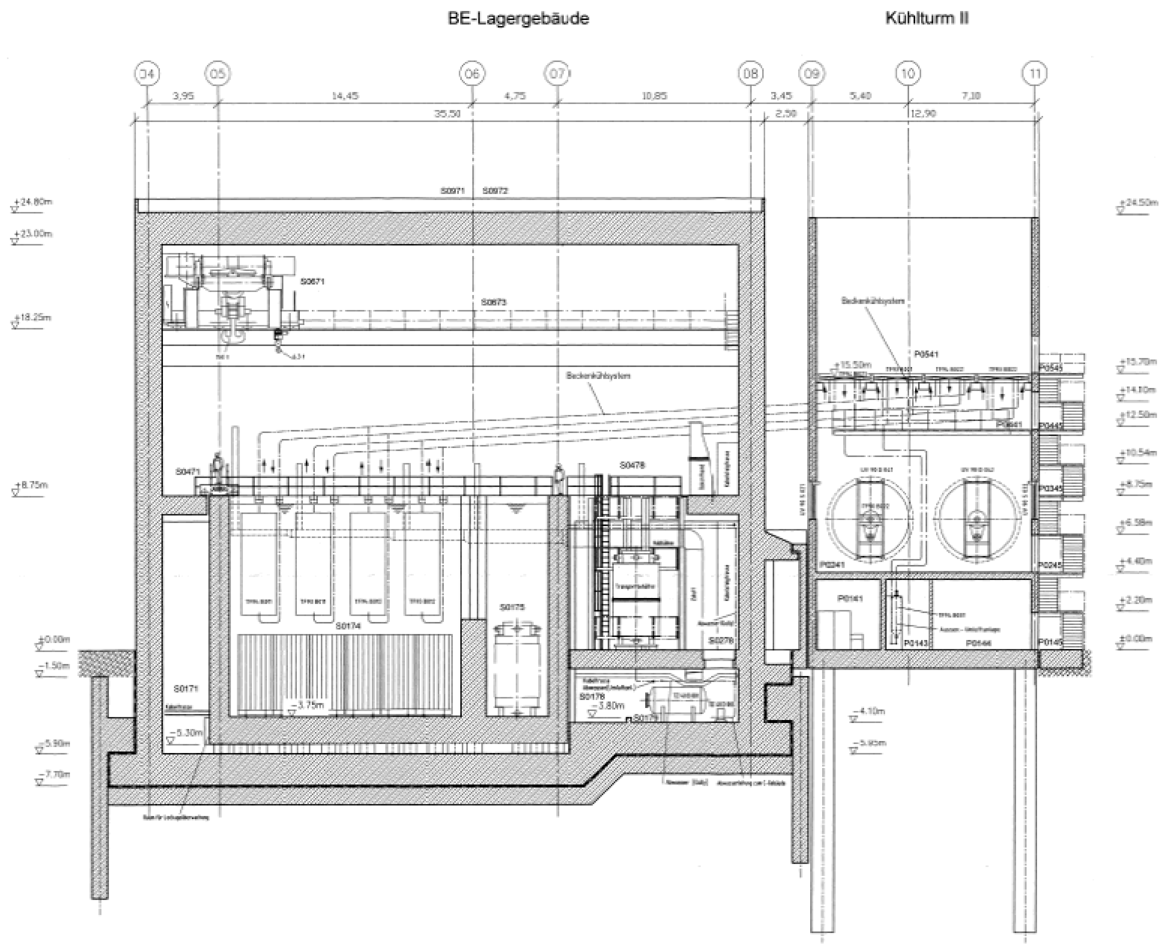



**FIGURE 45: CROSS-SECTION OF THE GOSGEN FUEL STORAGE BUILDING AND COOLING TOWERS**

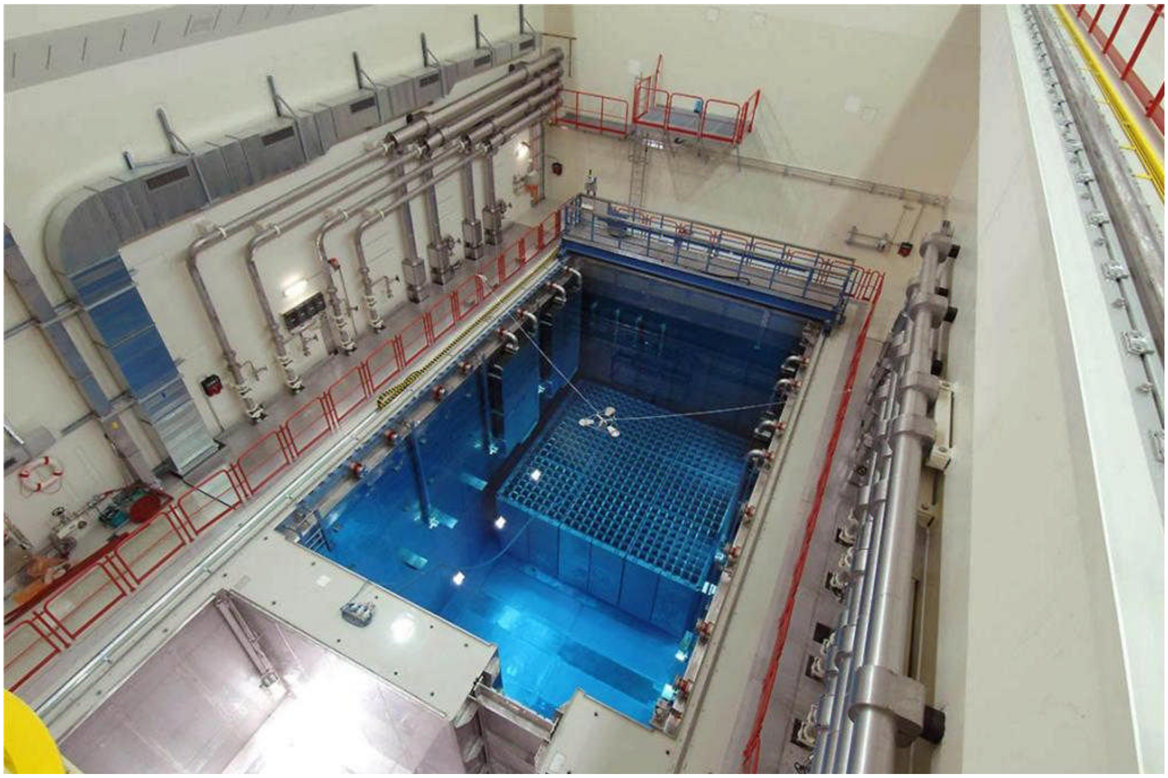


**FIGURE 46: FUEL POOL AND COOLING SYSTEM AT GOSGEN**

**Underwater Spent Fuel Storage in La Hague**

AREVA's facility at La Hague stores spent nuclear fuel underwater in 4 interconnected storage pools with a total capacity of 14,000 tU (tonnes of uranium). These facilities have been operational for over 20 years and consist of:

- An underwater unloading facility (as discussed above) connected to a pool with a storage capacity of 2,000 tU. This facility supplies spent fuel to the reprocessing plant;
- The 'C' pool with a capacity of 3,600 tU;
- The 'D' pool, connected to the T0 dry unloading facility (as discussed above) and designed for a 3,500 tU capacity. The D pool also supplies spent fuel for the reprocessing plant;
- The 'E' pool with a storage capacity of 4,900 tU.

These storage pools are designed to allow flexible management of spent fuel assembly storage to meet the needs of the reprocessing operations while maximising safety.

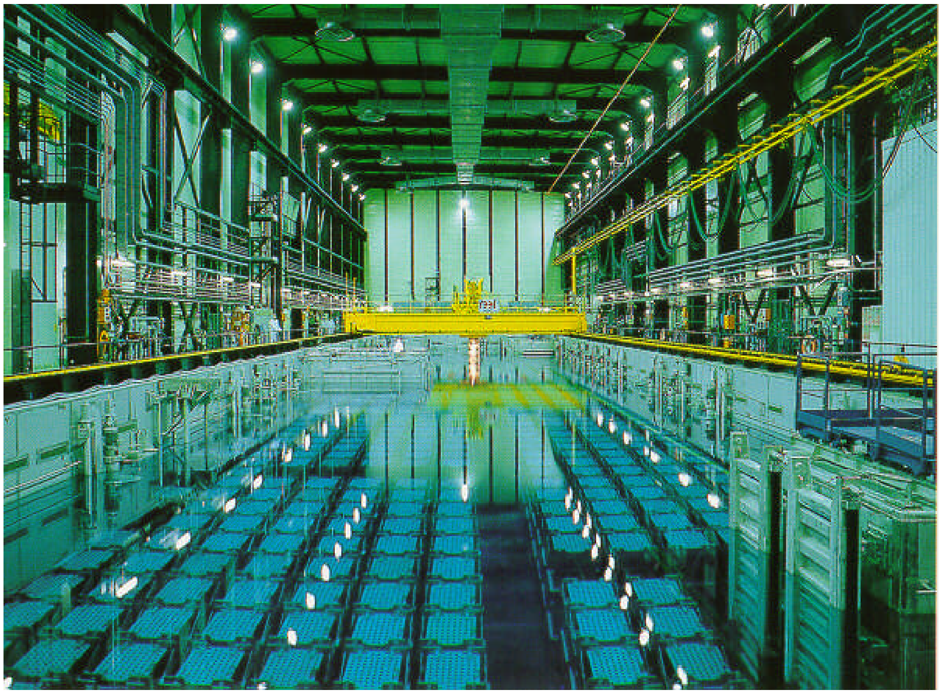
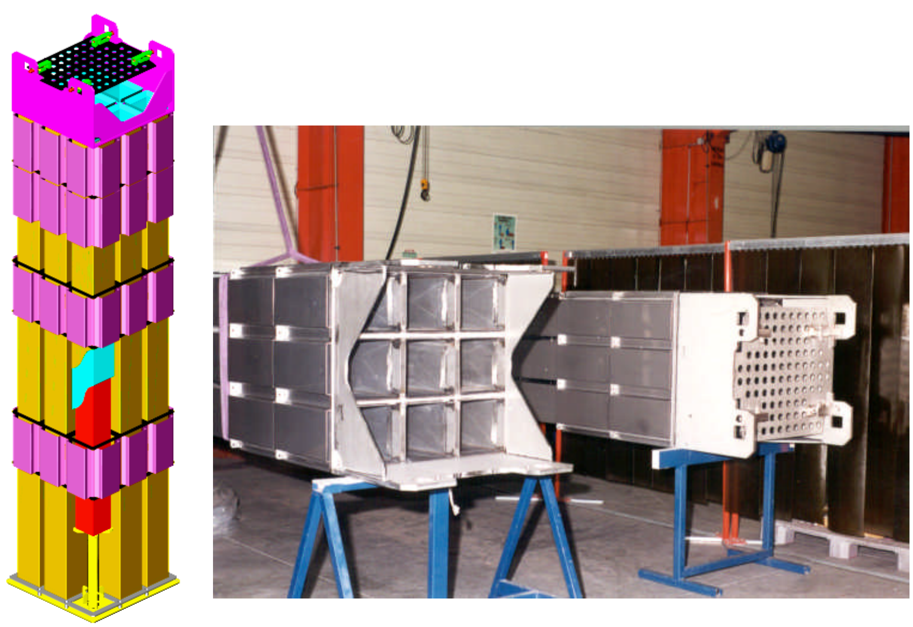


**FIGURE 47: THE D STORAGE POOL**

**FIGURE 48: PWR STORAGE BASKET**



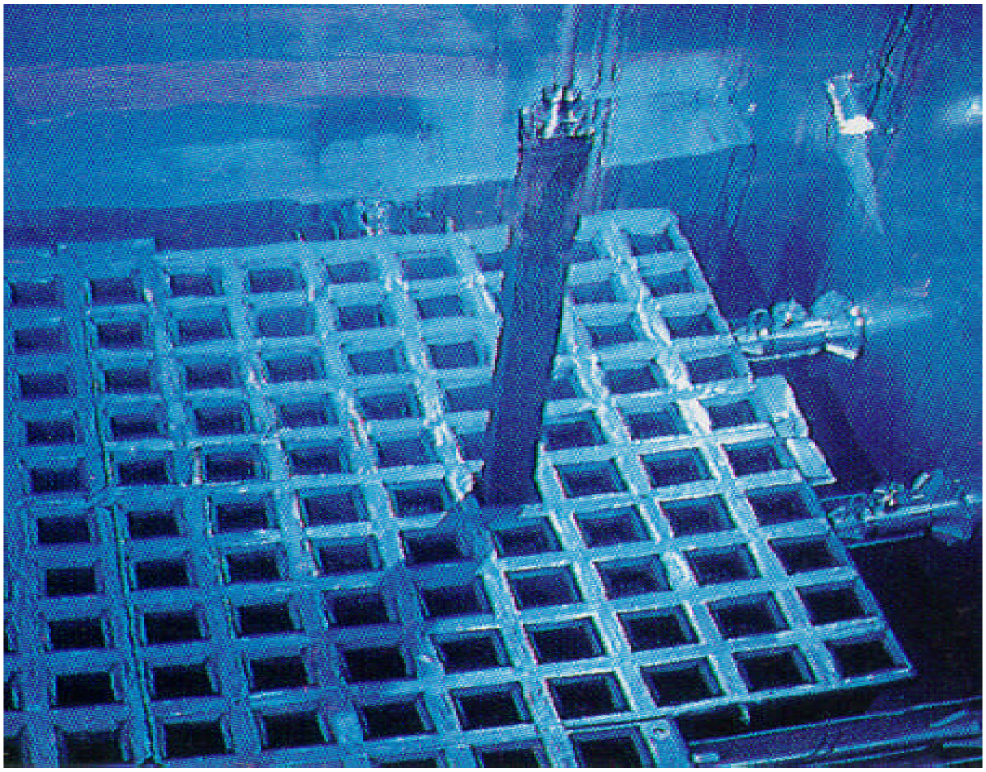


**FIGURE 49:FUEL ASSEMBLY STORED IN RACK IN REACTOR POOL**

The storage pools are isolated from the environment by impermeable barriers (e.g. stainless steel liner) and by a ventilation system incorporating high efficiency particulate in air (HEPA) filters. Additional systems are provided to detect any leaks during pool operation. The pools are protected against earthquake damage by neoprene pads which support the pool and isolate it from the building structures.

The build-up of heat and radiochemical contaminants within the pool is prevented by the incorporation of heat exchangers and ion exchangers. Both of these are submerged in the pool and together they ensure that optimum conditions are maintained. Locating these facilities within the pools also minimises the infrastructure requirements and the risk of leaks through external pipework.

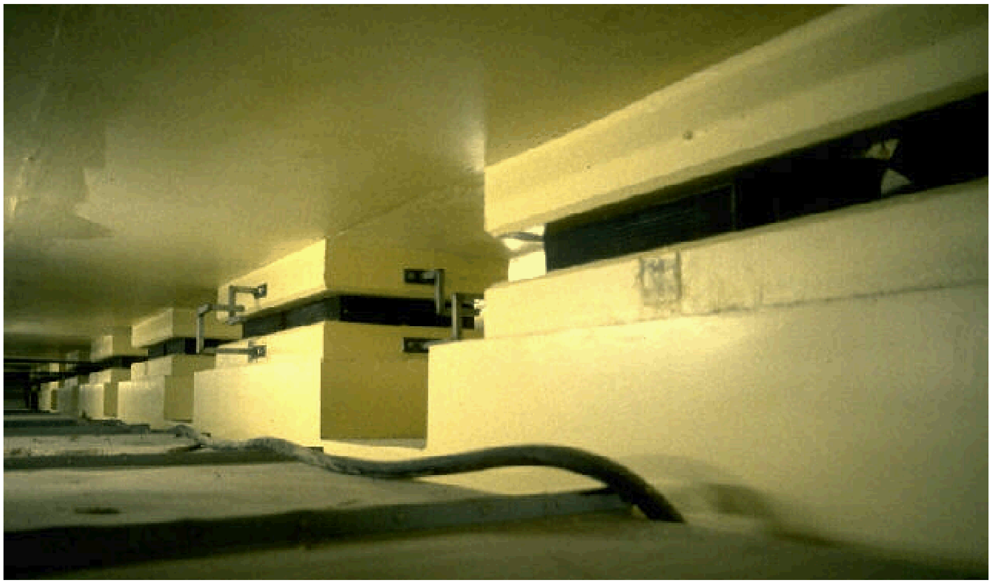


FIGURE 50: POOL SUPPORTS

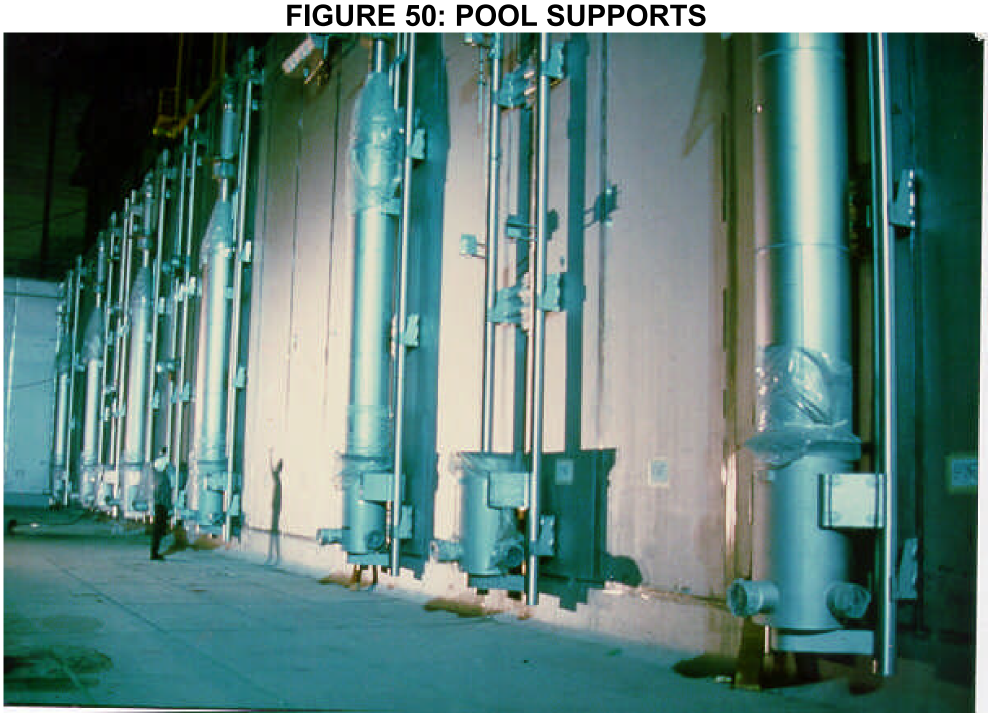


FIGURE 51: SUBMERGED HEAT AND ION EXCHANGERS

### 9.3.1.2.2 Spent Fuel Dry Storage

A number of dry storage technologies have been developed in the past 15 years. This report describes four potential options that have been identified for possible use in the UK EPR however, as technology in this area develops new storage products may become available and be considered for use in the EPR. The four potential options described in this report are as follows:

- Dry storage in a metal flask;
- Dry shielded canisters in concrete storage modules (e.g.NUHOMS);
- Dry shielded canisters in vertical storage modules (TN NOVA);
- Dry storage in a vault.

These dry storage options are discussed in more detail below.

**Dry Storage in a Metal Flask**

The metal flask is a dual purpose technology for both spent fuel transport and interim storage. This technology is used in Germany, Switzerland and Belgium.

The steel containment vessel provides the main gamma shielding and an external layer of resin covered by an outer steel shell provides shielding against neutrons. Criticality is controlled by an internal basket made of boronated alloys.

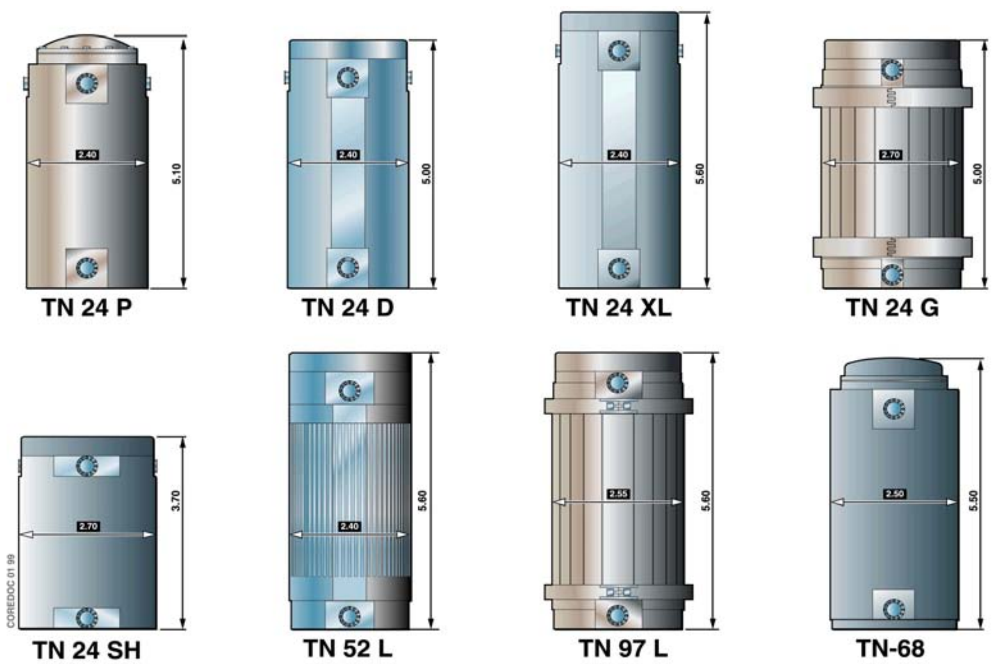



**FIGURE 52: DUAL PURPOSE STORAGE/TRANSPORT METAL FLASKS**

**Dry storage in concrete storage modules (e.g. NUHOMS® )**

NUHOMS® is a proven system for dry storage which has been in use at reactor sites since March 1989. The NUHOMS® system provides containment, shielding, criticality control and passive heat removal independent of any other facility structures or components.

NUHOMS (NUTECH Horizontal Modular Storage) technology involves:

- A Dry Shielded Canister (DSC) is inserted inside a site transfer cask;
- Placing spent fuel in a Dry Shielded Canister (DSC) underwater;
- The DSC is drained, sealed, vacuum-dried and filled with helium (line removed);
- The transfer cask is placed in a horizontal position on a transport trailer and moved to the HSM site;
- The cask including the loaded DSC is used to transport the spent fuel to the interim storage facility;
- The site transfer cask is aligned with a Horizontal Storage Module (HSM). The HSMs are massive concrete structures and provide shielding and security for safe interim storage;
- The DSC is pushed out of the site transfer cask and into the HSM;
- The HSM is sealed and the site transfer cask reused to transfer other DSCs.

The HSM provides radiological shielding and physical protection for the Canister against a wide range of postulated natural hazards. The Module has internal air flow passages to provide natural convection cooling for removal of decay heat from the Canister’s spent fuel.

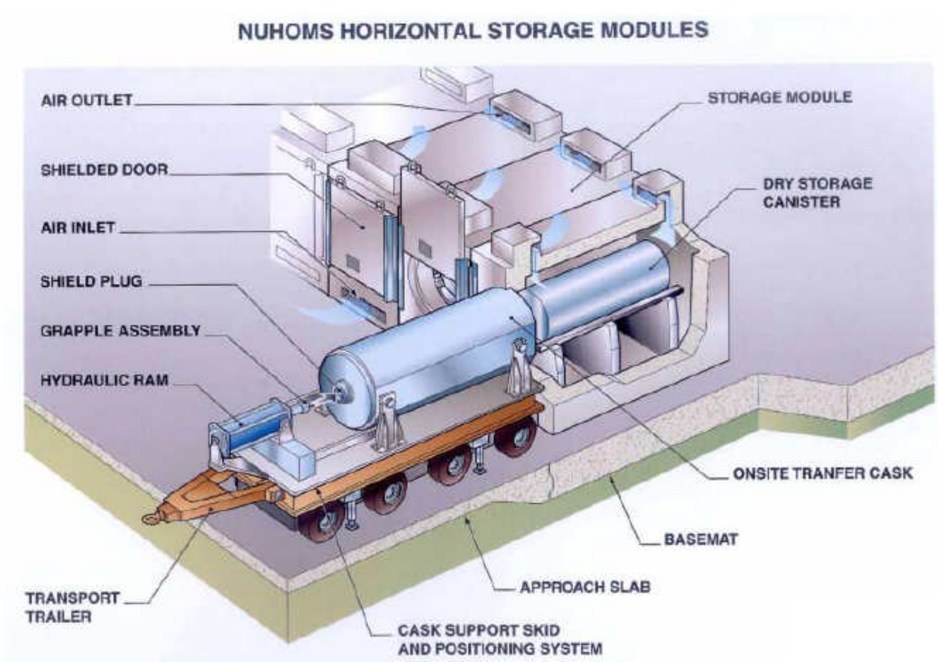



**FIGURE 53: THE NUHOMS SYSTEM**

**FIGURE 54: HORIZONTAL STORAGE MODULES**

**Dry storage in TN NOVA (vertical storage modules)**

The TN NOVA system is evolved from the NUHOMS® canister system, It differs with the NUHOMS® system only by a vertical metallic storage overpack instead of the horizontal concrete module which surrounds the canister in the storage configuration. At the storage facility location, the loaded transfer cask is aligned with the TN NOVA overpack and the DSC canister is then transferred to the overpack in a horizontal position. Once the transfer is complete, the TN NOVA overpack is placed upright in a vertical position for storage. Once inside the TN NOVA overpack, the DSC is in a safe, passive dry storage configuration.

The TN NOVA overpack provides radiological shielding and physical protection for the Canister. The TN NOVA overpack is constructed primarily of carbon steel. It has internal air flow passages to provide natural convection cooling for removal of decay heat from the Canister's spent fuel.

The overpack is well shielded to keep doses ALARA. Additionally, the TN NOVA provides aircraft impact and earthquake resistance.

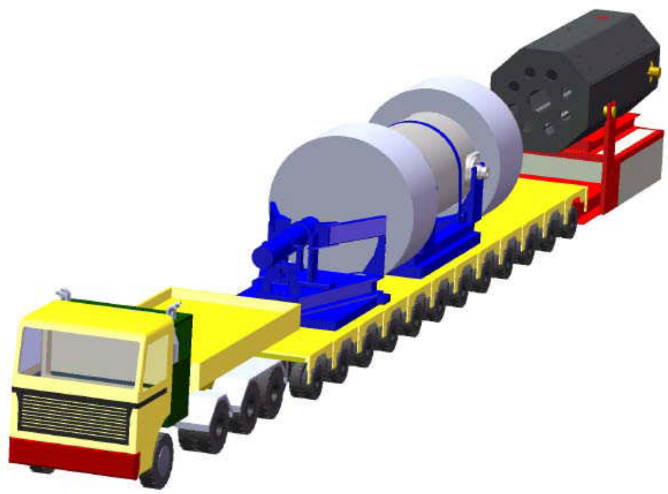


FIGURE 55: TRANSFER TRAILER WITH CASK SUPPORT SKID

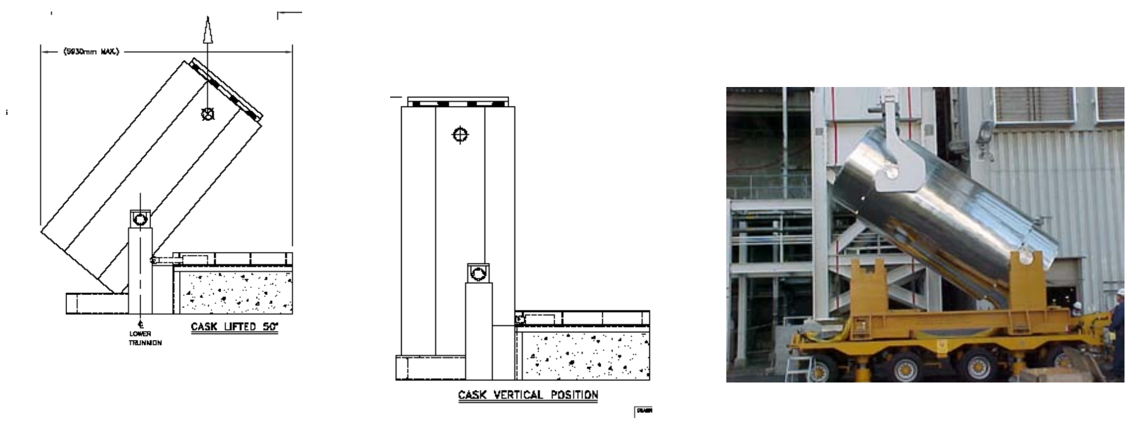



FIGURE 56: TN NOVA UPRIGHTING OPERATIONS

### Spent Fuel Dry Storage in Vault

The storage of spent fuel in a dry vault ensures safe and sustainable storage conditions. It involves the sealing of spent fuel in canisters and the placement of those canisters in storage wells set into a reinforced concrete slab. The operations within the facility include:

- Transport container receipt and preparation for unloading;
- Dry unloading of fuel assemblies and placement into canisters;
- Canister welding and transfer to the storage vault;
- Placement of the canister into the storage well and well closure.

The concrete structure of the vault ensures the protection of fuel assemblies from external events and protects workers and local residents from irradiation. The sealed canisters and the closed wells prevent radioactive releases to the environment.

In France, the CASCAD facility, designed for 50 years of interim storage, was the first to use natural convection to cool the storage wells where the canisters containing spent fuel are kept.


**FIGURE 57:CASCAD DRY STORAGE FACILITY: CANISTER LOADING**

This technology is also used in the Netherlands in COVRA's Habog facility (see Figure 58). This facility accommodates not only fuel canisters but also glass canisters and intermediate level waste containers.

**FIGURE 58:GENERAL VIEW OF COVRA HABOG FACILITY, NETHERLANDS**

A multipurpose interim storage facility using vaults has been designed for ENRESA, in Spain. This facility provides storage for spent fuel (with characteristics similar to EPR fuel) in vaults cooled by natural air ventilation and also provides separate storage for solid waste stored in bunkers cooled by forced air ventilation. This type of multipurpose dry vault storage system offers a number of significant advantages including:

- Multiple containment systems, with the sealed canister, the closed storage well and the building ventilation system all serving to prevent radioactive releases to the environment;
- Safe and secure containment of spent fuel within the canister during any necessary handling operations. The canister also incorporates features to facilitate secure movement by overhead crane, thereby minimising the risk of a dropped load;
- Passive cooling of stored spent fuel assemblies by natural air convection. The storage vault temperature is kept low so that the cladding temperature can be maintained under 380°C, even under extreme conditions.380°C, even under extreme conditions.

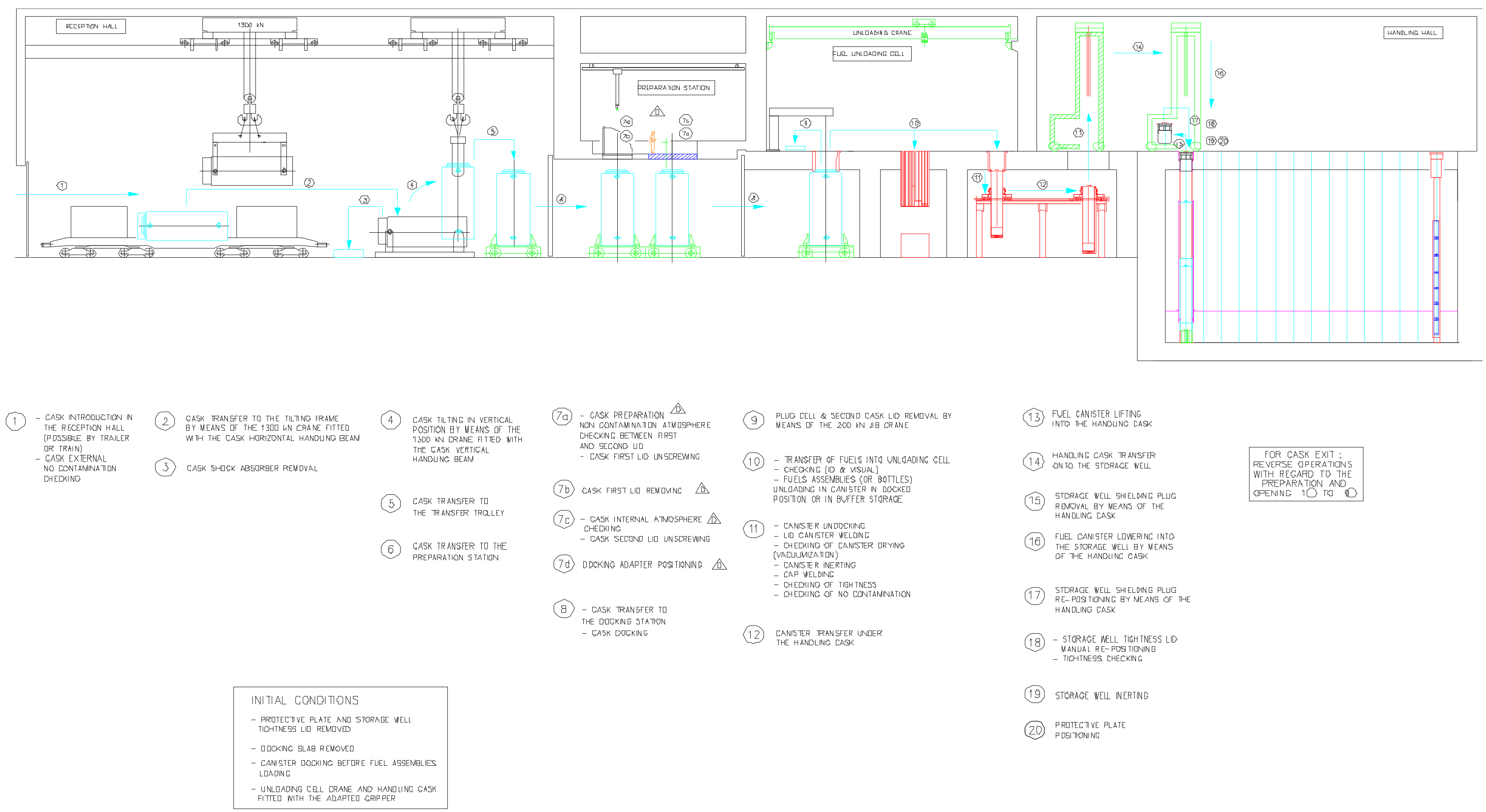



Figure 59: General Operating Diagram for a Standard Vault Type Facility

#### 9.3.1.2.3 Comparison between Wet and Dry Spent fuel Interim Storage Technologies

The dry interim storage technologies have a number of advantages with regards to operational flexibility. These include:

- The dry storage system uses passive cooling systems;
- The modular design of a dry interim storage system means that the storage facility can start operating with a reduced number of modules and additional modules can be built according to the need;
- In the case of dual purpose casks the spent fuel is conditioned for transport off-site without any additional constraints for retrievability;
- Spent fuel assemblies are stored in sealed canisters offering a high level of containment;
- The fuel storage cask or the vault storage building can be designed to withstand external hazards such as earthquake, aircraft crash, explosion;
- The need for secondary waste treatment, particularly for contaminated water, is significantly reduced;
- Decommissioning of a dry store is much easier and there is less equipment to dismantle, decontaminate and dispose of.

It should be noted however that dry interim storage is less flexible to accommodate spent fuel with very high heat dissipation and unforseen HLW with dimensions greater than the facility can handle.

Underwater storage has a number of advantages with regard to technical performance. These include:

- It is well adapted to high heat production fuel and so the initial cooling period at the reactor pool can be reduced;
- There is a large amount of experience in the design and safe operation of these facilities as all PWR operators use them;
- Pools can be designed to withstand external hazards (e.g. earthquake, aircraft crash, explosion) and internal hazards (e.g. incorporation of a leakage system to reduce water loss and maintain watertight facility);
- It simplifies the management of leaking spent fuel;
- Its main concern is the necessity to have active systems to permanently maintain water cooling and to keep water depth sufficient to provide shielding and ensure safe, long term storage of all types of fuel assemblies.

A drawback of underwater storage is the production of quantities of liquid and solid secondary waste (resins, etc.).

#### 9.3.1.2.4 Different scenarios for Spent Fuel interim storage

The different scenarios envisaged for spent fuel interim storage can be adapted to:

- A facility at an individual reactor site for a single NPP;
- A facility at a reactor site dedicated to several NPPs;
- A facility located off-site but providing interim storage facilities for several NPPs.

Locating an interim storage facility on each reactor site would minimise the risks associated with handling and transporting the spent fuel. However, a storage facility serving several reactor sites would require less plant and equipment overall. For the purpose of this study we have considered the design of facilities for a single unit and multi units.

## 9.4 Spent Fuel Disposal

According to base case requirements, direct spent fuel disposal is the ultimate solution to be considered. Therefore, after on-site storage, spent fuel is evacuated to the disposal site where it will be buried in deep underground disposal.

Whatever the previously described interim storage technology selected, key features are that:

- They keep the fuel integrity by maintaining appropriate storage conditions (maintain inert atmosphere, maintain fuel temperature below acceptable limits);
- They allow for interim store retrievability and the eventual loading in a flask for final transportation to the disposal site. (Note that use of the dual purpose cask avoids rehandling of spent fuel before ultimate transport).

The generally agreed approach on safety of deep geological facilities relies on 3 barriers to prevent any impact on the environment. These are as follows:

1 The waste package.

At the disposal site, an Encapsulation plant will be required to allow the spent fuel to be conditioned in canisters. Different configurations of canister are today envisaged in different countries. They differ depending on their requirements and characteristics. As an example, the SKB design (in Sweden) is based on a robust and thick canister made of stainlees steel and copper, to accommodate up to 4 fuel assemblies. In USA, a thinner canister, holding up to 21 Fuel Assemblies is foreseen. The main functions of this Encapsulation facility are similar to that of the receiving building of the interim store (Flask reception and unloading, conditioning of spent fuel in canisters). Differences are on the canister /package characteristics and the transfer to the Geological Disposal Facility.

2 The repository engineered barrier.

The repository engineered barriers depend on design of the repository. Several approaches are possible (horizontal or vertical emplacements of packages, single package or groups of packages). A key aspect is related to the ability to retrieve or not the packages, at least during the operational phase which can last several decades.

3 The host rock.

Characteristics of the host rock (granite, clay, …), the presence of water, acceptable heat load, etc generate constraints and requirements on the package criteria. A key driver of the overall repository design is the host rock maximum allowable temperature.

Most of the disposal requirements are not yet known, but both UK EPR spent fuel characteristics and interim storage facility design do not compromise the disposability of spent fuel in the future. The following figures illustrate the currently envisaged disposal facilities in USA, Sweden and Finland and France.

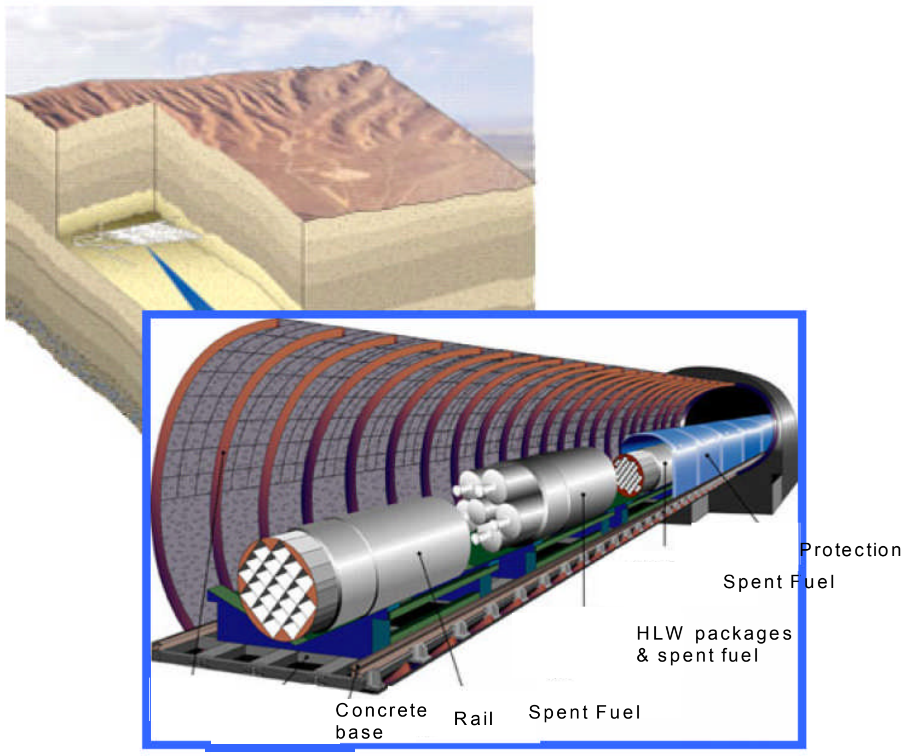



**FIGURE 60: DISPOSAL CONCEPT IN USA (YUCCA MOUNTAIN PROJECT)**

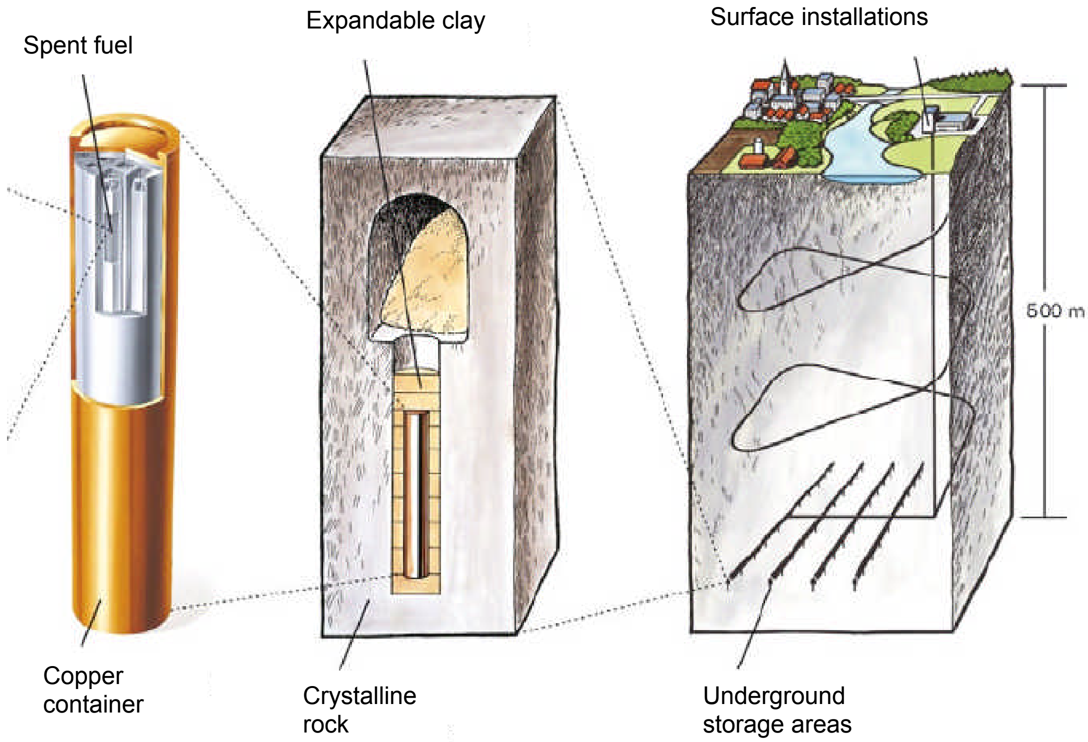



**FIGURE 61: KBS-3V DISPOSAL CONCEPT IN SWEDEN AND FINLAND ( VERTICAL OPTION)**

In France, for the studies performed by ANDRA for the “Dossier Argile” relating to a clay formation, 2 types of spent fuel packages have been determined to comply with host rock temperature requirements:

- A large diameter (1250 mm) package with 4 fuel assemblies. The maximum heat load of the package is 1600 W;
- A small diameter (600mm) package, limited to a single fuel assembly, with a maximum heat load of 1100 W.

These 2 packages are illustrated hereafter (see Figure 62 and Figure 63).

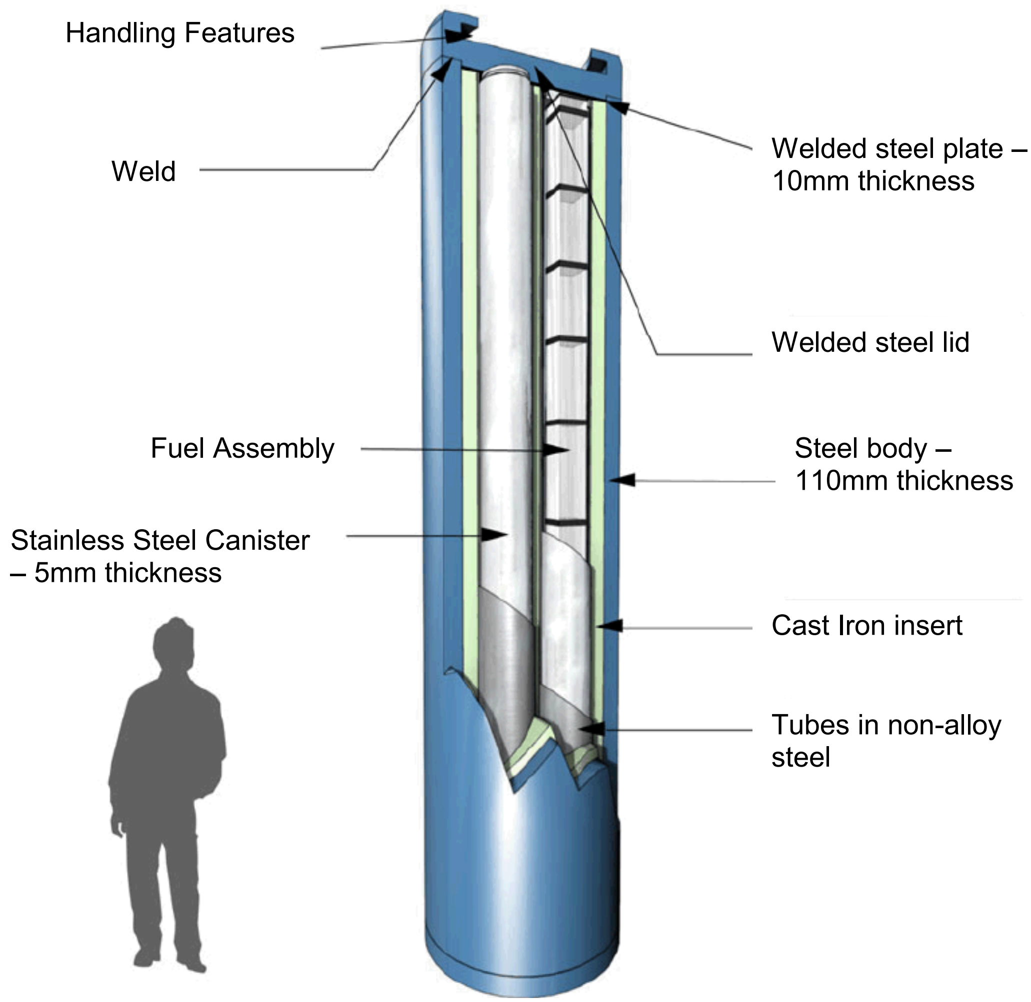



**FIGURE 62: DISPOSAL CONTAINER IN FRANCE (EXAMPLE WITH 4 SFA)**

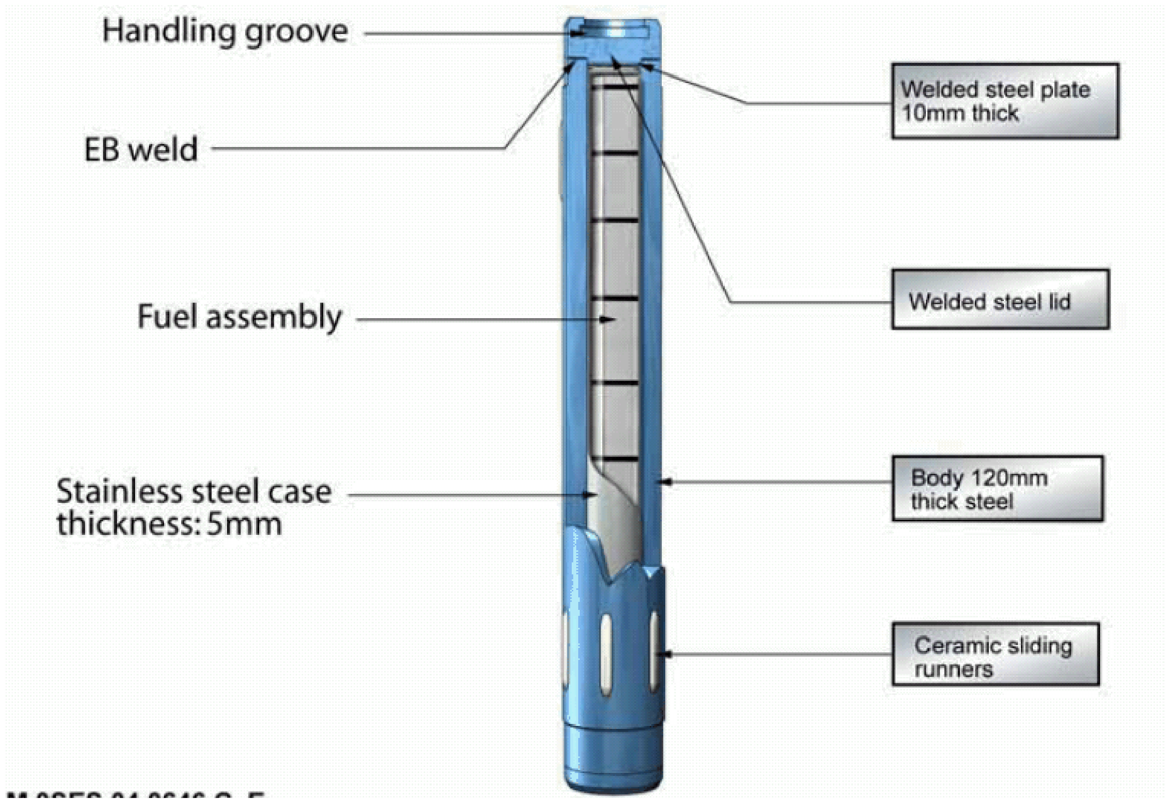



**FIGURE 63: SMALL DIAMETER SPENT FUEL DISPOSAL PACKAGE (ONE SFA)**

# 10 REGULATORY AND SAFETY OVERVIEW OF WASTE TREATMENT AND INTERIM STORAGE FACILITIES

## 10.1 Regulatory Baseline

This section describes the UK regulatory requirements that apply to operations occurring in the Waste Treatment Building (WTB) and the Interim Storage Facility (ISF). In the UK the regulator responsible for overseeing the safety of operations on a nuclear licensed site is the Health Safety Executives (HSE's) Nuclear Installations Inspectorate (NII).
The NII has four fundamental expectations with regard to radioactive waste. AREVA has taken these expectations into account in the development of this waste strategy. These expectations are listed below in regard to this strategy [Ref. 48]:

1. Production of radioactive waste will be avoided. Where radioactive waste is unavoidable, its production will be minimised;
2. Radioactive material and radioactive waste will be managed safely throughout its life cycle in a manner that is consistent with modern standards;
3. Full use will be made of existing routes for the disposal of radioactive waste generated by the EPR;
4. Remaining radioactive material and radioactive waste will be put into a passively safe state for interim storage pending future disposal or other long-term solution.

### 10.1.1 Nuclear Site Licence Conditions

Regulation of radioactive waste will be achieved via compliance with the Nuclear Site Licence and its associated 36 Nuclear Site Licence Conditions many of which are directly applicable to radioactive waste processing operations taking place in the WTB and the ISF. The following Licence Conditions are applicable to the control of nuclear matter (a definition which includes radioactive waste) on licensed sites [Ref. 48]:

Licence Condition 4: Restrictions on nuclear matter on the site.
Licence Condition 6: Documents, records, authorities and certificates.
Licence Condition 11: Emergency arrangements.
Licence Condition 14: Safety Documentation.
Licence Condition 15: Periodic review.
Licence Condition 17: Quality Assurance.
Licence Condition 23: Operating rules.
Licence Condition 25: Operating records.
Licence Condition 26: Control and supervision of operations.
Licence Condition 28: Examination Maintenance Inspection and Testing.
Licence Condition 32: Accumulation of radioactive waste.
Licence Condition 33: Disposal of radioactive waste.
Licence Condition 34: Leakage and escape of radioactive material and radioactive waste.
Licence Condition 35: Decommissioning.
The Licence Conditions and their applicability to operations occurring in the WTB and ISF are discussed in more detail below.

**Licence Condition 4: Restriction on Nuclear Matter on the Site**
This Licence Condition ensures that there are adequate arrangements, (approved by the NII) to control the introduction and storage of nuclear material on a nuclear licensed site. The Licence Condition specifies that any alterations to these arrangements must be approved by the NII and provides the power for the NII to specify that certain types of nuclear material cannot be bought on site without consent. This will allow the NII to confirm the adequacy AREVA's arrangements prior to the material being introduced onto site for the first time. To demonstrate compliance with this Licence Condition, AREVA will therefore produce arrangements for all buildings handling and storing nuclear material including the WTB and ISF.

**Licence Condition 6: Documents, Records, Authorities and Certificates**
This condition requires that adequate records be maintained for a period of 30 years or, such other time period as the NII may approve, to demonstrate compliance with the Site Licence Conditions. AREVA's arrangements for compliance and any subsequent modifications to the arrangements will require approval from the NII. With respect to WTB and ISF operations the arrangements will include all records relating to the generation, treatment, conditioning and storage of waste including isotopic information, monitoring records, storage records, transport, receipt and dispatch records.

**Licence Condition 11: Emergency Arrangements**
The purpose of the Licence Condition is to ensure that there are adequate arrangements in place for emergency situations. AREVA will meet this requirement by developing and implementing the measures necessary to respool effectively to any incident ranging from a comparatively minor on-site event to a significant release of radioactive material and ensure adequate planning and rehearsal.

With respect to the WTB and ISF operations this Licence Condition requires the identification of reasonably foreseeable accident scenarios by AREVA (this is usually carried out using the risk assessment presented within the safety case for these operations) and the development and rehearsal of appropriate contingency plans to respond to these incidents when the site is commissioned.

**Licence Condition 14: Safety Documentation**
This Licence Condition requires that AREVA establish arrangements for the preparation and assessment of safety cases. A safety case is required to justify the safety of all operations occurring on a nuclear licensed site. AREVA will develop a safety case for those operations in the WTB and ISF to fulfil this requirement.

**Licence Condition 15: Periodic Review**
This Licence Condition requires that the plant remains adequately safe and that the safety cases are kept up to date throughout the lifetime of the plant. In the UK this process is generally referred to as a Periodic Review of Safety (PRS). The PRS reviews the safety case against modern standards and identifies reasonably practicable improvements to demonstrate that the plant is safe to continue to operate for the next defined period (usually ten years). The WTB and ISF alongside the rest of the EPR plant will be subject to the PRS process required under this Licence Condition.

**Licence Condition 17: Quality Assurance**
This Licence Condition requires that licensees make and implement adequate quality assurance arrangements in respect of all matters that affect safety. This has been implemented as part of

the EPR design. High standards of QA will continue to be applied throughout the lifetime of the installation and in the preparation and review of safety documentation.

Adequate QA arrangements will be used for all operations in the WTB and ISF including receipt, monitoring, buffer storage, treatment conditioning, storage and disposal/ transfer to an Off-site treatment location. The application of an appropriate QA system is particularly important for record keeping and will ensure that appropriate records are generated and maintained for all wastes processed and held in the WTB and ISF.

**Licence Condition 23: Operating Rules**

This Licence Condition requires that all operations that may affect safety are supported by a safety case (as Licence Condition 14) and that the safety case identifies the limits and conditions (operating rules) within which the plant can safely operate. The output of the safety case production process for the WTB and ISF will identify a set of Operating Rules for the processes occurring within the buildings.

**Licence Condition 26: Control and Supervision of Operations**

The purpose of the Licence Condition is to ensure that safety related operations (as identified from the assessments presented in the safety case) are only carried out under the control and supervision of Suitably Qualified and Experienced Persons (SQEP).

**Licence Condition 28: Examination, Maintenance, Inspection and Testing**

This Licence Condition ensures that all plant that may affect the safety of operations is scheduled to receive regular and systematic examination, maintenance, inspection and testing (EMIT) and this should be carried out by SQEPs under the control of a SQEP. In addition the Licence Condition requires that records of any work that has been carried out are generated and maintained. This Licence Condition will be applicable to any equipment that has the potential to affect safety in the WTB and the ISF and has been taken into account as part of the GDA submissions.

**Licence Condition 32: Accumulation of Radioactive Waste**

This Licence Condition is described earlier in this report. The condition requires the development and implementation of adequate arrangements for minimising so far as is reasonably practicable the rate of production and total quantity of waste accumulated on the site at any time and for maintaining records of accumulated waste. This Licence Condition is of particular relevance to the operations that will take place in the WTB and the ISF and is addressed in this document.

**Licence Condition 33: Disposal of Radioactive Waste**

This Licence Condition is reproduced in an earlier section of this report. The Licence Condition provides the NII with the power to direct the disposal of waste. Any disposal of waste is required to be in accordance with a discharge authorisation granted under the Radioactive Substances Act 1993 (RSA 93) (regulated by the Environment Agency). A discharge authorisation under the RSA 93 will be sought for all wastes arising from EPR operation including those generated in the WTB and ISF.

**Licence Condition 34: Leakage and Escape of Radioactive Material and Radioactive Waste**

This Licence Condition is reproduced in full earlier in this report. To meet this Licence Condition AREVA will ensure so far as is reasonably practicable that radioactive material and waste on the site is al all times adequately controlled and contained so that it cannot leak or otherwise escape from the containment and in the event of any fault or accident which results in a leak there are adequate means for detecting the leak and reporting to NII. This Licence Condition is of particular importance to operations in the WTB, which will be the location of the majority of waste treatment and conditioning for the EPR. This principle has been applied in the EPR design development.

**Licence Condition 35: Decommissioning**

This Licence Condition requires that adequate provisions be made for decommissioning and gives discretionary powers to the NII to direct the decommissioning of any plant or process to be commenced or halted. AREVA has incorporated decommissioning requirements into the EPR generic design. The WTB will be the last building in the UK EPR installation to be subject to decommissioning. This has been taken into consideration in the modular design of the WTB which will permit replacement of existing treatment and conditioning equipment with temporary equipment that can be removed from the site once decommissioning of the WTB has been completed.

In addition to the Site Licence Conditions the NII are also responsible for ensuring compliance with the Ionising Radiation Regulations (1999) (IRR99) on a Nuclear Licensed Site. These regulations provide a regulatory framework for ensuring that doses to workers arising from operations are As Low As Reasonably Practicable. The principles and requirements of IRR99 are taken into account in the generic design for the UK EPR.

Further information of the regulatory context to the Management of Radioactive Waste on nuclear licensed sites can be found in the Technical Assessment Guide (TAG) that the NII have produced for use by their inspectors [Ref. 48].

### 10.1.2 Environment Agency Regulatory Requirements

The NII works closely with the Environment Agency (EA) in regulating the generation, treatment and disposal of radioactive waste, under the provisions of RSA 93 [Ref. 10]. This partnership is in accordance with a Memorandum of Understanding that aims to facilitate effective and consistent regulation between the two nuclear regulators. In this context the term disposal includes discharges into the atmosphere, sea, rivers, drains or groundwater, disposal to land and disposals by transfer to another site. The key condition of discharge authorisations for radioactive waste granted under the RSA 93 is the application of Best Practicable Means to minimise the volume and activity of waste produced. This term is being increasingly replaced by the term Best Available Techniques which the Environment Agency believe to be broadly equivalent to the use of Best Practicable Environmental Option (BPEO) and BPM [Ref. 49]. AREVA has taken account of the principles of waste minimisation in developing this waste strategy.

The Environment Agency has recently consulted its inspectors on draft guidance entitled 'Radioactive Substances Regulation Environmental Principles' [Ref. 16]. The guidance consists of a series of fundamental principles for Radioactive Substances Regulation supported by a number of topic and sector developed principles. A key condition of the discharge authorisation under RSA 93 relates to the use of Best Available Techniques (BAT). AREVA will meet these conditions by applying and implementing the following environmental principles [Ref. 16]:

- The best available techniques will be used to ensure that production of radioactive waste is prevented and where that is not practicable minimised with regard to activity and quantity [Principle RSMDP3 – Use of BAT to minimise waste];
- The best available techniques will be identified by a process that is timely, transparent, inclusive, based on good quality data and properly documented [Principle RSMDP4- Process for identifying BAT];
- In all matters relating to radioactive substances, the 'best available techniques' will mean the most effective and advanced stage in the development of activities and their methods of operation [Principle RSMDP6- Application of BAT];
- When making decisions about the management of radioactive substances, the best available techniques will be used to ensure that the resulting environmental risk and impact are minimised [Principle RSMDP7 – BAT to Minimise Environmental Risk and Impact];
- The best available techniques will be used to prevent the mixing of radioactive substances with other materials, including radioactive substances, where such mixing may compromise subsequent effective management or increase environmental impacts or risks [Principle RSMDP8- Segregation of Wastes];
- Radioactive substances will be characterised using the best available techniques so as to facilitate their subsequent management, including waste disposal [Principle RSMDP9- Characterisation];
- Radioactive substances will be stored using the best available techniques so that their environmental risk and environmental impact are minimised and that subsequent management, including disposal is facilitated [Principle RSMDP10- Storage];
- The best available techniques, consistent with relevant guidance and standards will be used to monitor and assess radioactive substances, disposal of radioactive wastes and the environment into which they are disposed [RSMDP13-Monitoring and Assessment].

The above principles highlight the importance of the consideration and demonstration of BAT for the processes and techniques selected for use in the WTB and ISF in ensuring compliance with the site discharge authorisation granted under the RSA 93.

#### 10.1.3 Overview Of Quality Assurance Arrangements In The Context Of WTB And ISF

As part of the overall process leading to an acceptable design, its evolution and the supporting rationale will be clearly and adequately documented and kept readily available for future reference.

All activities related to the UK EPR will be subject to a quality assurance program encompassing the entire procurement cycle including the selection process and the various stages such as the detailed design, construction and the operation.

The objective of the quality assurance is to ensure with confidence that the installation will perform satisfactorily during operations to the end-point of decommissioning. To that end, quality assurance will include all planned and systematic actions necessary to assure that all aspects of the project, covering activities, systems, components and materials meet the quality requirement. The quality assurance requirement shall always be commensurate with the safety and licensing requirements. Relevant government and international guidance detailing the quality assurance needs for the EPR project will be taken into account in developing a quality assurance programme [Ref. 50 and 51].

### 10.1.4 Radioactive Waste Record Keeping

Information that might be required now and in the future for the safe management of radioactive waste arising from the EPR will be recorded and preserved. Waste records will thoroughly document the waste history from the generation process to the final disposal (or final solution) location including transfers and treatment operations.

#### 10.1.4.1 General Requirements For Record Keeping

The waste generation, characteristics, technical information, processing, conditioning, packaging characteristics and disposal records will be retrievable and traceable. The scope and contents of waste records will be pre-determined and approved for each phase of the wastes lifecycle. Procedures will be developed to control the correction, supersedance or voidance of records. Furthermore records will be kept regarding incidents and non-conformances.

#### 10.1.4.2 Generation Of Records

The generation of waste records will be:

- Specified by administrative and operating procedure, design specifications and procurement;
- Legible, amendable and reproducible;
- Completed and approved by two authorised personnel;
- Valid only if dated, stamped, initialled, signed or otherwise authenticated by authorised personnel;
- Assigned a unique identification number to make them traceable to the waste processes and product.

#### 10.1.4.3 Record Maintenance

Records of waste history will be:

- Long-standing and sophisticated to adequately capture the waste generation, treatment and disposal process information;
- Maintained as a hard copy, digital records (software will be verified and validated) and filmed reproductions;
- Protected from physical and environmental influences to prevent degradation;
- Where possible dual records will be maintained in separate locations;
- Consistent and coordinated between interacting organisations;
- Documented at each stage of the wastes life.

#### 10.1.4.4 Records

The information specified below will comprise the waste record [Ref. 48]:

- Details of ownership;
- The origin of the waste and the processes that generated it;
- Physical, radiological and chemical inventories of the waste;
- Any further technical data available to enable further waste characterisation at a later date;
- Form and content of the waste (e.g. radioisotopes and hazardous materials);
- Waste assay and monitoring records;
- Pre-treatment of the waste;
- Treatment and conditioning of the waste;
- Packaging type;
- Design of the containers and/or packages and of equipment, structures, systems and components for the pre-treatment, treatment, conditioning and storage of the waste;
- Modifications to waste packages;
- Discharge of the waste;
- Records generated during storage of the waste;
- Data needed for a national inventory of waste;
- Records of non-conformances and corrective actions on the waste;
- Information on incidents and defects;
- Assessment, inspection and verifications relating to all activities;
- Non-conformances and corrective actions relating to all activities.

Records of the following will also be retained to assist with the waste history [Ref. 49]:

- Operating procedures;
- Staff training;
- Records from the control process for generation, treatment, packaging and conditioning;
- Procurement of the containers and/or packages and of equipment, structures, systems and components for the pre-treatment, treatment, conditioning and storage of the waste;
- Trends in operating performance;
- Environmental monitoring;
- Authorisations (e.g. licenses).

#### 10.1.4.5 Packaging Records

All packaging will be suitable for the waste that it stores. Packaging labels are the primary means of linking waste packages to the associated records. Packages will be:

- Provided with a unique identification number;
- Marked with a permanent legible means of identification;
- Labels will be qualified to their intended storage environment and life;
- Metal tags, bar codes or optical character recognition using laser etching will be used (consistent with the intended environment and lifetime);
- Check digits will be used to validate package identifiers.

### 10.1.4.6 Records for Decommissioning

Throughout the EPR's life-cycle, information required for decommissioning purposes will be identified, prepared, up-dated and maintained. The information retained for decommissioning operations will include:

- Facility design and subsequent modifications;
- Operational history;
- Incidents and associated response or remedial actions;
- Radiological surveys;
- Radioactive substances, waste quantities, locations, condition and ownership;
- Safety cases;
- Physical condition of the EPR and site, including evidence from the Examination, Maintenance, Inspection and Testing record;
- Knowledge of relevant staff.

Documents related to decommissioning, including the decommissioning plan, end-state and decommissioning history will be generated and retained in a manner consistent with the predicted timescales of the decommissioning operations.

### 10.1.4.7 British Radwaste Management Information System (BRIMS)

In addition to the radioactive waste records described above. Information on wastes can also be stored in the Britisih Radwaste Management Information System (BRIMS). BRIMS is a suite of databases used to store and compile data for the Radioactive Waste Inventory that is used for the short and long-term management of waste. This is conducted on behalf of the Crown with the Intellectual Property Rights belonging to Defra. It is, however, maintained by the nuclear industry as a whole, including the regulators and the government, and also receives financing from the HSE. The system has become the industry wide tool and will be maintained for the foreseeable future.

Defra has proposed that by 2010 it could become a "live" inventory with it being able to be accessed and/or amended whenever necessary rather than at the current three year intervals [Ref.52]. The individual waste packages stored on-site will be marked/labelled (i.e. bar codes, engraving) and these will be related back to BRIMS where data concerning that specific waste package will be kept [Ref. 53].

## 10.2 Strategy And Assumptions

The following strategic baseline and assumptions underline this solid operational waste management strategy:

1) The operational WTB and Intermediate Waste Storage Facility (ISF) are separated and independently operated.
2) If possible the liquid and solid waste treatment and conditioning systems are of modular design (e.g. skid mounted) permitting upgrade and replacement as new technologies become available.
3) VLLW and LLW that are exempt from regulatory control are immediately shipped off-site. The WTB storage area provides a buffer for waste packages conditioned. The buffer zone will have sufficient capacity to meet production rates over a cycle.
4) VLLW can be disposed of in its immediate state. No treatment, conditioning or special packaging will be used for VLLW. Resins will be dewatered.
5) VLLW and LLW disposal sites will be available at the time of NPP start up.
6) The utility may choose to perform only basic treatment steps for solid VLLW/LLW on site and send these wastes directly to centralised off-site facilities for conditioning.
7) Combustible wastes can be directly sent to central (off-site) incineration facilities. At the time of the NPP start up such facilities will be internationally or nationally available and accessible. The waste building storage area provides buffer storage for waste generated during a reactor cycle to be sent to an incinerator.
8) Wastes that are suitable for melting will be directly sent to central (off-site) smelting facilities after pre-treatment, for example sorting and compaction. At the time of NPP start up such facilities will be internationally or nationally available and accessible. The waste building storage area provides buffer storage for packages generated over 1 reactor cycle.
9) Residues from off-site processing such as incineration and melting (e.g. slag) are not returned to the NPP but subject to further treatment in the off-site processing facility.
10) ISF design life will be 100 years and all ILW will be stored there until such time as the GDF becomes available.
11) ILW is transferred to the ISF immediately after treatment and/or conditioning. The waste building provides an 18 month buffer capacity. The waste building provides buffer storage for 1 reactor cycle.
12) The utility could decide to condition waste in its primary packaging and store for extended periods until sufficient decay has taken place or until off-site shipment for final disposal is scheduled.
13) ILW wastes that go into on-site storage will be immediately retrievable. If following decay the activity of such packages is below the threshold for ILW, these packages will be removed from storage and sent off-site as LLW.
14) The NPP will operate its own active laundry. As such these effluents are routed to the waste treatment systems prior to discharge.
15) The discharge limits set for plant operations in the context of the OSPAR treaty do not exceed the discharges that are based in current treatment technology and best practice in 2008.

16) To allow individual segregation and/or treatment of ILW/LLW after long-term storage (decay, application of future conditioning methods), the utility may select to store passively safe ILW wastes (e.g. metals, dried materials) uncompacted and unconditioned. The conditioning / compaction will be performed prior to shipment off-site.
17) Decommissioning will commence immediately after operations have ceased[4].
18) The intermediate waste storage facility (ISF) is a fully functional self-sufficient facility separate to the other buildings.
19) The ISF is elevated above the 100 year flood level and has restricted access routes.
20) The Construction of the ISF could be executed in several phases. About 60 years after commencing operations the storage area could be extended to accommodate the ILW from decommissioning.
21) Storage space that has not been used for operational waste or has been freed due to packages removed following decay could be utilised for storing decommissioning waste.
22) The overall design life of the interim store is 100 years from the introduction of the first waste package. The facility can be expanded with inspection, monitoring repackaging, and treatment facilities to ensure that future waste management criteria and regulations can be met. Such an expansion is optional.
23) The ISF will be actively operated during operation and decommissioning as waste packages are placed for storage. During the time span between end of decommissioning and prior removal of the packages the storage facility will be placed in a maintenance and care regime.
24) The waste packages are segregated and placed in individual storage concrete compartments that are partially or fully covered during loading/unloading operations.
25) The ISF will be designed to protect the waste packages against external events.
26) The ISF will be designed so that doses to workers and public are as low as reasonably practicable during handling and storage operations.
27) The design of the ISF will facilitate monitoring, inspection and retrieval of waste packages.
28) The design of the ISF will allow for the future remediation / reworking of waste packages.
29) Concrete storage compartments that have been loaded with waste packages are closed, disconnected from ventilation and humidity controlled.
30) As the rest of the site is returned to an agreed end state with regulators and the local planning authority, the operations and security of the ISF is continued within a fenced area.

## 10.3 Generic Safety Aspects of Radioactive Waste and Spent Fuel Facilities

This section provides some background information on the selection of safety features for radioactive waste and spent fuel storage. A top-level description of the main hazards and a description of generic safety features that could be incorporated into the design of the building and process equipment to control the hazards are presented. This section does not present detailed information on the safety systems that will be provided in specific items of process plant as these will be dependent on the detailed design and will be assessed in detail in the safety case. The final processes will be selected following an options assessment process to ensure that the technologies selected are the most appropriate.

[4] A conservative assumption of this strategy.

### 10.3.1 Identification of Safety Features

Prior to selection of a safety system the first step in the selection process will be to determine whether a hazard can be eliminated. If the hazard cannot be eliminated then further assessment shall determine if the hazard can be reduced, for example, by reduction of the hazardous inventory used in a process. If a hazard cannot be reduced then an isolation measure shall be incorporated where practicable to isolate the operator from the hazard for example by the use of remote operations. If 'eliminate', 'reduce' or 'isolate' are judged not to be practicable only then should a control measure or engineered safety measure be deployed.

During the design of the waste and spent fuel management facilities we shall apply the following hierarchy of controls is applicable:

a) Accident Prevention

    a. Passive safety measures that do not rely on control systems, active safety systems or human intervention;
    b. Automatically initiated active engineered safety measures;
    c. Active engineered safety measures that need to be manually brought into service in response to the fault;
    d. Procedural safety measures that act to prevent a fault from progressing.

b) Consequence Reduction

    a. Safety measures that act to mitigate the consequence of a fault.

From the list above it can be seen that the design will incorporate passive safety systems in preference to active systems. Reliance will be placed on procedural controls only if provision of an engineered system is not practicable. Therefore the emphasis will be placed on hazard avoidance and then control rather than mitigation of an event once it has occurred.

### 10.3.2 Generic Hazards from Operations of the Waste and Spent Fuel Management Buildings

In the paragraphs that follow, the generic hazards that relate to the waste and spent fuel management buildings are described, together with a description of the safety features that may be used to control or mitigate the hazard. Details of safety features peculiar to a particular waste management facility or spent fuel facility are outlined later with the description of that facility.

#### 10.3.2.1 Direct radiation

Radioactive waste (in particular ILW) and spent fuel emit penetrating gamma rays and neutrons. Employees and members of the public must be protected from exposure.

Doses from direct radiation can be minimised by reducing exposure time, increasing distance from the radiation source and provision of shielding. In reducing exposure time and increasing distance from the source there is a preference for provision of engineered controls and design features where reasonably practicable. Remote operations may be implemented where appropriate to remove the operators from the radiation source. Both fixed and mobile shielding may be provided. In practice dose reduction will be achieved by a combination of the above measures.

### 10.3.2.2 Radioactive Contamination

Radioactive contamination presents a radiological hazard to operators through inhalation, ingestion or ingress through open wounds resulting in internal irradiation of organs. Internal irradiation may also result from contamination of the skin with beta or gamma emitting radioisotopes. There is also the potential for increased public dose following release to environment. In normal operations the operator will be isolated from contamination through primary containment systems provided by the equipment or waste packaging. Further control of radioactive contamination is provided by the provision of appropriate local extract and ventilation systems to ensure that where there is a likelihood of generation of free particulates these will be entrained into the building ventilation system and captured in abatement systems, such as HEPA filtration. The resulting air flow will then be monitored, to confirm that any residual activity levels are within the discharge limits specified in the plant RSA authorisation prior to discharge into the atmosphere. In addition to the capture of contamination at source, regular radiological surveys will be carried out to monitor for the presence of any contamination.

Abnormal events such as dropped loads may result in operator injury and / or radiological release if containment fails. Mechanical handling equipment will be designed in accordance with relevant standards with appropriate safety systems to prevent dropped loads. The number of waste or spent fuel package movements shall be minimised as far as practicable and where necessary carried out at minimum lift heights. Where waste or fuel is containerised the package itself will afford varying degrees of protection to the waste package against dropped load/impact scenarios. Operators will be trained in the safe connection and use of lifting equipment.

Another abnormal event may be the failure of a vessel or service line containing liquid. All equipment will be built to approved UK and international standards and commissioned (i.e. thoroughly and methodically tested) prior to use. The waste and spent fuel management facilities will be designed to provide containment in the event of a spill of material (such as resulting from a pipe or tank rupture or internal flooding) by for example the use of bunds and impermeable floors where appropriate.

Each area of the waste and spent fuel management facilities will be appropriately designated and the contamination control measures put in place will be commensurate with that designation. An example of such a control measure for a contamination hazard is to maintain the area at a negative pressure to the surrounding areas. This ensures that in the unlikely event of a release of contamination it is retained in the area of origin. There is also the provision of continuous air monitoring which alarms on detection of mobile airborne contamination.

### 10.3.2.3 Criticality

Criticality can only occur if there is an accumulation of fissile material with appropriate geometry and neutron moderation to sustain a nuclear chain reaction. Whilst accidental nuclear criticality incidents are, by their nature rare and short lived, they do result in the generation of pulses of high levels of gamma and neutron radiation which are harmful to unshielded personnel in the immediate vicinity.

It is not anticipated that any of the solid operational or decommissioning wastes generated during the reactor lifecycle will contain more than trace amounts of fissile material. All waste streams will be monitored to ensure that they do not contain unexpected fissile material. The waste management buildings therefore do not require any specific safety features for the prevention of criticality accidents.

By its very nature, spent fuel, even when depleted after service in a reactor, contains a significant amount of fissile material. With spent fuel, the fissile material is held within the structure of the fuel assemblies. Spent fuel handling and storage facilities have specific safety measures to ensure that the risk from accidental criticality is minimal. These safety features are discussed further in the specific section concerning spent fuel management facilities.

#### 10.3.2.4 Dropped Load

The processing, handling and transport of waste and spent fuel often requires lifting operations to be performed. The movement of waste packages, spent fuel assemblies and transport containers therefore present risks associated with dropped loads or collisions between transport vehicles, which may result in damage and/or injury, as well as potential radiological releases. These risks can be eliminated or reduced by:

- The design of handling routes and setting of specific operating procedures to minimise the number of lifting operations and lift heights, particularly in the spent fuel unloading cells where bare fuel assemblies are moved individually between containers;
- Lifting and handling equipment is designed in accordance with relevant standards with appropriate safety systems against dropped load. The equipment will also be designed to withstand design basis impacts and earthquakes;
- Provision of mechanical locking systems and measures to prevent derailment or overbalancing of cranes, trolleys, etc;
- Waste packages, spent fuel canisters, and transport containers are highly robust and have been designed and substantiated to withstand drop accidents without compromising containment of radioactive material or shielding;
- Radioactive waste packages and spent fuel containers and transport containers are heavy. Where lifting operations take place, the facility itself will be robustly constructed to withstand the impact of a dropped load. In particular, safety related equipment will not be positioned where it could be affected by a dropped load or impact by a moving load.

The facility safety case will address in detail all situations where dropped loads could occur and will assess the risks and the need for safeguards and mitigation measures on a case by case basis.

#### 10.3.2.5 Fire and Internal Explosions

Fire has the potential to result in both environmental radiological release and standard industrial consequences such as operator injury. Fire may be initiated through a number of sources such as electrical equipment, hot surfaces, vehicles, flammable chemicals and lightning strike. Measures to reduce the likelihood of fire include strict fire zoning, good housekeeping, removal of ignition sources, reduced fire loading in design and materials of construction, and testing of electrical equipment.

The waste and spent fuel management facilities will be fitted with appropriate fire detection and fire mitigation systems in accordance with UK regulatory requirements.

Restriction of petroleum based fuel inventories on site will prevent more severe fires. Where such materials need to be stored on site they will be sited away from buildings containing radioactive materials. Additionally suitable precautions will be taken to manage inventories of fuel tanks in waste and spent fuel transporters.

#### 10.3.2.6 Chemicals

Chemicals are used as part of the waste treatment and conditioning process and during maintenance of plant and equipment. Depending on the chemical properties, these chemicals may cause operator injury through skin contact / inhalation or by fire or corrosive hazards. Where possible, hazardous chemicals shall be substituted for chemicals with less harmful properties. Other controls shall include isolating the operator and environment from the chemical, minimising the held inventory of stored chemicals, providing suitable storage arrangements such as physical segregation of incompatible chemicals with bunding, fire protection and access controls, ventilation arrangements and provision of personnel protective equipment.

The use of chemicals will be in accordance with the requirements of the Control of Substances Hazardous to Health (COSHH) Regulations 2002 (as amended).

#### 10.3.2.7 Machinery

Many of the items of equipment that will be used for the treatment and conditioning of solid wastes will involve fast moving and/or sharp parts or application of high compressive forces which may present a hazard to operators both through industrial injury and / or radioactive contaminated wounds. Examples of such items of equipment include shredders and compactors.

Some of the items of equipment that will be used for the handling of waste packages, spent fuel assemblies and their containers may present a hazard to operators both through industrial injury and / or radioactive contaminated wounds. Examples of such items of equipment include lifting equipment and transfer trolleys. The operator will be isolated from the hazard by carrying out the majority of operations remotely.

Machinery that presents a physical injury hazard will be appropriately guarded and/or interlocked to meet the requirements of the Provision and Use of Work Equipment Regulations (1998).

#### 10.3.2.8 Access

The use of equipment such as fork lifts and trolleys shall be restricted to minimise vehicle movements and reduce the potential for interaction with personnel routes. Likewise, where practicable the process flow shall be optimised to ensure segregation of waste and spent fuel movements from personnel access routes.

Movement of vehicles and personnel within reception halls shall be restricted to minimise vehicle movements and reduce the potential for interaction with personnel. Emergency lighting will be provided in all occupied areas to reduce the likelihood of industrial incidents such as slips, trips and falls resulting in operator injury.

Access to areas where there is the potential for radiological exposure will be controlled.

Access to areas where there is the potential for high dose uptake will be interlocked to prevent inadvertent exposure.

### 10.3.2.9 External Events

External events are challenges to a facility that could originate outside of a facility and therefore cannot be prevented by safety systems within the facility itself. The facility, therefore, must be robust against those challenges that could lead to a release of radioactive material. The waste and spent fuel management buildings and the operations contained within could be challenged by a variety of external events resulting in a range of scenarios including loss of containment or fire. These include aircraft crash, seismic, lightning strike, flooding, extreme weather and external explosion.

#### 10.3.2.9.1 Seismic Event (Earthquake)

The waste and spent fuel management building structures will be designed to withstand a design basis earthquake in accordance to the relevant engineering standards.

#### 10.3.2.9.2 Lightning Strike

The waste and spent fuel management facilities will include a lightning protection system in accordance with relevant engineering standards.

#### 10.3.2.9.3 Flooding

The facilities will be designed and located to minimise the likelihood and consequences of flooding.

The risk is also mitigated by the appropriate design of the surface water drainage system to protect against extreme rainfall.

#### 10.3.2.9.4 Loss of Electrical Power Supply

The waste and spent fuel management facilities will use, where possible, passively safe systems that do not rely upon electrical power supplies to perform their safety functions. In the event of a loss of the main electrical power supply, secondary electrical supplies and emergency electrical supply will provide, if required, power to safety critical systems, instruments, utility service systems and operating systems to allow safe conditions to be maintained at all times.

Discharge monitoring equipment will be connected to emergency electrical supplies to ensure their continued operation.

#### 10.3.2.9.5 External Explosion / Missiles

Explosions external to the waste and spent fuel management buildings may subject these buildings to shock wave and/or missile impacts.

To reduce the possibility of explosion risk, the use of pressurised gas bottles will not be used or stored within the waste and spent fuel facilities. Pressurised gas supplies will be piped into the facilities from an external source where their use is required (e.g. for instrumentation or inerting spent fuel storage containers).

#### 10.3.2.9.6 Aircraft Crash

The frequency of aircraft crashes and the potential consequences on the facilties will be analysed to ensure the risks are minimised.

# 11 WASTE TREATMENT BUILDING

## 11.1 Introduction

The WTB is the single interface for the processing of all radioactive operational waste materials that will be generated by the operation of the UK EPR.

The building is designed to perform all waste management functions to house the treatment and conditioning systems that are necessary to meet the utility and regulatory baseline. As a result of this the building will accommodate a wide variety of complex functions for safe handling, treatment, conditioning, buffer storage, packaging and monitoring of wastes.

The building layout and systems provide the following key waste management functions:

1. Treatment of radioactive wastewater and effluent;
2. Treatment of solid waste;
3. Conditioning of solid / liquid waste.

As such it provides the storage capabilities for incoming raw waste (liquid /solid) to be treated as well as the buffer space needed prior to shipping of conditioned wastes for intermediate storage or transfer to the off-site disposal location.

The liquid and solid waste treatment and conditioning systems provide the capability to minimise the interim storage and disposal volumes while satisfying the storage and disposal requirements. In addition the handling and transportation requirements for the waste packages are met.

Following monitoring, clean liquids will be discharged in accordance with the discharge authorisation issued under the RSA 93 and decontaminated parts are measured and released off-site as the material exempt from regulatory control limits are met.

In addition the WTB provides all functions required for the protection and safe handling of radioactive material. As such it is designated a controlled area under the Ionising Radiation Regulations (1999) and incorporates measures to ensure that the doses received by operators are reduced in accordance with the requirements of these regulations.

The raw waste volumes expected during normal operations and outages will determine the sizing of the treatment systems. To allow maximum flexibility in the waste treatment and conditioning techniques the equipment used will be mobile, for example skid mounted units and modular in its construction where possible.

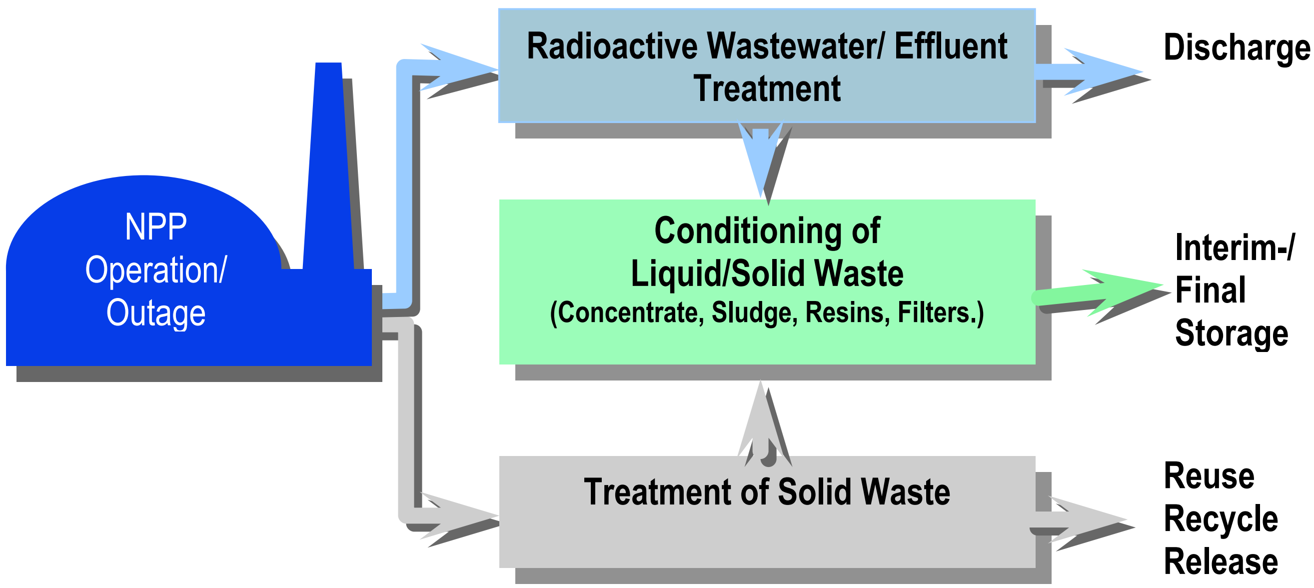



**FIGURE 64: FLOWCHART SHOWING TREATMENT AND DISPOSAL ROUTES OF SOLID WASTE AND LIQUID EFFLUENT**

## 11.2 Building Functions and Process Systems

### 11.2.1 Waste Treatment Building Functions

The WTB will house the following waste treatment and conditioning systems. Where possible the systems will be designed as individual (e.g. skid mounted) modules that can be relatively easily exchanged or replaced during the operation of the plant as waste treatment technology develops. The treatment and conditioning systems that will be incorporated into the WTB have been described in detail earlier in this report. In addition to treatment and conditioning of waste, the WTB will also incorporate buffer storage, additional waste management functions and various auxiliary systems. These functions are described below.

Buffer Storage will include:

- Buffer tanks for liquid effluent awaiting treatment;
- Buffer tanks for concentrates awaiting conditioning;
- Buffer tanks for treated water destined for discharge;
- Buffer areas for raw solid waste;
- Buffer areas for treated solid waste prior to conditioning;
- Buffer areas for conditioned wastes prior to transfer on-site and shipment off-site.

The additional waste management functions that will be performed by the waste treatment building will include:

- Shipping and receiving areas;
- Effluent monitoring;
- Waste container monitoring;
- Contamination monitoring and survey;
- Controlled clearance of solid waste;
- Decontamination System;
- Active Laundry;
- Transfer Station(s);
- Hot workshop

To support operations in the WTB the following auxiliary systems will be provided:

- Cranes, hoists;
- HVAC;
- Cooling, Chilled Water;
- Demineralised water;
- Compressed air;
- Electricity.

The treatment and conditioning systems are installed in three main building sections that are connected to the buffer area, as shown in Figure 65.

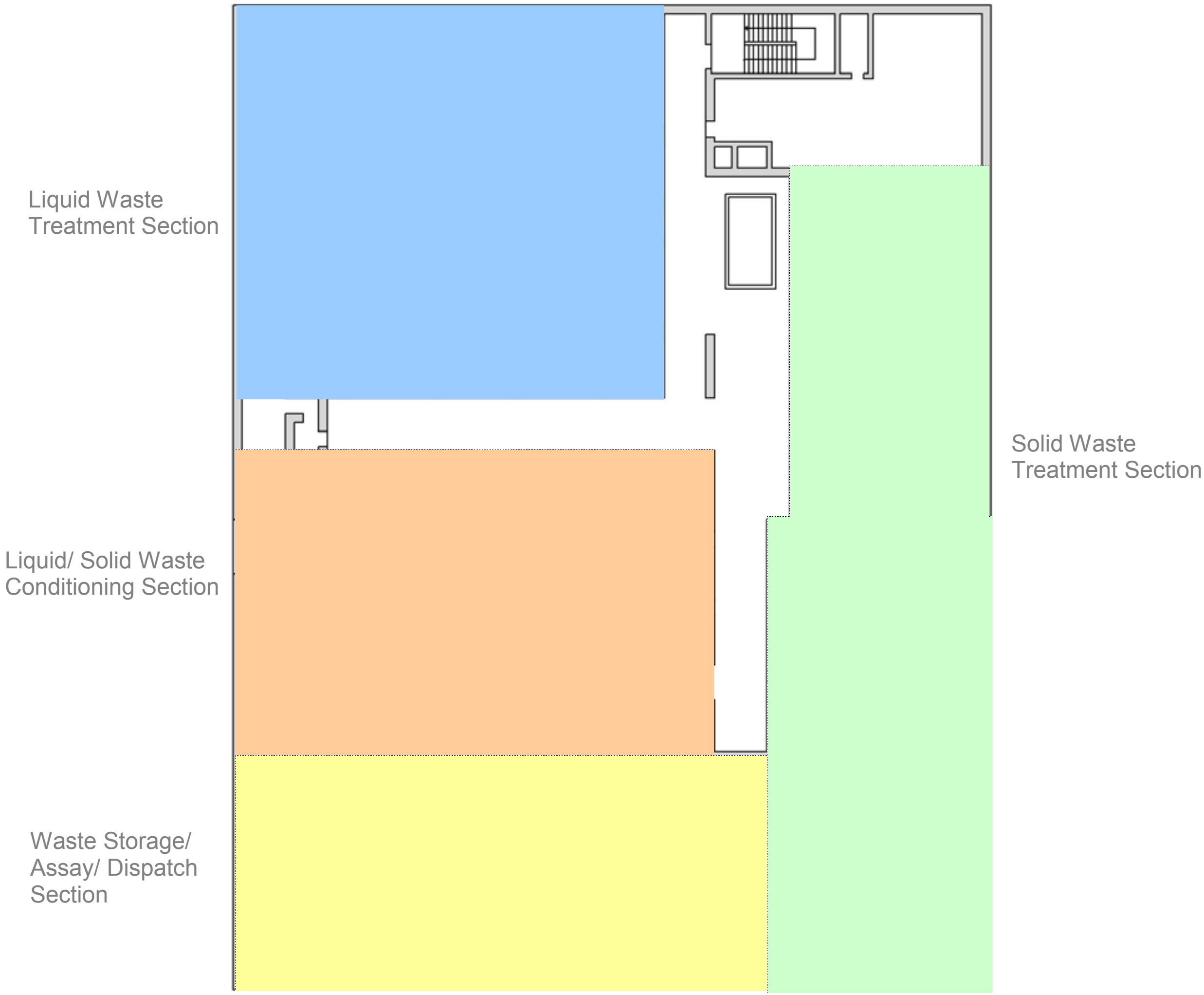



**FIGURE 65: FLOOR LAYOUT OF A WTB SHOWING SOLID AND LIQUID TREATMENT, AND WASTE STORAGE AREAS.**

### 11.2.2 Solid Waste Treatment

The operational solid radioactive waste generated by the EPR will include paper, plastic, metal waste and components arising from maintenance operations in the reactor, concrete and electrical parts. The waste will be collected and stored according to waste type in marked bins at designated locations within the WTB. Typically plastic bags, drums, containers or a combination of these items will be used to store waste.

Various wastes are shipped from different locations of the controlled area to the WTB:

- VLLW and LLW;
- ILW.

Typically, the WTB will be directly connected to the nuclear auxiliary building of the UK EPR.

All untreated solid wastes will be placed into temporary buffer stores prior to treatment. The waste will be separated into the different fractions dependent on the eventual treatment method. Waste will be stored in these areas until a sufficient quantity has accumulated for a treatment campaign to commence or for shipment for off-site treatment. Once a sufficient volume has been accumulated the waste is routed to the treatment system and the appropriate treatment is initiated.

Records will be maintained describing the contents and composition of the waste at each treatment stage. This information will be stored in a central database and used to complete the appropriate storage, transfer and disposal documentation.

During all handling operations the radiological properties of the wastes will be carefully considered and monitored and shielding and other measures provided to ensure that doses to operators are reduced. Examples of the types of safety features that will be incorporated into the design of the WTB are provided in this report.

The layout of the WTB will be designed to facilitate contamination control. Areas with a high probability of the presence of loose contamination, for example inside the waste sorting box, will be individually connected to the building Heating Ventilation and Air Conditioning (HVAC) system via a pre-filter and other contamination control equipment. Other areas, for example the decontamination booth, will be routed via a series of High Efficiency Particulate in Air (HEPA) filters prior to the connection to the main WTB building HVAC. The airflow direction within the building will be designed to direct potential contamination away from operators towards the HVAC systems. Process layout will be optimised to ensure that handling operations are located away from access routes.

The solid waste arisings will initially be collected in-situ in plastic bags. The bags will then be sealed and shipped in either 200 litre drums or other appropriate containers to the buffer areas. From the buffer areas the packages will be retrieved, transported and opened at the appropriate treatment location. These locations will include areas for segregation, sorting, shredding, decontamination, compaction and the controlled area workshop.

#### 11.2.2.1 Segregation Of Solid Waste

Prior to treatment by super compaction, the dry solid radioactive waste will be pre-treated to grade and size reduce large items of waste. The first step will be sorting smaller items of solid waste. This will be carried out in the sorting box. Larger items of solid waste will be collected and stored separately to facilitate easier handling of the waste. It is not anticipated that this waste will be sorted using the sorting box.

Moist solid waste will be dried to prevent degradation of the waste container during storage. Solid non-compactable waste will be collected in 200 litre drums, stored in the room used for raw waste and treatment by size reduction, decontamination or dismantling in the active workshop.

### 11.2.2.2 Sorting Of Solid Waste

The waste contained in bags and drums will be monitored at the point of generation and again upon receipt in the buffer store using portable monitoring equipment to identify items with high activity levels. Following monitoring, the waste will be moved to a manual sorting facility. The drums containing the waste to be sorted will be opened, placed on a tilting device and emptied into the sorting box. Following emptying, the drums will be removed to a storage area and then sent back to the operational areas for re-use. The bags will be loaded into the sorting box through a loading flap.

The waste will then be extracted onto a sorting table inside the sorting box and sorted manually. The segregation of the waste into different waste groups will be carried out on the basis of different physical and chemical properties, e.g. wet solid waste, combustible and compactable waste, and non-compactable waste. The sorted waste will be collected in drums arranged below the sorting table. Compactable drums will be used for collecting compactable and burnable waste. Non-compactable waste (such as metals) as well as moist solid waste will be collected in 200 litre drums. Bulky, combustible and compactable waste will either be collected in 200 litre drums or in suitable containers for further shredding. The operator will visually monitor filling of the drums.

The sorting box has a number of discharge holes below which waste drums will be located on transport trolleys. Following filling of the drum the discharge hole will be covered, the drum lowered and the drum lid manually installed. The drums will then be transported to the buffer storage room pending further treatment.

As stated earlier in this description moist compactable solid wastes will be dried prior to further treatment. Following sorting in the sorting box the moist wastes will be transferred to a drum and dried. Monitoring of the temperature/moisture in the ventilation extract pipe will be used to determine when the drying process has been completed. Following completion of the drying process the waste will be monitored prior to placement into compactable drums and super compaction of the waste.

### 11.2.2.3 Shredding

Bulky solid combustible and compactable waste will be size reduced by shredding prior to further treatment. Waste is transported to the shredder and inserted in the loading part of the shredder. The waste is size reduced by the use of a rotating blade assembly. The shredded material then falls through a duct into a compactable drum located directly below the shredder. Once full, the drum will be returned to the storage area and temporarily stored until a sufficient volume of waste for treatment or disposal is collected.

#### 11.2.2.4 Super Compaction

Combustible and compactable waste to be super compacted will be transferred to the super compactor in compactable drums.

The compaction process can be subdivided into the following steps:

1. Supply of compactable drums;
2. Feeding of drums into the super compactor;
3. Compaction of drums;
4. Removal, sorting and storage of compacted drums (pellets);
5. Loading of pellets into a 200 litre drum;
6. Capping and lidding of the drum.

The filled compactable drums are positioned on a feeding conveyor and routed to the compactor. The drums are placed into the super compactor using the drum pellet grab. A ram cover separates the compression region of the compactor from the environment so that in the event of a release of contamination during compaction the contamination is contained within the compactor. The compression region is extracted via a filter to the HVAC system.

During super compaction the pressing ram driven by hydraulic force is lowered on the compactable drum and the drum, including its contents, is compacted to a smaller volume using a compaction force of about 2000 tonnes. During the compaction process a guide mould is used to constrain the compacted drum ensuring a constant diameter is maintained. Prior to compaction the drum is pierced by a hydraulically driven spike, which is incorporated into the bottom plate of the super compactor. Any liquid released during super compaction is collected in a small collection tank equipped with a pump that is part of the super compactor.

After the compaction process is finished, the drum pellet grab will remove the compacted drum (‘pellet’) and place it onto the pellet conveyor. Here the weight, height and dose rate of the drum is measured and recorded. Before the pellets are inserted into the 200 litre drums, the pellets are sorted on the conveyor into height order, to maximise the use of space in the drum. After the pellets are sorted they are picked up from the buffer by means of a pellet-handling device and inserted into a 200 litre drum. The drums are then manually capped and transferred by crane into the drum store where they are stored awaiting disposal.

### 11.2.3 ILW Streams

ILW streams will arrive at the WTB. These will be placed in the appropriate buffer areas, treated and conditioned prior to shipment to the Interim Storage Facility. These waste streams will include:

- Spent filter cartridges;
- Spent resins;
- Higher activity solid wastes.

Spent filter cartridges will be inserted directly into 200 litre drums by the filter changing equipment. The drums containing spent filter cartridges will then be inserted into a shielded cask and transported from the nuclear auxiliary building to the WTB where they will be stored for several years (e.g. 5 years) to allow decay of the radioactivity in the tubular shaft store. Once the filters have decayed sufficiently they will be retrieved, conditioned and sent to the ISF.

Spent resins will be placed into specially designed transportation containers in the nuclear auxiliary building. The containers will then be transported to the WTB for treatment and conditioning. Upon arrival in the WTB the spent resin container will be connected to the spent resin tank and its content is flushed into the tank awaiting conditioning.

Higher dose solid waste parts will be placed into shielded transport containers at the place of origin. A transportation vehicle will be used to transfer these containers to the appropriate treatment section (e.g. the decontamination booth).

### 11.2.4 Active Laundry

If the Active Laundry Facility is located in the WTB a buffer store will be provided for all items of laundry for example personal protection equipment, over shoes etc. Any items containing higher than expected levels of activity will be segregated and placed into a designated area for decay storage. Items will be retrieved from the store and checked for damage prior to industrial washing and drying. Once washed the items will be monitored and placed into the buffer storage area pending shipping for re-use. All effluents will be sent to the liquid storage tanks prior to treatment and processing.

### 11.2.5 Handling And Shipping Of Final Packages

Following treatment the waste will be placed in an appropriate container for transport or disposal. Following sealing the containers will be monitored for the presence of external contamination prior to transfer out of the WTB.

Waste containers awaiting transfer off-site will be placed in buffer stores and transferred into transportation containers (e.g. IP2 containers) prior to loading onto the transportation vehicle.

Wastes that will require conditioning on site will also be placed in a buffer store prior to sentencing for conditioning at an appropriate time.

Conditioned packages that will be sent to the interim buffer store, ISF or off-site will be weighted and monitored to determine the dose rates and contamination levels at the time of shipping. All data collected will go into the waste tracking system in conjunction with other package specific data (e.g. content, treatment, conditioning, nuclides).

Waste containers leaving the building will be transferred to a number of locations including:

- Off-site treatment facilities;
- VLLW and LLW disposal facilities;
- ISF.

Prior to shipping the data that has been collected for the waste package will be verified for completeness and the appropriate transportation, storage and/or disposal documentation prepared. Waste packages will not be transported until this paperwork has been verified to meet the required standards. Following thorough verification some materials will be exported as ‘exempt from regulatory control’ from the WTB. These materials will be shipped to another location, for example for recycling.

### 11.2.6 Liquid Waste Treatment System

Liquid waste will be transferred via pipework into the WTB.

During operation of the nuclear power plant, liquid effluent (water) and liquid waste of different origins is produced by the drains system, leakage, flushing etc.

The function of the liquid waste treatment system and the nuclear island liquid radwaste monitoring and discharge system is to collect, process and decontaminate all radioactive liquid waste and liquid discharge from the controlled area of the power plant prior to discharge from the EPR.

The system will collect, store, process and clean the liquid radioactive waste produced by let-down, drainage, purge, venting, or leakage from systems in the controlled area. All liquid effluent generated in the controlled area will be collected and treated in the liquid waste processing system prior to discharge.

The functions of the system include:

- Liquid effluent collection and storage;
- Liquid effluent treatment;
- Liquid effluent release including control of liquid effluent release to the environment;
- Liquid waste concentrate collection and transfer to the solid waste treatment system.

The main objective of the system is to provide a solid waste form suitable for conditioning and a liquid component suitable for discharge to the environment which is within the limits specified on the discharge authorisation granted under the RSA1993 [Ref. 10].

#### 11.2.6.1 Liquid Effluent Collection

The system will provide the following functions:

- Selective collection and segregation of liquid effluents produced by the reactor coolant treatment system, nuclear auxiliary systems, reactor cavity and spent fuel pool, as well as all potentially-contaminated liquids produced in the controlled area (such as floor drains, laundry and decontamination waste);
- Routing of the collected waste to the storage and treatment facilities.

The total storage capacity of the liquid effluent collection system will correspond to the maximum quantity of effluents produced as a daily maximum during outages. Mobile decontamination equipment will be available for the removal of any sludge that has accumulated in the sumps of the radioactive liquid waste systems.

Each subsystem will have sufficient process flow and storage capacity to process the daily inputs produced during normal operation and during anticipated operational occurrences, including refueling, plant shutdowns, maintenance and startup operations, as well as initial startup.

#### 11.2.6.2 Liquid Effluent Treatment

The following functions will be provided:

- Selective front-end storage of potentially-contaminated liquid effluent (water) according to the chemical composition and radioactivity of the various waste streams. In particular the effluent from the equipment drains (higher activity) and from adjacent floor drains (high conductivity, organics, low activity) will be separately collected;
- Analysis of the contents of each storage tank and confirmation that subsequent adequate treatment (adjustment, evaporation, separation, filtration, etc.) will be performed so that the treated liquid can be discharged to the environment in accordance with the requirements of the plant discharge authorisation;
- Treatment of the liquid effluent to separate it into clean liquid for discharge and waste concentrate for conditioning;
- Corresponding transfer of the clean liquid to the monitoring system to monitor the activity and check for non-radioactive impurities prior to discharge;
- Receipt and back-end storage of the concentrated liquid waste product;
- Transfer of concentrated waste produced (evaporator concentrates, sludge etc.) to the radioactive concentrates processing system for conditioning.

#### 11.2.6.3 Liquid Discharges

The following functions will be provided:

- Receipt and collection of treated liquids after appropriate treatment (adjustment, evaporation, separation, filtration, etc.) prior to discharge;
- Measurement of the volume and activity level of liquid effluents prior to release;
- Determination and recording of release rates;
- Automatic isolation of the discharge line, if an authorised limit is exceeded.

Treated water discharged to the environment via the liquid waste processing/storage system will comply with the requirements of the discharge authorisation granted under the RSA 93. The monitoring and accounting of the effluent released to the environment will be in accordance with this and other applicable legislation.

The Liquid Waste Storage and Treatment Systems are designed to store and process liquid effluent accumulated in the controlled area during power operation, maintenance, overhauls and refueling, to discharge cleaned liquids into the environment and to provide back-end storage for concentrated liquid waste prior to conditioning.

The systems will be controlled from a local control station. The rate of processed liquid discharge may be set as a function of cooling water volume.

### 11.2.7 Wet Solid Waste Conditioning

The wet solid waste conditioning system serves to condition liquid radioactive waste generated and stored during operation of the nuclear power plant.

Sludges and concentrates will be pumped from the concentrate tanks into the concentrate buffer tank for homogenisation, chemical pretreatment/pH-adjustment and mixing with other waste types.

The function of the liquid waste conditioning is to receive, process and condition liquid and solid radioactive waste into a form that is suitabe for either long-term storage and/or disposal and is compliant with the relevant approvals.

The types of waste that will be processed by this system include evaporator concentrates and sludge and other types of solid waste from the solid waste treatment system including spent resins and spent filter cartridges.

The conditioning process (drying, cementation, etc.) for the treatment or solidification of the waste into the final form and the waste package itself will be compliant with the requirements of of the relevant disposal authorities for example the LLWR and RWMD.

The evaporator concentrates and the sludge as well as the solid waste will be conditioned and converted into a solid and long-term stable product (e.g. salt block, encapsulated product, etc.).

The spent resins will be treated by dewatering in drums, in a shielded cast iron cask or solidification. The resulting waste form will be a disposable product which meets the final disposal regulations.

The spent filters from the water treatment system will be treated by drying or embedding in grout to form a compliant solidified waste package. The resulting waste form will meet the requirements for final disposal.

The compliant product is packaged in accordance with the relevant transport and disposal regulations. The waste package is either directly transportable and disposable, or the filled waste drum will be inserted into an overpack or be embedded in a disposal container in concrete or bitumen depending on the transport, intermediate and final storage regulations, and the radiation dose levels of the product.

### 11.2.8 Building Layout

The floor plan and dimensions below illustrate the layout typical for a waste building of the EPR. The layout and dimensions are subject to change based on the required facilities, packages and footprint of the selected site.

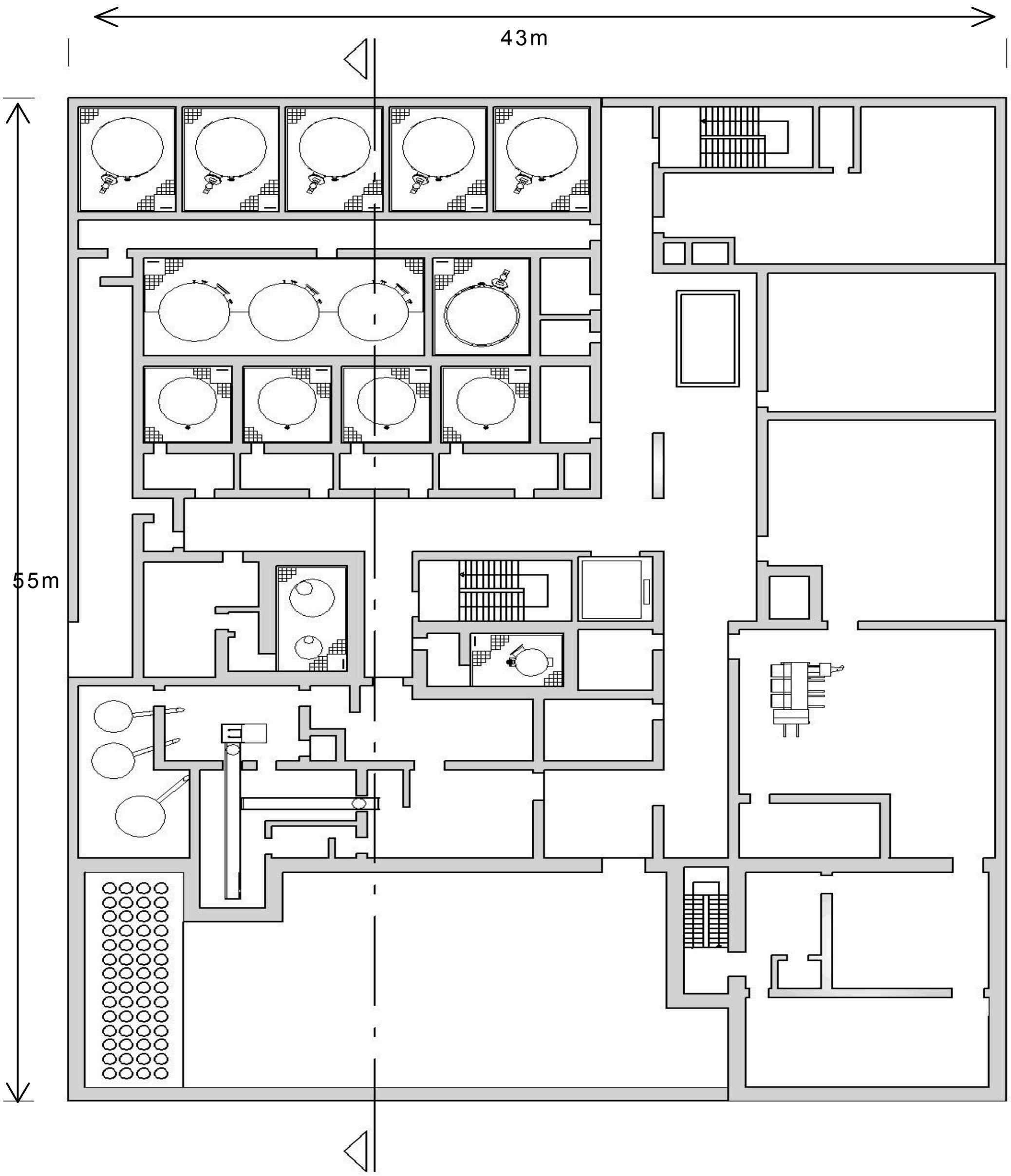



**FIGURE 66: FLOOR LAYOUT OF A WTB SHOWING TYPICAL DIMENSIONS**

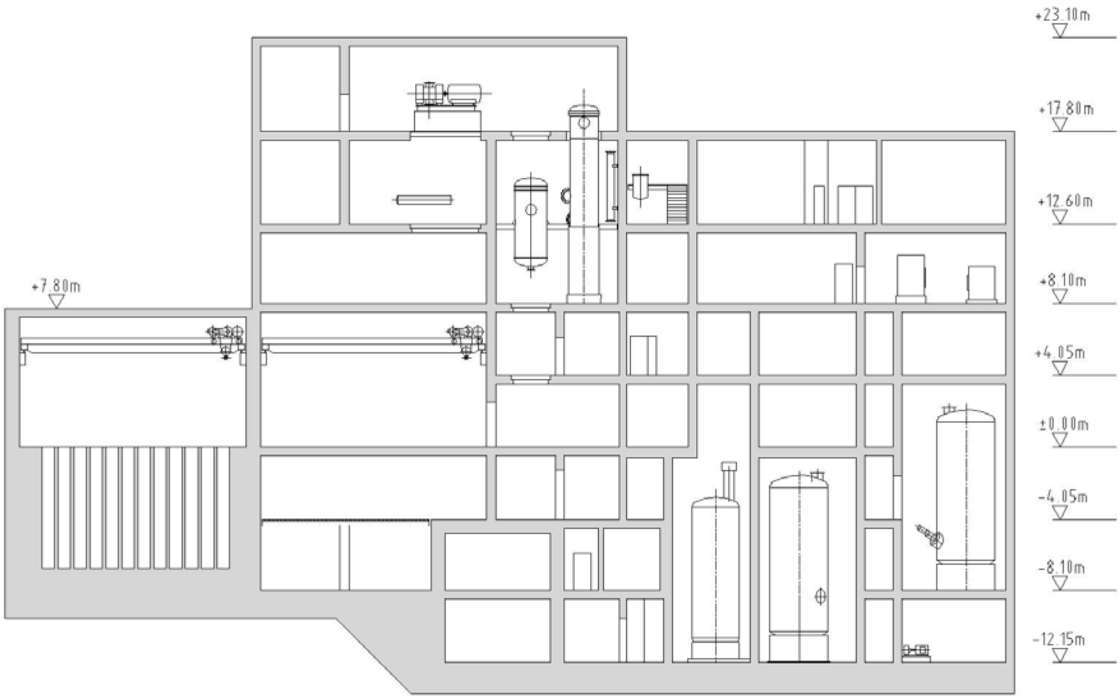



**FIGURE 67: SIDE VIEW OF WTB SHOWING FLOOR LEVEL DIMENSIONS.**

### 11.2.9 Specific Safety Features

#### 11.2.9.1 Hot Surfaces

Some of the treatment and conditioning processes (for example drying) will result in the creation of hot surfaces with the associated potential for operator injury. Additionally, hot surfaces are a potential source of fire. Appropriate engineered protection will be provided to safeguard operators from this hazard.

#### 11.2.9.2 Fire

A large proportion of combustible waste will be categorised as LLW. This will be packaged and exported to an LLWR management route as soon as practicable. Appropriate fire protection measures will be installed.

### 11.2.10 Building Operations

The WTB will support all phases of UK EPR operation including:

1. Treatment and conditioning of wastes from UK EPR operations;
2. Treatment and conditioning of wastes from UK EPR decommissioning[5].

The raw waste quantities and composition for each phase will be significantly different.

The WTB will be the last nuclear building to be decommissioned during the UK EPR dismantling. As such it must be able to support its own declassification to an undesignated area and removal of the stored waste and its treatment and processing systems. For these reasons the WTB must be able to accommodate the changes required during the transition from the treatment processing of operational wastes to decommissioning wastes.

#### 11.2.10.1 EPR Operational Support

The operations of the WTB must accommodate the treatment and conditioning facilities of all operational waste produced from the NPP during its operations.

The treatment areas for the proposed mobile modular (e.g. skid mounted) systems will be provided with connections for all services (for example pressurised air, de ionised water, HVAC) as required but will maintain the flexibility to exchange, modify or replace the treatment systems as required.

#### 11.2.10.2 EPR Decommissioning Support

Following the transition into EPR decommissioning the liquid and solid waste volumes, composition and processing scope will change. The liquid effluent volumes will be reduced and the volumes of dismantled components that must be treated as solid wastes will increase.

As the Reactor and auxilliary buildings are decommissioned the resulting wastes, in particular LLW that cannot be treated within the reactor and auxillary building, will be routed to the WTB.

[5] Construction of a dedicated building for the processing of decomissioning wastes is an alternative option.

Liquid and solid waste treatment will therefore continue with additional personnel and an increase in the rate of treatment, conditioning, monitoring and shipping.

It is intended that segregation of decommissioning waste will occur at the point of generation i.e. in the area that is being decommissioned. In practice there may be a number of areas of the plant that are being decommissioned in parallel resulting in the establishment of a number of disassembly and segregation areas. The process of decommissioning will result in an increased number of heterogeneous, larger components arising from dismantling of the primary circuit. To accommodate these increased waste volumes rooms originally assigned as buffer stores during the operational phase of the EPR will be modified as required.

A key objective of the decommissioning process will be to maximise the volume of concrete and metals that can be sentenced and disposed of as exempt from regulatory control.

#### 11.2.10.3 WTB Decommissioning Operations

Following completion of decommissioning of the controlled areas of the reactor and other buildings the WTB itself will require decommissioning. The decommissioning of the WTB itself will also result in the generation of radioactive waste. This will require careful planning and the phased introduction of temporary equipment to replace the existing waste treatment equipment, permitting decommissioning of the existing equipment. The modular nature of the WTB will facilitate the installation of temporary treatment equipment.

# 12 ILW INTERIM STORAGE FACILITIES

## 12.1 Introduction

ILW generated by the operation of the EPR will be stored in an Interim Storage Facility (ISF). The ISF is located separately from the WTB and will continue to provide interim storage for ILW following delicensing and return of the site to an agreed end state with the regulators and local planning authority.

The ISF will be used to store operational ILW that has accumulated during the 60 years of plant operation and can be adapted to store any additional ILW that is generated during decommissioning of the plant after final shut down. The expected lifespan of the facility is 100 years.

As such the site selected for the construction of the EPR and the building design itself will permit the expansion of the building for storing decommissioning wastes as they are generated. The facilities will be designed to handle the different waste packaging geometries that will be stored in the ISF.

The ILW will be stored in fully covered, concrete storage compartments that provide individual storage volumes. The fully concrete storage compartments permit operator access, loading and unloading, visual inspection and ventilation of the concrete storage compartments amongst other functions.

The ISF will consist of areas performing the following functions:

- Receipt and dispatch area;
- Interim storage space for all operational ILW and possibly all decommissioning ILW until a final disposal / solution facility becomes available (up to 100 years);
- Package Inspection area;
- A storage area that permits removal of ILW that may become LLW following a period of decay storage.

**FIGURE 68: EXAMPLE OF THE RECEPTION AREA OF A STORAGE FACILITY FOR RADIOACTIVE WASTE**

## 12.2 Building Functions And Process Systems

The ISF building will be a fully functional, self-contained building that is located separately from other building structures at the site. As such it will be surrounded by a fence and have a separate single point of entry.

The building could be constructed on an elevation to reduce the risk of flooding and the building structure and walls will serve as shielding to reduce the radiation emitted from within the building.

The ISF is designed to allow two future expansions if required to receive and store decommissioning wastes and to add packaging and conditioning facilities for the waste prior to disposal of the waste to the proposed GDF.

It is anticipated that the first extension will be constructed (after shut down of the EPR 60 years after commencement of operations) in the EPR at the start of the decommissioning phase.

If required, due to changes in national standards and practices a second extension may be constructed after approximately 100 years. This extension would be used to house state of the art packaging/conditioning/monitoring facilities for the waste prior to shipment to the final disposal/solution facility.

The ISF will provide the following functional areas:

- Reception and shipping area;
- Storage compartment loading/ unloading area;
- Covered ILW concrete storage compartments;
- Crane maintenance;
- Control room;
- Monitoring area;
- Auxiliary areas and rooms (e.g. crane maintenance, electrical, HVAC, controlled entrance area);
- Extension to accommodate decommissioning wastes;
- Potential for extension to accommodate further packaging/conditioning/monitoring facilities if required.

### 12.2.1 Main Systems And Facilities

#### 12.2.1.1 Reception And Shipping Area

The on-site transport of the waste packages will be carried out using a shielded transport vehicle, if required.

The reception and shipping area will be designed to receive both shielded and unshielded waste packages from the WTB. The ISF crane will be able to access the receipt and dispatch area via a hatch for loading/unloading the packages from and to a truck. Handling of the waste packages inside the storage area will be performed remotely.

#### 12.2.1.2 Shielded Transport Container Loading/Unloading Section

The shielded container will be transferred from the transport vehicle by crane to the shielded container loading and unloading section, which will be located adjacent to the concrete storage compartments.

Prior to removal of the drums and containers from the shielded transport container the lid will be removed from the shielded container and laid down.

#### 12.2.1.3 Storage Compartment Loading And Unloading Area

The storage compartment loading and unloading area is located directly above the concrete storage compartments. This area will be supplied with an overhead crane that permits access to the individual concrete storage compartments.

The storage compartment loading/unloading area extends above the receipt/dispatch area and all other auxiliary supporting areas/rooms.

### 12.2.1.4 Covered ILW Storage Concrete Compartments

The storage area is divided into individual compartments. These divisions allow the flexibility to segregate the waste packages according to their properties, content, decay characteristics, etc. as required.

All compartments will normally remain closed with concrete slabs (an example of insertion openings is shown in Figure 74). However, the compartment that is currently used for loading or reloading is covered with an iron frame slab. Within this slab is an iron frame including access holes that permit the use of the compartment for emplacement and retrieval of drums and boxes.

The slab thickness reduces the dose in the loading/unloading area above permitting unrestricted, direct human access.

The preliminary design of these compartments allows stacking of the following (for example): 500 litre NDA drums in 4 layers, 3 m³ NDA Boxes in 3 layers, 200 litre drums in up to 6 layers or cast iron containers in up to 4 layers, or 4m boxes. Other waste package geometries can also be accommodated if required.

**FIGURE 69: 500 LITRE DRUMS**

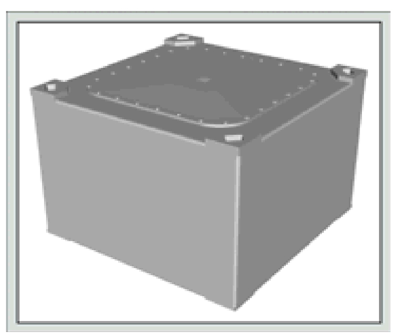


**FIGURE 70: 3M$^3$ BOX – CORNER LIFTING VARIANT**

**FIGURE 71: 3M$^3$ BOX – MID SIDE LIFTING VARIANT**

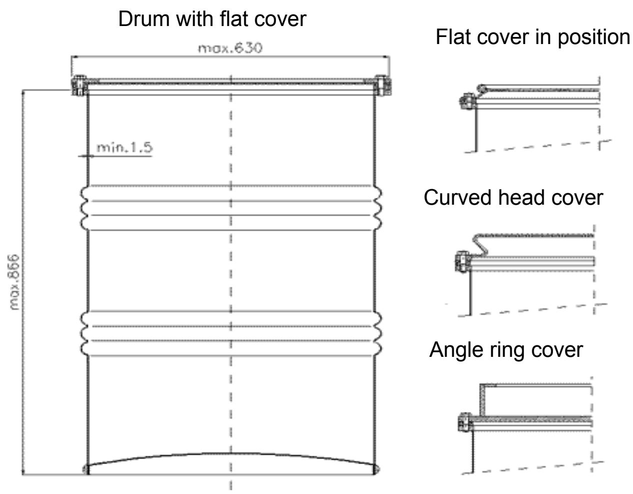



**FIGURE 72: 200 LITRE DRUM**

**FIGURE 73: 4M$^3$ BOX**

The sectioned concrete slabs contain individually plugged openings (e.g. for inspections). The following picture shows the openings and plugs for the loading and unloading of drums.

**FIGURE 74: EXAMPLE OF INSERTION OPENINGS IN A STORAGE FACILITY**

The storage concrete compartments will be of different sizes but can be subdivided into sections to accommodate different packages. Thus the storage compartments can be loaded with different waste package geometries and stacking characteristics. The compartments are designed so that additional stacking aids can be installed if the packaging requires additional support.

The compartments are individually connected to an active ventilation system. The active ventilation system connects the compartments and routes the exhaust air to the ventilation/air conditioning system.

Due to the expected life span of 100 years the ventilation ducts could be integrated in the concrete structure.

Whilst the active ventilation is primarily used during the loading/unloading (and infrequent inspection procedures) all compartments that are not actively operated are left in a passive state.

During this passive stage the compartment remains fully enclosed and access of air that can result in changes of humidity within the compartment will be minimised though the closure of all openings.

## TABLE 41: CONDITIONED WASTE VOLUMES AND NUMBER OF CONTAINERS FROM OPERATIONAL AND DECOMMISSIONING ACTIVITIES

| Year | Waste Transferred to Interim Storage | | | | | Decayed ILW | | | | Ramaining Waste | Packages in Interim Storage | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Ion Exchange Resins (IER) | Sludge | Water Filters | Technological Operational Waste | Total in | Sludge | Technologicl waste | Water Filters | Total Decay | Total Waste in Store | 200 l drums | Cast Iron Container | 500 l NDA drum |
| **200 l Drum Concept:<br>Interim Storage - 200 l drum<br>Final Repository - 200 l drum macro-encapsulated in 500 l NDA drum** | | | | | | | | | | | | | |
| 10 | 33 | 9 | 50 | 15 | 108 | 6 | 4 | | 10 | 98 | 491 | - | - |
| 20 | 67 | 19 | 100 | 30 | 215 | 16 | 14 | | 30 | 185 | 934 | - | - |
| 30 | 100 | 28 | 150 | 45 | 323 | 26 | 24 | | 50 | 273 | 1377 | - | - |
| 40 | 133 | 37 | 200 | 61 | 431 | 36 | 34 | | 70 | 361 | 1819 | - | - |
| 50 | 167 | 46 | 250 | 76 | 539 | 46 | 44 | | 90 | 449 | 2262 | - | - |
| 60 | 200 | 56 | 300 | 91 | 646 | 56 | 54 | | 110 | 536 | 2705 | - | - |
| 70 | | | | | 646 | 66 | 64 | 25 | 155 | 491 | 2483 | - | - |
| 80 | | | | | 646 | 76 | 74 | 50 | 200 | 446 | 2262 | - | - |
| 90 | | | | | 646 | 86 | 84 | 75 | 245 | 401 | 2041 | - | - |
| 100 | | | | | 646 | 96 | 94 | 100 | 290 | 356 | 1820 | - | - |
| **Cast Iron Container Concept:<br>Interim Storage - 200 l drum, Cast Iron Container, 500 l NDA drum<br>Final Repository - 200 l drum macro-encapsulated in 500 l NDA drum, Cast Iron Containers** | | | | | | | | | | | | | |
| 10 | 113 | 25 | 50 | 15 | 203 | 6 | 4 | | 10 | 193 | 250 | 93 | 18 |
| 20 | 226 | 50 | 100 | 30 | 406 | 16 | 14 | | 30 | 376 | 500 | 178 | 18 |
| 30 | 339 | 75 | 150 | 45 | 610 | 26 | 24 | | 50 | 560 | 750 | 263 | 17 |
| 40 | 452 | 100 | 200 | 61 | 813 | 36 | 34 | | 70 | 743 | 1000 | 348 | 17 |
| 50 | 565 | 125 | 250 | 76 | 1016 | 46 | 44 | | 90 | 926 | 1250 | 433 | 17 |
| 60 | 678 | 150 | 300 | 91 | 1219 | 56 | 54 | | 110 | 1109 | 1500 | 518 | 16 |
| 70 | | | | | 1219 | 66 | 64 | 25 | 155 | 1064 | 1375 | 499 | - |
| 80 | | | | | 1219 | 76 | 74 | 50 | 200 | 1019 | 1250 | 480 | - |
| 90 | | | | | 1219 | 86 | 84 | 75 | 245 | 974 | 1125 | 461 | - |
| 100 | | | | | 1219 | 96 | 94 | 100 | 290 | 929 | 1000 | 443 | - |

### 12.2.2 Crane Maintenance

The crane maintenance area will be located directly above the auxiliary rooms and systems.

### 12.2.3 Control Room

The control room will be located next to the shipping and receiving area. This provides remote control and status information for all systems (e.g. crane, ventilation, monitoring area).

### 12.2.4 Monitoring/Packaging/Inspection Room

The monitoring area will provide facilities for nuclide determination, dose rate monitoring, contamination monitoring and inspection of waste packages before and after storage if required. Suitable mobile monitoring equipment will be available and placed in a shielded room.

The radiological status of waste packages that have been stored to permit radioactive decay below the ILW limits can be determined prior to removal and shipment to a LLW facility, if appropriate. This room will also be used to undertake any repackaging that is required for these waste streams to permit their disposal as LLW.

In addition to the functions described above the room can also be used for the repacking of any waste packages that may have become damaged during handling operations.

### 12.2.5 Expansion For Decommissioning Wastes (Phase 2)

After approximately 60 years of operation or as required the ISF can be extended to permit storage of decommissioning ILW. This will require extension and the creation of additional storage compartments. It is possible that the additional storage area will be served using the existing crane and facilities. An alternative option would be to construct a separate stand-alone storage facility for decommissioning wastes.

### 12.2.6 Expansion For Packaging/Conditioning/Monitoring Facilities (Phase 3)

At such time as the final disposal facility becomes available, the ISF may be expanded to include treatment and conditioning facilities that might be required to meet the monitoring/conditioning/packaging requirements that are in place at this time. Potential facilities that may be provided in this extension include state of the art systems to:

- Perform Non Destructive Assay services;
- Provide over packaging to meet the transport/disposal requirements;
- Recondition wastes e.g. to separate nuclides and minimise the disposal volume.

An alternative option would be to construct separate stand-alone facilities to accommodate such systems.

## 12.3 Building Layout

### 12.3.1 Building

The arrangement of the ISF is illustrated and shown below:

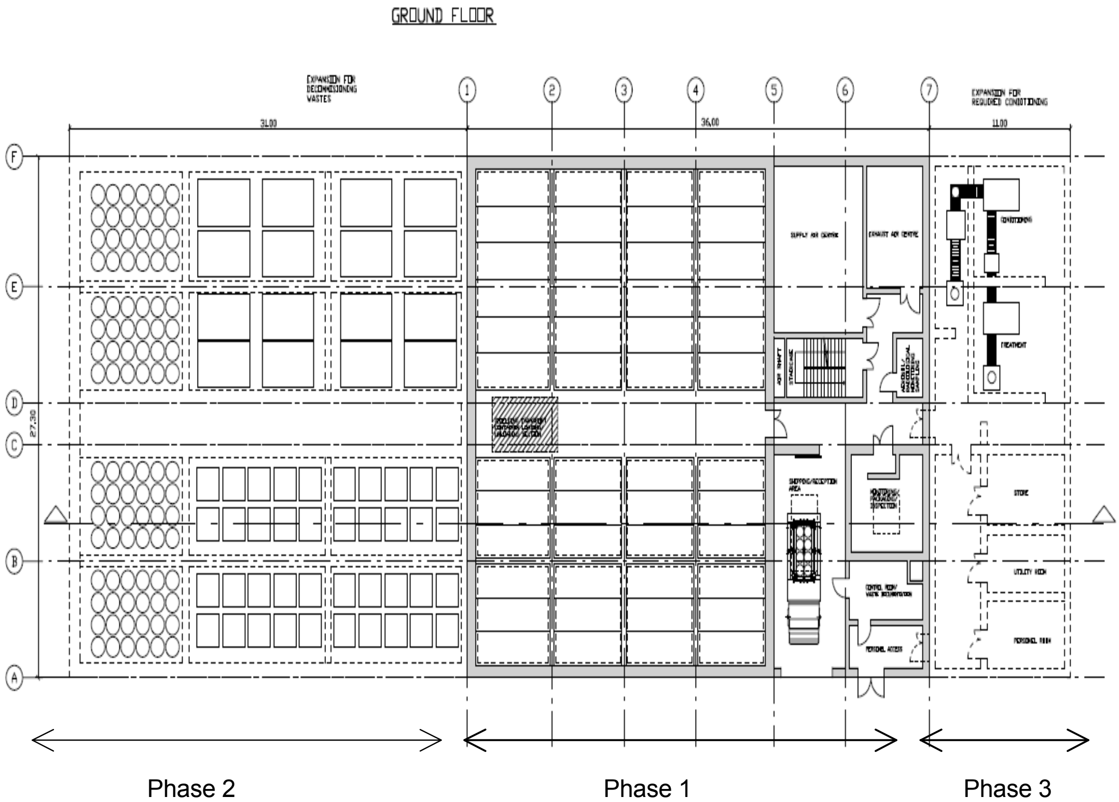



**FIGURE 75: OVERVIEW LAYOUT OF ISF SHOWING THE EXTENSIONS FOR DECOMMISSIONING WASTE (PHASE 2) AND THE OPTIONAL ADDITIONAL TREATMENT AND PACKAGING FACILITIY (PHASE 3)**

GROUND FLOOR:

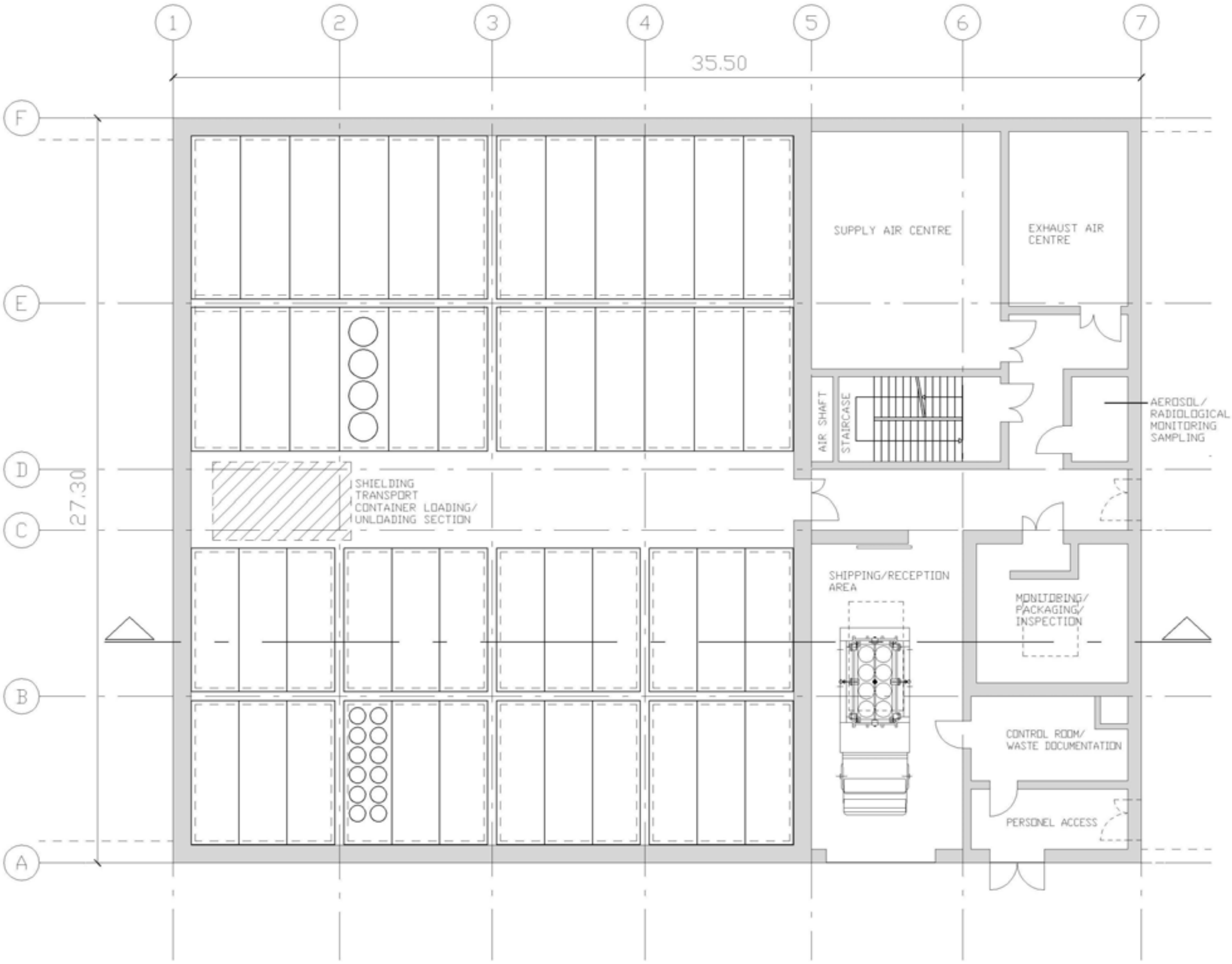



**FIGURE 76: GROUND FLOOR LAYOUT OF ISF**

BASEMENT:

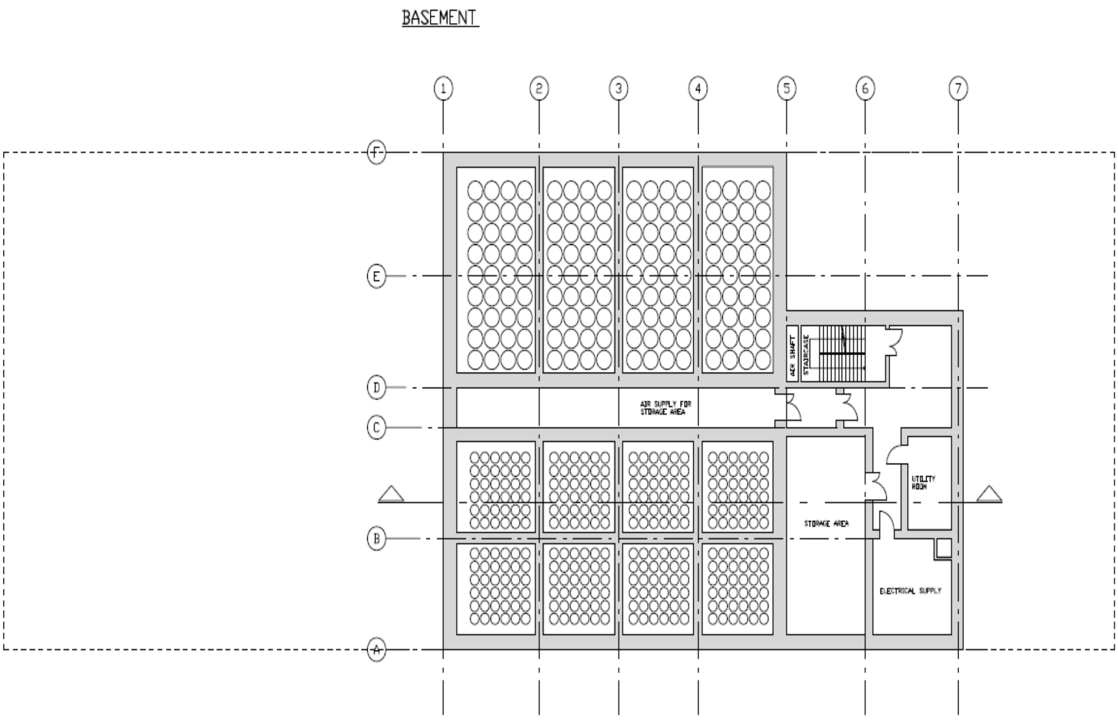



**FIGURE 77: LAYOUT OF STORAGE COMPARTMENTS OF THE ISF**

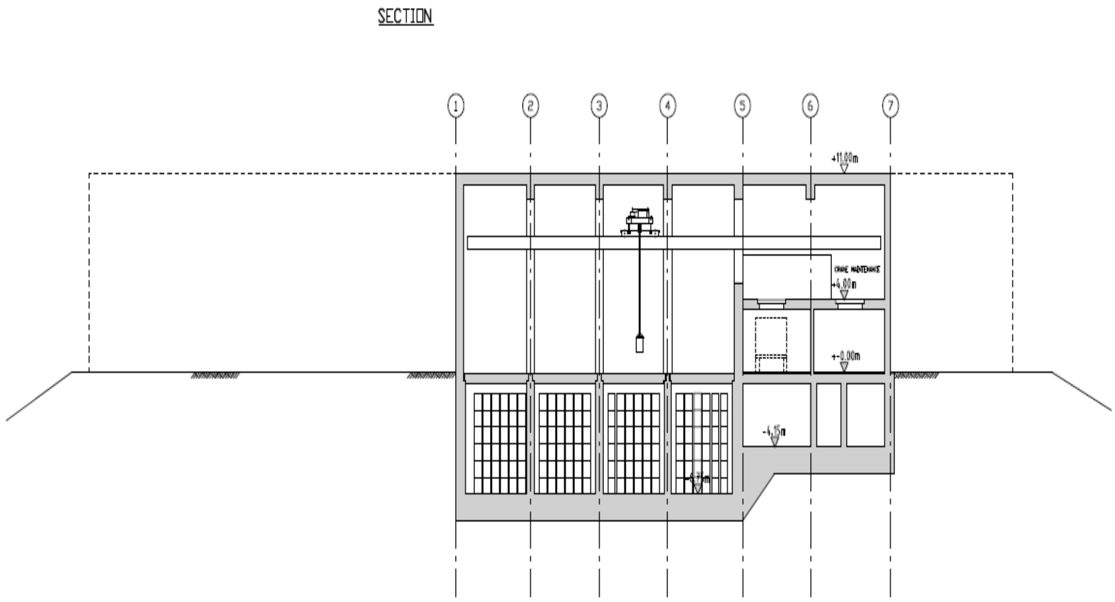



**FIGURE 78: SECTION PLAN OF ISF**

The Interim Storage Facility is a reinforced concrete structure. The storage area is divided into different compartments (compartments with dimensions of approximately 10 m x 5 m and compartments with dimensions of approximately 5 m x 5 m).

The bigger sized compartments will be used for the storage of larger and shielded packages such as cast iron containers. Up to five containers can be vertically stacked in these compartments. The smaller compartments can be equipped with stacking aids and are used for storage of drums.

The compartments are covered using concrete slabs, which forms an access level above the compartments. The slabs are equipped with small concrete plugs, which can be removed to allow for camera or other monitoring equipment access for in situ inspection of the drums and container.

The container truck access control and the transfer area are located in close proximity to the storage area. The service area comprises rooms for access control, monitoring/measuring instrumentation, air supply and exhaust systems and a staircase.

A remote-controlled crane is designed to stack the containers in each compartment.

From ground level the building can be constructed out of pre-cast concrete elements minimising the time required to construct the building and facilitating future expansion of the building if required. The roof slab will be installed at a gradient of not less than 2% to ensure that water does not collect on the roof of the building.

The pre-cast roof girders will span the building in the transverse direction and be supported on the columns which are connected to the braced basement floor.

The outer walls will have a thickness of approximately 60 cm and provide radiation protection to persons outside of the building. The walls will consist of wall beam elements stacked in a mortar bed and connected to the monolithic columns by re-bars anchored in poured joints.

Replaceable thermal insulation and weather protection will be installed on the outer walls. If required this can be replaced as the building ages.

The integrity of the roof waterproofing will be checked by periodic inspection.

No loose contamination is expected to be present in the ISF. The transportation container with the drums or cask iron containers inside will be hoisted completely over the dividing wall, into the adjacent storage area where they will be unloaded. Therefore no decontaminable coating will be installed in the reception area. In the unlikely event of a spill it is expected that this will be confined to a localised area and will be relatively easy to decontaminate.

### 12.3.2 Specific Safety Features

#### 12.3.2.1 Control of Contamination

Waste packages will be stored in closed, lidded compartments within robust containers. The waste packages themselves provide the primary barrier against contamination spread. Only the compartment being accessed for waste emplacement, inspection or retrieval will be open. This will minimise the potential for contamination spread from the storage compartments in the unlikely event of waste package leakage.

All waste package inspection, assay and remediation activities will take place remotely in a shielded cell.

Waste packages are approved containers designed for long term storage. Each waste package is inspected for integrity and monitored for contamination on the external surfaces prior to and during passive storage.

The active ventilation and air conditioning system will be connected to the concrete storage compartments. This system will provide a suitable dry environment for storage and if contamination is released this will be entrained into the building ventilation system and captured in abatement systems, like HEPA filtration. The extract air flow will be monitored for activity and to confirm that the discharge meets the plant RSA authorisation prior to discharge into the atmosphere.

Regular radiological surveys will be carried out by Health Physics personnel to monitor for the presence of any contamination.

#### 12.3.2.2 Seismic Event

The waste packages and stacking regimes are designed such that under design basis earthquake conditions the stability of the packages stacked on top of each other is not endangered. This robustness may be provided by means of fixed waste package support columns.

The ISF building will be seismically qualified in order that it maintains a containment function during and after a seismic event. The ventilation system must close and maintain its integrity.

#### 12.3.2.3 Flooding

The whole site housing the ISF will be elevated above the groundwater table and the periodic high flood level to eliminate the risk of water intrusion. The foundation slab will be designed to prevent water entering the building and concrete storage compartments.

#### 12.3.2.4 Aircraft Crash

The frequency of aircraft crash and potential consequences on the ISF will need to be analysed.

#### 12.3.2.5 Missiles and Explosion Pressure Waves

There is the potential for impacts to the ISF from external events. The ISF will be a fully functional, self contained building that is located separately from other building structures at the site. The design of ISF will minimise the consequences of such an event by placing the storage area under the ground.

## 12.4 Building Operations

### 12.4.1 Normal Operation of ISF

#### 12.4.1.1 Waste Package Transportation

The on-site shipment of the waste packages will be performed by means of a Container Transport Truck. The truck will be equipped with a shielded transport container, if required into which the waste package will be placed. A cover will be fitted over the shielded transport container during transportation. For transportation from WTB to the ISF existing roads inside the supervised area of the NPP will be used.

### 12.4.2 Receipt Of Waste Packages

The serial numbers of the waste packages will be documented via the transport papers on arrival at the ISF.

The data for each waste package (e.g. dose rate, serial number, batch number, analysis of the waste batch, etc.) and storage of the waste package will be documented at the package measuring station. The location of a waste package within the ISF will be unequivocally linked to the waste package number. The method of linkage can be achieved by two means:

1. Input of the information into the PC system in the WTB;
2. Input on arrival at the ISF.

Both methods will ensure that the package data and storage location are appropriately logged for future reference.

The waste packages will be stored in accordance with a predetermined storage plan. The final storage position in the ISF is selected by the ISF operator as required under an optimised storage concept.

### 12.4.3 Waste Package Handling

As the ISF will receive different packages during operations and decommissioning two handling scenarios are considered:

- Handling of packages that can come within a shielded container;
- Handling of packages that are handled without shielding.

The waste packages are managed in the ISF in such way that at all times radiation protection and protection of environment is assured. This is achieved by isolation of the ISF building by means of shield walls, multiple barriers and the confinement of radioactive waste in packages.

The truck bay of the ISF will be separated from the storage area by a concrete wall in order to ensure proper shielding and minimal radiation exposure in the truck passage. For the transfer of the waste packages to the storage position the truck will drive rearwards into the unloading position and the bridge crane will be used to lift the waste packages over the top of the wall.

The truck carrying the loaded and covered waste package in the transport container will enter the ISF via the truck access. After opening the transport container the waste package will be lifted using the appropriate grapple or spreader. If a shielded container is received a traverse will be used.

The waste package will then be transferred using the bridge crane to the final destination. The crane will be remotely operated and its progress visually monitored using cameras in the control room. Only one waste package can be transported at the same time.

Waste packages that are received in shielded transport containers will be removed from the truck and placed at the loading/unloading section above the concrete storage compartments. Here personnel will release the lid of the shielded container and leave the area.

Once the area has been cleared, the crane will remove the shielding lid and place it in the lay down area. If necessary the grapple will be installed to be able to handle the waste package.

The crane will remotely connect to the waste package and places it at the pre-determined storage location.

After the waste package has been unhooked the crane will return and replace the lid on the shielded container. Operations staff will then secure the lid and using the crane place the empty shielded container on the truck.

If required the crane will also place a lid section or cover above the waste package storage location until the next loading.

### 12.4.4 Waste Package Storage

Waste packages will be sorted by pre-determined characteristics such as content, expected decay characteristics or dose rate within the concrete storage compartments.

Waste packages with a higher dose rate will be placed in the middle of a compartment and waste packages with a lower dose rate shall be put in the outer space of the compartments providing shielding to the higher dose packages.

The waste packages will be placed in each compartment and section of the ISF storage area in accordance with the predetermined loading plan.

Where a mix of shielded casks and 200 litre drums is used, the small compartments will be used to store the 200 litre drums in stacks and the large compartments will be used to store shielded casks. The ISF will be filled compartment by compartment. First a complete layer of waste packages will be placed prior to addition of the next layer. Each compartment of drums will consist when filled of six rows and six columns in six layers. Each compartment of the shielded cask will consist when filled of eight rows and eight columns in five layers.

#### 12.4.4.1 Monitoring And Inspection

The lidded storage compartment design with its inspection plugs will at all times allow direct camera access to visually inspect the waste packages at the storage location for signs of leakage.

If waste packages require further inspection they can be removed from the storage position following relocation of the packages that have been stacked above. Using the crane the package can then be shipped directly to the monitoring and inspection room where detailed inspection can be carried out. When required the package can be placed in an over pack and returned to the storage location.

#### 12.4.4.2 Operational Control Of The ISF

Operational control of the ISF will be performed from the control room. Here an operator will coordinate the activities in the ISF and control the systems.

The following systems are linked to the control room:

- Doors and Entrance Area;
- Crane;
- Cameras;
- Ventilation;
- Radiation Monitoring equipment;
- Package Inspection;
- Waste Inventory Database.

The systems are controlled using individually assigned control panels.

### 12.4.5 Radiological Monitoring Of The Facility

#### 12.4.5.1 Contamination Control

Although no loose contamination is expected during normal operations routine surveys will be undertaken to confirm the absence of contamination.

If expansion of the facility is required to incorporate advance treatment technologies for conditioning of wastes prior to disposal in the disposal facility the extended area will be fitted with appropriate control equipment and systems.

#### 12.4.5.2 Monitoring of the Truck

No loose contamination is forseen during normal operation of the ISF therefore no monitoring of the truck is expected.

#### 12.4.5.3 Monitoring of Local Dose Rate

Constant dose rate monitoring will be performed to warn personnel of high dose rates during transfer operations.

Each dose rate detector head and its associated visual alarm lamp will be located together.

#### 12.4.5.4 Sampling Of Activity Release

A noticeable activity release due to a dropped load or container degradation is not expected to occur during normal operations. However, monitoring systems are provided in the WTB to monitor for contamination if required.

In the event of a release measures can be taken to divert the airflow via a HEPA filter prior to discharge to the atmosphere.

### 12.4.6 Decontamination and Decommissioning Of ISF

The waste packages stored in the ISF will be removed from the building before the start of decommissioning. It is not anticipated that the ISF itself will be contaminated. It is intended that its building and its foundations will be demolished using conventional methods and the resulting waste disposed of in accordance with relevant UK legislation.

If an unexpected local contamination event were to occur during the life span of the storage facility the area affected would be decontaminated and the resulting waste sent to the WTB or off-site for conditioning as appropriate.

Following removal of the radioactive inventory the site occupied by the ISF would be delicensed and restored to an end state agreed with regulators and the local planning authority.

# 13 SPENT FUEL INTERIM STORAGE FACILITIES

## 13.1 Introduction

### 13.1.1 General considerations

As already outlined in Chapter 9, a variety of techniques are available to safely store EPR spent fuel assemblies for a period of 100 years. Three of these options are discussed and compared in this chapter, based on an assumption in line with the BERR base case that the facility would be dedicated to a single reactor and would accommodate all fuel assemblies from that reactor.

### 13.1.2 Spent Fuel Export From an EPR Reactor

After 10 years storage in the reactor fuel building spent fuel pool, the fuel assemblies have to be exported to the interim storage facility. This transfer is performed using a shielded transport container. The following main operations are performed for fuel removal from the reactor spent fuel pool:

- An empty transport container arrives at the reactor fuel building on a trailer;
- After shock absorber removal, it is rotated to a vertical orientation and placed on a transfer machine. This machine is used to move the container to the various operating stations where various container operations are performed;
- At the handling opening station, the container cover is removed and the container is fitted with a docking flange. The container is filled with water and is then moved to the biological lid handling station where its lid is removed;
- The transfer machine is then moved beneath the penetration in the lower part of the loading cell;
- The penetration is placed in contact with the container using a sealing device;
- Once the penetration opened, the fuel assemblies are loaded into the container. The transfer machine is moved back to the various stations where container post-loading operations (draining, drying, lid positioning, checks) are performed.

Finally, a loaded container, fitted with shock absorber is exported on a trailer to the interim storage facility.

### 13.1.3 Spent fuel Interim Storage Facilities

The following three spent fuel storage technologies have been considered:

1. A wet interim storage facility: fuel assemblies stored in a pool.
2. A dry interim storage facility: fuel assemblies stored in metal casks.
3. A dry interim storage facility: fuel assemblies stored in vault type storage.

All facilities described here are based on available and proven technologies. It should also be noted that it is assumed that the interim storage facility will be provided with dry unloading installations, whether the storage system itself is wet or dry. The purpose of the unloading installation is to remove spent fuel from its transport container. (Note that metal casks do not require unloading as they form both their own transport and storage container).

Dry unloading will ensure:

- Increased speed of transport container turnover resulting from fewer handling and decontamination operations;
- Reduced liquid effluent and waste production as a result of simpler operations prior to and following transport container unloading;
- No requirement for a specific unloading pool which reduces the overall height of the facility;
- Ability to undertake operations remotely resulting in reduced personnel exposure.

The baseline strategy is that the interim storage facility will be located on the reactor site. However, given the flexibility of the modular construction of a spent fuel storage facility, further consideration will be given elsewhere to the option of a centralised facility to accommodate spent fuel from several reactors, either on the site of one of the reactors or at a separate site. For example, where the spent fuel back end solution could be implemented (reprocessing or disposal). For that purpose and as an example, the vault storage technology is presented and sized for both one or several NPPs.

### 13.1.4 General Design data

Considering the 241 fuel assemblies contained in the reactor core and the core renewal rate over the 60 years of the reactor service life, including periodic shutdowns for maintenance, about 3400 spent fuel assemblies will be produced and will require interim storage. This equates to a mass of about 1800 tonnes of enriched uranium to be stored per reactor.

Taking into account a 10 year cooling period inside the reactor pool, the thermal output of a single spent fuel assembly arriving at the interim storage facility is calculated to be 1400 Watts (2000 Watts if using reprocessed fuel).

Whatever technology and/or location is selected, all spent fuel storage facilities must meet the following safety requirements:

- Ensure criticality control;
- Provide radiation protection;
- Provide safe and secure containment of radioactive material;
- Ensure the integrity of the spent fuel assemblies;
- Maintain adequate cooling of the spent fuel assemblies;
- Facilitate retrieval of the spent fuel assemblies.

This chapter assumes that a dry unloading technology would be implemented, although wet unloading remains a possible option and further analysis will be required before the final selection is made.

In addition to spent fuel, the interim storage facilities may be used for activated core components such as the RCCas. RCCAs can be transferred from the reactor pool to the spent fuel interim storage facility using the same route as per the spent fuel for further decay.

## 13.2 Wet Interim Storage Facility

### 13.2.1 Design Data

The design of the wet storage facility for EPR spent fuel is based on the last generation of La Hague complex storage pools. The PWR spent fuel assemblies stored in each of the four La Hague pools are loaded into individual 9-cell storage baskets. To assist in criticality control, the baskets are made of boronated stainless steel. They are handled using a mast crane. The description of the wet storage technology provided hereafter is based on the use of spent fuel baskets. Rack technology could also be used which may slightly modify facility dimensions and handling requirements but would not result in a significant change to the overall process description.

The interim storage pool for one EPR reactor shall be designed on the baseline of 380 baskets, each containing 9 fuel assemblies and generating a total less than 5 MW of residual heat.

The main design objectives of the interim storage pool are:

- To ensure containment of radioactive materials;
- To ensure the removal of heat generated by the stored fuel;
- To ensure the continuous cleaning of the pool water.

### 13.2.2 Functions and Process systems

The wet interim storage facility is composed of three main building areas:

- A reception hall for fuel assembly transport container receipt and preparation for unloading and shipping;
- A process and utilities building for loading fuel assemblies into baskets and providing all necessary utilities for the storage process;
- A pool building, including an entry channel for the baskets.

The main mechanical functions performed in the facility are described below.

#### 13.2.2.1 Transport Container Reception, Preparation And Shipping

The reception hall takes delivery of the transport container, internal transfer of the container to the preparation areas before fuel unloading and stores or exports the empty transport containers for return to the reactor fuel building. When a fuel assembly is to be exported from the facility (i.e. for final disposal) the reverse operation will take place after loading the fuel assembly into a transport container. This part of the facility comprises the transport container reception hall, the preparation rooms and the process air lock area and is connected to the fuel assemblies unloading unit.

a) <u>*Transport container arrival*</u>

The transport container carriage or trailer will arrive in the reception hall. The reception hall doors, which can only be operated after clearance from the control room, will be opened to receive the transport container and the transport container will be transferred to its specific re-handling location where an external contamination check will be performed.

b) *Transport container re-handling and tilting in vertical position*

In the existing designs, all transport containers arrive in a horizontal position. The transport container is therefore tilted into a vertical position using the reception hall crane, fitted with a specific handling beam.

c) *Transport container positioning onto the transfer trolley*

Once in a vertical position, the transport container is placed onto a transport container transfer trolley. The trolley is designed to ensure that the transport container is held securely and that it will remain safe in the event of an earthquake.

After local clearance by the operator, all the transfer operations are monitored from the control room. An emergency stop can be ordered locally if needed. After the door is opened, the trolley moves to the preparation station.

d) *Transport container preparation*

In the transport container preparation room, before transfer of the transport container to unloading area, the following operations are performed:

- Monitoring of the atmosphere between the inner and outer transport container lids to confirm that the transport container inner lid remains tight;
- Removal of the outer lid;
- Monitoring of the transport container inner atmosphere for krypton to confirm that no fuel has been leaking during the transport. If krypton is detected a specific procedure will be carried out;
- Unscrewing of the inner lid;
- Fixing of a docking adapter onto the inner lid.

e) *Transport container return*

The operations described above are performed in reverse order when preparing fuel assemblies for export from the facility.

#### 13.2.2.2 Fuel Assemblies Handling And Loading In Baskets

These operations take place in the process building and involve loading the fuel assemblies into the storage baskets. A fuel assembly is placed into each of the 9 partition cells of the storage baskets using a crane provided in the unloading cell.

*a) New basket introduction*

New baskets are introduced in the process building via the reception hall and are moved to the spent fuel pool. They are placed into the spent fuel pool using the pool overhead crane.

b) *Transport container inner lid opening*

This operation is performed remotely. After preparation, the transport container is transferred to the transport container docking station and the transport container is docked to the unloading cell. The transport container docking system ensures that the connection between the transport container internal cavity and the spent fuel unloading cell is fully sealed. The transport container inner lid is removed by means of the in-cell jib crane.

c) *Fuel assemblies unloading from the transport container*

Fuel assemblies are unloaded by the unloading cell handling crane. The fuel assemblies are first transferred to a cooling and rinsing enclosure before being transferred into the storage baskets.

Once spent fuel assemblies are unloaded, the internal cavity of the transport container is checked using a camera mounted on the crane. Once it has been confirmed that the transport container is empty the transport container is closed by the jib crane of the docking station.

d) *Fuel assembly cooling and rinsing*

The unloaded fuel assemblies are cooled and rinsed by immersing them in a water-filled enclosure. This operation ensures:

- The detection of leaking fuel by monitoring the water for radioactivity;

  *Note: If a leaking fuel assembly is detected, it can be stored in a dedicated bottle which prevents the spread of contamination in the storage pool. In that case, specific baskets holding 4 bottles are used. No other specific measure is taken once the basket holding the bottles is stored in the pool until it is evacuated.*

- The removal of corrosion and activation products from the fuel elements.

e) *Fuel assembly loading in storage baskets*

The fuel assemblies are introduced one by one into the 9-cell storage basket. The top of the basket is level with the unloading cell floor to avoid unnecessary lifting height of the fuel assemblies. A metallic transfer device, fitted with a lateral door houses the storage baskets and moves them from their loading position in the cell to the reception position at the bottom of the transfer channel.

#### 13.2.2.3 Handling And Storage Of Baskets

The transfer channel from the unloading cell to the spent fuel storage pool is closed by a trap-door to protect the channel from any tools or equipment dropped during maintenance operations inside the unloading cell.

The storage and retrieval of baskets are undertaken with a remotely controlled mast crane. It should be emphasized at this point that handling of bare fuel assemblies inside the spent fuel pool should not be required. Baskets are moved inside the pool building from the transfer device to the basket storage position in the pool, and from their storage position in the pool to the transfer device upon retrieval.

To ensure criticality control, baskets are stored in fixed locations to maintain a safe geometry. This fixed layout also ensures that there is adequate space for moving baskets as a part of normal storage and retrieval operations.

#### 13.2.2.4 Cooling Of Storage Pool

The decay heat produced by the stored fuel assemblies is dissipated by the pool water. To maintain the pool water temperature below 45°C during normal operating conditions (and below 70°C during accidental modes) the interim storage pool includes a cooling system. The system is based on the AREVA-designed heat exchangers composed of:

- A set of heat exchangers submerged in the storage pool (each having a nominal power dissipation of 1 MW);
- A set of compact forced draught external cooling towers dissipating the heat (with an individual power dissipation of 2 MW).

A key feature of this design is that the radioactive portions of the cooling network remain within the pool and the use of closed-circuit systems drastically limits the amount of effluent generated. The base case of storing EPR spent fuel from a single reactor in an interim storage pool has been calculated to generate up to 5 MW of residual heat. The pool will therefore contain 5 heat exchangers and 3 external cooling towers. The cooling towers will be constructed outside the spent fuel facility.

a) *Heat exchanger functional description*

The heat exchangers are submerged vertically and fixed on the storage pool side walls. Each is composed of 3 removable and distinct parts:

- A supporting structure, including a stirring mechanism;
- A bundle of tubular heat exchangers;
- A pump (above the water level) to circulate the pool water inside the heat exchanger tubes.

b) *Equipment outside the pool*

The cooling towers are located outside the pool building. Their tubular water/air heat exchangers have fans to induce forced cooling. A set of pumps circulates the cooling water between the heat exchanges and the cooling towers.

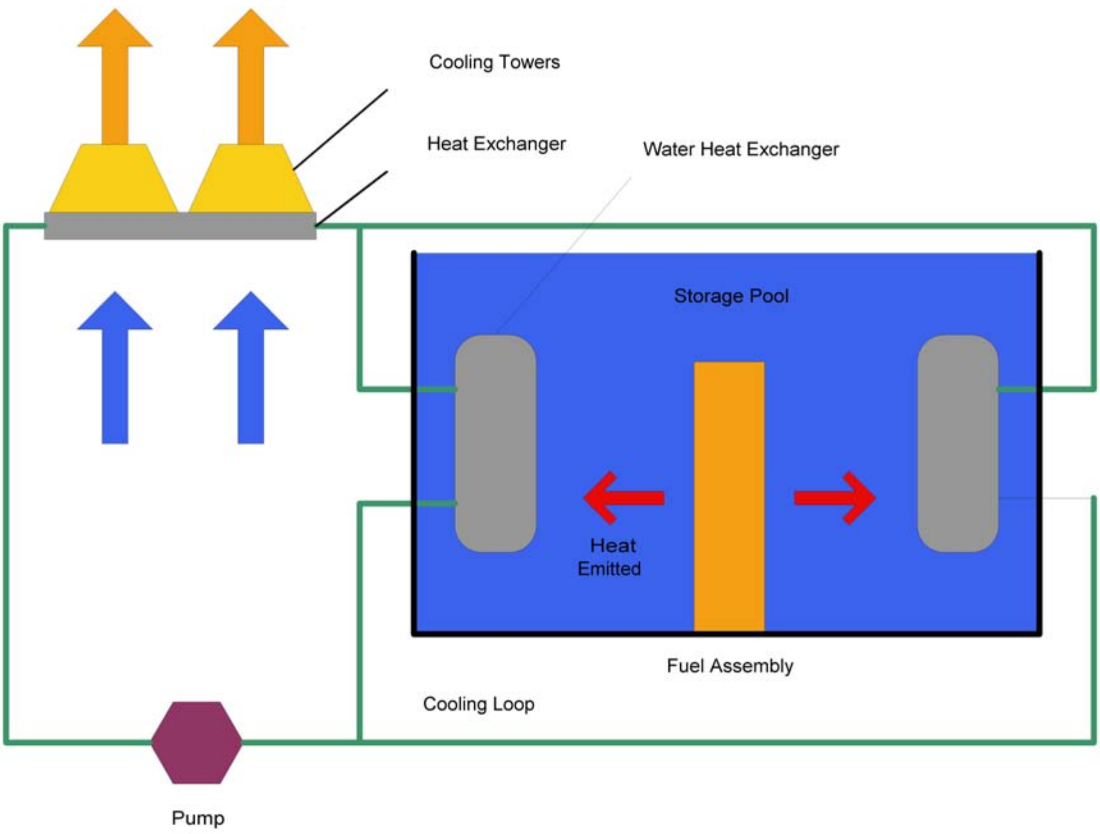



**FIGURE 79:TYPICAL COOLING SYSTEM OPERATING DIAGRAM**

#### 13.2.2.5 Purification Of Storage Pool Water

The chemical contamination of the water must be controlled to ensure the integrity of the fuel cladding, the pool steel lining and the process equipment. Pool cleaning also serves to keep the water clear and to prevent or minimise the build up of material on the bottom of the pool.

a) *Ion exchanger Units - functional description and sizing*

To maintain acceptable chemical conditions in the pool, water is continuously purified by AREVA-designed ion exchangers. These are submerged in the storage pool and each unit is able to purify the storage water of the equivalent of 500 tonnes of uranium fuel. For the UK EPR spent fuel interim storage pool, assuming an average of 1800 tonnes of uranium produced, a total of 3 or 4 units would be needed.

As with the heat exchangers, the ion exchangers are submerged vertically and fixed on the side walls of the storage pool. Each is composed of:

- A supporting structure with water inlet and outlet;
- An above water pump to circulate water.
- Anionic resin and cationic resin, each conditioned in removable cartridges for ease of maintenance.

This design also improves operating conditions since all radioactive portions of the purification network remain within the pool. The ion exchange cartridges form a solid radioactive waste and they need additional treatment before final disposal.

b) *Maintaining water transparency*

Pool water is periodically filtered to remove suspended particles. The filtration network is located outside the pool and is composed of pumps and screens of different sizes. Filtered water is injected back into the bottom of the pool. Skimming of the pool water surface is also undertaken when necessary.

c) *Pool bottom and sides cleaning*

The bottom and sides of the spent fuel pool are cleaned as and when necessary using a mobile suction cleaner. This sucks in water and filters out any solid material for treatment and disposal.

### 13.2.3 Building Layout

#### 13.2.3.1 Applicable Data For Facility Sizing

The baseline facility design to accommodate spent fuel from a single EPR reactor would be approximately 63 m long, 45 m wide and 25 m high. The pool internal dimensions would be about 42 m long, 16 m wide and 9 m deep.

**FIGURE 80: TYPICAL POOL STORAGE FACILITY LA HAGUE**

#### 13.2.3.2 Building Layout Drawings

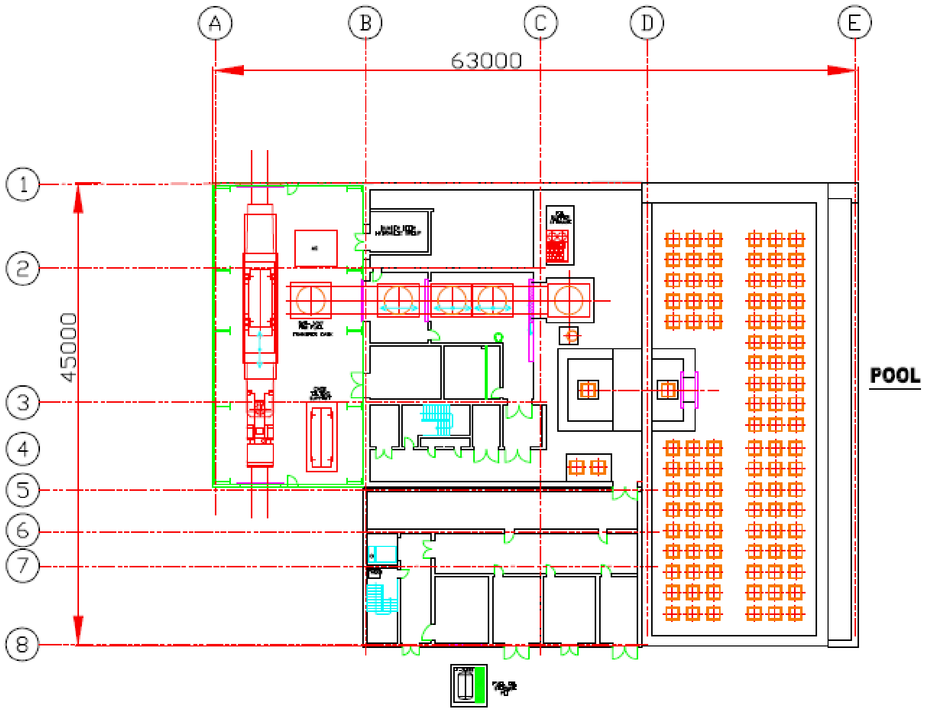



**FIGURE 81: WET STORAGE FACILITY PLAN VIEW**

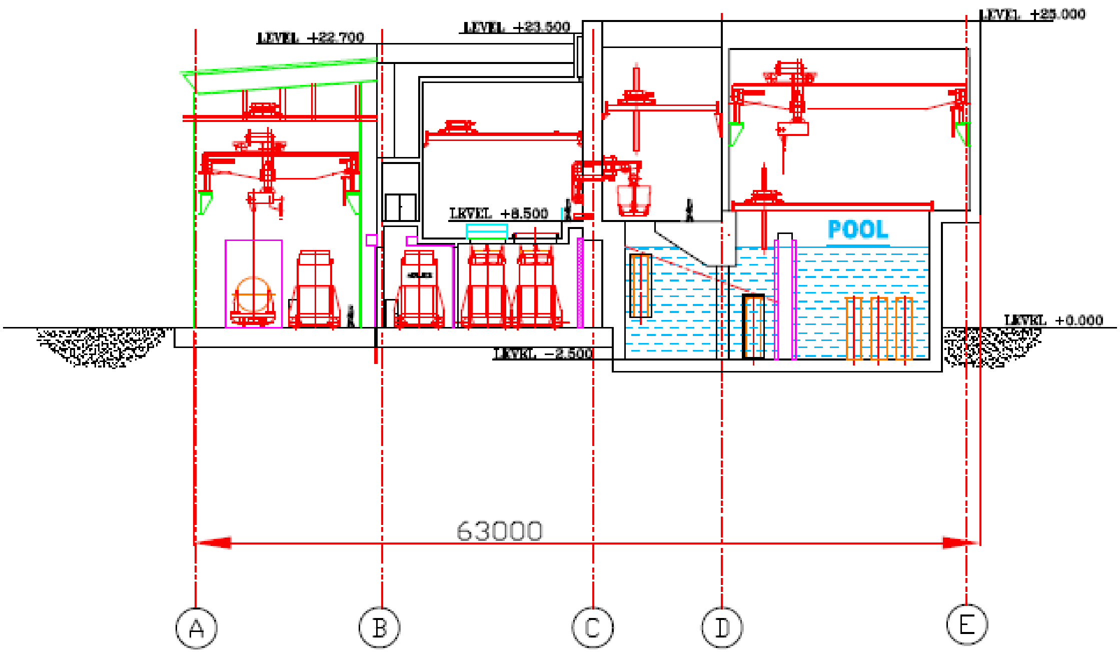



**FIGURE 82: WET STORAGE FACILITY ELEVATION VIEW**

### 13.2.4 Specific Safety Features Wet Storage

Generic safety aspects of spent fuel storage facilities are provided in Chapter 10.3. Only specific safety aspects related to wet storage are described hereafter.

**Direct Radiation**

Specific features that have been incorporated into the design to control dose include the following:

- Shielding provided by the spent fuel transport container;
- Unloading spent fuel from the transport container and loading into the Fuel Storage Pool within a shielded cell and using remote operations to remove the operators from the radiation source;
- Shielding provided by the Fuel Storage Pool water. At all times, the water level is well above the top of the fuel.

Operator doses will be monitored and operations in the facility optimised to ensure compliance with the IRRs. Areas in the facility will be designated in accordance with the IRRs based on the radiological hazard present in that area.

**Radioactive Contamination**

In normal operations the operator shall be isolated from contamination by at least two containment barriers during each stage of the process of fuel unloading and storage. These barriers are as follows:

- Fuel cladding (1st barrier) and for the 2nd barrier:
    - The spent fuel transport container during transport;
    - The unloading cell structure and the transport container connected via the docking system during fuel unloading;
    - And during the interim storage period:
        - The fuel storage pool water;
        - The fuel storage pool liner;
        - The storage pool building.

Further control of radioactive contamination is provided by the provision of an active ventilation system with HEPA filtration abatement system.

Conditioning systems will be provided to remove suspended or dissolved contamination from the Fuel Storage Pool water, and to control water chemistry so as to minimise corrosion of the fuel cladding.

Leakage of water from the Fuel Storage Pool could result in spread of contamination within the facility and in extremis in loss of shielding and cooling for the spent fuel. Containment of the water will be ensured by the stainless steel liner and support framework and by the reinforced concrete structure housing it. It should be noted that the liner will undergo a full radiographic examination during manufacture and that any leakage would be collected through a network of leak pipes incorporated in the pool framework and routed to a leak detection cabinet.

**Criticality**

Criticality safety is achieved by ensuring that the geometrical arrangement of spent fuel is safe throughout the handling and storage operations. The fuel assemblies are delivered in a safe configuration within the transport container. They are then individually transferred to and retained in storage baskets, which maintain sufficient separation between the assemblies to ensure there will be no criticality when immersed in water in the Fuel Storage Pool. In addition to this, the use of boron in the storage baskets will contribute to increasing criticality safety margins.

**Overheating**

The heat generation of spent fuel stored in the Fuel Storage Pool is not considered to be a fire risk, however the fuel and water temperatures need to be maintained within operating parameters to prevent excessive evaporation of the water, spread of contamination and eventual loss of cooling and shielding functions. The heat exchangers provided in the Fuel Storage Pool and the associated air coolers will ensure that the heat generated by the spent fuel is removed and dissipated into the atmosphere. Redundant cooling circuits will be provided: this redundancy, together with the thermal inertia of the Fuel Storage Pool, will provide protection from overheating.

**Internal Flooding**

Internal flooding risks arise from the cooling water in the Fuel Storage Pool, water from the cooling/rinsing enclosure and water in the general building services. These risks are controlled by the following measures:

- Limited volume closed circuits for rinsing, cooling and clean up water systems;
- High-integrity level control to prevent overfilling the Fuel Storage Pool;
- Integrity of the Fuel Storage Pool liner and structure and lack of penetrations at low level - the framework and supporting structure will be designed to withstand earthquake and thermal expansion;
- Leak detection systems incorporated in the Fuel Storage Pool framework with alarms linked to the central control room;
- Ensuring that any credible failure takes water away from areas where water could present a safety issue;
- Safe geometry of arrays of spent fuel assemblies in the process in the event of a flood.

**External Events:**

*Earthquake*

The facility will be designed to withstand a design basis earthquake (DBE). All equipment with a nuclear safety function will be designed to maintain the safety of the facility after the DBE. For example, one important feature of the seismic design is the Fuel Storage Pool, which must perform its safety functions following a DBE. It will be considered as a separate structural unit from the remainder of the building, with the uncoupling of the two structures being achieved through the use of neoprene bearing pads. The water supply network would also be designed to withstand the DBE and be able to maintain the appropriate water level above the spent fuel assemblies.

*Aircraft crash*

Facilities are designed against an aircraft crash.

### 13.2.5 Building Operations

#### 13.2.5.1 General Considerations

There will be three broad operational periods for the storage facility. The first period will be when spent fuel is being received and placed into the interim storage pools. This period will start within 10 years of the reactor start date, since the fuel will need to be stored in the reactor cooling pool for up to 10 years before it can enter interim storage. This first period will finish about 10 years after the reactor ceases operations and the last batch of fuel has completed its initial cooling period. (Note, the reactor is designed for an operational lifetime of 60 years.). There will then be a period when there are no loading or unloading operations, although monitoring of storage conditions will continue. This is expected to last 30 or 40 years, depending upon developments in the UK radioactive waste strategy and the availability of the Geological Disposal Facility. The final period is when the stored fuel assemblies are retrieved and exported to another facility. After this the facility will be decommissioned. The facility is therefore expected to have an operational life of around 100 years.

An EPR reactor core has 241 fuel assemblies and it is assumed that the core renewal is completed every 18 months, with about one third of the fuel assemblies being replaced each time.

The baseline design assumption for the interim storage facility is that it will serve a single reactor with an average of 3400 fuel assemblies requiring storage.

#### 13.2.5.2 Operational Data

Operations in the transport container reception, preparation and shipping building are performed manually by operators, using overhead cranes for transport container handling and a protective platform for transport container preparation operations before docking.

In the process building, all operations are performed remotely from a centralised control room, namely:

- Transport container docking/undocking operations;
- Spent fuel unloading operations and subsequent cooling/rinsing;
- Loading of baskets into the transfer device and movement through the transfer channel.

The mast crane operations in the pool storage building are also fully automated and remotely controlled from the centralized control room although all systems can be operated manually if necessary.

It should be noted that optimal operating conditions are maintained by routine maintenance of pool equipment. The exposure of workers to radiation is minimised by appropriate ventilation, shielding, minimisation of waste, routine monitoring and maintenance of dose records.

## 13.3 Dry Interim Storage Facility

### 13.3.1 The Metallic Flasks Technology – The TN DUO

Spent fuel assemblies can be safely stored in a dual purpose metallic flask, the TN DUO, which is designed for both transport and storage configurations. (TN DUO is an evolution of the TN 24 cask family).

#### 13.3.1.1 Functions and Process system

Dry flask storage allows spent fuel that has already been cooled in the spent fuel pool to be surrounded by inert gas inside a container called a flask. The flasks are typically steel cylinders that are either welded or bolted closed. The steel cylinder provides a leak-tight containment of the spent fuel. Each cylinder is surrounded by additional steel, concrete, or other material to provide radiation shielding to workers and members of the public. Some of the flask designs can be used for both storage and transportation.

The dry flasks could be stored outside or inside a dedicated building.

From the technical point of view, the safety of the storage facility is based on the flask. Indeed the flask has been designed to withstand all accident conditions without taking into account complementary protection system such as a building

#### 13.3.1.2 General description of the TN DUO dual purpose flask

The TN DUO dual-purpose flask is mainly constituted of:

- Two truly redundant and independent containment barriers: a canister (first containment barrier), and a flask body (second containment barrier) closed by two bolted lids;
- A basket to host the spent fuel assemblies;
- The shock absorbing covers in transport configuration, an anti-aircraft crash cover in storage configuration.

The safety is mainly ensured by the mechanical properties of the flask body equipped with its shock absorbing covers. It insures shielding and the transfer/dissipation of the decay heat.

**FIGURE 83: SKETCH VIEW OF THE TN DUO FLASK IN STORAGE CONFIGURATION AND IN TRANSPORT CONFIGURATION**

The canister (the first independent containment barrier) contributes also to the mechanical strength of the flask and to the shielding.

In its storage configuration, a monitoring system including three pressure sensors located on the side of the flask overcomes the requirement to remove the anti-aircraft crash cover in case of repair.

An anti-aircraft crash cover is fitted on the top of the flask and protects the package in case of aircraft crash.

For transport, the TN DUO is equipped with shock absorbers which are suitable for rail and road utilisation.

**Containment**

The TN DUO solution offers two independent containment barriers.

The primary (inner) containment boundary consists of:

- A rolled and welded steel shell;
- A thick bottom steel plate which is welded to the cylindrical body;
- A forged steel top flange which is welded to the cylindrical body;
- A top lid is bolted to the flange with bolts. This lid is equipped with two gaskets (metallic gaskets and/or elastomer gaskets).

The secondary containment boundary is designed on the same concept as the primary, with the following differences:

- Two rolled and welded steel shells (instead of one) are shrink-fitted together;
- The top lid is bolted to the flange with bolts. This lid is equipped with two gaskets (metallic gaskets and/or elastomer gaskets).

During storage, the secondary lid forms a line of defence in case of failure of the primary lid. The inter-lid space is pressurised with helium and continuous monitoring of the pressure allows detection of any decrease of leak-tightness performance long before any radioactive release is possible. The pressurisation of the monitored space ensures that no gas can flow from the inner cavity to the atmosphere.

Both primary and secondary lids are equipped with a single orifice (used for draining water from the cavity, venting, drying, vacuum drying and back-filling with helium). Multiple penetrations are thus avoided.

The flask cavity is filled with helium under atmospheric pressure.

All the confinement sealing surfaces are covered with stainless steel overlay for enhanced corrosion protection.

The two containment boundaries are independent of each other. All challenges that could result from accident conditions would be prevented by the two-layer-thick secondary barrier. The primary containment would thus be subjected to reduced stress therefore ensuring leak-tightness.

**Criticality control**

The flask is fitted with a basket designed to fulfil three functions:

- Mechanical support for the spent fuel assemblies;
- Conduction of decay heat to the cavity surface;
- To ensure sub-criticality of the spent fuel.

The basket consists of slotted aluminium and steel plates. These two types of plates are stacked in a criss-crossed structure to form an array of cells. Basket-length boron aluminium plates are inserted flush with the side surface of these lodgements to ensure sub-criticality. Assembly of the basket is thus simplified.

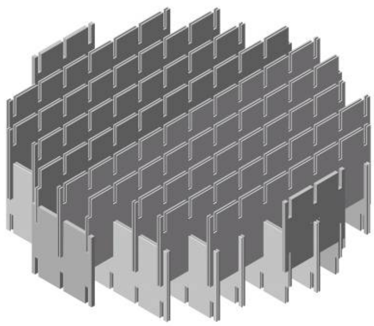


**FIGURE 84: CONSTRUCTION OF THE TN DUO BASKET**

**Shielding**

The TN DUO design features have been chosen to comply with the following storage and transport requirements:

**TABLE 42: DOSE RATES FROM THE TN DUO FLASK**

| Activity | Contact (mSv/h) | 2 meters (mSv/h) |
|---|---|---|
| Storage | < 0.50 (average) | < 0.1 (maximum) |
| | < 10 (maximum) | |
| Transport | < 2 (maximum) | < 0.1 |

The radial shielding has the following features:

- Neutron shielding is mainly provided by a layer thick resin poured outside the secondary containment barrier, between the aluminium heat exchangers;
- Gamma shielding is provided by the rolled and welded steel shells of the two containment boundaries and also by the high density resin neutron shielding;
- Additional neutron shielding is provided near the hollowed plates of the handling device to reduce doses to operators.

The axial shielding is assured by the following features:

- The neutron shielding layer is added to the primary lid to reduce dose rates to operators during handling operations;
- Gamma shielding is provided by the steel material of the two lids and the flask base.

**Heat dissipation**

The TN DUO has been designed to cater for the general trends of spent fuel in Europe: higher enrichment and heat power, higher burn-up. The heat dissipation capacities of the TN DUO flask and variations will evolve accordingly with the operators' fuel characteristics over the life of a power plant.

The basket aluminium plates build heat conducting pathways to the cavity inner surface. The cavity is backfilled with helium - a gas with a high heat conduction coefficient (10 times that of nitrogen). The hottest fuel rod cladding temperature will thus be always kept below the maximum allowed temperature in transport and storage configurations.

Aluminium heat exchangers are pinned on the outer surface of the secondary containment shell to conduct decay heat to the outer surface of the flask. These heat exchangers are manufactured in wave corrugated aluminium plates to ease fabrication.

The outside paint coating is specified for high thermal emissivity.

## Handling and operations

The TN DUO has been designed to offer, as much as possible, similar handling procedures and tools as the TN 24 flask family. The TN DUO handling procedures will never prove to be an issue.

The primary lid is equipped with a single orifice allowing all the operations needed after the fuel assemblies have been loaded into the flask: draining water from the cavity, venting, drying, vacuum drying and back-filling with helium. This orifice is connected to a draining tube reaching the lowest part of the cavity.

The primary lid is positioned precisely to ensure the connection of the orifice tool with the draining tube.

The flask primary lid has been designed with a conical form, to reduce the risk of jamming the lid in the cavity during in-pool operations.

Loadings and unloadings should be performed underwater or in a dry environment hot cell. For loadings performed in a pool, a gasket will seal the gap between the primary and secondary containment barrier. Furthermore this space between these barriers will be pressurised in order to prevent water ingress into this gap.

The TN DUO incorporates an innovative handling system. Trunnions are no longer located on the flask but on a handling tool serving as interface between the flask and the lifting beam. These trunnions will be fitted to hollow cavities designed on the top and bottom of the flask, for vertical or horizontal handling and tilting operations. The validity of this solution has been verified with competent experts.

**FIGURE 85: THE TN DUO WITH ITS INTERFACE TOOLS AND TRUNNIONS**

The flask is designed so as to minimise surface contamination and to facilitate the decontamination of what remains after loading:

- Outer surfaces are treated with a paint that can be decontaminated;
- The design of the flasks eliminates as far as possible retention zones where contamination might become trapped. Where a potential retention could not be eliminated through design, tools and/or methods to protect the zone are indicated in the operating instructions;
- The TN DUO also allows the trouble-free retrieval of spent fuel after the interim storage period by simply opening the 2 bolted lids.

**Shock absorbing covers and anti-aircraft crash cover**
The shock absorbing covers provide the protection required for transport licensing.

The anti-aircraft crash cover provides the protection against an aircraft crash on the storage facility. It is made of carbon steel with paint protection against corrosion.

#### 13.3.1.3 TN DUO Characteristics

All data presented below are only for information use and could be modified after exhaustive studies taking into account the specific data like the environmental conditions, the interface data, shielding criteria, thermal criteria.

**Dimensions and Weights (subject to change according to specific requirements)**

**TABLE 43: DIMENSIONS OF TN DUO FLASK**

| | |
|---|---|
| Flask diameter in transport configuration | < 3200 mm |
| Flask diameter in storage configuration | < 2500 mm |
| Flask height (without shock absorbing covers) | < 5600 mm |
| Flask height (with shock absorbing covers) | < 7500 mm |
| Flask height (in storage configuration) | < 6000 mm |

**TABLE 44:WEIGHTS OF TN DUO FLASK**

| | |
|---|---|
| Flask loaded with the primary lid | < 120 te |
| Loaded flask in transport configuration | < 128 te |
| Loaded flask in storage configuration | < 130 te |

**Flask performance (subject to change according to specific requirements)**
The main characteristics of the flask are presented below; it should be noted that flask performance could be improved by additional aluminium heat exchangers on the outer surface, to account for higher heat dissipation requirements.

**TABLE 45: FLASK PERFORMANCE REQUIREMENTS**

| | |
|---|---|
| Flask capacity | Up to 21 assemblies |
| Type of assembly | EPR 17x17 |
| Average burn up | 65 000 MWd/tU |
| Maximum 235U enrichment | 5 % (in weight) |
| Cooling time | 10 years |

### 13.3.2 Building layout

**Interim storage facility: outside or inside a building**

When the flasks are stored outside, the storage facility consists of a concrete storage pad for the flasks which store the spent fuel. The facility would be located within a protected area of the plant site.

The storage building consists on a storage hall which has the main characteristics: thick concrete structure, wall thickness approximately 0.8-1.2 m, roof thickness 0.6-1 m, and one or two naives in the building.

*Doel Storage Facility, Belgium*

*Surry Storage Facility, US*

The storage facility is designed to operate for 100 years.

Some of the reasons to prefer a building over a dry storage system are:

- An overbuilding will help to reduce the dose rates by almost eliminating skyshine dose and also reduce direct dose. However the flask has been designed to meet any dose rate criteria without a building;
- Nevertheless, the storage of flasks would require that the site fence were some distance from the flasks in order to satisfy off-site dose limits;
- Having a building will protect better from weather conditions. The flask equipped with the protective cover will also provide weather protection.

It can be noted that having a building over the dry storage system has some disadvantages such as the impact on the natural convection heat transfer due to the effect of enclosure from the building.

**Storage building**

The schematic drawing of the storage building for metallic flasks is presented below. Interim storage of an average of 3400 spent fuel assemblies yields to about 170 to 220 flasks, depending on the flask capacity. The approximate dimensions of the storage building are 100 m x 40 m.

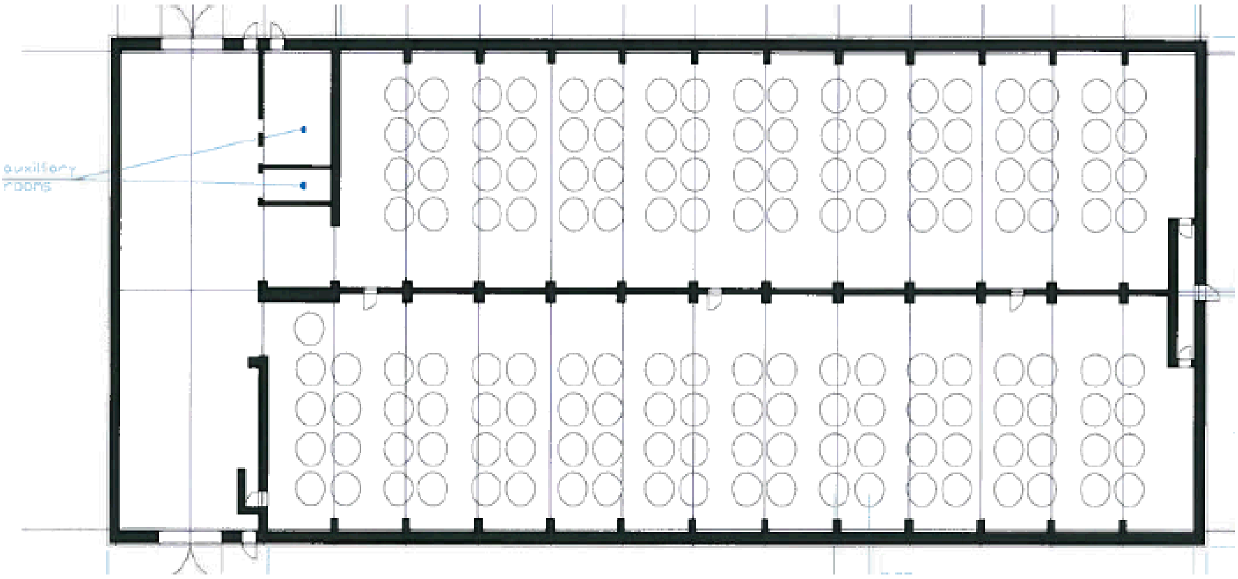


**FIGURE 86: SCHEMATIC DRAWING OF THE STORAGE BUILDING FOR METALLIC FLASKS**

**Outdoor storage facility**

The schematic drawing of the outdoor storage pad for metallic flasks is not presented but the location is the same as the flasks stored inside a building. To store an average of 3400 spent fuel assemblies, there will be around 170 to 220 flasks, depending on the flask capacity.

### 13.3.3 Specific Safety features Dry Storage in Flasks

The purpose of the storage system is to protect persons, property and the environment from the effects of the storage of radioactive material.

This protection is achieved by providing:

- containment of the radioactive contents;
- control of external radiation levels;
- prevention of criticality;
- prevention of damage caused by heat.

The flask complies with normal and accidental conditions as described below.

The flask is designed to ensure the safety of the spent fuel during their storage period and their transportation.

The flask has been designed to meet type B package requirements of the transport regulations issued by IAEA. It is in accordance with TS-R-1, 2005 edition.

The design of the proposed transport and storage systems shall meet all safety requirements imposed by UK Regulations. The flask is designed to meet type B(U)F package requirements. As a consequence, its compliance with the IAEA normal and accident transport requirements is ensured.

**Radioactive contamination**

The flask must provide the containment of the nuclear material retained in the package (§ 656, TS-R-1, 2005 edition).

In normal conditions of storage, the maximum allowed leakage rates for the total primary confinement boundary and redundant seal in normal conditions will be defined so as to ensure that thank to the fact that the volume between the redundant seal is pressurized and monitored, no leakage to the environment occurs.

The leakage analysis will be consistent with an internationally recognized standard, such as the “Safe transport of radioactive materials – Leakage testing on packages” (ISO 12807 - 1996) or the "ANS for Leakage Tests on Packages for Shipment of Radioactive Materials" (ANSI N14.5 - 1997).

**Shielding**

The flask is designed to meet the IAEA radiation levels (§ 572, TS-R-1, 2005 edition):

- Maximum dose rate in normal transport conditions:
    - 2 mSv/h on the external surfaces of the package,
    - 0,1 mSv/h at 2 meters from the vehicle lateral surface.
- Maximum dose rate after regulatory accidental test:
    - 10 mSv/h at 1 meter from surface of the flask.

In the storage configuration, the effective dose rate around the flask will meet the limits required by the regulations.

The flask has either a stainless steel or painted decontaminable surface so that surface contamination should not exceed the limits defined in paragraph 656, TS-R-1, 2005 edition.

**Criticality**

The flask must maintain the fuel in a subcritical condition under all credible normal, off-normal and accident conditions (§ 671 to 682, TS-R-1, 2005 edition).. Analysis is performed for a flask in any position under normal or accident conditions.

**Heat dissipation**

The design will assure that the flask and the fuel material temperatures will remain within the allowable values and criteria for normal and accident conditions.

Fuel cladding maximum temperature at the time of initial flask loading shall be below the anticipated damage-threshold temperatures for normal conditions.

**Tip over**

Tip over is postulated to occur during handling. Handling equipment will be interlocked to not exceed the maximum safe transfer speed.

**Fire**

The design will take into account all fire-related structural considerations, including increased pressures in the containment vessel, changes in material, stresses caused by different coefficient of thermal expansion and/or temperature in interacting materials in order to preserve the safety function of the storage system.

The flask has been designed to withstand a fire caused by an aircraft fuel tanker releasing 6000 litres of kerosene. This fire scenario is considered to be more severe than a fire of 800°C lasting for 30 minutes.

**External events:**

***Flooding***

The flask withstands loads from forces developed by the probable maximum flood including hydrostatic effects and dynamic phenomena.

***Earthquakes***

Flask systems shall be designed against a suite of vibratory ground motions, which together define the Design Basis Earthquake (DBE).

***Aircraft crashes***

Flask equipped with a dedicated cover is designed against an aircraft crash.

***Flask burial after over building collapse***

The safety functions of the flask shall be assured even assuming a complete collapse of the sheltering building onto the flask.

The flask integrity and the heat removal conditions demonstrate that the consequences of collapse over building are acceptable when compared with those for other accident conditions.

#### 13.3.3.1.1 Building Operations

**Operations to be performed at the Reactor Building**

The main operations to be performed at the reactor building are outlined in the following paragraphs.

The main operations to be performed on the TN DUO flask are the typical of those already performed with the TN24™ family flasks, as shown on the following photos.

<u>Preparation of the flask before tilting to receive spent fuel</u>

- Transport of the flask to reactor building (Figure 87);
- Remove half bearing for handling trunnions;
- Remove shock-absorbing covers.

**FIGURE 87: FLASK ON LORRY**

Transfer the flask to preparation area

- Tilt flask to an upright position (Figure 88);
- Transfer of the flask from the vehicle to the preparation area (Figure 89).

**FIGURE 88: TILTING OF TN FLASK TO UPRIGHT ORIENTATION**

**FIGURE 89: LIFTING OF TN FLASK**

Transport container preparation before loading

- Secondary (outer) lid removal (Figure 90);
- Install top protection plate;
- Primary (inner) lid removal.
- Depending on the design of the interim store the flask is either lowered into the fuel loading compartment (Figure 91) is moved under the spent fuel pool to interface with a fuel transfer portal

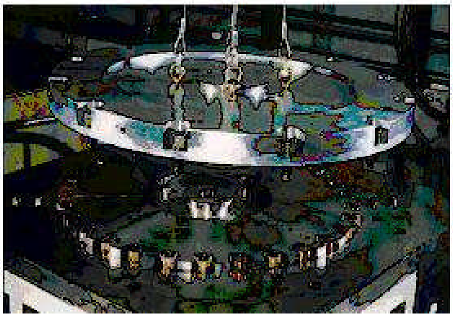


**FIGURE 90: SECONDARY LID REMOVAL**

**FIGURE 91: LOWERING OF FLASK INTO FUEL LOADING COMPARTMENT**

TN Flask Loading Operations

- Fill loading pit with water;
- Load fuel assemblies (Figure 92);
- Install primary lid;
- Drain loading pit.

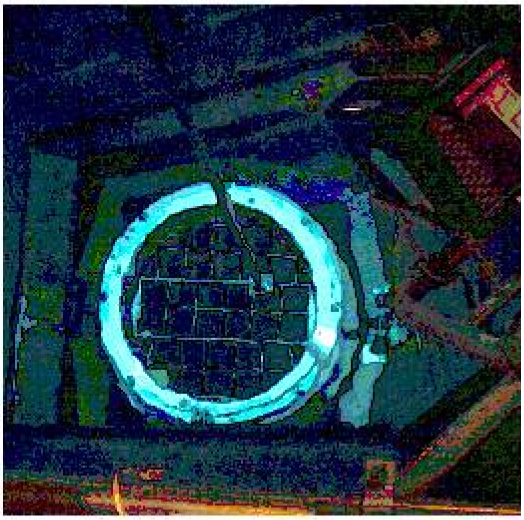


**FIGURE 92: LOADING SPENT FUEL ASSEMBLIES INTO A TN CASK**

Preparation of the loaded TN Cask

- Secure the primary lid with bolts;
- Transfer of the TN casks to the decontamination enclosure;
- TN cask closure and preparation for transfer to the operating place;
- Preliminary leak test of the primary lid before draining of the cavity;
- Drain and dry cavity (Figure 94);
- Further leak tests of the primary lid (Figure 93);
- Install secondary lid;
- The level of external contamination on the TN cask is measured and verified to be acceptable for export from the building.

**FIGURE 93: LEAK TESTING OF PRIMARY LID**

**FIGURE 94:DRAINING AND DRYING OF CAVITY**

Preparation before shipment

- Flask transfer to the trailer;
- Install half bearing for handling trunnions;
- Install shock-absorbing covers;
- Check dose rate and contamination levels.

**Operations to be performed at the interim storage facility**

The main operations to be performed at the interim storage facility would be as follows:

- Receipt and general check of the loaded flask upon arrival at the interim storage facility;
- Perform the regulatory controls and documentation checks;
- Removal of the shock absorbing covers;
- Tilting of the packaging and transfer to the storage area;
- Setting and checking of the monitoring system;
- Setting the protective cover (or anti aircraft cash cover).

**FIGURE 95: DOEL STORAGE FACILITY, BELGIUM**

**Maintenance program of flasks in the interim storage facility**

Maintenance during normal storage is expected to be minimal. Apart from visual inspection of the condition of the flasks and checking of pressurisation of overpressure tank, no further examination is anticipated.

**Visual surveillance**

Every 6 months an external visual inspection of the outside surface of the flask is performed.

**Pressurisation of overpressure tank**

When the pressure in the overpressure tank reaches 4 bars absolute pressure., the tank must be refilled with helium so that the pressure is set again to 7 bars abs., after having checked is the flask for no abnormal leakage.

Due to the specified leak rate of the flask monitoring system, the duration between two refillings of the tank is greater than the storage time 100 years.

### 13.3.4 Dry Storage Vault

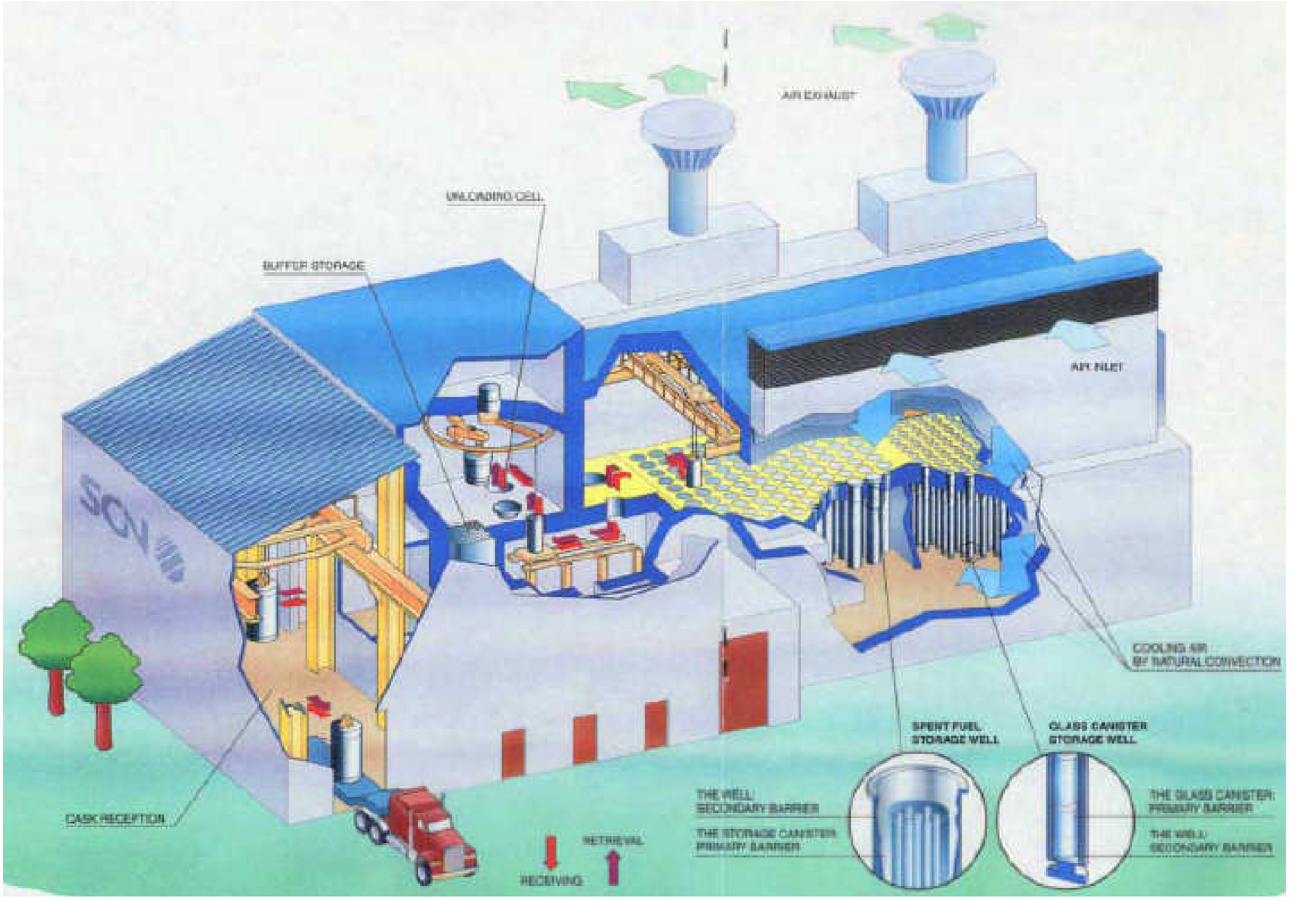



**FIGURE 96: DRY STORAGE VAULT**

#### 13.3.4.1 Design Data

A key safety feature of the AREVA vault storage design is the canistering of fuel assemblies upon receipt. This provides a containment barrier and prevents the spread of contamination in the facility.

The canisters are stainless steel containers provided with aluminium partitions to house fuel assemblies and ensure heat dissipation. The canisters are sealed by welding, filled with inert gas and then lowered into the wells of the vault. For the UK EPR there would be six fuel assemblies per canister and two canisters in each storage well.

Although not discussed in this chapter, it is worth noting that a stand-alone vault storage facility would also be capable of accommodating modules for the storage of Intermediate Level Waste.

### 13.3.4.2 Functions and Process systems

Whether designed for one or several reactors, the vault-type facility is composed of three main zones:

- A reception hall for fuel assembly transport container receipt and preparation for unloading and shipping;
- A process and utilities building for:
  - Loading fuel assemblies into storage canisters; and
  - Providing all necessary utilities for the storage process, in particular the cooling of storage vaults.
- A vault storage building.

The main mechanical functions performed in the vault-type facility are described below:

**Transport container reception, preparation and shipping**

This part of the facility takes delivery of the transport containers, transfers the transport containers to preparation areas and stores or exports the empty transport containers for return to the reactor. When a fuel assembly is to be exported from the facility (i.e. for final disposal) the reverse operations will take place after loading the fuel assembly into a transport container. The unit comprises the transport container reception hall, the preparation rooms and the process air lock area and is connected to the fuel assemblies unloading unit.

a) *Transport container arrival*

The transport container carriage or trailer will arrive in the reception hall. The reception hall doors, which can only be operated after clearance from the control room, will be opened to receive the transport container and the transport container will be transferred to its specific re-handling location where an external contamination check will be performed.

b) *Transport container re-handling and tilting in vertical position*

Transport containers arrive at the facility in a horizontal orientation. As transport container unloading is carried out in vertical position, the horizontal transport container has to be tilted using the reception hall crane, fitted with a specific handling beam.

c) *Transport container positioning onto the transfer trolley*

Once in a vertical position, the transport container is placed onto a transfer trolley. The trolley is designed to ensure that the transport container is held securely and that it will remain safe in the event of a credible seismic event.

After local clearance by the operator, all the transfer operations are monitored from the control room. An emergency stop can be ordered locally if needed. After the door is opened, the trolley moves to the preparation station.

d) *Transport container preparation*

In the transport container preparation room, before movement of the transport container to unloading area, the following operations are performed by operations staff standing on a shielded platform:

- Monitoring of the atmosphere between the inner and outer lids to confirm that the transport container inner lid remains tight;
- Removal of the outer lid;
- Monitoring of the transport container inner atmosphere (krypton detection), to confirm that no fuel has been leaking during the transport. In case of krypton detection a specific procedure will be carried out;
- Unscrewing of the inner lid;
- Fixing of a docking adapter onto the inner lid;

e) *Transport container return*

The operations described above are performed in approximately reverse order when preparing fuel assemblies for export from the interim storage facility.

**Fuel assemblies unloading and placing into canisters**

This part of the facility removes the fuel assemblies from the transport containers and places them into the fuel storage canisters. The fuel assemblies are sealed in air-tight storage canisters which are then moved to the storage vault.

a) *New canister introduction*

An empty canister is introduced into the fuel assemblies transfer tunnel using a handling crane operated from the handling hall.

b) *Transport container inner lid opening*

This operation is remotely controlled. After preparation of the transport container, it is transferred to the docking station and docked to the fuel assemblies unloading cell. The transport container docking system ensures that the connection between the transport container internal cavity and the fuel unloading cell is sealed to avoid the spread of contamination. The transport container inner lid is removed by means of the in-cell jib crane.

**FIGURE 97: TRANSPORT CONTAINER DOCKED TO DRY UNLOADING CELL**

c) *Fuel assemblies unloading from the transport container*

Fuel assemblies are unloaded by means of the in-cell handling crane, fitted with a specific gripper adapted to the fuel assemblies to be handled. The fuel assemblies are transferred to an adapted fuel assembly canister.

Once the spent fuel assemblies have been unloaded a camera mounted on the crane is used to confirm that the transport container is empty and the transport container is closed.

**FIGURE 98: DRY UNLOADING CELL INTERNAL VIEW**

d) *Fuel assembly identification*

Before being placed into a storage canister, the spent fuel assemblies are transferred to a checking station located in front of the unloading cell. The fuel assembly identification is confirmed and a visual check of the fuel assembly integrity is completed.

e) *Fuel assembly canisterization*

The fuel assemblies are removed from the transport container and placed in a storage canister, with six fuel assemblies being placed into each storage canister.

A transfer trolley performs the following tasks:

- carries the empty fuel assembly canister to the canister docking system;
- lifts the fuel assembly canister under the docking station;
- rotates the fuel assembly canister during the welding and contamination; control operations; and
- transfers full canisters to the handling hall.

The fuel assemblies canister docking system ensures an airtight connection between the fuel assemblies canister internal cavity and the fuel assembly unloading cell. The fuel assemblies canister lid is then welded shut, with the welding parameters and the rotation speed being monitored continuously during welding.

**FIGURE 99: WELDING MACHINE**

The canister external docking area is checked for possible spread of contamination with a smear sample after completion of the welding operation.

The canister conditioning operations (evacuation of the air within the canister and filling with inert gas) are then performed. After checking the weld tightness, the fuel assembly canisters are transferred to the storage vault.

Note: As the canister provides a containment barrier, there is no need to check cladding integrity at the receipt of fuel assemblies. Leaking fuel, if any, are placed in canister in the same way as intact fuel

#### 13.3.4.2.1 Handling and storage of fuel assemblies

This part of the facility provides the main storage function for the spent fuel assemblies. It includes all the handling equipment required to place the canisters into the vault and to retrieve canisters and return them to the unloading cell (e.g. for final disposal).

a) *Fuel assemblies canisters storage operations*

All storage canisters are placed into vault storage wells using a handling machine. Each well is closed with a lid and a shielding plug. This ensures the radiological protection of the facility operators. Up to two storage canisters may be stacked into a single storage well. The storage wells are cooled by natural convection of air driven around their external surfaces by the decay heat of the spent fuel.

b) *Storage well preparation*

Storage wells are fitted with a tight lid and a shielded plug. The lid has two seals to ensure an airtight seal and the shielding plug ensures the radiological shielding function for personnel entering in the handling hall. A protective plate creates a flat and uniform area on the whole slab.

Before a canister arrives, the storage well to be used is identified and prepared by:

- Removal of the protective plate removal, unscrewing and removal of the lid. These are undertaken manually by an operator;
- Operators leave the vault;
- The vault door is locked;
- The shielding plug is removed by the remote handling machine.

The well is then ready to receive a storage canister.

c) *Canister handling and loading in wells*

The canisters are conveyed on a transfer trolley through the transfer tunnel to the handling station. The canister is gripped by the handling machine fitted with a canister gripper and is automatically transferred to the allocated storage well location. The handling machine lowers the canister into the well. The stacking level is checked before the storage canister is released. The canister can only be released when a “no load” sensor is activated, thereby preventing a canister being dropped into a well.

**FIGURE 100: HANDLING MACHINE**

d) *Storage well closing*

Once the canister is positioned in the well and the well is full (i.e. contains two canisters), the handling machine replaces the shielding plug.  Operators can then re-enter the vault and replace the lid and the protective plate.

e) *Storage wells internal atmosphere checking*

Periodic monitoring of the primary and secondary barrier integrity is undertaken by analysing the internal well atmosphere. If a leak is detected from one of the storage canisters, both canisters are retrieved from the well, transferred to the unloading cell and opened. The fuel assemblies are placed into two new canisters docked in the transfer tunnel. Once emptied, the failed canisters are dismantled, placed into a specific canister and stored in a dedicated well. The new canisters are processed in the normal way and returned to the vault.

*Note: if canisters are known to hold leaking fuel, they are subject to more frequent controls in the storage well.*

f) *Storage of fuel assemblies canisters*

Each dry storage facility module consists of two storage vaults. The dry storage vault modules are constructed as separate units with separate ventilation systems. This modular approach makes the design flexible and allows the facility to respond to a need for increased storage capacity.
Each vault includes a number of wells to accommodate the fuel assembly canisters. The canisters are stored vertically, stacked on 2 levels, inside the air-cooled storage wells.

g) *Canisters retrieval*

At the end of the storage period, canisters will be retrieved and transferred into the dedicated unloading cell before being exported for final disposal or another long-term storage facility.

**Cooling of storage vaults**

Each storage module (comprising 2 vaults) is equipped with an independent cooling circuit. The air cooling system is based on natural convection through a jacket around each vertical storage well and up a stack. Fresh air is supplied through air inlets into the vault bottom plenum. An intermediate plate is located inside each vault to ensure the separation between the low inlet plenum and the vault internal volume. Cooling air is circulated in the well tube annulus and warm air is removed through the module upper plenum before being exhausted through the stack.

All discharges will be continuously monitored. The cooling system is designed to maintain concrete walls and stored package temperatures below the maximum permissible values.

**FIGURE 101: VIEW OF COVRA HABOG FACILITY, NETHERLANDS**

### 13.3.4.2.2 Building layout and description

**Applicable data for facility sizing**

The typical thermal output of 1400 W per EPR spent fuel assembly after ten years cooling means that canisters can be loaded with up to 6 spent fuel assemblies whilst maintaining acceptable conditions for safe fuel storage. For one EPR storage facility, the overall dimensions are approximately 85 m long, 42 m wide and 25 m high.

The stacks for the storage building are likely to be about 45 m high, although a detailed site specific assessment will be required to establish the stack height required to achieve adequate dispersion.

A storage facility designed for fuel assemblies arising from five EPR reactors could have one of several configurations. For example, a facility could be designed lengthwise with modules constructed side by side as needed. The configuration would be determined partly by the constraints imposed by a specific site.

In addition to the process rooms and the vault store there will be a utilities and services building dedicated to auxiliary functions such as ventilation, control, electricity supply, personnel; access, etc.

All the buildings, except for the reception hall building, are made of reinforced concrete for the walls and the roof.

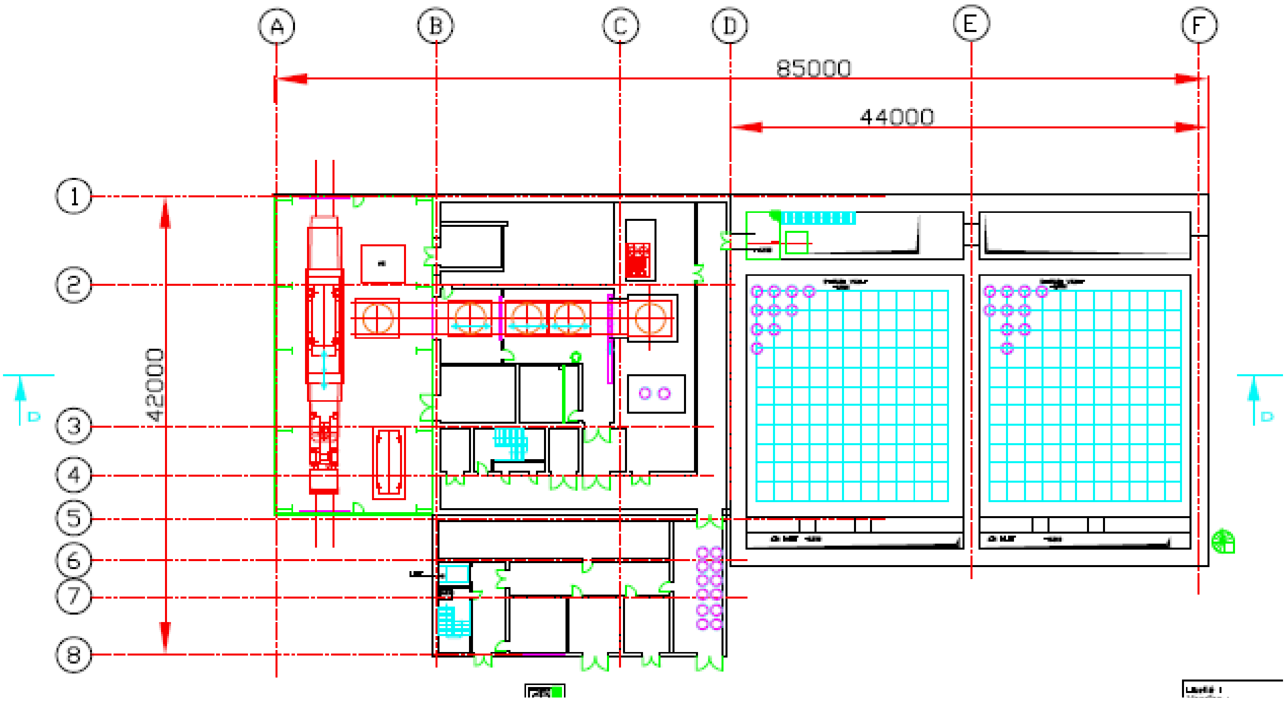



**FIGURE 102: VAULT STORAGE BUILDING – PLAN VIEW**

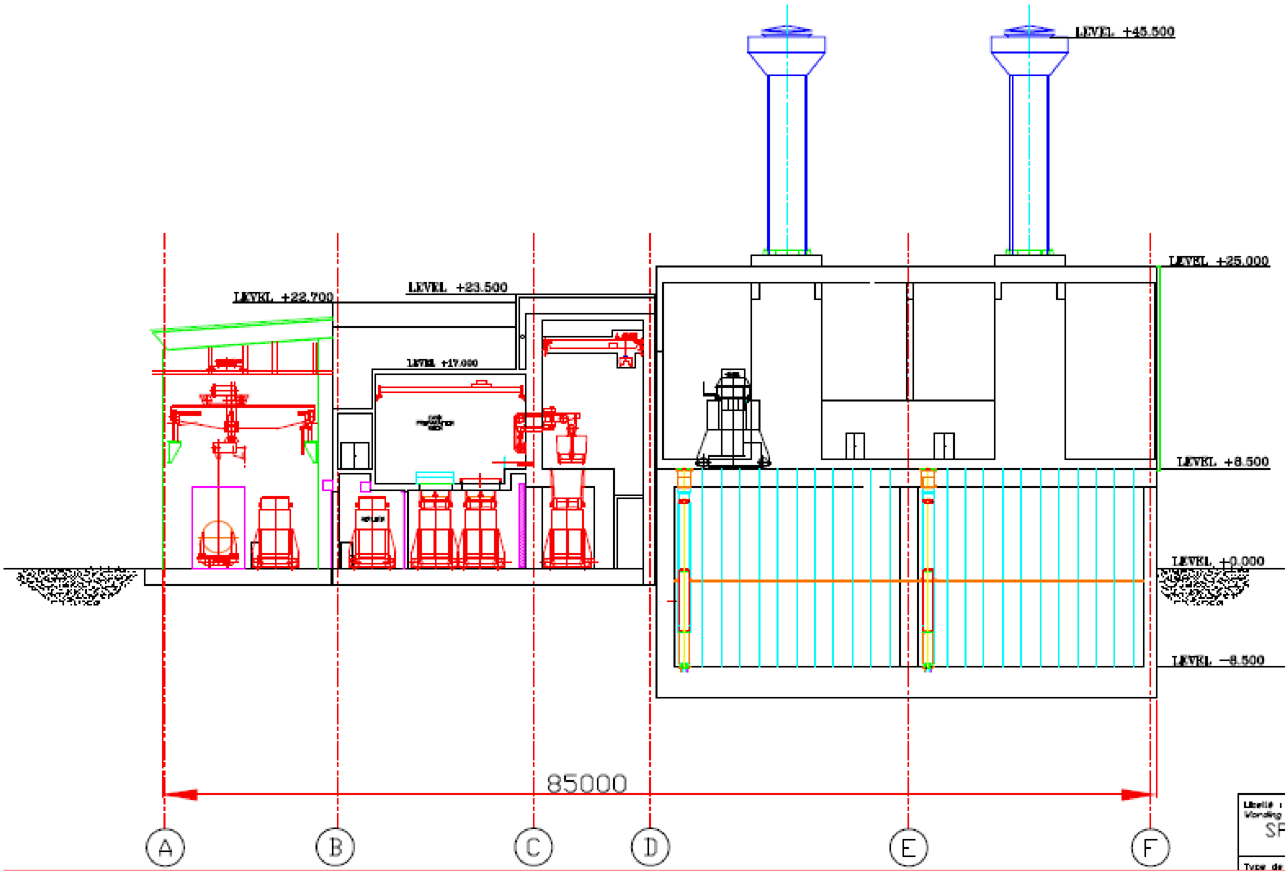



FIGURE 103: VAULT STORAGE BUILDING – SIDE ELEVATION

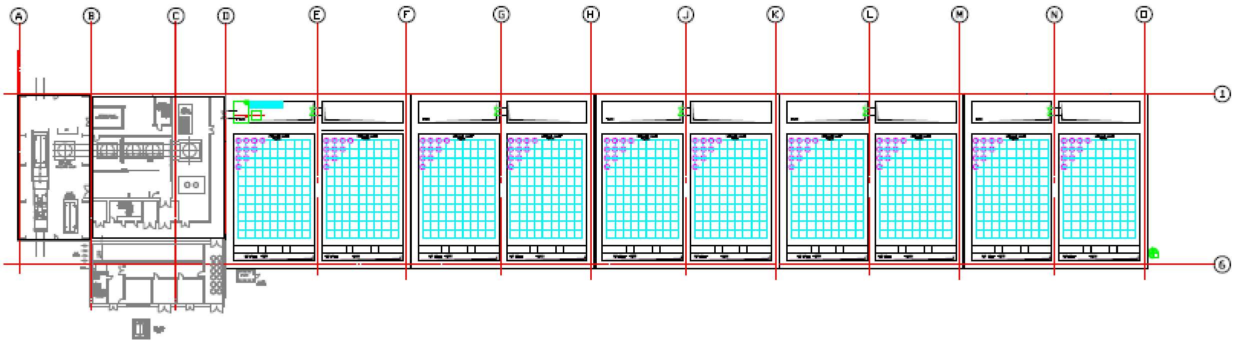


FIGURE 104: VAULT STORAGE BUILDING – PLAN OF A FACILITY SERVING MORE THAN ONE SITE

#### 13.3.4.2.3 Building Operations

There will be three broad operational periods for the spent fuel storage facility.

The first period will be when spent fuel is being received and placed into the storage vaults. This period will start within 10 years of the reactor start date, since the fuel will need to be stored in the reactor cooling pool for up to 10 years before it can enter dry interim storage. This first period will finish about 10 years after the reactor ceases operations and the last batch of fuel has completed its initial cooling period. (Note, the reactor is designed for an operational lifetime of 60 years.)

There will then be a period when there are no loading or unloading operations, although monitoring of storage conditions will continue. This is expected to last 30 or 40 years, depending upon developments in the UK radioactive waste strategy and the availability of facilities for further storage or disposal.

The final period is when the stored fuel assemblies are retrieved and exported to another facility. After this the facility will be decommissioned. The facility is therefore expected to have an operational life of around 100 years.

An EPR core reactor has 241 fuel assemblies; it is assumed core renewal is completed every 18 months, with about one third of the fuel assemblies being replaced each time.

The baseline design assumption for the storage facility is that it will serve only a single reactor with an average of 3400 fuel assemblies requiring storage.

Discharges arising from the ventilation of building rooms (i.e. personnel access areas and fuel handling cells) will be filtered.

### Mode of Operations

Operations in the transport container reception, preparation and shipping building are performed manually by operators, using overhead cranes for handling and a protective platform for transport container preparation operations before docking.

In the process building, all operations are performed remotely from a centralised control room, namely:

- Transport container docking/undocking operations;
- Spent fuel unloading operations and identification;
- Canister loading and transfer for storage in the vaults.

#### 13.3.4.2.4 Safety Features for a Vault-Type Dry Interim Storage Facility

Generic safety aspects of spent fuel storage facilities are provided in previous section. Only specific safety aspects related to vault storage are described hereafter.

**Direct Radiation**

Specific features that have been incorporated into the design to control dose include the following:

- Shielding provided by:
  - the spent fuel transport container;
  - the remotely-operated unloading cell;
  - the transfer tunnel;
  - the remotely-operated canister handling machine;
  - the storage vault structure.
- Fully shielded, interlocked features for interfacing:
  - the spent fuel transport container with the unloading cell;
  - the canister handling machine with the transfer tunnel;
  - the canister handling machine with the storage well in the vault.

Operator doses will be monitored and operations in the facility optimised to ensure compliance with the IRRs. Areas in the facility will be designated in accordance with the IRRs based on the radiological hazard present in that area. Some areas of the facility (for example welfare areas) will not require designation in accordance with the IRRs.

**Radioactive Contamination**

The spent fuel storage facility must provide an environment which protects the long term integrity of the fuel assemblies. This is important as the spent fuel will need to be transported and repackaged prior to disposal.

In normal operations the operator and environment will be isolated from contamination by at least two containment barriers during each stage of the process of fuel unloading and storage. These barriers are as follows:

- Spent fuel transport container during transport;
- Unloading cell structure and the transport container connected via the docking system during fuel unloading;
- Seal-welded spent fuel storage canister, filled with inert gas;
- Tight storage wells in the vault;
- Facility buildings.

Assurance of containment integrity after closure is provided by scheduled checking and monitoring.

All equipment, including the spent fuel transport container, will be manufactured with surface finishes that aid decontamination.

Further control of radioactive contamination within buildings is provided by an active ventilation system with HEPA filtration abatement system.

**Criticality**

Criticality safety is achieved by ensuring that the geometrical arrangement of spent fuel is safe throughout the handling and storage operations.

Fuel assemblies are individually located in the transport container and in the storage canister by fixed partitions, which maintain sufficient separation between the assemblies to ensure there will be no criticality under any conceivable conditions,. Fuel assemblies in the unloading cell are handled in an adapted canister, with a similar arrangement of partitions.

Procedural controls will be put in place to prevent fuel assembly handling errors such as loading extra assemblies into the unloading cell.

**Overheating**

The facilities will be designed to maintain the temperature of spent fuel assemblies within the identified safe limits under all expected combinations of radioactive decay heating and ambient temperature. During storage and transport, spent fuel cooling is achieved entirely by passive processes, aided by features such as:

- heat conducting design of canister partitions;
- selection of a gas with good heat transfer properties for backfilling canisters;
- vault design with air channels to provide good convection in all conditions.

Long term temperature control in the fuel storage vaults is important for preventing the progressive oxidation of $UO_2$ in the fuel to $U_3O_8$ which has a much lower density. Conversion of the uranium dioxide in such a manner could lead to failure of fuel cladding due to internal expansion of the fuel.

**Loss of electrical power supply**

With the spent fuel cooling system being passive by design, the continued cooling of spent fuel will not be affected by a power failure.

**External events**

Flooding

The dry storage vault is such designed that it will remain sub-critical even if flooded.

Earthquake

The facility will be designed to withstand the design basis earthquake (DBE). All equipment with a nuclear safety function will be designed to maintain the safety of the facility after the DBE.

The process building, in particular the unloading cell, and its associated vessels and pipework will be designed to withstand the DBE without failure of containment.

Storage vaults will be designed to withstand seismic loads while retaining their shielding and cooling functions, and protecting the canister, fuel and handling features.

Structures will be designed with sufficient rigidity to limit displacements in the DBE so that the canister remains recoverable. The most onerous case is likely to be an earthquake during spent fuel transfer between transport container and unloading cell, or into or out of the canister handling machine. Positioning features will be designed accordingly.

Aircraft crash

Facilities are designed against an aircraft crash.

# 14 TRANSPORTATION

## 14.1 Regulatory Baseline

The British Regulations are based upon the International Atomic Energy Agency's (IAEA) Regulations for the Safe Transport of Radioactive Materials.

The objective of these Regulations is to protect persons, property and the environment from the effects of radiation during the transport of radioactive material. This protection is achieved by requiring:

(a) containment of the radioactive contents;
(b) control of external radiation levels;
(c) prevention of criticality; and
(d) prevention of damage caused by heat.

These requirements are satisfied firstly by applying a graded approach to contents limits for packages and conveyances and to performance standards applied to package designs depending upon the hazard of the radioactive contents. Secondly, they are satisfied by imposing requirements on the design and operation of packages and on the maintenance of packaging, including a consideration of the nature of the radioactive contents. Finally, they are satisfied by requiring administrative controls including, where appropriate, approval by competent authorities.

It can be noted that according the "Memorandum of Understanding on Approval of Certificates relating to the Safe Transport of Radioactive Material for Civil Purposes" issued on February 2006 by the Department for Transport of the United Kingdom and the Directorate general for nuclear safety and radiation protection of the Republic of France, each competent authority accepts approvals issued by the other competent authority as evidence that the design of a package meets the requirements for package design contained in the IAEA Regulations. In other words, approvals granted in France by TN International would be accepted in the United Kingdom.

### 14.1.1 Basic Regulations

The transport solutions shall comply with the following main current transport regulations:

- Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations 2007 (CDG 2007);
- "Regulations for the Safe Transport of Radioactive Materials" of the International Atomic Energy Agency (IAEA) - TS-R-1 (Transport Safety Rev.1) Edition 2005;
- European Agreement concerning the International Carriage of Dangerous Goods by Road (ADR);
- Regulations concerning the International Carriage of Dangerous Goods by Rail (RID),
- International Maritime Dangerous Goods (IMDG) code – IMO.

### 14.1.2 Guides

The following safety guides are used for the transport of radioactive material:

- Guide to an Application for UK Competent Authority Approval of Radioactive Material in Transport, Department of the Environment, Transport and the Regions, January 2001;
- Safety Guides N° TS-G-1-1, "Advisory Material for safe Transport of Radioactive Materials", International Atomic Energy Agency (IAEA), Edition 2002.

## 14.2 Waste And Spent Fuel Transport Arrangements

### 14.2.1 Waste Package Transportation

TNI has also developed and implemented various solutions for nuclear waste packages which require specific transportation arrangements.

The following are examples of suitable containers for different types of irradiated waste. Other types of packages, which also comply with international transport regulations could be chosen by a utility.

### 14.2.2 Spent Fuel Transportation

For the transport of spent fuel offsite, an IAEA type (B) transport container is required.

TN International (TNI) has already designed and implemented various solutions that comply with transport regulations. With reference to the specific case of EPR spent fuel, the first solution that would be considered would be the TN-DUO type transport and storage cask, described earlier in this document, which is a storage solution compatible with transport requirements. It is important to note that other types of package designs may be employed.

For transportation purposes only, TNI has designed and implemented a number of transportation casks which have been routinely used since the 1980s. More than 200 transports of spent fuel are undertaken every year in Europe.

As shown below, the TN$^{TM}$13-2 cask can be adapted for the carriage of EPR spent fuel.

**Main design characteristics of the TN$^{TM}$13-2:**

Overall length with shock absorbers: 6670 mm
Maximum diameter with trunnions:2500 mm
Maximum weight loaded with shock absorbers: 113.5 tons
Internal diameter of the cavity: 1220 mm
Weight empty: 103.4 tons
Pay-load: 12 PWR 15x15 or 17x17 (reactor 1300 or 1450 MW)

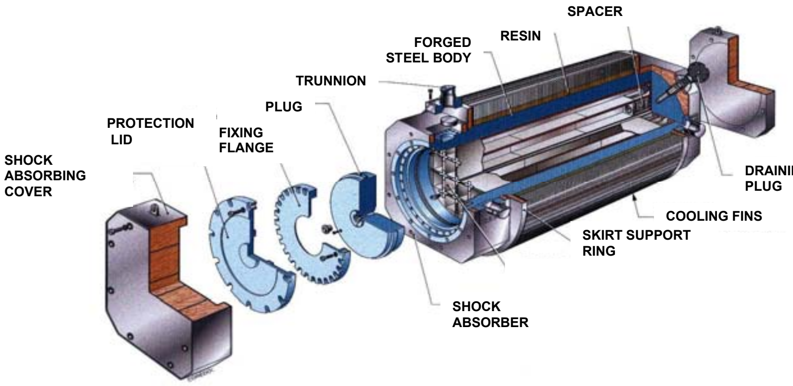



**FIGURE 105: TN™13-2 CASK**

Cask performance:

- PWR 15x15 or 17x17 (reactor 1300 or 1450 MW);
- Maximum thermal power: 64.5 kW without canopy;
- Maximum enrichment in U5: 4.3%;
- Maximum burn-up: 52,500 MWd/tU;
- Minimum cooling time: 120 days.

The TN™13-2 is composed of a body in forged steel structure designed for fire resistance and containment integrity, both after 9 m regulatory drop tests and the punch test.

The subassemblies of the cask satisfactorily perform the various functions that are essential to safety. The transport cover acts as a mechanical shock absorber during severe impacts or shocks, and thermal insulation for the cask lid in case of fire. The body and the closing system guarantee that radioactive materials will be confined, reduction of radiological radiation emitted by radioactive materials and dissipation of thermal power released by radioactive materials.

This cask is composed of the following subassemblies:

- The basic structure is a cylindrical receptacle formed mainly from a thick shell made of carbon steel comprising an internal cylindrical cavity in which the spent fuel to be transported are placed. The thickness of the forged steel shell forms the main gamma shielding;
- The body is equipped with bolted removable trunnions for handling, tilting and stowing,
- Inside the cavity, an aluminium basket is fitted and provides a structural support for the spent fuel;
- Surrounding the cylindrical forged body resin provide neutron shielding features of the cask. The lid is also equipped with a similar neutron shielding. Shaped copper plates separating resin compartments, crimped along their length by welding cover plates forming the outer envelope, and in tangential contact with the forged shell, ensure the thermal evacuation of the heat power released by the spent fuel to the outer shell, itself in contact with ambient air;
- Two shock absorbers are placed on the top and the bottom of the cask for lateral and axial drops.

The TN™13-2 belongs to the Mark2 family cask designed by TN International, which includes also the TN™12-1 and the TN™12-2. These designs have been licensed for several years in France and they have been also licensed for use in Germany, Belgium, the Netherlands, Japan, Switzerland and Sweden. Other types of casks are available for use by the utility.

### 14.2.2.1 DV 78 Package

DV 78 is the standard transport unit for LLW (Low Level Waste) transfer from AREVA's plants to the ANDRA disposal sites.

The design is based on a 20' ISO hard top container with a single locking pad for both doors and removable roof. It also has a shock absorbing floor and interchangeable racks for securing various contents.

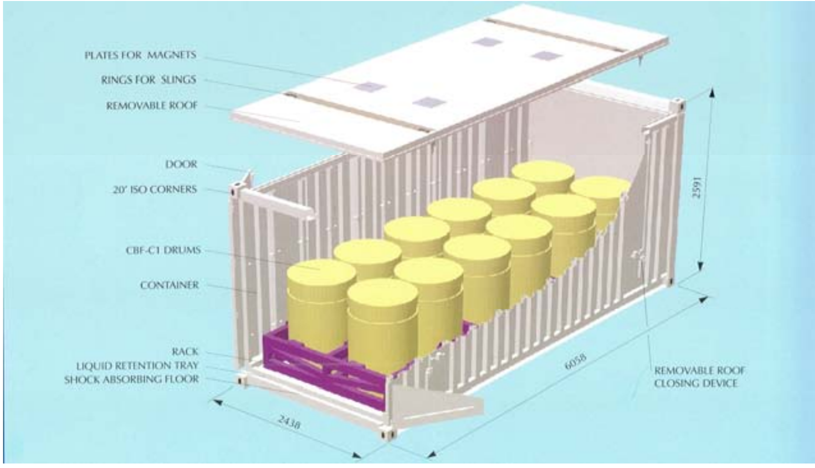



**FIGURE 106: DV 78 PACKAGE**

#### 14.2.2.1.1 Package performance

Content:

- Alpha waste in industrial drums;
- Radiolysable and non radiolysable technological waste;
- Low activity waste < $10^{-4}$ $A_2$/g.

Waste packaged in:

- Intermediate boxes or fuse boxes;
- 118, 200, 213, 223 litres metallic drums or 200 litre drums;
- Concrete shell;
- Metallic boxes;
- Storage containers.

Package characteristics:

- Overall dimensions 6058 x 2438 x 2591 mm;
- Total gross weight 24 000 kg;
- Maximum payload 19 000 kg.

#### 14.2.2.2 RD 26 package

The RD 26 package is a B(U) packaging which was developed for the transport of a single 118 litre drum.

The RD 26 package has a cylindrical shape and is mainly composed of:

- a body made of an internal stainless steel containment, a specific material acting as a neutron shielding and a thermal protection, and an outer stainless steel envelope;
- a lid bolted onto the body and equipped with an orifice in order to take a sample of the gas cavity before opening the cask;
- two elastomer gaskets between which a check of the leaktightness can be performed before a transport operation;
- a shock absorber cover bolted onto the body and made predominantly of wood;
- a short frame for handling operations.

The maximum gross weight is 610 kg for the package containing a 118 litre drum of 150 kg. The height is 1145 mm (with the shock absorber) and the outer diameter is 860 mm. During transport, each drum is placed in a RD 26 cask and twelve of them can be transported by road at the same time in a DV 78 container.

##### 14.2.2.2.1 Package performance

Content:

- Alpha technological waste;
- Uranium and plutonium oxide (UO2, UO3, UO4, U3O8, PuO2).

Waste packaged in:

- 100 or 118 litres metallic drums;
- The oxide is packaged in 118 litres metallic drums in the form of powder, pellets, liquids or sections of fuel rods.

#### 14.2.2.3 CC102 over pack

The CC102 package is an example of a package with possibility of automatic loading in the reprocessing plant.

The CC102 package has an IP2 package conformity certificate.

##### 14.2.2.3.1 Package performance

Content:

- Technological waste, radioactive resin or fissile material (< 15g per package);
- low activity waste < $10^{-4}$ $A_2$/g.

Waste packaged in:

- concrete primary packages CBF-C2.

Package characteristics:

- Overall dimensions: 2480 x 2246 x 2128 mm;
- Total gross weight: 23 450 kg;
- Maximum payload: 11 480 kg;
- Content: 3 CBF-C2 (height: 1504 mm – diameter 1004 mm).

#### 14.2.2.4 TN™833

The TN™833 can transport 12 bitumen drums (equivalent to 200 l drums) by cask, arranged in two layers. This transport container design will be available from 2010 and will be a type B(U) package. It will be transportable by road or rail, and is vertically tied-down on a flat ISO 20’ frame that is attached to a wagon or a truck.

The TN™833 cask is developed to transport bituminized residues arising from spent fuel reprocessing to their owners (Spain, etc.).

Its design has consequently to be compliant with the AREVA-La Hague DE/EB facility mechanical interfaces: mass less than 40 tons, loading / unloading operations with trunnions, for instance.

This cask is made of stainless steel with external heat insulator on the body and inside the lid. The shock absorber cover, on the top, is made of aluminium and contains insulating material.

- a cylindrical body (external diameter : 2330 mm), equipped with two trunnions;
- the lid consists of a thick disc equipped with an orifice and a handling feature;
- one top shock absorber made of aluminum with internal heat insulating material (external diameter : 2700 mm);
- two internal baskets, each for 6 bitumen drums (around 215 l);
- Total height: 2680 mm;
- Total gross weight: 40 tons;
- Content weight: 6.8 tons;
- Lid cover weight: 2.5 tons.

#### 14.2.2.5 TN™837

The TN™837 cask is being developed to transport technological waste immobilised in a concrete matrix, in a packaging called CBFC’2. The TN™837 cask is mainly being developed in order to allow AREVA to return waste from AREVA La Hague plant to Spain.

Its design, therefore, has needed to be compliant with the AREVA-La Hague plant DE/EDS facility mechanical interfaces: mass less than 50 tons, loading / unloading operations with trunnions, for instance.

The TN™837 will be available when the first waste return is required and is a type B(U) package. It will be transportable, by road or rail, and is vertically tied-down on a flat ISO 20’ frame attached to a wagon or a truck. This transport frame has the same handling design as for the TN™833.

The $TN^{TM}$837 capacity is of 3 CBFC'2 by cask. This cask is made of stainless steel, with a top shock absorber made of aluminum.

- 1 cylindrical body (external diameter: 2530 mm), equipped with 2 trunnions;
- 1 lid: thick disc equipped with an orifice and an handling device;
- 1 top shock absorber made of aluminum (external diameter: 3000 mm);
- 1 internal basket for 3 CBFC'2;
- Total height: 2355 mm;
- Total gross weight: 46 tons;
- Content weight: 12 tons;
- Lid cover weight: 3 tons.

## 15 CONCLUSIONS

This SRWSR sets out management arrangements for processing and interim storage of waste and spent fuel generated by the UK EPR in accordance with the UK Government policy and regulatory constraints. The SRWSR provides a high degree of confidence that the challenges associated with the management of solid waste and spent fuel from the UK EPR are fully understood and that solutions are available with the envelope of current UK and international experience. Specifically it demonstrates that:

- The production of radioactive waste will be avoided and where this is not reasonably practicable, the quantity of waste produced will be minimised;
- The radioactive wastes generated by the UK EPR are similar of those wastes generated by operating PWRs and all waste streams have process routes to interim storage or final disposal/solution;
- A number of waste and spent fuel options to the FA3 Reference Case which are based on nationally or internationally proven technologies and the experience from EPR and AREVA projects are available for the UK EPR;
- The dominance of short-lived radionuclides in some ILW will enable it to be declassified to LLW within the on-site interim storage period.

The base case waste treatment facilities for each UK EPR site include:

- Waste Treatment Building for the receipt, segregation, treatment and conditioning of solid radioactive wastes;
- Interim Storage Facility for solid ILW packages;
- Spent fuel Interim Storage Facility for receipt, packaging and safe interim storage of spent fuel.

The facilities are modular in design and may be adapted to cope with future process changes and to maintain an appropriate capacity storage volume.

It is assumed that the decommissioning of the UK EPR will commence immediately after shutdown and de-fuelling. The decommissioning of the UK EPR can be performed in approximately 12 years. This duration is indicative and depends on the dismantling scenario and programme adopted by each utility.

## 16 REFERENCES


1 Managing Radioactive Waste Safely. A Framework for Implementing Geological Disposal. Cm 7386, June 2008.

2 DTI, The Future of Nuclear Power. The Role of Nuclear Power in a Low Carbon UK Economy: Consultation Document, May 2007.

3 BERR, Meeting the Energy Challenge. A White Paper on Nuclear Power, January 2008.

4 The Stationary Office, Review of Radioactive Waste Management Policy: Final Conclusions Cm2919, July 1995.

5 Fundamentals of the Management of Radioactive Waste. An introduction to the management of higher level radioactive waste on nuclear licensed sites, December 2007.

6 Energy Bill 2008 [HL Bill 52]. 2007-2008.

7 The Energy Bill 2008. Consultation on Funded Decommissioning Programme Guidance for New Nuclear Power Stations, February 2008.

8 The Energy Bill 2008. The Government response to the consultation on funded decommissioning guidance for new nuclear power stations, September 2008.

9 The Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations 2007. SI No.1573.

10 Radioactive Waste Substances Act, Crown Copyright, Her Majesty's Stationary Office, May 1993.

http://www.opsi.gov.uk/acts/acts1993/ukpga_19930012_en_1 .November 2008.

11 HSE, Safety Assessment Principles for Nuclear Facilities. 2006 Edition, Revision 1, 2006.

12 Memorandum of Understanding Between the Health and Safety Executive and the Environment Agency on Matters of Mutual Concern at Nuclear Sites Licensed by HSE in England and Wales.

13 The Management of Higher Activity Radioactive Waste on Nuclear Licensed Sites. Part 1 The Regulatory Process. Guidance from the Health and Safety Executive, the Environment Agency and the Scottish Environment Protection Agency to nuclear licensees, December 2007.

14 Health and Safety Executive. Nuclear Power Station Generic Design Assessment- Guidance to Requesting Parties. Version 3, August 2008.

15 Radioactive Substances Regulation Environmental Principles (Interim). Version 1, 1st November 2005.

16 Environment Agency, Radioactive Substances Regulation Environmental Principles Consultation Draft, June 2008.

17 CoRWM, Response to the Report and Recommendations from the Committee of Radioactive Waste Management (CoRWM) by the UK Government and devolved administrations, October 2006.

18 Radioactive Waste Management Directorate. Disposability of Solid Waste Arisings from New Build, 25th April 2008.

19 NDA, Nirex report Generic Repository Studies. Generic Waste Package Specification Volume 1 – Specification. N/104, June 2005.

20 NDA, Foundations of the Phased Disposal Concept, RADIOACTIVE WASTE TRANSPORT, Nirex Report N/022, October 2002.

21 NDA, Nirex Report Generic Repository Studies. Generic Transport Safety Assessment. Volume 1 Main Report. UK Nirex Limited. N/078, July 2003.
22 Clearance and Exemption Principles, Processes and Practices for Use by the Nuclear Industry A Nuclear Industry Code of Practice, Interim Issue, May 2003
23 Policy for the Long Term Management of Solid Low Level Radioactive Waste in the United Kingdom by Defra, DTI and the Devolved Administrators, 26th March 2007.
24 Radioactive Waste Management, Harwell General Guidance for Letter of Compliance Submissions. RWPR60-WI01. Revision 0, 1st April 2007.
25 NDA, Foundations of the Phased Disposal Concept, Radioactive Waste Transport, Nirex Report N/022, October 2002.
26 NDA, The Viability of a Planned Geological Repository for the Long-Term Management of the UK's Radioactive Waste. Nirex Report N/122, November 2005.
27 UK-EPR fundamental Safety Overview, Volume 3 Environmental Impact, Chapter D, Section D.7.1
28 SSER - UK EPR PCSR, Chapter 6.3, April 2008.
29 NDA, Briefing Note on Sealed Sources, Document number 376338, UK Nirex Ltd, 5 April 2002.
30 The Waste Framework Directive (75/442/EEC), July 1975.
31 Revision of the Waste Framework Directive (91/156/EEC). March 1991.
32 Consolidated revision of the Waste Framework Directive (2006/12/EC), April 2006.
33 LLWR, 2008. Conditions for Acceptance by LLW Repository Ltd of Radioactive Waste for Disposal at the Low Level Waste Repository.
34 IAEA. Innovative waste treatment and conditioning technologies at nuclear power plants. IAEA-TECDOC-1504, 2006b.
35 IAEA, Policy for the Long Term Management of Solid Low Level Radioactive Waste in the United Kingdom. Technical Reports Series No 456, 2007.
36 IAEA, Handling and Processing of Radioactive Waste from Nuclear Applications. Technical Reports Series No 402, 2001.
37 IAEA, Application of Membrane Technologies for Liquid Radioactive Waste Processing Technical Reports Series No 431, 2004.
38 Mann N. R. and Todd T. A., Crossflow filtration testing of INEEL radioactive and non-radioactive waste slurries. International Conference on Solid-Liquid Separation Systems, Oahu, Hawaii (April 1999). Chemical Engineering Journal (ISSN 1385-8947), vol. 80, no 1-3, pp. 237-244, 2000.
39 IAEA, Application of Ion Exchange Processes for the Treatment of Radioactive Waste and Management of Spent Ion Exchangers. Technical Reports Series No. 408, 2002.
40 Defra, Policy for the Long Term Management of Solida Radioactive Waste in the UK, 2007
41 Buckley M., Handy B. and Hillis Z. K., Litreature Review of the Potential Application of Metal Melting in the UK Nuclear Sector. NNC Report 11426/TR/001 Issue 3, 2004.
42 Hillis Z.K. and Buckley M., Summary of the Workshop on the Potential Application of Metal Melting in the UK Nuclear Sector held on the 6th October 2004. NNC Report 11426/TR/002, 2004.
43 Clearance and Exemption Working Group, Nuclear Industry Safety Directors Forum, Clearance and Exemption Principles, Processes and Practices for Use by the Nuclear Industry. A Nuclear Industry Code of Practice, 2006.
44 IAEA, Application of Thermal Technologies for Processing of Radioactive Waste. IAEA-TECDOC-1527, 2006a.
45 Oldiges, O. and Blenski, H-J., A New, Small Drying Facility For Wet Radioactive Waste And Liquids. Waste Management Symposium, 2003.

46 IAEA, Application of Ion Exchange Processes for the Treatment of Radioactive Waste and Management of Spent Ion Exchangers. Technical Reports Series No. 408, 2002.

47 Nirex (NDA), Generic Repository Studies Generic Waste Package Specification Volume 1 - Specification. Nirex Report no. N/104, 2005.

48 Management of Radioactive Materials and Radioactive Waste on Nuclear Licensed Sites. T/AST/024. Issue 3, Match 2001.

49 Radioactive Substances Regualtion Environmental Principles. Assessment Guide No 1. Assessment of Best Available Techniques (BAT). Consultation Draft, June 2008.

50 IAEA, Quality Assurance for Safety in Nuclear Power Plants and other Nuclear Installations. Code of Safety and Guides. Q1-Q14. IAEA Safety Series No. 50-C/SG-Q. 1996.

51 IAEA, Selection of Away-from-Reactor Facilities for Spent fuel Storage. A Guidebook. IAEA-TECDOC-1558. September 2007.

52 Defra, The Radioactive Waste Management Advisory Committee's: Advice to Ministers on The United Kingdom Radioactive Waste Inventory, November 2008. http://www.defra.gov.uk/rwmac/reports/inventory/rwmac_inventory.pdf

53 HSE, Technical Assessment Guide. Control, Storage, Handling and Transport of Nuclear Matter including Fissile Materials, November 2008.

http://www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/tast041.pdf