Form10/00

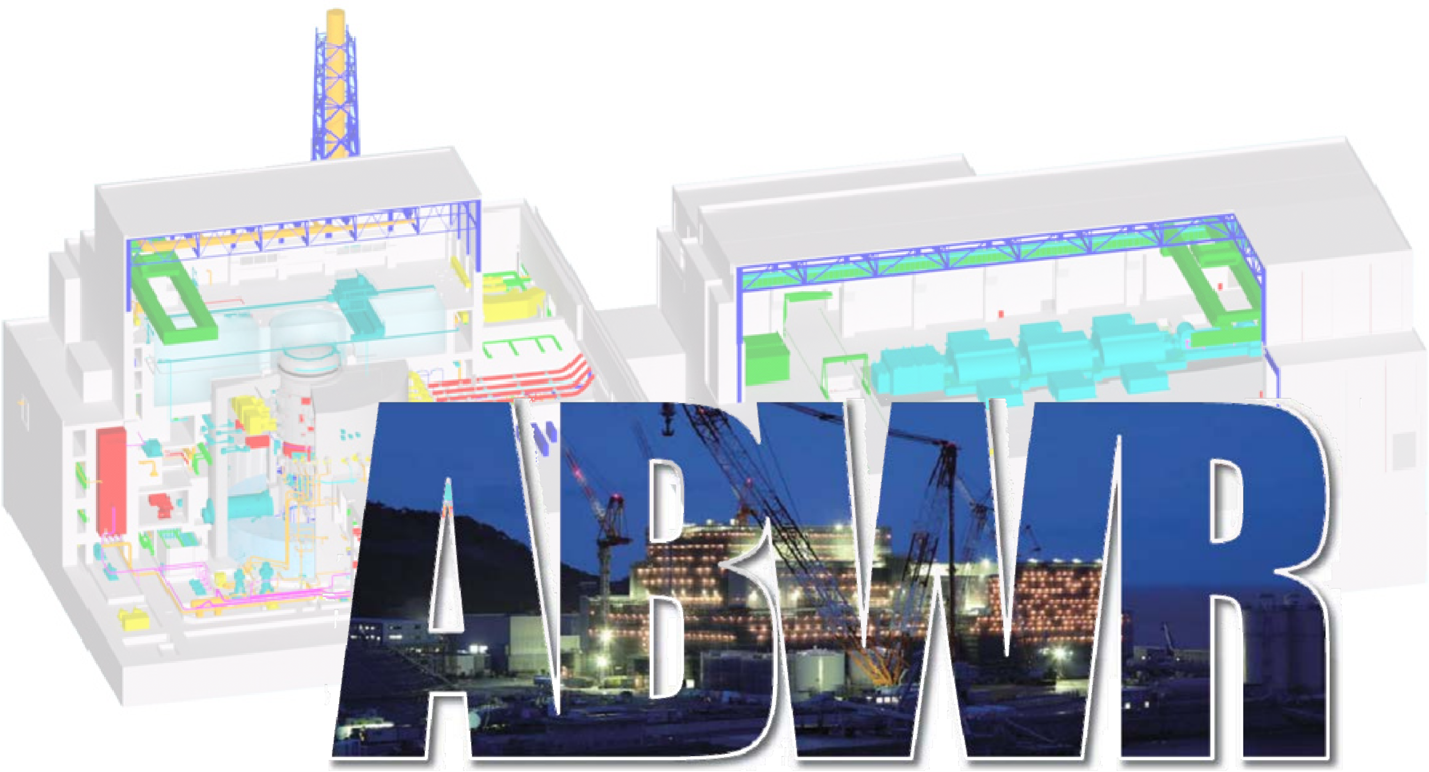
UK ABWR

HITACHI

Document ID	:	GA91-9901-0045-00001
Document Number	:	XE-GD-0155
Revision Number	:	A

UK ABWR Generic Design Assessment

Preliminary Safety Report on Spent Fuel Interim Storage



Hitachi-GE Nuclear Energy, Ltd.

Form10/00

UK ABWR

HITACHI

DISCLAIMERS

Proprietary Information

This document contains proprietary information of Hitachi-GE Nuclear Energy, Ltd. (Hitachi-GE), its suppliers and subcontractors. This document and the information it contains shall not, in whole or in part, be used for any purpose other than for the Generic Design Assessment (GDA) of Hitachi-GE's UK ABWR. This notice shall be included on any complete or partial reproduction of this document or the information it contains.

Copyright

Hitachi-GE Nuclear Energy, Ltd.

Table of Contents

Abbreviations and Acronyms

Abbreviations and Acronyms	Description
ABWR	Advanced Boiling Water Reactor
ALARP	As Low As Reasonably Practicable
BAT	Best Avairable technology
GDA	Generic Design Assessment
GDF	Geological Disposal facility
MVDS	Modular Vault Dry Store
R/B	Reactor Building
SFP	Spent Fuel Pool
SQEP	Suitable Quarified Experienced Person
UK	United Kingdom
U.S.	United States

1. Introduction

In the UK, the licencee should demonstrate that spent fuels discharged from reactors can be stored safely until final disposal.

In this document, the requirements for safety evaluations, conditions for storage design, management strategy for the optioneering process to determine the suitable storage process along with the possible design options (e.g., dry type, wet type, etc.) and the parameters to evaluate those options are presented.

To demonstrate that the waste and spent fuels from a UK ABWR can be safely managed until disposal, Hitachi-GE will propose a “preferred” interim storage option during the GDA. This does not necessarily mean that the technology or the approach will be adopted by future licensees, but will provide assurance that the spent fuels can be safely stored on site prior to final disposal.

2. Proposal of Decision Making Process

Hitachi-GE is in the process of considering the options available to them for the spent fuel interim storage on the Generic Site for a single UK ABWR unit. At this stage of GDA (Step 2), a final decision is yet to be made and instead a description of the available options is provided (see Section 6).

The final decision on the most appropriate approach to spent fuel interim storage will be taken by future licensees. However, in order to demonstrate that waste and spent fuels from a UK ABWR can be safely managed until disposal, Hitachi-GE will select a “preferred” option during the GDA.

Hitachi-GE can confirm that the “preferred” option will meet all license conditions and be consistent with the plant design and licensed operation. As far as possible, Hitachi-GE will avoid design and operational actions that would foreclose onward transport or processing of spent fuel. If the plant design limits future options for interim storage, those limitations will be discussed.

For Hitachi-GE to select the “preferred option” for GDA, a defined “Decision Making Process” will be followed that takes into account UK Government policy, legislation, and relevant International and UK National guidance and regulations.

It is vitally important that the decision making process involves the correct people. Hitachi-GE has a well-defined process for identifying the Suitably Qualified and Experienced Personnel (SQEP) and will use this process to ensure the correct people are involved at the various stages of the project. This will include, for example:

- Fuel Engineers
- Spent Fuel specialists
- Radiation Protection Advisors
- Radioactive Waste Management Advisors

- Spent Fuel Pool Chemists
- Security personnel

As a first step in the process, the design requirements will be specified (see Section 3). Following this, all input information and assumptions (see Section 4) will be documented and justified. This information includes spent fuel characteristics, conditions, and quantities as well as the generic site conditions. Some of the key assumptions to be considered are as follows:

- It is assumed that the Geological Disposal Facility (GDF) will be available at the end of the site life and that 140 years of cooling is required to meet the disposal canister exterior temperature requirement.
- The repackaging of the spent fuel into the eventual disposal canister may be required and, therefore, the interim storage process must provide the flexibility to accommodate repackaging.

In Section 6, the available interim storage options to be considered are described. Each of the options considered has been used worldwide to safely provide interim storage of spent fuels.

To allow a transparent decision to be made within this decision making process, the parameters to be considered for all options described in Section 6 are described in Section 7. A judgement will be made on the relative significance of each of the parameters and data will be collected to underpin the process.

Following this process, Hitachi-GE will determine the preferred option.

3. Design Requirement

3.1. General Safety Requirements for Spent Fuel Interim Storage

Major requirements for safe storage of spent fuels are as follows:

3.1.1. General Requirements

The spent fuel interim storage facilities should be designed to the same high safety level applied to the nuclear power plant. The dose rate from spent fuel storage facilities and casks should be designed in accordance with the ALARP/BAT policy and have sufficient capacity to withstand the generic site envelope conditions and fault schedule.

3.1.2. License Conditions

To ensure that the licensee will be able to comply with the standard license conditions that may be applicable to the interim storage facility, the following standard license conditions, at a minimum, shall be highlighted in the evaluation of the interim storage facility design:

- License Condition 8: Warning Notices
- License Condition 11: Emergency arrangements
- License Condition 14: Safety documentation
- License Condition 16: Site plans, designs and specifications
- License Condition 18: Radiological protection
- License Condition 21: Commissioning
- License Condition 23: Operating rules
- License Condition 27: Safety mechanisms, devices and circuits
- License Condition 28: Examination, inspection, maintenance and testing
- License Condition 32: Accumulation of radioactive waste
- License Condition 34: Leakage and escape of radioactive material and radioactive waste
- License Condition 35: Decommissioning

3.1.3. Requirements for Final Disposal

The spent fuel storage facilitiesa and casks should be designed to ensure spent fuel retrievability for eventual transfer to the final disposal site or to accommodate actions necessary in the event of accidental damage.

The strength of spent fuel cladding should be maintained during the interim storage. For damaged fuel, the interim storage design should not cause further degradation of the fuel cladding.

3.1.4. Requirements for Spent Fuel

Considering the assumed fuel rod internal pressure of spent fuels discharged at design basis burn-up and the maximum decay heat emittance, the interim storage approach, including all operations to transfer and load the spent fuels such as draining, drying, etc., needs to maintain the appropriate environment to prevent fuel cladding damage.

3.1.5. Requirements for Damaged Spent Fuel

The interim storage facility will be designed to safely store damaged fuel. The classification of damaged fuel and its condition will be included as part of the evaluation of the preferred storage option.

3.2. Final Disposability Consideration

At this moment, it is necessary to design the storage system on the assumption that stored spent fuels may be repackaged into final disposal canisters after the interim storage period because the design concept of the final disposal system (type of canister, handling system etc.) and the limitations of transportation to the final disposal facility have not been determined.

If repackaging is required, the repackaging into the final disposal canisters or maintenance of the interim storage casks can be conducted in the Reactor Building as long as the R/B is operational. However after concluding operation and prior to the start of the R/B decommissioning, an additional facility for repackaging or maintenance will be needed.

If required, future licensees can deploy repackaging/maintenance facilities because similar facilities have already been put in use around the world. Because the conclusion of the first UK ABWR will be around 2080, future licensees should design the repackaging/maintenance facilities based on the ALARP/BAT at that time.

4. Design Assumption

Major assumptions for the design of the spent fuel interim storage facilities include the spent fuel type, quantity of the spent fuels generated during the plant operation and the site conditions.

4.1. Spent Fuel

Though various nuclear fuels can be used for the UK ABWR, in the UK ABWR GDA, the GE14 will be adopted as a base condition of the studies. The GE14 fuel assembly consists of a fuel bundle (composed of fuel rods, water rods, spacers, and upper and lower tie plates), and a channel that surrounds the fuel bundle (Figure 4.1-1). GE14 has 92 fuel rods (14 partial length rods and 78 full length rods) in a 10x10 rod array and two water rods. The fuel rods and water rods are spaced and supported by the upper and lower tie plates with intermediate spacing provided by eight spacers. The upper and lower tie plates are fixed by eight tie rods, which hold the fuel bundle together. The upper tie plate has a handle for lifting and transferring the fuel assembly.

General information for the spent fuel is shown in Table 4.1-1 and an example of the preliminary evaluation for the transition of decay heat of GE14 spent fuel is shown in Figure 4.1-2.

Table 4.1-1 General information of GE14

Item	Value(2)
Fuel bundle length (mm)	4468
Overall weight (includes channel weight) (kg)	300
Weight of UO_2 (kg)	200
Average discharged fuel burn-up (GWd/t)	50
Total number of fuel assemblies discharged over 60 years operation	9600(1)

(1) Note that this value does not include the potential increase in the quantity of spent fuels due to premature discharge of damaged fuel. During the evaluation, this value should be increased based upon industry experience (i.e., use probability of fuel damage) with suitable margin to obtain a bounding total number of discharged fuel assemblies.
(2) All values are approximate.

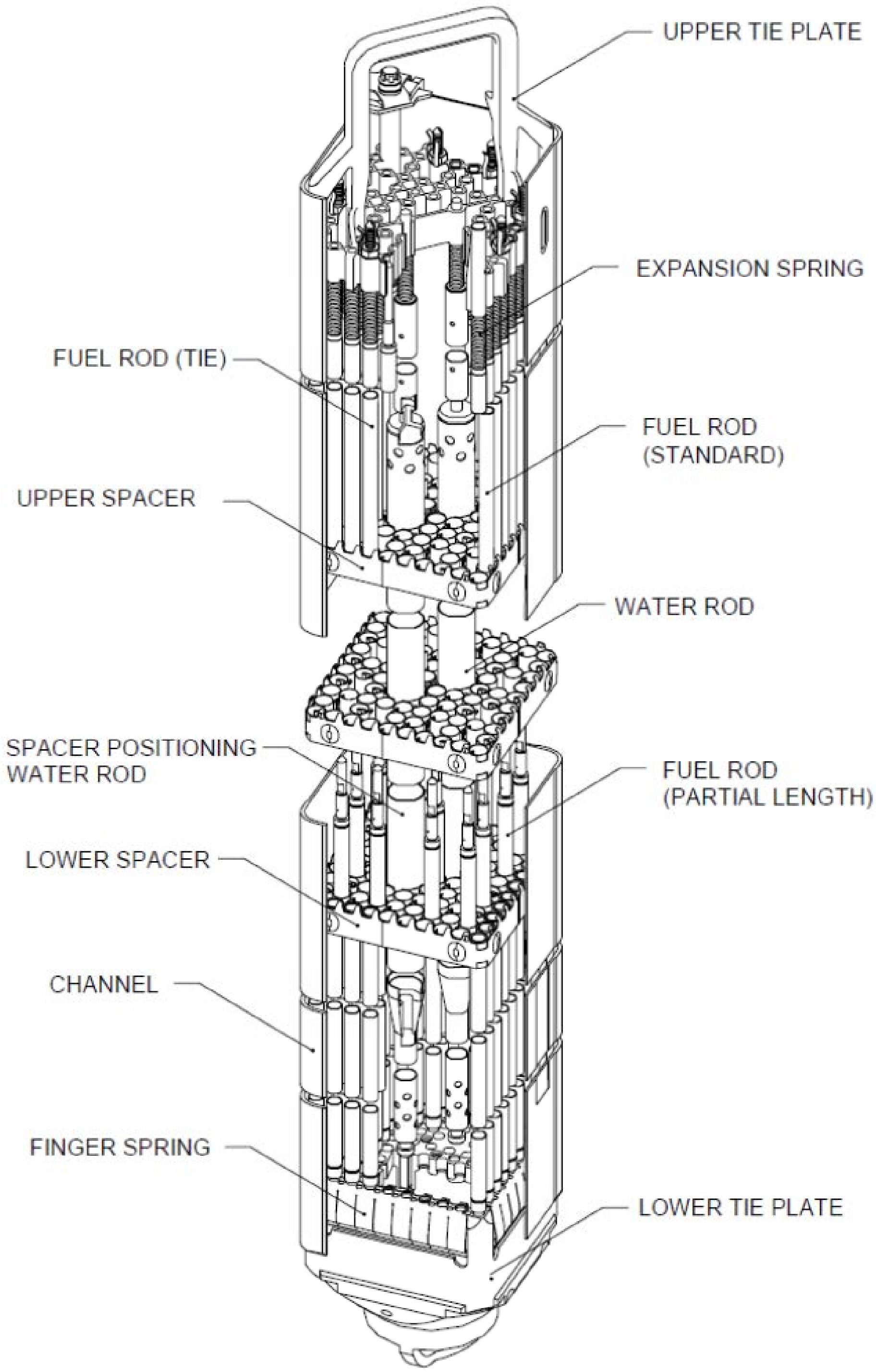


Figure 4.1-1 GE14 Fuel Assembly

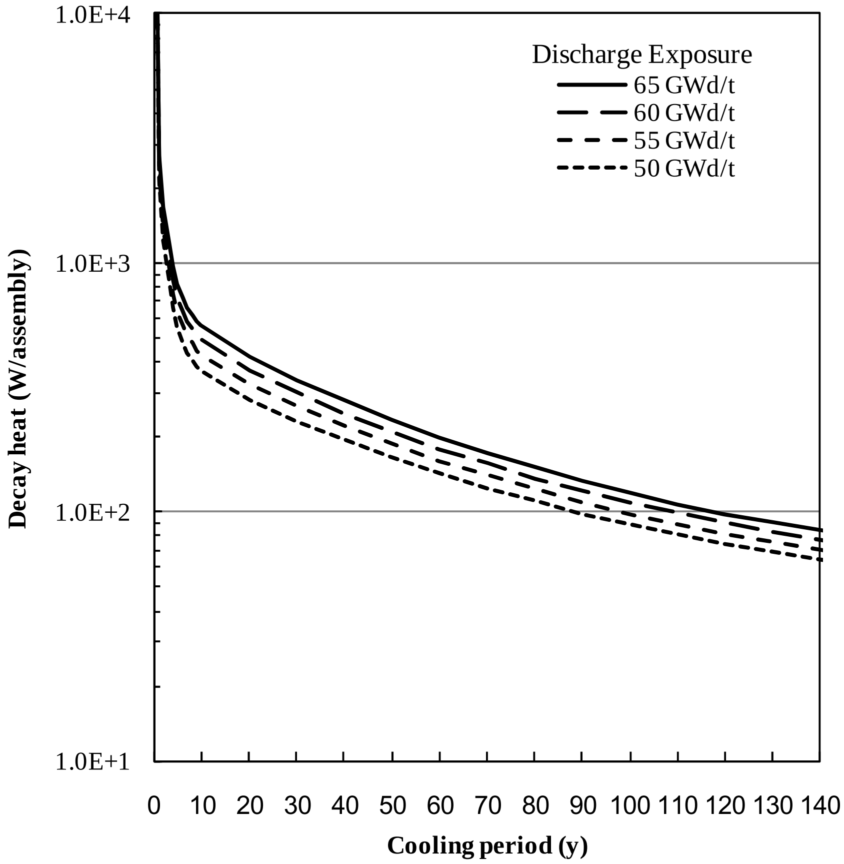


Figure 4.1-2 Transition of decay heat of GE14 spent fuel (Preliminary)

4.2. Generic Site Conditions and Faults

The generic site conditions and faults to be considered for the design of spent fuel interim storage facilities will be equivalent to that of the nuclear power plant. See Generic Site Envelope (Document ID GA91-9901-0010-00001) [Ref-1] and Fault Studies to Discuss Deterministic Analysis, PSA and Fault Schedule Development (Document ID GA91-9901-0009-00001) [Ref-2], respectively for a detailed listing of the generic site conditions and faults.

5. Management Strategy for Spent Fuels

5.1. Spent Fuel Management Plan

Spent fuels which are discharged during operation are temporarily stored in the spent fuel pool of the reactor building (SFP) and pre-cooled for a certain period before being transported to the interim storage facility. After interim storage, spent fuels will be shipped to a final disposal site. Spent fuels may be repackaged into appropriate casks, if needed, depending on final disposal facility and transportation requirements..

This section describes the minimum pre-cooling period and the maximum period the spent fuel can be stored in the interim storage facility.

The specific details on the fuel design including the constraints imposed by the total fuel life cycle are described in documents relevant to fuels and include major constraints imposed by the interim storage approach.

5.1.1. Pre-cooling Period

When considering the pre-cooling period from the viewpoin of ALARP, a shorter pre-cooling period is beneficial to mitigate the severity of a SFP fault because spent fuels can be removed earlier and the number of spent fuels in the SFP can be decreased. However, the overall decay heat value of the SFP mainly depends on the number of spent fuels which are cooled less than 3 years because of their relatively large decay heat emittance. Therefore, the contribution of a shorter pre-cooling period on improving the SFP safety seems to be small. On the other hand, a short pre-cooling period means that the operator must handle hot, highly activated spent fuels, which increases the worker dose rate and increases the dose rate of a potential accident.

Therefore, from an ALARP consideration, the longer pre-cooling period is desirable since it will reduce the risk of radiation exposure on workers and public.

5.1.2. Pre-cooling Period Limitation from Spent Fuel Pool Capacity

The capacity of the UK ABWR SFP will be designed to store the amount of spent fuels generated from 10 years of commercial operation with adequate margin. Therefore, the interim storage system for UK ABWR should be designed to store spent fuel with 10 years pre-cooling, at a mimimum. The ability of the interim storage facility to store spent fuel with less pre-cooling will provide operational flexibility and additional capacity margin. This will be included in the evaluation of the potential storage options.

5.1.3. Pre-cooling Period Limitation on Wet Storage (External Pool)

In the case of wet storage in an external storage pool, spent fuels are transported from the SFP to the external pool using a wet transport cask, unpacked in the external pool, and stored under water in spent fuel storage racks. In this case, the allowable thermal load and shielding ability of the wet transportation cask and heat removal ability of the external pool are major determining factors for the pre-cooling period. However, the thermal capacity and shielding ability of the wet transportation cask are typically the limiting factors because the heat removal capacity of the external pool can be adjusted.

The evaluation will determine the minimum cooling time required transferring the spent fuel to the external pool and this value will be considered in the evaluation of the potential storage options.

5.1.4. Pre-cooling Period Limitation on Dry Storage

In the case of dry storage using a metal cask or canister, the spent fuels are packed under water, sealed into the storage vessel, and are dried in the reactor building before transportation to the interim storage facility. For dry storage, the heat removal and shielding ability of cask/canister and shielding ability of the storage facility are the limiting factors.

The evaluation will determine the minimum cooling time required storing the spent fuel in the metal cask and concrete casks and these values will be considered in the evaluation of the potential storage options.

5.1.5. Maximum Duration of Spent Fuel Storage in Interim Storage

To ensure that the interim storage facility can meet the required 140 year spent fuel cooling time. The maximum duration that the spent fuel can be stored in each of the potential interim storage options will be determined. The evaluation will also confirm the maximum design life of each of the potential interim storage options. The maximum design life and spent fuel storage duration will be compared to determine the limiting maximum duration.

5.1.6. Maximum Duration of Spent Fuel Storage in SFP

The maximum duration of storage in the spent fuel pool is discussed in Initial Safety Case Report on Spent Fuel Pool (Document ID GA91-9901-0003-00001) [Ref-3]. This sets the maximum cooling time that should be considered in the storage option evaluation.

5.1.7. Wet Storage Maximum Duration of Spent Fuel Storage

The maximum duration of storage in the spent fuel external spent fuel pool will be determined as part of the evaluation. The maximum duration will consider the impact of the previous pre-cooling in the SFP on the maximum duration. Additionaly, the design life of the external spent fuel pool will also be determined as part of the evaluation.

5.1.8. Dry Storage Maximum Duration of Spent Fuel Storage

The maximum duration of storage in dry storage will be determined as part of the evaluation. The maximum duration will consider the impact of the previous pre-cooling in the SFP on the maximum duration. Additionaly, the design life of the dry storage facility will also be determined as part of the evaluation.

6. Interim Storage Processes to be Evaluated

Both, dry and wet storage processes for spent nuclear fuels, are in use around the world.

Wet storage is a storage process that stores spent fuels under water in an external storage pool and the main feature of wet storage is to keep the fuel cladding temperature low enough to avoid degradation of the cladding material by cooling water.

In regard to the dry storage process, there are various methods in use, however, the basic principle is to avoid fuel cladding degradation by using an inert gas to facilitate cooling in the sealed storage cask.

The most common methods of interim storage in use today are described in the following sections.

6.1. Concrete Cask (canister packed in concrete overpack) Storage Process

The concrete cask storage system is one of the more typical dry storage systems in use today. Spent fuels are packed at the SFP into a canister with a lid that is sealed by welding. The canister is transferred outside of the reactor building and stored within a robust concrete overpack. For transportation of the canister between the SFP and the concrete overpack, a designated transfer cask is used. Air inlet vents at the bottom of the concrete overpack and exit vents at the top allow natural convection of air through the space between the canister and the overpack providing heat removal to the environment. Furthermore, the concrete overpack also provides shielding against radiation from the canister. Generally, the concrete overpacks are stored in the open on a concrete pad in both the vertical and horizontal positions. Concrete cask storage systems are used around the world to store BWR fuel and it has become the most popular process in the U.S.

Typical components of the concrete cask storage system consist of;

- Canister (shell, lid and basket assembly)
- Cask for onsite transfer
- Concrete overpack
- Vehicle for transfer cask and concrete overpack transportation
- Equipment for canister lid welding and drying

Methods for ensuring safety functions are as follows;

- Containment: canister shell, canister lid and lid welding are designed to maintain containment capability
- Subcriticality: basket assembly assures that sub-criticality is maintained
- Heat removal: natural convection of air between canister and concrete overpack assures heat removal
- Shielding: concrete overpack provides adequate radiation shielding
- Protection against external hazards: concrete overpack is adequately designed against hazards

At the end of the intrim storage period, repackaging of the spent fuels from the storage canisters into final disposal containers will make the interim storage canisters low level radioactive waste.

Other components such as concrete overpacks will also have some components that will be classified as low level radioactive waste. If repackaging is required, a repackaging facility will also be classified as radioactive waste.

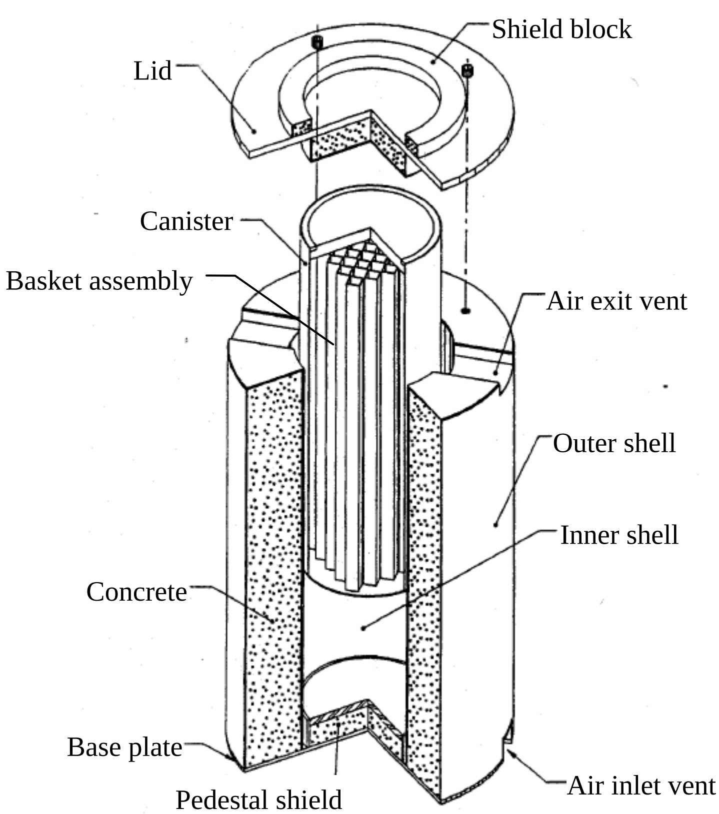


Figure 6.1.1-1 Schematic view of a typical concrete cask (Example)

6.2. Metal Cask Storage Process

The metal cask storage system is one of the more common dry storage systems. Spent fuels are packed at the SFP into an inner shell of a containment vessel. Containment is provided by cask lids (typically double lids) with metal gaskets which are bolted to the top flange. Metal casks can be designed either for storage only or for both storage and transportation (dual-purpose cask). The dual-purpose cask can be transported with suitable high performance shock absorbers, and can be stored without repackaging spent fuels into another storage vessel. Metal casks are stored in the open on a concrete pad or within a building in both vertical and horizontal positions. Metal cask storage systems are used widely in many countries; examples include Dresden in the U.S., Gundremmingen in Germany and Tokai in Japan.

Typical components of the metal cask (dual-purpose cask) storage system are;

- Metal cask (inner shell, lid, outer shell, basket assembly, neutron shield material and metal gaskets)
- Shock absorber
- Vehicle for metal cask transportation
- Storage facility (if needed)

Methods for ensuring typical safety functions are as follows;

- Containment: inner shell, bolted lids with metal gaskets are designed to maintain containment integrity
- Subcriticality: basket assembly assures the sub-criticality can be maintained
- Heat removal: Natural convection of air on the outer surface of the cask assures heat removal
- Shielding: inner shell, lid, outer shell and neutron shield material provides adequate radiation shielding
- Protection against external hazards: structure of metal cask provides protection

At the end of the intrim storage period, repackaging of the spent fuels from the metal casks into the final disposal canisters will make the casks low level radioactive waste. Other components such as cask anchoring devices and the cask storage building will be non-radioactive waste. If repackaging is required, a repackagintg facility will also be classified as radioactive waste.

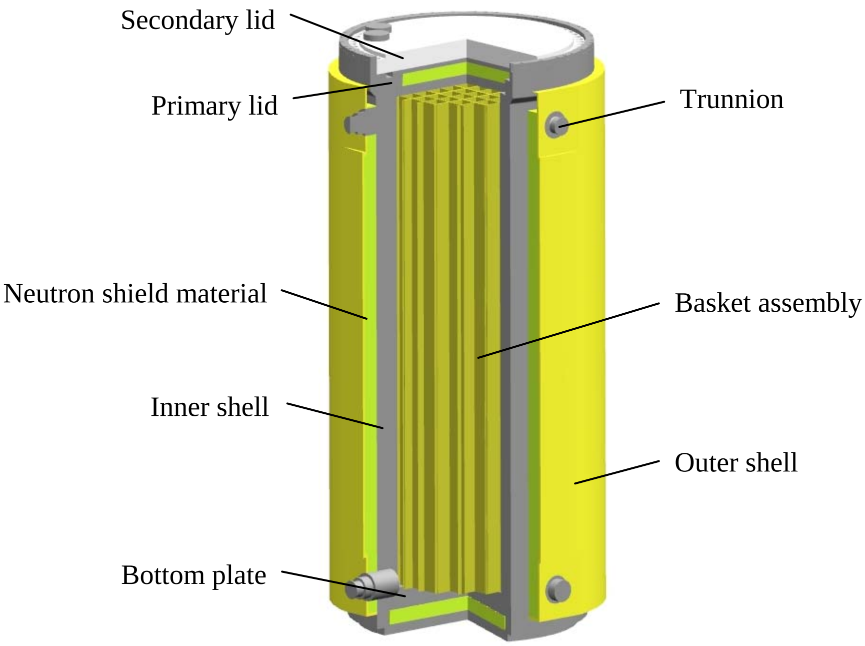


Figure 6.1.2-1 Schematic view of a typical metal cask (Example)

6.3. Wet Storage Process

The wet storage process is a storage system that stores the spent fuels in the external fuel storage pool separate from the SFP of the reactor building. The basic technology is essentially the same as the SFP in the reactor building. Wet storage is effective for sites where the available area for storage is limited because it provides higher storage density in comparison with other storage system.

In wet storage, cooling is provided by demineralised water and integrity of the fuel cladding material will be maintained by keeping the temperature of the fuel cladding low.

The fuels which are pre-cooled for a shorter term can be stored by designing sufficient heat removal capacity of the pool cooling system, but the limitations of the pre-cooling period are typically driven by the thermal capacity and the shielding ability of the wet transportation cask, which is used to shuttle the spent fuels from the reactor building SFP to the external storage pool.

Major components of wet storage are described below;

- Spent fuel pool
- Spent fuel storage rack
- Cooling water circulation, cooling and clean-up system
- Fuel handling equipment
- Cask handling equipment (e.g., overhead crane)
- Spent fuel transportation cask
- Emergency power supply system

Methods for ensuring safety functions are as follows;

- Containment: spent fuel is maintained and cooled in pool of demineralised water
- Subcriticality: spent fuel storage rack assures that sub-criticality is maintained
- Heat removal: water circulation through a heat exchanger provides cooling
- Shielding: spent fuel maintained at an adequate depth under water
- Protection of fuel clad: demineralised water provides cooling and maintains low temperature
- Protection against external hazards: facility structure is designed to withstand hazards

After the interim storage period, the pool liner, storage racks and circuration, cooling and clean-up water systems will be low level radioactive waste. Filters for the circulation system will be intermediate or higher level radioactive waste. Other components such as buildings will be non-radioactive waste.

Figure 6.1.3-1 shows the schematic view of the wet storage facility installed in Switzerland (called CLAB). The facility has been in operation since 1985. External wet storage facilities are also in operation at the Rokkasho Nuclear Fuel Reprocessing Facility in Japan and the La Hague Nuclear Fuel Reprocessin Facility in France.

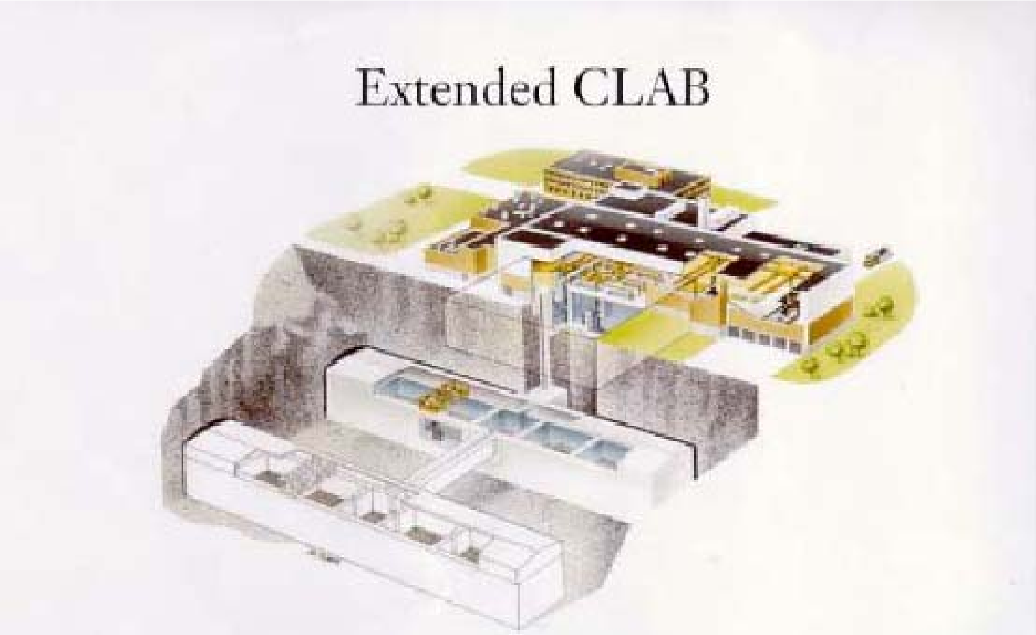


Figure 6.1.3-1 Schematic view of Wet Storage Facility (CLAB)

6.4. Vault Storage Process

A vault is a reinforced concrete structure containing an array of storage cells built either above or below ground. Shielding is provided by the surrounding structure. Commercially available vault systems are located above the ground level and the heat is generally transferred to the atmosphere by natural convection of air over the exterior of the cells. Each storage cell or cavity can contain one or more spent fuel assemblies stored in metal tubes or storage cylinders. Each metal tube or storage cylinder is sealed to provide containment of the spent fuel.

Spent fuels are loaded into these tubes either on-site with fuel handling machines in a charge hall or off-site at the reactor pools. The vault itself can be a relatively simple design, but requires additional infrastructure for the reception and handling of the spent fuel assemblies. The storage concept permits modular construction and incremental capacity extension.

A vault storage system is suitable for storing relatively small canisters, however the facility inself is relatively large.

After the interim storage period, the canisters will be low level radioactive waste. Most of the other components such as the canister custody building will be non-radioactive waste, but structures, systems, and/or components around the canisters will be activated during operation and will be classified as low level radioactive waste.

Examples of vaults located at reactor sites are the Wylfa facility (Magnox dry storage) in the UK, the Modular Vault Dry Store (MVDS) facility at PAKS in Hungary, and the Fort St. Vrain MVDS in the U.S.

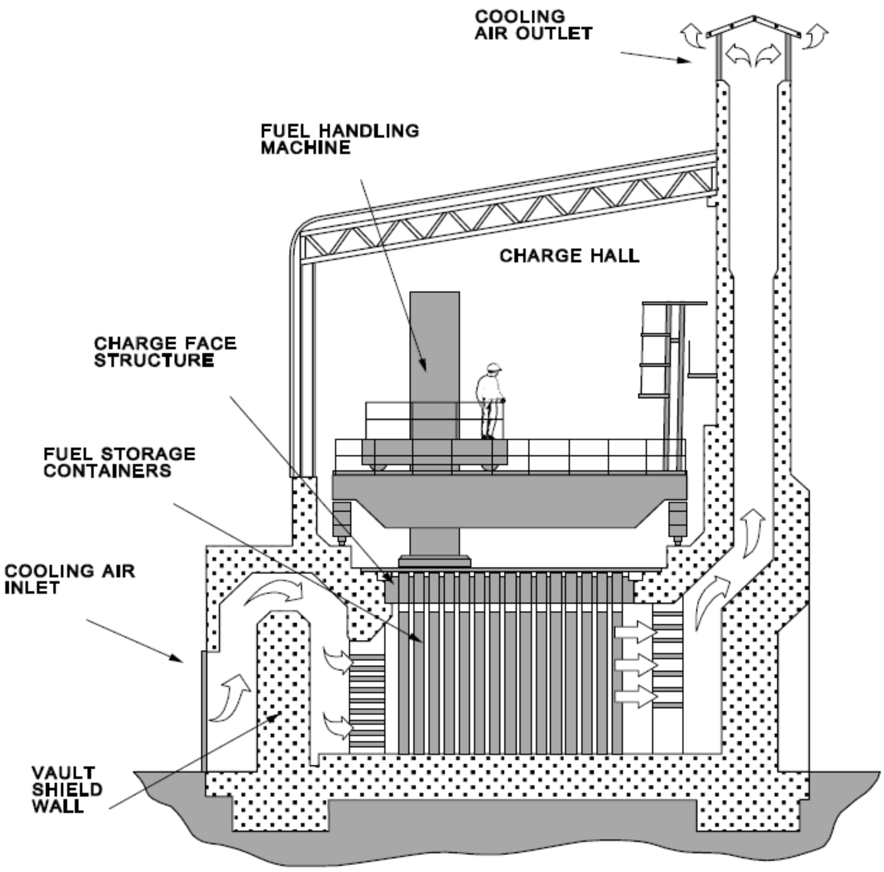


Figure 6.1.4-1 Schematic view of a modular vault dry storage (MVDS)

7. Evaluation Parameters

The evaluation parameters have been chosen to evaluate the options against the following broad requirements:

- The radiaton dose (including any committed effective dose, if applicable) to operators and members of the public is minimized.
- The risk of radiological exposure to operators and members of the public is minimized.
- Radioactive material and radioactive waste will remain adequately controlled or contained so that it cannot leak or otherwise escape from such control or containment.
- The rate of production and total quantity of radioactive waste shall be minimized.
- Adequate arrangements for the regular and systematic examination, inspection, maintenance and testing of all systems and components that may affect safety are provided.

7.1. Parameters

The following parameters will be utilized in the evaluation of the potential storage options described in Section 6.

7.1.1. Initial Screening

The parameters included in the initial screening set the minimum requirements each of the potential interim storage options must meet. If a potential storage option cannot meet one of the initial screening parameters, the potential interim storage option will not be considered.

7.1.1.1. Proven and Practical

Each of the potential interim storage options must have been used to store similar light water reactor fuel for a significant duration to demonstrate its technical adequacy. If any of the potential interim storage options have any technical issues, those issues will be evaluated for their potential application to the UK ABWR. If they do not apply or can be adequately resolved, this will be discussed in the evaluation.

To the extent that public information is available, operating history of each of the interim storage options will be reviewed to identify lessons learned. These lessons learned will be discussed in the evaluation along with their potential implication on the UK ABWR.

7.1.1.2. Compliance with Design Requirements

Each of the potential interim storage options will be reviewed against the Design Requirements in Section 3. If a potential storage option cannot meet one of the design requirements, the potential interim storage option will not be considered. If proven design enhancements and/or changes can be implemented to meet the Design Requirements, they this will be discussed in the evaluation.

7.1.2. Direct Radiological Impact

Normal operation of the potential interim storage options will result dose to the operators and may result in potential dose to the public. Note that during normal operation, no release of radioactivity is expected and, therefore, it will not be included in the direct radiological impact. Potential release of radioactivity will be included in the radiological risk section.

7.1.2.1. Dose to Public from Normal Operation

When conservatively assuming that public members are located at the site boundary for extended durations on the order of 2,000 hours per year, it is possible for members of the public to receive a dose. The relative dose rate at a specified distance will be compared between potential interim storage options based upon normal operations with all spent fuels located in the interim storage facility.

7.1.2.2. Dose to Operators from Normal Operation

The relative dose to operators will be compared between potential interim storage options based upon normal operations including the dose to operators during the transfer of the spent fuels from the R/B SFP to the interim storage facility.

7.1.3. Radiological Waste

During normal operation of the potential interim storage options, radiological waste will be generated. The quantity and characteristics of the radiological waste will be evaluated along with its containment.

7.1.3.1. Radioactive Waste Generation

The quantity and characterization of the radiological waste for each potential storage option will be quantified and characterized. The quantity of waste generated will be based on normal storage operations including the operations to transfer the spent fuels from the R/B SFP to the interim storage facility.

7.1.3.2. Radioactive Waste Containment

The containment of radiological waste will be compared between potential storage operations. It is expected that all potential storage operations will adequately contain the radioactive waste. The method of containment and its robustness will be evaluated.

7.1.4. Radiological Risk

The relative radiological risk of each potential interim storage option will be evaluated by comparing the designed barriers for containment, radiation shielding, and subcriticality. The strength of each barrier will be evaluated and passive barriers will be given preference over active barriers.

7.1.5. Operation and Maintenance

For each of the potential interim storage options, the amount of operator actions necessary for normal operation and maintenance will be evaluated. The potential interim storage option with the least amount of operation and maintenance activities will be valued highest.

7.1.6. Generic Site Conditions and Faults

The performance of each potential interim storage option will be evaluated against each generic site condition and fault using the approach to value the option that provides a solution as near to the top of the following list as possible: avoid the hazard, design to achieve fault tolerance, maintain safe conditions by passive means rather than active systems, initiate protection automatically in preference to manually, and mitigate fault consequence.

7.1.7. Economic

For each of the potential interim storage options, the capital, operating, and decommissioning costs will be estimated. Using this cost information and an assumed schedule for implementation, the present value cost for each options will be determined.

7.2. Weighting

As listed above in Section 7.1, the parameters are grouped. The grouping of the parameters is listed in order of their relative significance and this same grouping will be used to weigh the evaluation results of each parameter.

The initial screening is used as a pass/fail. If the potential interim storage technology cannot pass the initial screening the option will be eliminated. Included in the initial screening is that the potential technology has been used to perform the same function previously and has been found to adequately perform its design function.

The highest weighting will be given to the direct radiological impact, which are the expected radioactive doses to be expected to the public and the operators. These are the doses that would be the result of normal operations.

8. Conclusion

As mentioned above, there are various interim safe interim storage systems for nuclear spent fuels which are in use around the world. Hitachi-GE is in the process of considering the options available to them for the spent fuel interim storage on the Generic Site for a single UK ABWR unit. At this stage of GDA (Step 2), the final decision is not yet to be made. The final decision on the most appropriate approach to the spent fuel interim storage will be taken by future licensees. However in order to demonstrate that the waste and spent fuels from a UK ABWR can be safely managed until disposal, Hitachi-GE will select a "preferred" option during GDA following the process described above. Hitachi-GE have committed that this "preferred" option will meet all license conditions and be consistent with the plant design and licensed operation to provide a safe method for the interim storage of the spent fuels.

9. References

Ref-1	Generic Site Envelope (GA91-9901-0010-00001, Hitachi-GE Nuclear Energy, Ltd)
Ref-2	Fault Studies to Discuss Deterministic Analysis, PSA and Fault Schedule Development (GA91-9901-0009-00001, Hitachi-GE Nuclear Energy, Ltd)
Ref-3	Initial Safety Case Report on Spent Fuel Pool (Document ID GA91-9901-0003-00001)
Ref-4	Summary of the Generic Environmental Permit Applications (GA91-9901-0019-00001, Hitachi-GE Nuclear Energy, Ltd)
Ref-5	Demonstration of BAT (GA91-9901-0023-00001, Hitachi-GE Nuclear Energy, Ltd)
Ref-6	Radioactive Waste Management Arrangements (GA91-9901-0022-00001, Hitachi-GE Nuclear Energy, Ltd)
Ref-7	Storage of Spent Fuel from Power Reactors 2003 Conference
Ref-8	IAEA Costing of Spent Nuclear Fuel Storage (No. NF-T-3.5)
Ref-9	Safety Assessment Principles for Nuclear Facilities (2006 Edition, Revision 1, Health and Safety Executive)
Ref-10	Technical assessment guides NS-TAST-GD-081 (Rev. 1) SAFETY ASPECTS SPECIFIC TO STORAGE OF SPENT NUCLEAR FUEL (2013/179146 issued June 2013, reviewed June 2015, Office for Nuclear Regulation)
Ref-11	Licence condition handbook (October 2011 , Office for Nuclear Regulation)
Ref-12	Radioactive Substances Regulation Environmental Principles
Ref-13	Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs