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Principles of Nuclear Safety

**Maurizio CUMO
Universita' di Roma "La Sapienza"
Palazzo Baleani
Corso Vittorio Emanuele II, 244
00186 Roma
ITALY**

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PRINCIPLES OF NUCLEAR SAFETY: SPENT FUEL MANAGEMENT

Maurizio Cumo

University of Rome “La Sapienza”
Department of Nuclear Engineering and Energy Conversion
Corso Vittorio Emanuele II, 244 - 00186 Rome, Italy
E-mail: maurizio.cumo@uniroma1.it

1. General

Depending on a number of conditions and considerations, two options are available for the spent fuel management.
The spent fuel contains relevant quantities of fissionable isotopes, which can be further used to produce energy. For this reason the spent fuel is not generally considered as a waste, which does not contain any valuable material, but it is a category on its own. And thus the first management option is the reprocessing, with the aim of separating its main components, i.e.:

- fission products, produced in reactor as consequence of the fission process; they have a relatively short life and are disposed of as medium level waste;
- actinides, produced in reactor by neutron absorption in heavy metals; they usually have an extremely long life (even > 100,000 years) and are disposed of as high level wastes;
- uranium, depleted in respect to the initial enrichment in ^{235}U; it may be used for further enrichment and may have a commercial value;
- plutonium, particularly ^{239}Pu and ^{241}Pu; it is fissionable, and may have a commercial value, since it may be recycled in the mixed oxides (MOX) fuel.

The second option is the direct disposal (without chemical reprocessing) as high level waste in geological disposal sites, after an adequate cool down period, which depends on the disposal site specification and may range from 10 to 100 years.
A new strategy is currently in research and development phase, i.e. partitioning and transmutation (P&T). This strategy includes partitioning, which is a way to separate the constituents of spent fuel in preparation for the transmutation and the transmutation itself. The transmutation is the transformation of long lived actinides into different nuclides by neutron absorption. Transmutation may be achieved in critical reactors, thermal and fast, in Accelerator Driven Systems (ADS), including a subcritical reactor, or in fusion reactors. This subject is further developed later.

In all the strategies described above there is the need for a final disposal facility, since there will be in all cases some quantities of long lived actinides. The quantities, however, starting from the same amount of initial spent fuel, are decreasing from direct disposal to reprocessing and finally to P&T. A flow chart describing the first two strategies - that are the only available currently at an industrial and commercial level - is included in Fig. 1 and 2.
Since there is a diffuse rethinking about the economic viability of fuel reprocessing the need for spent fuel storage is increasing all over the world. This is true both for operating plants, that might have their storage facilities with a limited capacity and might stop operations, and for plants in decommissioning, for which the removal of spent fuel from the plant is precondition for plant dismantlement. Some quantitative figures on this issue are reported in Tables 1 and 2, where there are also some future projections (values in thousands of tons of uranium and other heavy metals) (source IAEA, 1997).

Fig. 1 – Spent fuel management strategies.

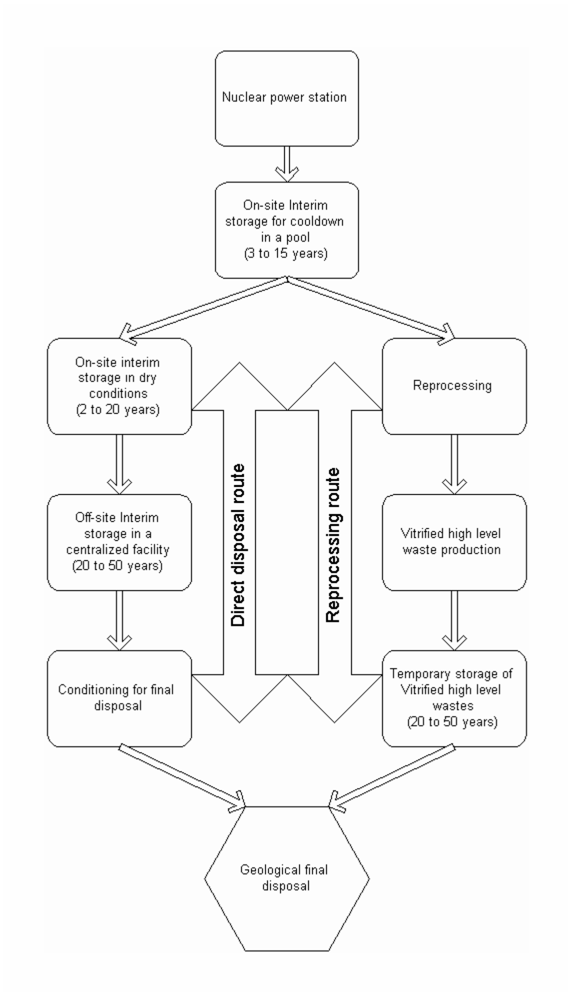


Fig. 2 – Spent fuel management strategies.

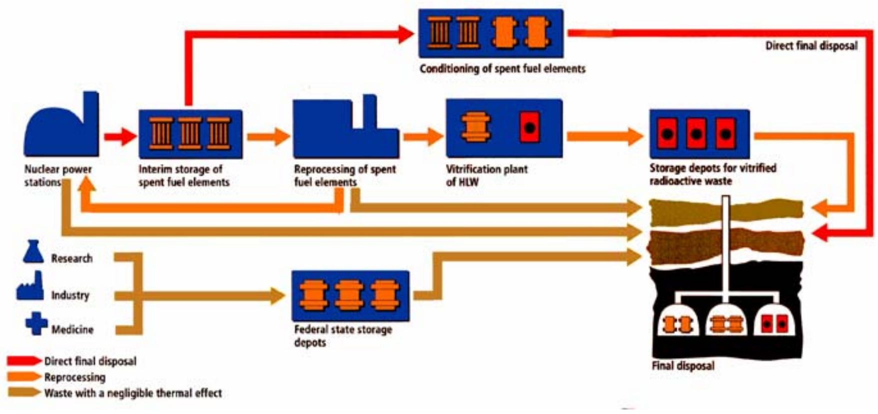


Table 1 - World commercial reprocessing capacity (tonnes per year). Sources: OECD-NEA 1999 Nuclear Energy Data, Nuclear Engineering International handbook 2000.

LWR fuel	France, La Hague	1600	
	UK, Sellafield (THORP)	1200	
	Russia, Chelyabinsk (Mayak)	400	
	Japan	90	
	Total		3290
Other types of nuclear fuels	UK, Sellafield	1500	
	France, Marcoule	400	
	India	200	
	Total		2100
Total civil capacity			5390

Table 2 - Spent fuel (heavy metal tons) stored in world areas.

Region	1997	2005	2010	2015
West Europe	34,2	40,1	38,9	36,4
Asia & Africa	12,5	27,6	38,6	50,2
East Europe	18,0	31,1	39,4	47,9
North and South America	64,6	91,3	108,4	125,9
Total world	129,3	190,1	225,3	260,4

2. Development of storage solutions

Dry storage facilities have been initially conceived as single purpose facilities as buffer short-time (10 to 20 years) storage before transporting the spent fuel to reprocessing or final storage or interim centralised storage. Storage systems were not initially qualified also for transport off-site and, therefore, the spent fuel had to be transferred into transport casks at the time of off-site delivery.
Afterwards It came evident that it would have been convenient avoiding spent fuel transfer from one storage system into a transport system at the time of its off-site delivery. This because the transfer operation is always a complex operation, because there might not be enough space in the plant pool, or because the pool does not exist at all, since the plant has to undergo a decommissioning operation. Typically for the double function (storage and transport) the metallic casks are well suited, since they are mainly transport casks with additional design provisions to be able to remain in the long term storage also. However, also some other technologies have developed special systems to allow off-site transport without transferring the spent fuel.
Recently in the USA also Multipurpose systems have been proposed, i.e. containers that would be qualified for on-site storage, transport and also for final geological disposal. The qualification, of course, would require the knowledge of the technical requirements of the geological disposal and the acceptability of final disposal without spent fuel conditioning. For these reasons in Europe there is not at the moment a container which is intended to be multipurpose.
At the same time the on-site storage periods became longer and longer, since both interim storage facilities and disposal sites were either not available or delayed (e.g., the case of the Yucca Mountain site for final disposal and the German case of Gorleben). Also for these reasons the licenses have been asked for longer periods up to 50 years. Moreover there are research programs to check the limiting storage time for these systems, identifying the expected degradation in the storage components and in the spent fuel itself up to 100 and even 300 years. The first results show that there are no strong indication of degradation for any components. It is therefore expected that in the future the licensing for these systems can extended.
A general world-wide overview of the situation is reported in Table 1 including both operating and under construction dry storage facilities. To complete the table it has to be mentioned that in Germany, after off-site spent fuel transportation has halted by a political decision, practically all nuclear power plant sites have started construction of dry storage. In Italy SOGIN decided to use the metallic dual purpose cask containers as the dry storage technology.

3. Available and future technologies

3.1 Reprocessing of spent fuel

Over the last fifty years the principal reason for reprocessing has been to recover unused uranium and plutonium in the spent fuel as valuable energy sources. A secondary reason is to reduce the volume of material to be disposed of as high-level waste. In addition, the level of radioactivity in such waste (fission products) after about 100 years falls much more rapidly than in spent fuel itself.
A typical balance for PWR fuel way may be the following:

- among recyclables there are 95% of uranium and 1% of plutonium;
- among the waste 4% of fission products.

The contribution of minor actinides is of the order of 0.1%, but they account for 90% of waste radiotoxicity.
In the last decade interest has grown in separating ('partitioning') radionuclides both to reduce long-term radioactivity in residual wastes and to transmute separated long-lived radionuclides into shorter-lived ones.
Reprocessing to recover uranium and plutonium avoids the waste of a valuable resource because most of the spent fuel can be recycled as fresh fuel, saving some 30% of the natural uranium otherwise required. It also avoids leaving the plutonium in the spent fuel, where in a century or two the built-in radiological protection will have diminished, allowing it to be recovered for weapons production (risk of proliferation).
So far, more than 75,000 tons of spent fuel from commercial power reactors have been reprocessed for U and Pu recovery, and the annual capacity is now some 5000 tons per year.
Spent fuel assemblies removed from a reactor are very radioactive and produce heat. They are therefore put into large tanks or "ponds" of water which cool them, and at least three meters of water over them shields the radiation to reasonable levels. Here they remain, either at the reactor site or at the reprocessing plant, for a number of years as, in this period, the level of radioactivity decreases considerably. For most types of fuel, reprocessing occurs anytime from 5 to 25 years after reactor discharge.
All commercial reprocessing plants use the well-proven hydrometallurgical PUREX process. This involves dissolving the fuel elements in concentrated nitric acid. Chemical separation of uranium and plutonium is then undertaken by solvent extraction steps. (Neptunium can also be recovered, if required, and maybe used for producing ^{238}Pu for thermo-electric generators for spacecraft.) The Pu and U can be returned to the input side of the fuel cycle - the uranium to the conversion plant eventually prior to re-enrichment, and the plutonium straight to fuel fabrication .
The remaining liquid after Pu and U are removed is high-level waste, containing about 4% of the spent fuel in the form of fission products and minor actinides (Np, Am, Cm). It is highly radioactive and continues to generate a lot of heat. It is conditioned by calcining and incorporation as dry material into borosilicate glass, then stored pending disposal. In principle any compact, stable, insoluble solid is satisfactory for disposal.
Another version of PUREX has the minor actinides (americium, neptunium, curium) being separated in a second aqueous stage and then directed to an accelerator-driven system cycling with pyroprocessing for transmutation (see below). The waste stream then contains largely fission products.
The PUREX process may also be modified to enable recovery of iodine by volatilisation and of technetium by electrolysis. French CEA research has shown 95% and 90% recoveries respectively. The same research effort has demonstrated a significant separation of caesium.
Recently (1999) a variation of PUREX has been proposed by the US Dept. of Energy for civil wastes. In this, only uranium is recovered (hence UREX process) for disposal as low-level waste, with iodine and technetium also being recovered at the head end. The residual is treated by pyroprocessing to recover transuranic elements (actinides), including plutonium for transmutation. The fission products become the main high-level waste. A major goal of this system is to keep the plutonium with the other transuranics to be destroyed by transmutation. Nevertheless, one version also has Pu recovered for commercial recycling as fuel, as in Europe, solution at present contrary to US policy.

3.1.1 History

A great deal of reprocessing has been going on since the 1940s, mainly for military purposes, to recover plutonium for weapons. In the UK, metal fuel elements from the first generation gas-cooled commercial reactors have been reprocessed at Sellafield for about 40 years. A 1500 t/y plant has been successfully developed to keep abreast of evolving safety, hygiene and other regulatory standards. From 1969 to 1973 oxide fuels were also reprocessed, using part of the plant modified for this purpose. A new 1200 t/yr thermal oxide reprocessing plant (THORP) was commissioned in 1994 and the corresponding MOX fabrication plant in 2001.

In the USA, no civil reprocessing plants are now operating, though three have been built. The first, a 300 t/yr plant at West Valley, NY, was operated successfully from 1966 to 1972. However, escalating regulations required plant modifications which were deemed uneconomic, and the plant was shut down. The second was a 300 t/yr plant built at Morris, Illinois, incorporating new technologies which, although proven on a pilot-scale, failed to work successfully in the production plant. The third was a 1500 t/yr plant at Barnwell, South Carolina, which was aborted due to a change in government policy which ruled out all US civilian reprocessing as one facet of US non-proliferation policy. In all, the USA have over 250 plant-years of reprocessing operational experience, the vast majority being at government-operated defence plants, since the 1940s.

In France one 400 t/yr reprocessing plant is operating for metal fuels from gas-cooled reactors at Marcoule. At La Hague, reprocessing of oxide fuels has been performed since 1976, and two 800 t/yr plants are now operating. India has a 100 t/yr oxide fuel plant operating at Tarapur with others at Kalpakkam and Trombay, and Japan is building a major plant at Rokkasho, while having most of its spent fuel reprocessed in Europe meanwhile. It has had a small (100 t/yr) plant operating. Russia has a 400 t/y oxide fuel reprocessing plant at Ozersk, Chelyabinsk.

3.1.2 Partitioning goals

Several factors give rise to a more sophisticated view of reprocessing today, and the simple introduction of the term partitioning reflects this. Firstly, new management methods for high and intermediate-level nuclear wastes are under consideration, notably partitioning-transmutation (P&T) and partitioning-conditioning (P&C), where long-lived radionuclides are the prime objects to separate out. Secondly, new fuel cycles such as those for fast neutron reactors (including lead-cooled ones) and fused salt reactors, and the possible advent of accelerator-driven systems, require a new approach to reprocessing. Here the focus is on pyrometallurgical processes (pyroprocessing) in a molten salt bath, with electrochemical separation.

The main radionuclides targeted for separation for P&T or P&C are the actinides neptunium, americium and curium (along with U & Pu), and the fission products iodine (I^{129}), technetium (Tc^{99}), caesium (Cs^{135}) and strontium (Sr^{90}). Removal of the latter two significantly reduces the heat load of residual, conditioned wastes. In Japan, the platinum group metals are also targeted, for commercial recovery. Of course, any chemical process will not discriminate different isotopes of any element.

Efficient separation methods are needed to achieve low residuals of long-lived radionuclides in conditioned wastes and high purities of individual, separated ones in transmutation targets. Otherwise, any transmutation effort is a random process with uncertain results. In particular, fertile U isotopes have to be avoided in a transmutation target ,or it will generate further highly radiotoxic transuranic isotopes.

To achieve effective full separation for any transmutation program is likely to mean pyroprocessing of residuals from the PUREX or similar aqueous processes.

A BNFL-Cogema study in 2001 reported that 99% removal of actinides, Tc^{99} and I^{129} would be necessary to justify the effort in reducing the radiological load in a waste repository. A U.S. study identified a goal of 99.9% removal of the actinides and 95% removal of technetium and iodine. In any event, the balance between added cost and societal benefits is presently subject of considerable debate.

3.1.3 Pyro-processing

Pyrometallurgical processing to separate nuclides from a radioactive waste stream involves several techniques: volatilisation, liquid-liquid extraction using immiscible metal-metal phases or metal-salt phases, electrorefining in molten salt, fractional crystallisation, etc. They are generally based on the use of either fused (low-melting point) salts such as chlorides or fluorides (e.g., LiCl + KCl or LiF + CaF_2) or fused metals such as cadmium, bismuth or aluminium.

Pyroprocessing can readily be applied to high burn-up fuel and fuel which has had little cooling time, since the operating temperatures are high already. However, such processes are at an early stage of development compared with hydrometallurgical processes already operational.

Separating (partitioning) the actinides contained in a fused salt bath involves electrodeposition on a cathode, extraction between the salt bath and a molten metal (e.g., Li), or oxide precipitation from the salt bath.

Many pyroprocessing techniques are at an early stage of development, and only one has been licensed for use on a significant scale. This is the US IFR process developed by Argonne National Laboratory and used for pyroprocessing the spent fuel from EBR-II experimental fast reactor which ran from 1963 to 1994. This application is essentially a P&C process, because neither plutonium nor other fissile transuranics are recovered for recycle. The process is used to facilitate the disposal of a fuel that could not otherwise be sent directly to a geologic repository. The uranium metal fuel is dissolved in LiCl+KCl molten bath, the U is deposited on a solid cathode, while the stainless steel cladding and noble metal fission products remain in the anode and are consolidated by melting to form a durable metallic waste form. The transuranics and fission products in salt are then incorporated into a zeolite matrix which is hot pressed into a ceramic composite waste. The highly-enriched uranium recovered from the EBR-II driver fuel is down-blended to less than 20% enrichment and stored for possible use.

The PYRO-A process, being developed at Argonne to follow the UREX process, is a pyrochemical process for the separation of transuranic elements and fission products contained in the oxide powder resulting from denitration of the UREX raffinate. The nitrates in the residual refined acid solution are converted to oxides, which are then reduced electrochemically in a LiCl-Li_2O molten salt bath. The chemically more active fission products (e.g., Cs, Sr) are not reduced and remain in the salt. The metallic product is electrorefined in the same salt bath to separate the transuranic elements on a solid cathode from the rest of the fission products. The salt bearing the separated fission products is then mixed with a zeolite, to immobilise the fission products in a ceramic composite waste form. The cathode deposit of transuranic elements is then processed to remove any adhering salt and is formed into ingots for subsequent fabrication of transmutation targets.

Another pyrochemical process, the PYRO-B process, has been developed for the processing and recycle of fuel from a transmuter reactor. A typical transmuter fuel is free of uranium and contains recovered transuranics in an inert matrix, such as metallic zirconium. In the PYRO-B processing of such fuel, an electrorefining step is used to separate the residual transuranic elements from the fission products and recycle the transuranics to the reactor for fissioning. Newly-generated technetium and iodine are extracted for incorporation into transmutation targets, and the other fission products, with shorter life, are sent to waste.

3.2 Direct disposal

The direct disposal of the spent fuel requires a first period of interim storage in the plant pool, a second period in interim dry storage and finally the disposal in an adequate container.

3.2.1 Wet interim storage

For a long time this has been the only way to store the power reactors spent fuel after their defueling. Each nuclear power plant with LWRs has a pool with different sizing criteria, at least to assure the place to store in adequate racks for the spent fuel produced in 10 years

of operation plus a full core in case of the need to completely defuel the reactor.
The water of the pool has both the function of removing the decay heat through cooling systems and of beta-gamma and neutron shielding. The shielding function requires a minimum depth of several meters.
In many cases the need of spent fuel storing exceeds the former design values. This may happen mainly in relation to the difficulties experienced in most countries to send the fuel to reprocessing or to a final disposal site. This need generally does not apply to a decommissioning phase, because at the time of the final shutdown it is needed only to defuel an entire core and the lodgements in the racks shall be always available for an entire core. In addition, at an early stage of the decommissioning, it is necessary to remove the spent fuel from the pool to proceed with dismantlement.
However, if an extension of wet storage capacity is needed, several alternatives are available, including:

- recovery of spent fuel pool capacity by removal of non-fuel items;
- re-racking (i.e., increasing the fuel assemblies density, with added poisons, in the racks);
- rods consolidation;
- trans-shipping of spent fuel to another existing pool with free space available;
- construction of an additional spent fuel storage pool.

3.2.1.1 Pool re-racking

Subcriticality was originally maintained for LWR spent fuel (without credit to burnup) by spacing within the storage racks or baskets. However, with the need to store greater quantities of fuel, higher storage density has been achieved by the introduction of neutron absorbing materials in storage racks and baskets such as borated stainless steel or Boraflex. Reracking has been often considered the first choice and the least-cost alternative for expansion of at-reactor spent fuel storage capacity. Some utilities have done so several times.
Adoption of neutron absorbing materials increase rack costs and must be considered with care, taking account of existing operating experience. For example, some applications of Boraflex in existing racks have caused severe problems due to the contraction of such material under heavy irradiation and some consequential concerns have arisen about the maintenance of subcriticality calculated safety margins.
Long and very long wet storage is not really an issue for fuel and fuel cladding. Some zircaloy clad fuel has been wet stored satisfactorily for over 40 years. In this case care shall be given to maintain the required quality of the pool water.

3.2.1.2 Spent fuel consolidation

Rods consolidation involves the disassembly of the intact fuel assembly by removing the fuel rods from the assembly hardware (shrouds, grids, etc) and reconfiguring them in a metal storage container. Fuel rod volume reduction of up to a factor 2 has been achieved placing the rods from two FA’s into a container not larger than a single assembly. After the fuel rods are removed, the non-fuel bearing components (i.e. end fittings and spacers) can be compacted and placed in a separate container. In this latter case a volume reduction ratio of 6:1 can be accomplished.
Spent fuel consolidation demonstration programs have been conducted in the US (Oconee, Main Yankee, West Valley, Battelle Columbus, Millstone 2, and Prairie Island), but no utility has yet proceeded with rod consolidation on a full-scale basis.
Reasons for lack of interest include:

- need for license amendment to address criticality issues;
- seismic and structural analysis for the increased loads associated with storing consoli-

dated fuel;
- thermal hydraulic analysis to demonstrate the capacity of the pool cooling system to provide adequate coolant flow to remove decay heat from the consolidated fuel canisters;
- increase of release of corrosion products (crud) during the consolidation operations, with consequent spent fuel pool visibility problems and increase in duty and cost of pool water clean-up system;
- possible increase in worker exposure due to crud and failed fuel fission product release.

3.2.1.3 Independent wet storage pool

Independent, or Away-From-Reactor (AFR), pools are basically similar to At-Reactor (AR) pools. They receive fuel from AR pools in either wet or dry transport canisters and store it in stationary racks or movable baskets.
A wet storage facility may have the following features:

- cask reception, decontamination, unloading, maintenance and dispatch;
- underwater spent fuel storage (pool);
- auxiliary services (radiation monitoring, water cooling and purification, solid radioactive waste handling, ventilation power supply, etc.).

Usually this solution is reasonable for large quantities of spent fuel (usually for many reactors) and in a situation where an intensive nuclear program is underway and will be maintained for a long time. In addition, also in relation to the high operational costs, it is considered a rather short term buffer system for more long-term solutions and/or for reprocessing. AFR pools are in front of all reprocessing plants.

3.2.2 Dry interim storage

Since 80's the technologies of dry storing the spent fuel, after an initial cool down in a pool, have been gradually introduced for several applications.
This overview focuses on solutions already licensed and operated for storing spent nuclear fuel in a inert, dry environment. The feasibility of dry storage has been essentially demonstrated for all types of spent fuel. All types of storage concepts (metal casks, modular vaults, concrete casks, concrete silos and concrete canister-based storage systems) have exhibited good performances. In the following, the most significant concepts are outlined.
Dry storage facilities can be Away-From-Reactor (AFR) or also At-Reactor (AR). Dry storage alternatives include:

- metal casks;
- concrete casks;
- vaults;
- concrete modules.

3.2.2.1 Metal casks

Metal casks are massive containers used in transport, storage and eventual disposal of spent fuel. The structural material for the metal casks may be forged steel, nodular cast iron, or a steel/lead sandwich structure. They are fitted with an internal basket or sealed metal canister, which provides structural strength and also assures subcriticality. Metal casks may have a single or a double lid, generally bolted. In any case, they have a double seal and the volume between the seals is monitored for leak tightness.

Metal casks are usually transferred directly from the fuel loading area to the storage site. Some metal casks are licensed for both storage and off-site transportation. Fuel is loaded vertically into the casks, which are usually stored in a vertical position.
There are several vendors of metal casks. In Europe GNB (Germany) and Transnucleaire (France), while ENSA in Spain is developing a model on the basis of NAC design. In the USA, in addition to the European vendors, in the market there are also NAC and Westinghouse.
Metal casks are used in a number of countries such as Germany, the USA, the Czech Republic, Switzerland, Spain and Italy. In Europe most of the casks are licensed for both storage and transport.
In general, metal casks are expensive components. However, they provide the maximum flexibility, since, if they are licensed also for transport, they can be moved off-site without any further direct handling of the fuel and without leaving structures to be decommissioned. For limited quantities of fuel (e.g. up to 500 t HM) they may also be economically convenient. Another advantage is that they are highly modular and the supply may be adapted temporarily to the need. In the total cost, expenses for a storage building should be included.
As an example in the Table 3 main parameters of the metal cask CASTOR™ 440/84 (Fig. 3) supplied by GNB are given. This model is the most widely used for VVER fuel, including Greifswald (Germany) and Dukovany (Czech Republic).
Casks are protected from all design accidents (air crash included), and can be stored outdoor (Fig. 4). However, for practical reasons, in most countries they are hosted in a dedicated building. A typical layout of these buildings is presented in Fig. 5 and 6. Major functions of these buildings are to contribute to shielding and to enhance air circulation.

3.2.2.2 Vaults

A vault (Fig. 7) is an above- or below-ground reinforced concrete building containing arrays of storage cavities suitable for containment of more fuel units. Shielding is provided by the exterior structure. Heat removal is normally accomplished by forced or natural convection of air or gas over the exterior of the fuel containing units or storage cavities, and subsequently exhausting this air directly to the outside atmosphere, or dissipating the heat via a secondary heat removal system.
Typical features of the vaults are their modularity, which facilitates incremental capacity extension, separated shielding and containment functions, capability for containment monitoring, and a vertical fuel loading methodology.
Spent fuel is received (either dry or wet) at a vault facility using transfer or transportation casks. Spent fuel is removed from the cask, prepared for storage if needed, and placed in a metal storage tube (single fuel element) or a storage cylinder (single or multi-element canister) which is housed within a concrete storage cavity in the vault structure. The storage tubes or storage cylinders are sealed and may be backfilled with an inert gas to improve heat transfer from the fuel and prevent oxidation of spent fuel while in storage. They are usually fitted with connections to a continuous or periodic monitoring system.
In vaults using metal storage tubes, fuel assemblies are dried, as necessary, and transferred one by one directly into their storage locations. Typical components of this type of storage facility are the vault modules, the fuel handling machine operating in the charge hall, the cask receiving area and the auxiliary facilities (areas for plant control, maintenance, services, offices, etc.).
Single tube vaults are in place at Paks NPP in Hungary and at Wylfa NPP in UK.
Storage cylinders vault receive the fuel already sealed in containers. They are used in Gentilly-2 in Canada (CANSTORE application of the MACSTORE system), the CASCAD facility in France, Fort St. Vrain in the USA.

Fig. 3 – CASTOR 440/84 metal cask (GNB).

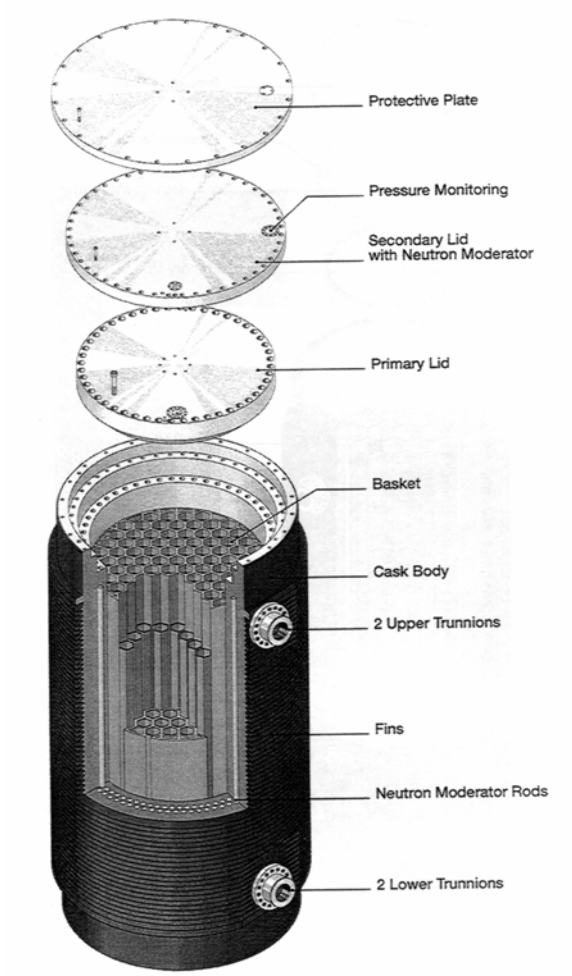


Table 3 - CASTOR™ 440/84 metal cask parameters (GNB).

Max heat output	*21 kW*
Max enrichment	*3,65 %*
Max burnup	*42 GWd/tU*
Typical cooling time	*60 months*
Cask length	*4.080 mm*
Cask diameter	*2.660 mm*
Wall thickness	*370 mm*
Cask body material	*Ductile cast iron*
Lid material	*Stainless steel*
Mass of transport configuration	*131 t*

Fig. 4 – Indoor and outdoor casks storage options.

Fig. 5 – Cask storage building (SOGIN).

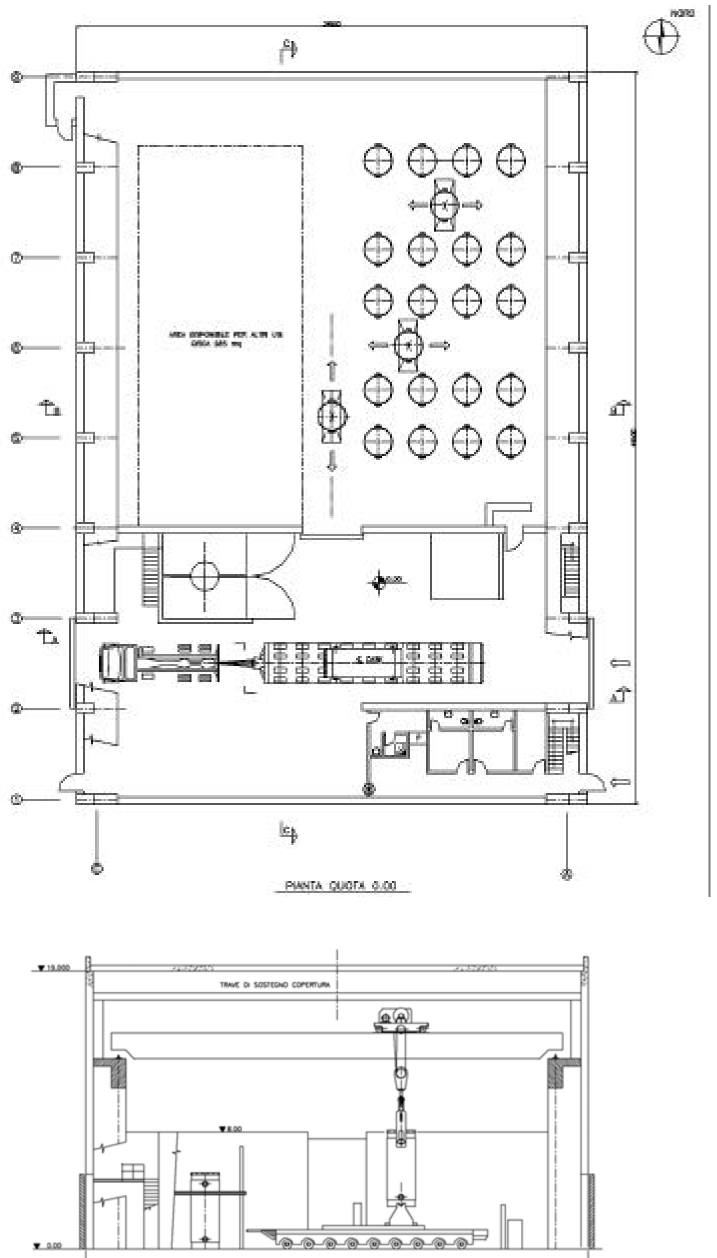
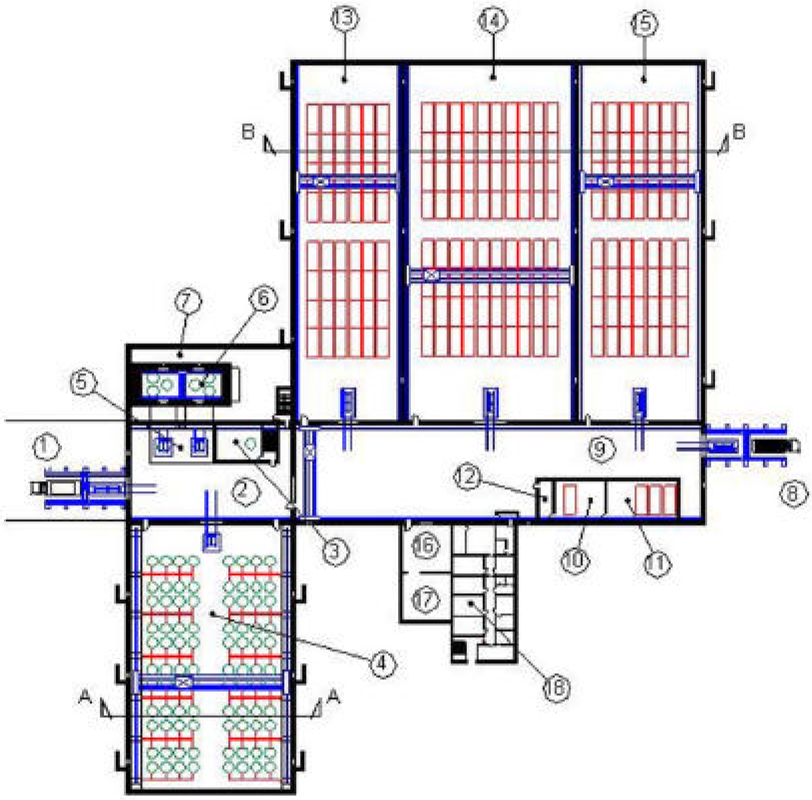
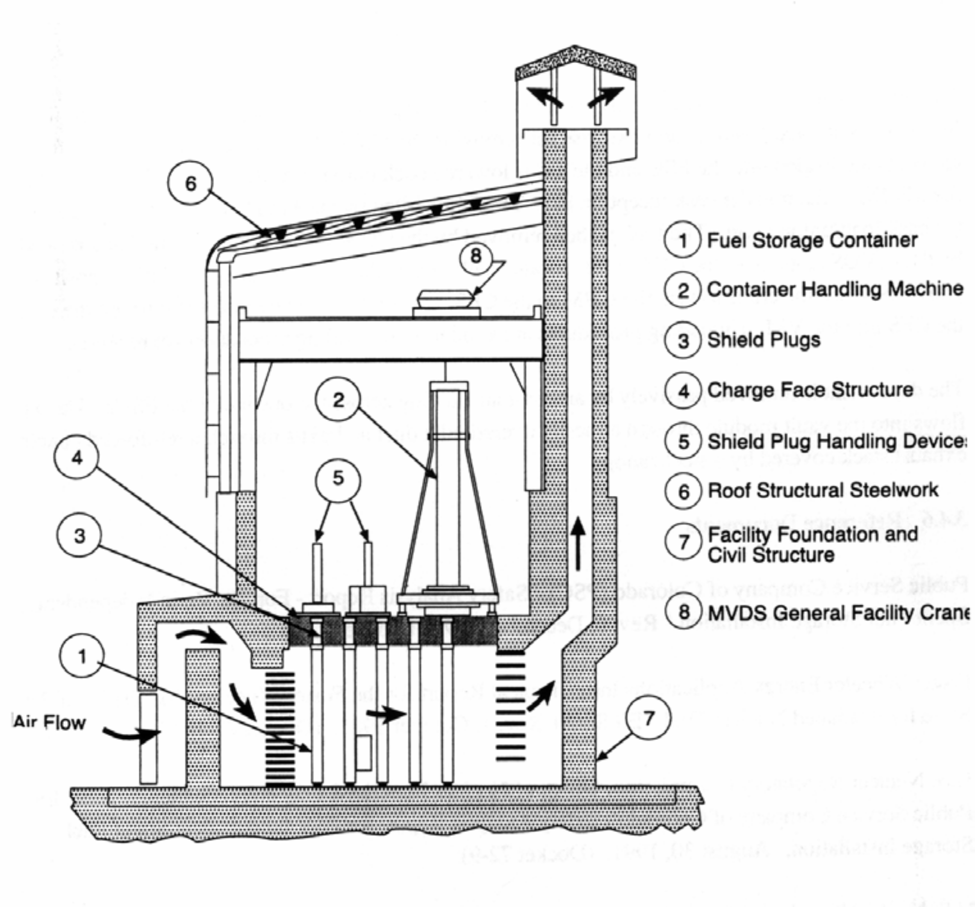


Fig. 6 – Combined waste and cask storage building (SOGIN).



SPENT FUEL AND VITRIFIED WASTE SECTION	OTHER RADIOACTIVE WASTE SECTION
1. Loading and unloading zone	*8. Loading and unloading zone*
2. Reception zone	*9. Reception zone*
3. Testing and preparation zone	*10. Testing zone*
4. Storage hall	*11. Out-of-standard storage hall*
5. Hot cell input zone	*12. Laboratory*
6. Hot cell	*13. Storage hall no. 1*
7. Hot cell control hatch	*14. Storage hall no. 2*
	15. Storage hall no. 3
	16. Workshop
	17. Warehouse
	18. Housing and offices

Fig. 7 – Typical vault facility (NRC).



Generally, vault facilities are more convenient for large inventories of spent fuel, since they require a number of auxiliary systems. They also need some efforts to be considered for decommissioning. One advantage is that they can be built in modules (all expenses for auxiliary systems are however upfront), and that the facility is completely independent from NPP facilities since it can provide also necessary equipment for off-site transportation of the fuel in transport casks.

3.2.2.3 Concrete casks

Concrete casks are movable structures with one storage cavity. They are used in storage and, in some cases, transport of spent fuel. Structural strength and radiological shielding are provided by reinforced concrete or high density concrete.

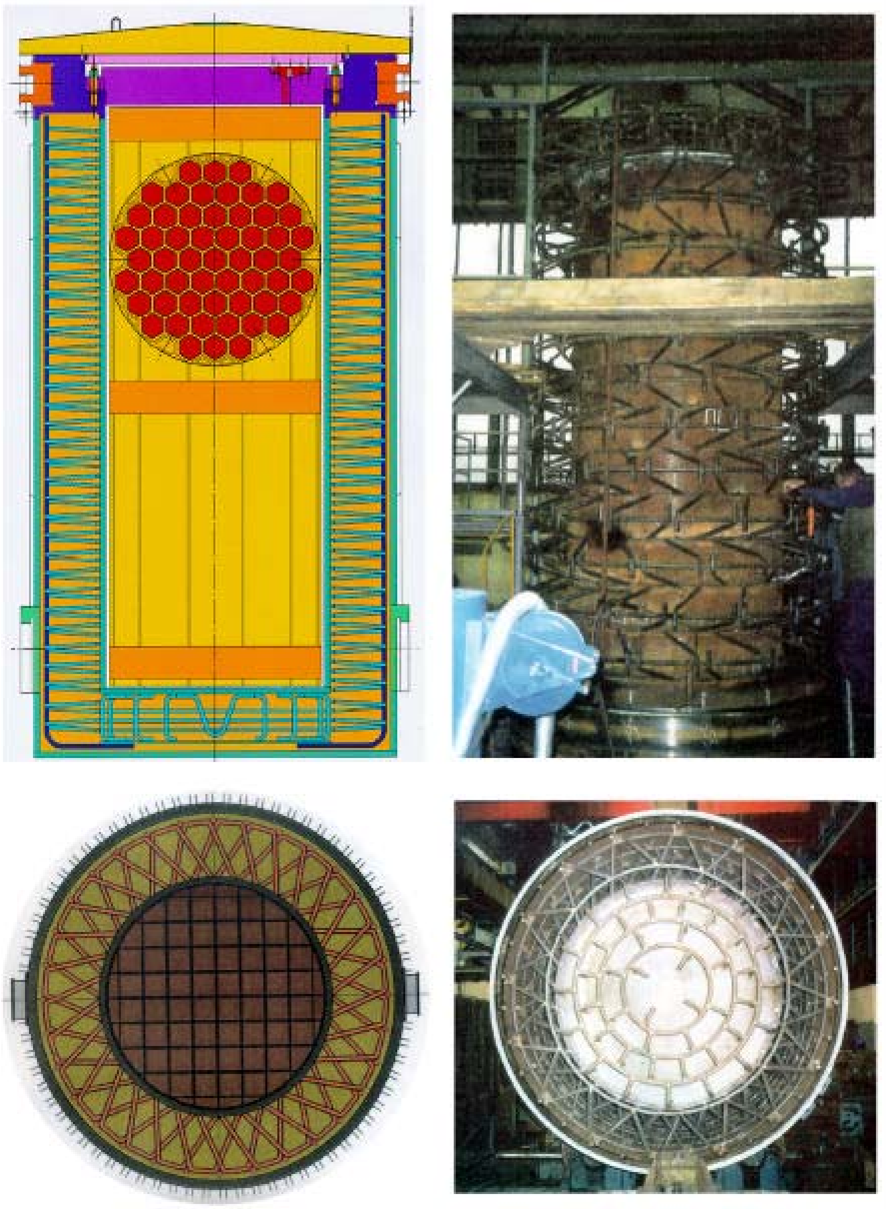
Concrete cask systems generally use sealed metal canisters housed inside the concrete storage cask to contain spent fuel. The metal canister may be cooled by natural convection of environment air and use a double lid closure system.

Sealed metal canisters may be contained in an on-site transfer cask for loading spent fuel from the fuel loading station and for transfer to the concrete storage cask.
Spent fuel may be loaded directly into a concrete cask in the fuel loading station and the concrete cask may be transferred directly to the storage site. Some sealed metal canisters may be licensed for transportation as part of an off-site transportation package.
Alternatively, concrete cask systems may use a metal liner in the cask cavity to contain spent fuel and a single lid closure system. Heat transfer may take place solely by conduction through the concrete structure.
Concrete casks that rely on conductive heat transfer have more thermal limitations than those using natural convection air passages.
Fuel is loaded vertically into the concrete casks and the concrete cask systems are stored in a vertical orientation.
Concrete casks use single or double lid systems, are welded closed and tested for leak tightness. Concrete cask systems may, or may not, be monitored for leak tightness.
Example of vertical concrete casks include Sierra Nuclear VSC cask and Ontario Hydro's Pickering concrete dry storage container, which is also designed for off-site transport.
GNB developed CONSTOR casks (Fig. 8) for VVER and RBMK fuel, which are made of a material sandwich. It consists of an outer steel shell, a reinforced heavy concrete layer and an inner steel shell (40mm / 340mm / 40mm). The aggregates for concrete are 40% barite and 60% steel balls. Inside the concrete, steel reinforcement is arranged to improve the strength and heat removal properties.
The lid system is designed as a multi-barrier system. The bolted primary lid fulfils strength and shielding functions. The sealing plate and the secondary lid are welded to the forged steel ring after loading and servicing the cask. A leak tightness monitoring is not needed, because there is a specially qualified technique for welding, approved also in Germany for the POLLUX cask, i.e. for a cask for final disposal.
One of the main goals was to use the CONSTOR as a multipurpose cask for both transport and dry storage and, in principle, also for final disposal.
A further goal was the efficient and cost effective manufacturing by using conventional mechanical engineering technologies and commonly available materials. Another intent was to fabricate CONSTOR casks in countries not having highly specialised heavy industries.
Nevertheless, the basic requirements for CONSTOR were to fulfil both the international IAEA test requirements for safe transport and safety criteria for long-term interim storage of spent nuclear fuel, including hypothetical storage site accident conditions (drop, fire, gas cloud explosion, side impact).
A version exists also for VVER 440 fuel, which can hold up to 54 fuel assemblies. The cask has a bolted primary steel lid, a welded sealing plate and a welded secondary lid. Shock absorbers are available for transport. In storage configuration, a protective cap is placed on top of the cask.
CONSTOR can contain 54 fuel assemblies because the allowable heat input is reduced and may be licensed also for transport. Since it can be manufactured by conventional methods with conventional materials, its cost is significantly lower than that of the CASTOR™ metal casks. CONSTOR version for VVER 440 fuel requires 7 years cooling time, at an average burnup of 42 GWd/t of HM and a maximum enrichment of 3,8%. The total mass of the loaded cask is 95 t.

3.2.2.4 Concrete silos

Silo systems are monolithic or modular concrete reinforced structures. The concrete provides shielding, while containment is provided by either an integral inner metal vessel (liner), which can be sealed after fuel loading, or by a separate sealed metal canister. In silos, spent fuel may be stored in vertical or horizontal orientation. Fuel loading into silos always takes place at the storage site.

Fig. 8 – Sketch wiews and construction phases of a CONSTOR 440/54 concrete cask (GNB).



A typical example of a silo system is AECL's concrete canister, which is built on-site, using regular reinforced concrete, and is fitted with a inner steel liner. Spent fuel is transferred in increments within sealed baskets, using a shielded transfer cask, and loaded vertically. Once loading operations are complete, a closure shield plug is placed and welded to the inner liner, to provide additional containment.

3.2.2.5 NUHOMS modular system

NUHOMS (NUteck HOrizontal Modular Storage) is a family of dry storage facilities which are basically formed by a concrete structure into which metallic welded canisters are placed in horizontal position (Fig. 9).
NUHOMS have been developed under the name of various companies (Nuteck, Pacific Nuclear, Vectra Technologies Inc.). Outside USA, fabrication and commercial license has been granted to FRAMATOME. In 1998, Transnucleaire (company 100% owned by French COGEMA) purchased the NUHOMS™ system and created Transnuclear West, taking over all Vectra assets.
So far, NUHOMS system has been adopted only in the USA and in Armenia at Medsamor plant. The main designs are known as NUHOMS-7P, NUHOMS-24P and the standardized NUHOMS-24P/52B (where P stands for PWR fuel, B stands for BWR fuel and the associated number stands for the quantity of fuel assemblies that can be stored in a single canister). Recently the MPC (Multi Purpose Canister) 187 has been approved by NRC, both for storage and for transport with the addition of an overpack.

Fig. 9 - Main components of the NUHOMS storage system.

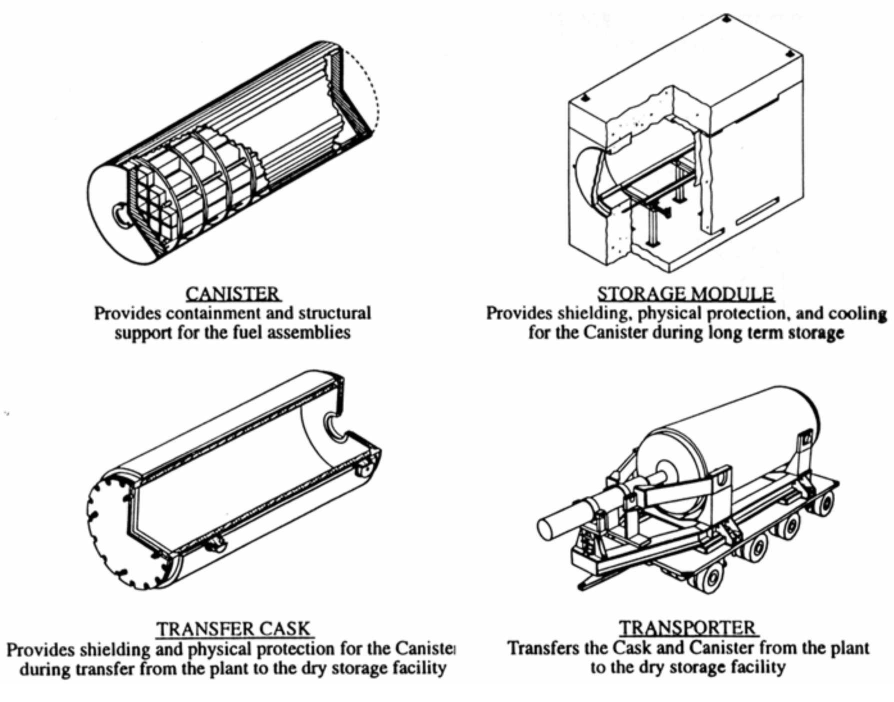


Table 4 - Design parameters for NUHOMS storage systems.

	Site specific licenses		*Standardized NUHOMS*	
	NUHOMS 7P	*NUHOMS 24P*	*NUHOMS 24P*	*NUHOMS 52B*
Initial enrichment ^{235}U	*3,5*	*4*	*4*	*4*
Burnup (MWd/t HM)	*35.000*	*40.000*	*40.000*	*35.000*
FA per DSC	*7 PWR*	*24 PWR*	*24 PWR*	*52 BWR*
DSC length (m)	*4,6*	*4,7*	*4,7*	*4,7*
DSC diameter (m)	*0,9*	*1,7*	*1,7*	*1,7*
DSC shell thickness (cm)	*1,3*	*1,6*	*1,6*	*1,6*
HSM length (m)	*5,9*	*6,1*	*6,1*	*6,1*
HSM heigth (m)	*3,7*	*4,6*	*4,6*	*4,6*
HSM width (m)	*1,7*	*2,7*	*2,7*	*2,7*
HSM concrete walls and roof thickness (cm)	*106*	*91*	*91*	*91*
HSM concrete interior walls thickness (cm)	*61*	*61*	*61*	*61*
Transfer cask length (m)	*4,60*	*4,75*	*4,75*	*4,75*
Transfer cask diameter (m)	*0,95*	*1,7*	*1,7*	*1,7*

The principal components of the NUHOMS are a stainless steel Dry Shielded Canister (DSC) with an internal fuel basket, a concrete Horizontal Storage Module (HSM) that protects the DSC and provides radiological shielding (overpack), a compatible Transfer Cask (TC) used to transfer the DSC from the spent fuel pool to the HSM, and a Hydraulic Ram System (HRS) used to insert the DSC into the HSM and TC (see Table 4).
The Dry Shielded Canister is designed to provide primary containment for 7 spent fuel assemblies. The DSC is a stainless steel cylinder. Stainless steel end plugs filled with lead are welded to the top and the bottom of the DSC with double seal welds. The canister contains a basket assembly made of guide sleeves consisting of borated stainless steel cladding (for criticality control) that are held by spacer disks. The basket geometry and the guide sleeves provide criticality control. The lower end of the DSC is coated with a lubricant to reduce friction when it is inserted and removed from TC and HSM.
The design includes three DSC types: one for fuel and control components, one for fuel only and one for failed fuel (which has not been licensed yet). The first may load 24 FAs with control components and has an internal cavity 4,4 m long. The other two have an internal cavity 4,2 m long and may load respectively 24 FAs or 13 failed fuel assemblies.
The HSM is constructed of reinforced concrete, structural steel and stainless steel. The HSM may be constructed as a single unit or as an array of modules (e.g. 2 x 20). Gamma and neutron shielding is provided by the HSM structure. A support rail structure anchored inside the HSM by the interior walls supports the DSC and extends to the access opening. Stoppers on the rails prevent horizontal movement of the DSC during a seismic event. A vertically sliding plate, consisting of thick steel and neutron-absorbing material, covers the entrance to the HSM and is tack welded close when the DSC is in place. Each HSM has two shielded air inlets on the front and two shielded air outlets on the roof.
NUHOMS-7P are in use in the US at H.B. Robinson (license 8/86). NUHOMS-24P are used at Oconee (license 1/90) and Calvert Cliffs 1&2 (license 11/92). Standardised HSM are used or planned to be used at Oyster Creek, Susquehanna 1&2), Davis Besse (license 1995).
More recently NUHOMS-MP187 dry storage canisters have been approved for off-site transport. The DSC is identical to the standardised NUHOMS-24P design, but includes borated neutron absorber panels. A license for use of NUHOMS-MP187 has been issued for

the Rancho Seco plant in decommissioning. It has to be underlined that the latter is the only plant in decommissioning using the NUHOMS technology, since an operational spent fuel pool is needed to prepare the fuel to transport off site. In fact the DSC shall be brought back to the pool in the inverse procedure. The DSC weld shall be cut, the fuel then transferred from the DSC into the transport cask.

3.2.3 Final disposal

Before the spent fuel may be finally disposed in a repository, it needs to be treated and conditioned to assure the highest safety levels of conservation in the very long period.
We might distinguish between preconditioning, which are processes that pre-treat spent fuel to make it suitable for conditioning and can take place also several years before conditioning for final disposal, and conditioning, which means the packaging of fuel in a containment suitable for final disposal.
The spent fuel, once it is declared a waste, must be isolated from the environment, and should meet the same safety and radiological protection standards required for the disposal of high level reprocessing waste. Guidance considerations are based on humans radiological protection and also environmental and natural resources protection. Protection levels are achieved by a combination of factors, i.e. the waste package, the engineered barriers within the repository and, mainly, the geological environment.
Several, specific technical solutions exist or have been proposed, but all of them include some steps listed below (some of them are optional and may depend on the final decision on the characteristics of the repository):

- Transport of the spent fuel to the conditioning facility;
- Unloading of the spent fuel;
- Disassembly of the fuel assemblies (*);
- Rods consolidation (*);
- Cutting of the rods (*);
- Determination of the radionuclide inventory and decay power;
- Loading of the fuel into the disposal container;
- Embedding of the fuel with matrix material (*);
- Sealing of the disposal container;
- Acceptance tests of the waste package.

(*) = If required, depending on the repository and specifications

Specifications are related to the following factors:

- Safety requirements - including containment, subcriticality, heat dissipation, operational doses limitation, protection against external events and external intrusions;
- Spent fuel types, properties and quantities - Size, design, structural materials, burnup and uranium chemical forms are important elements to be considered and challenges may be increased by the current trend of increasing fuel burnup (> 60 GWd/t);
- Geological setting of repository - including mechanical loads, groundwater flow regime and geochemical environment;
- Repository design - including acceptable dimensions and weight of the waste packages and their handling design, either horizontal or vertical;
- Quality control;
- Safeguards.

The waste package consists of the consolidated and/or unconsolidated spent fuel, including the non-fuel-bearing components (if present) and the container. The container consists of the external packaging of the waste and includes any internal structure for the positioning of

the waste form and for heat transfer to the surface of the container. In some cases, a neutron moderating material may be included.
In most conceptual designs under current consideration the waste package satisfies the following requirements:

- it is unshielded;
- it requires a transport cask;
- it is placed into concrete casks, metal casks or concrete vaults for interim storage.

For waste packages designs that meet the type B(U) criteria according to IAEA classification for transport of waste packages, the package becomes a cask with the following properties:

- it is self shielded;
- it does not need a transport cask;
- the cask can be used directly for interim storage.

The functional requirements of the waste packages are divided into two groups covering different periods of the waste package lifetime, i.e. the repository operational period and the repository post-closure period.
The first group can be specified as follows:

- To contain the spent fuel inventory during unloading, handling, storage, further packaging at the repository surface facilities, and emplacement into the repository facility;
- To control the release if containment is lost during a handling operation;
- To limit the potential for criticality within spent fuel waste packages;
- To provide for safe handling and shipping operations;
- To provide for unique identification;
- To provide for specific and legible markings.

Those related to the repository post-closure period can be specified as follows:

- To contain the radionuclides for a specific period of time;
- To contribute to controlling the rate of release of radionuclides after the containment period.

Among the various design concepts that have been proposed, or are at an advanced stage of implementation, here, as an example, the German proposal is described.
The repository that has been chosen in Germany in the 90s is a salt dome in the north of Germany (Gorleben). According to the criteria approved by competent authorities, a final disposal container must ensure safe isolation of radioactive material in the repository for at least 500 years and protection against mechanical loads as well as against an improbable attack from salt brines. The necessity for a corrosion protection layer will have to be determined in the licensing procedure of the repository.
The reference design of the containers is called Pollux and comprises a Pollux cask for disposal in drifts and a Pollux container for disposal in boreholes. The present version of Pollux container is designed for the final disposal of spent fuel in boreholes together with vitrified HLW. Transport and an optional interim storage, however, require an additional shielded cask. The dimensions (diameter 430 mm, length 1335 mm) correspond to the size of the vitrified waste canisters, which are delivered to Germany by Cogema of France. The container main host five PWR fuel assemblies cut into pieces. The container length could be increased to the full length of the fuel rods if a suitable repository were to become available.
The final disposal cask can accept four canisters with fuel rods. The cavity is filled with helium to reduce the fuel rods temperature enhancing the heat transfer. The gap between the outer shielding overpack and the cask is filled with a neutron moderating material and heat

transferring fins. The total weight amounts to 64 t. For fuel with a three years decay time, the radiation dose level at the surface is less than 0,2 mSv/h.

3.3 Actinides transmutation

Transmutation can be defined as the transformation of one isotope into another one by changing its nuclear structure. In the context of conditioning the constituents of spent fuel, transmutation converts plutonium and other actinides and long-lived fission isotopes into isotopes with more favourable characteristics (shorter life, lower radioactivity).

Nuclear transmutation can be achieved by exposing isotopes to neutrons flux in a nuclear reactor (a system designed to maintain a steady level of neutrons in a self-sustaining configuration) or in an accelerator-driven subcritical system. In the latter one, neutrons result when an accelerator beam of high-energy particles (protons, for example) collides with a dense, high-atomic-number target (this reaction and the resulting neutrons are referred to as "spallation"). These highly energetic neutrons are then multiplied through interactions with fuel materials in a surrounding blanket arrangement. In either case, the exposure of materials to neutrons results in their transmutation.

The most innovative systems are the ATW (Accelerator driven Transmutation technology for Waste), which are studied and developed in some countries around the world.

In an ATW a high-power particle accelerator produces energetic protons that react with a heavy metal target to produce neutrons. This target is situated at the centre of a "blanket" region filled with assemblies containing chemically separated long-lived transuranic and fission fragments. The target and blanket assemblies together are called a transmuter. The fissionable transuranics are arranged such that neutron chain reactions cannot be sustained without the introduction of an external neutron source, which is provided by the accelerator. Thus, the transmuter is "driven" by the neutrons produced when the accelerated protons strike the target (this is called subcritical operation, which means that the accelerator protons are necessary to keep the transmuter running).

The neutrons from the target, multiplied by neutrons from fissioning transuranics in the blanket, cause other transuranics to fission (releasing heat) and also transmute long-lived fission products into shorter-lived ones or stable products (i.e., Se^{79}, Zr^{93}, Tc^{99}, Pa^{107}, I^{129}, Cs^{135}). Materials separations processes are used to prepare (partition) spent fuel for introduction into an ATW system; materials separations are also used within an ATW system for material recycle.

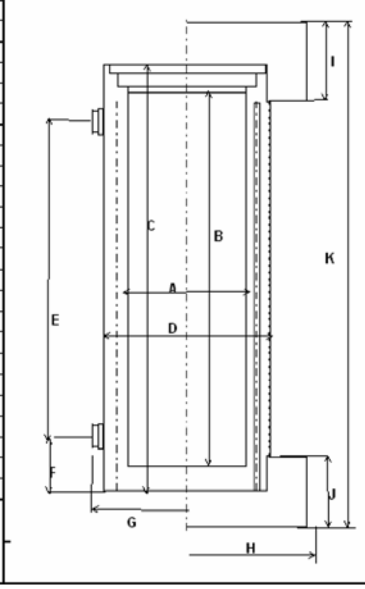
Operation of an ATW system produces sizeable amounts of heat that can be converted to electricity.

Discharged products from an ATW system must be packaged in customised waste forms and disposed of as radioactive waste, or in the case of uranium, may be stored for future uses. The ATW system is also described as partitioning and transmutation. Partitioning means that materials of interest are chemically separated from the much larger mass of discharged reactor spent fuel and prepared for transmutation, recycle, and/or disposal. Transmutation means that nuclear processes are used on the transuranics (actinides) and long-lived fission products from partitioning the spent fuel, to achieve objectives of further energy recovery and/or creation of less-radioactive by-products or shorter-lived by-products to be placed in a repository.

Efforts are being underway in Europe (France, Italy, Spain), Russia, Asia (Japan, Korea), and in other countries. These nations are pursuing ATW technology as a possible component of their long-term energy supply strategy. In the U.S., Los Alamos National Laboratory (LANL) has studied ATW technology since 1991 and has developed powerful accelerators for the Accelerator Production of Tritium program, that are of the class proposed for ATW. Argonne National Laboratory (ANL) has developed partitioning technologies designed to retain self-production of plutonium and transuranics at all times as part of the program of

Fig. 10 – Main characteristic of the CASTOR XB/52 cask developed to store the Caorso NPP (Italy) spent fuel (GNB-SOGIN).

Dimension [mm]		
Cavity diameter	A	1390
Cavity length	B	4510
Total bottom wall thickness		300
Total cask length w/o impact limiter	C	5152
Total side wall thickness		285
Outer diameter		1960
Trunnion center line distance	E	3800
Distance trunnion / cask bottom	F	660
Excentricity of bottom trunnion		30
Turning radius - bottom side		1194
Width across trunnions	G	2240
Diameter of outer recess of trunnion		280
Length of outer recess of trunnion		10[1] /5
Diameter of 1. recess of trunnion		250
Length of 1. recess of trunnion		85[1] /65
Diameter of 2. recess of trunnion		300
Length of 2. recess of trunnion		0[1] /55
Outer diamter impact limiter	H	2830
Length top impact limiter	I	935
Length bottom impact limiter	J	850
Total length with impact limiter	K	6082
[1] valid for 4 lid side trunnions		
Data Sheet	Var.	Rev.
Transport and Storage Cask	B	5
CASTOR X/B52	Date	02.02.00
	Author	Hüggenberg



Masses kg rsp. Mg	
Number of F/A	52
Number of CB	52
A. Mass of Components [kg]	
Cask body	49109
Basket	8829
Inventory	16848
Primary lid, bolts, moderator	3672
Secondary lid, bolts	2082
Cover plate, bolts	2173
Trunnions, bolts	775
Water during loading	3899
Impact limiter	8700
Bouyancy during loading	10238
Auxilary equipment	4500
B. Handling mass [Mg]	
At the pool: with Water, PL Aux. Eq., with Bouyancy	77,4
loaded, with PL, SL w/o water, Aux. Eq.	81,3
C. Transport configuration [Mg]	
loaded, with impact limiter without sec. lid, cover plate	87,9
loaded, with impact limiter without cover plate	90,0
D. Storage Configuration [Mg]	
loaded with cover plate, without impact limter	83,5

treating Experimental Breeder Reactor-II spent fuel for disposal.
Transmutation does not eliminate the need for a high deep final disposal repository for high level wastes, since there are always residual actinides that need to be disposed, even in much less quantities.

4. Metal containers design issues

4.1 Introduction

There is a variety of approaches to the task of demonstrating the structural adequacy of a cask. Additionally there are some requirements that differ for cask transport and cask storage. Differences are discussed in some details in the following and are particularly important for casks which should be qualified for both functions and therefore designed for the enveloping conditions. However, since casks have been primarily designed for transport and since in many cases the relevant requirements are the most demanding, this section is mainly oriented to the design of casks for transport.
One example of a metallic cask is CASTOR XB/52 that GNB developed for SOGIN to store the Caorso NPP spent fuel. Major characteristics of this cask are listed in Fig. 10.
Table 5 presents an example of a set of standards (reference country is Germany) to be applied to design different parts of the cask. Cask design is a highly iterative process, since many different and sometimes contrasting requirements must be complied with and optimised. A typical example of this iterative process is presented in Fig. 11.

Tab. 5 – Standards for cask design (Germany).

Component	KS	Project and manufacturing	Material	Welding	Control and testing
Cask body	1	IAEA TS-R-1 BAM Guidelines KTA 3201-2	WS 0.7040-04, Ind. 02	N/A	TRV 006 DIN EN 10204 BAM Guidelines WS 07040-04, Ind. 02
Internal basket	1/2	KTA 3201-2 KTA 3905	WS 1.4565-01, Ind. 00 WS 1.4462-01, Ind. 00 WS 1.4300-02, Ind. 02 WS 1.4306Bor-01, Ind. 01 WS Bor-Al-01, Ind. 00	DIN EN 25817 DIN EN ISO 13919	TRV 006 DIN EN 10204 WS
Primary lid	1	IAEA TS-R-1 BAM Prescriptions KTA 3201-2 FKM Standards	WS 1.4313-02, Ind. 01	N/A	TRV 006 DIN EN 10204 WS 1.4313-02, Ind. 01
Secondary lid	1	IAEA TS-R-1 BAM Prescriptions KTA 3201-2 FKM Standards	WS 1.4313-02, Ind. 01	N/A	TRV 006 DIN EN 10204 WS 1.4313-02, Ind. 01
Metallic sealing (aluminium)	1	BAM Prescriptions	WS 2.4969-01, Ind.00 (spur) WS 1.4307-01, Ind. 01 (inner) WS 3.0255-01, Ind. 00 (outer)	N/A	TRV 006 DIN EN 10204 WS
Metallic sealing (silver)	1	BAM Prescriptions	WS 2.4969-01, Ind.00 (spur) WS 1.4307-02, Ind. 00 (inner) WS Ag 99,99-01, Ind. 01 (outer)	N/A	TRV 006 DIN EN 10204 WS
Primary and secondary lid bolts	1	IAEA TS-R-1 BAM Prescriptions VDI 2230	WS 1.4313-01, Ind. 02	N/A	TRV 006 DIN EN 10204 WS 1.4313-01, Ind. 02
Pressure gage	1	IAEA TS-R-1	WS 1.4310-01, Ind. 00 (inner)	DIN EN 25817	TRV 006 DIN EN 10204 WS 1.4310-01, Ind. 00
Protection plate	2	FKM Standards	WS 1.0570-01, Ind. 01	N/A	TRV 006 DIN EN 10204 WS 1.0570-01, Ind. 01
Trunnions	1	BAM Prescriptions KTA 3905 FKM Standard	WS 1.4313-04, Ind. 01	N/A	TRV 006 DIN EN 10204 WS 1.4313-04, Ind. 01
Trunnion bolts	1	BAM Prescriptions KTA 3905 VDI 2230	WS 1.4313-01, Ind. 02	N/A	TRV 006 DIN EN 10204 WS 1.4313-01, Ind. 02
Shock absorbers	2/3	GNB Standards	DIN EN 10025 DIN EN 10029 WB52, WB54 (wood)	DIN EN 25817	TRV 006 DIN EN 10204 WB52, WB54
Moderating rods and plates	1	IAEA TS-R-1	WS 1.PE-HD-01, Ind. 02	N/A	TRV 006 DIN EN 10204 WS 1.PE-HD-01, Ind. 02

Tab. 5 (continued) – Legenda

KS:	*Component safety class in accordance with Technical Guide TRV-006*
IAEA TS-R-1:	*IAEA Safety Standards Series No. TS-R-1 (ST-1, Revised), "Regulations for the safe transport of radioactive material – 1996 Edition (Revised) – Requirements"*
BAM Prescriptions:	*Bundesanstalt für Materialforschung und -prüfung.*
BAM Guidelines:	*BAM, "Leitlinie zur Verwendung von Gußeisen mit Kugelgraphit für Transport- und Lagerbehälter für radioaktive Stoffe " Stand 4/8/2000* *BAM, "Gußeisen mit Kugelgraphit als Werkstoff für Transport- und Lagerbehälter bestrahlter Brennelemente", 1985*
VDI 2230:	*VDI Richtlinie 2230, "Systematische Berechnung hochbeanspruchter Schraubenverbindungen, Zylindrische Einschraubenverbindungen", 1986 (KTA)*
KTA 3201-2:	*KTA 3201.2, "Sicherheitstechnische Regel des Kerntechnischen Ausschusses (KTA) – Komponenten des Primärkreises von Leichtwasserreaktoren – Teile 2: Auslegung, Konstruktion und Berechnung", 1984.*
KTA 3905:	*KTA 3905, "Sicherheitstechnische Regel des Kerntechnischen Ausschusses (KTA) – Lastanschlagpunkte an Lasten in Kernkraftwerken", 1994.*
FKM Standards:	*Forschungskuratorium Maschinenbau, "Rechnerischer Festigkeitsnachweis für Maschinenbauteile – FKM Richtlinie", Frankfurt, 1998.*
GNB Standards:	*GNB B 01/94, "Festigkeitsverhalten von blechummantelten Probekörpern aus Fichten- und Buchenholz unter Schlag-Stauch-Bedingungen", Essen, 1994* *GNB B 35/94, "DROP – Dokumentation und Verifikation", Essen, 1994.*
TRV-006:	*Technische Richtlinie über Maßnahmen zur Qualitätssicherung (QM) und Überwachung (QÜ) für Verpackungen zur Beförderung Radioaktiver Stoffe, 1992.*
WS and WB:	*Material standards produced by GNB and approved by BAM (Bundesanstalt für Materialforschung und –prüfung).*
N/A:	*Not Applicable*

4.2 Structural analysis

Usually the structural design is performed in pieces: for example, the lift trunnions may be analysed by hand-calculations or a computer code, while the cask body shells are usually analysed by a finite-element computer code. This makes usually the task of integrating all these information into a Safety analysis Report difficult and requires attention to avoid inconsistencies.
The structural analysis for shipping casks requires special attention to two areas:

- calculation of the loads developed in impacts;
- the strength and stability of the structure that resists those loads.

Applied cyclic loads or vibrations must also be analysed, as well as thermal stress loads.
Similar questions exists for storage design, where different scenarios are considered, i.e. drops without shock absorbers and, in some countries, aircraft crash.
The materials which form the containment boundary of a cask must be qualified by elastic analysis: that is, they shall not yield in hypothetical accident scenarios. No containment components may function in the elastic-plastic regime.
Ductile materials are preferred for structural components, because they can absorb a significant amount of strain energy in an impact, without immediate failure of the component.
Material behaviour must be evaluated for temperatures as low as -40°C. Materials such as nodular cast iron, which may be less ductile at these temperatures, require special attention

during casting process to assure properties uniformity.
The locations of greatest attention for the structural design are the closure lids, the lid bolts and the cask mid-body, at which greatest bending moments may be applied.

Fig. 11 – Cask project flow-chart (NIS-SOGIN).

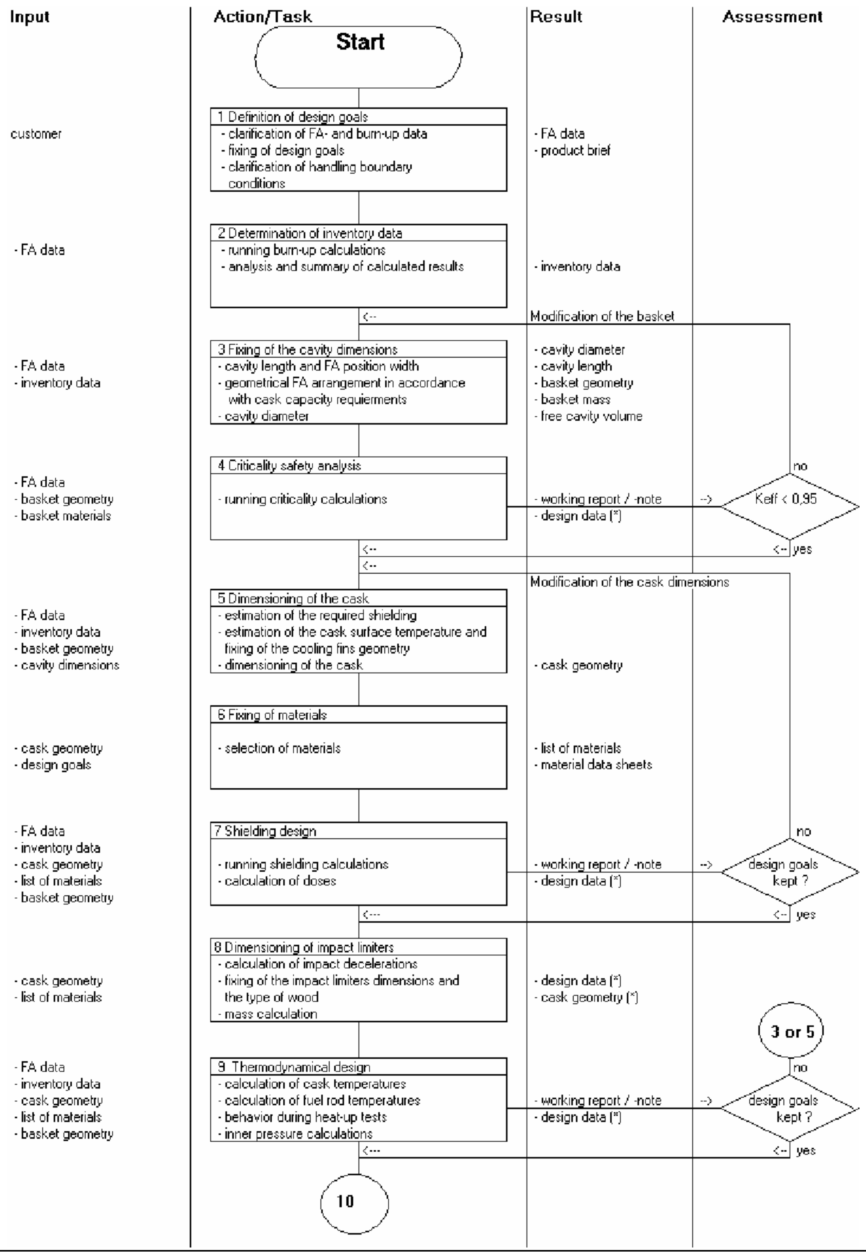
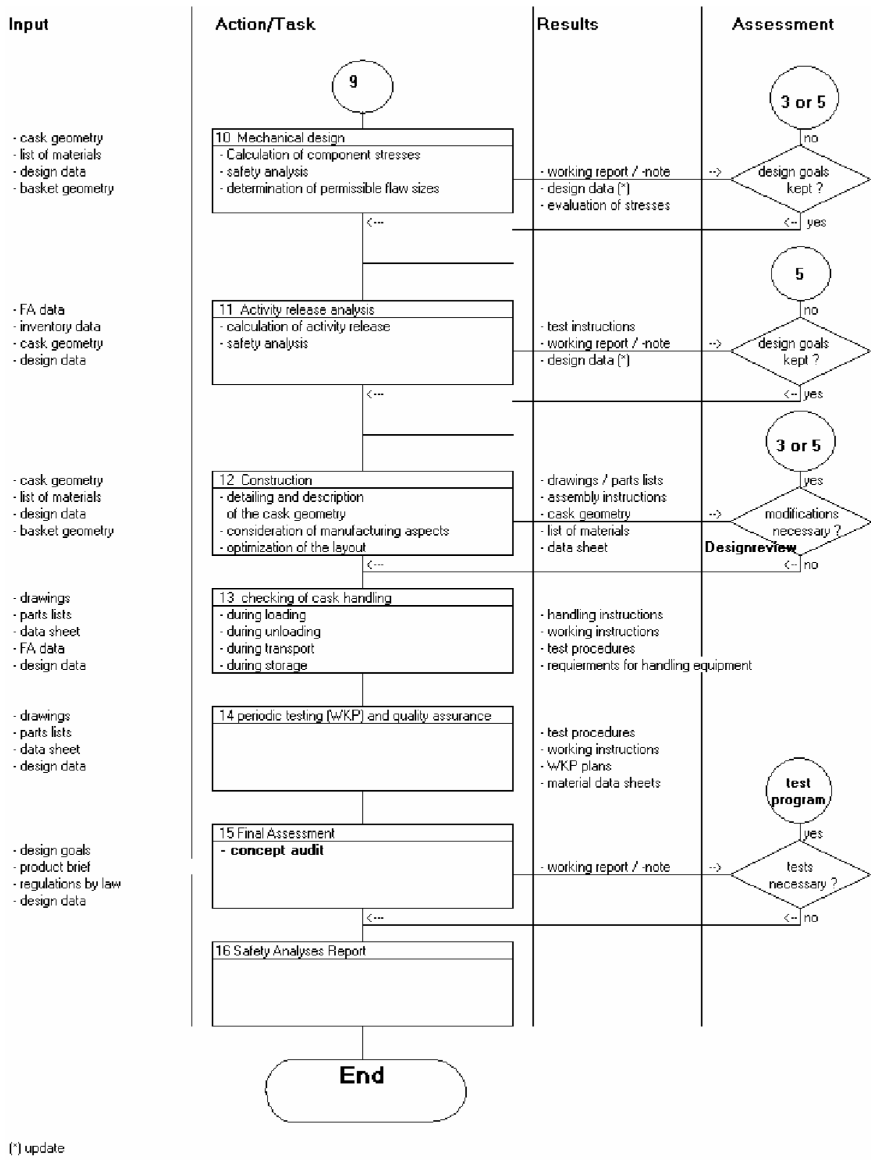


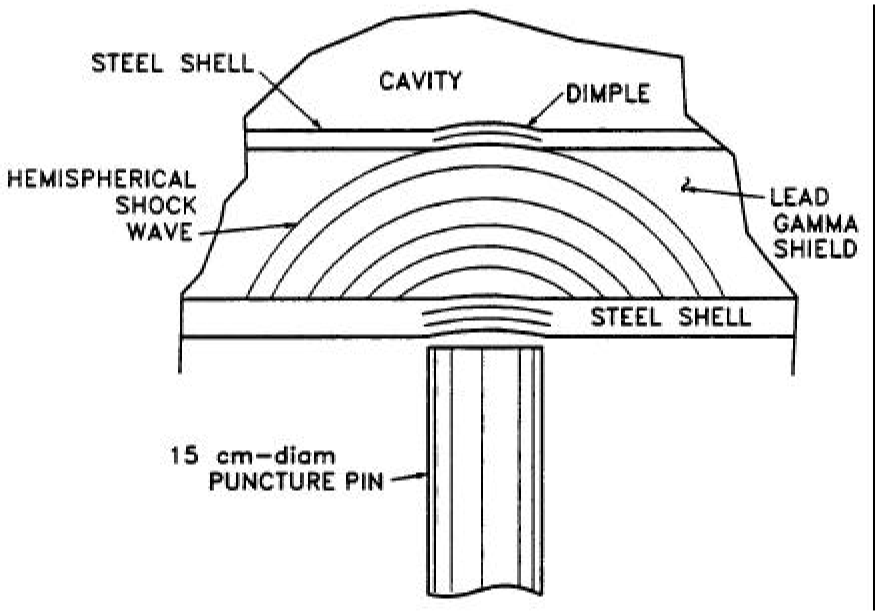
Fig. 11 (continued) - Cask project flow-chart (NIS-SOGIN).



Computer codes and models shall generally be qualified and validated by testing. Testing is treated in the following, but results (even in case of scaled tests) can be used by extrapolation also for analysis of untested drop orientations.

For example in the pin puncture test the energy in the shock wavesdecrease as the square of the distance traveled inward, as shown in Fig. 12.

Fig. 12 – Pin puncture event (ORNL).



A series of pin puncture tests were performed at the Oak Ridge National Laboratories to develop an empirical equation for the stress in the outer wall of a multi-wall casks a function of the mass of the cask and the thickness of the outer wall material. This equation (Nelm's equation) applies to steel-lead-steel construction and has been used for several licensed casks. Therefore for hot-rolled carbon steel and stainless steel outer shells, the minimum outer shell thickness required to withstand the punching action of a steel piston is given by Nelm's equation:

$$t = \left(\frac{W}{S}\right)^{0,71}$$

Where:

t	= shell thickness (in)
W	= cask weight (lb)
S	= ultimate tensile strength of the outer shell (psi)

Note that this equation gives acceptable results when the diameter of the package is greater than about 75 cm and when the impact location is not close to a stiffening structure such as a heat transfer fin; for packages having a diameter below 75 cm, a factor of 1.3 to the outer shell thickness shall apply.
Special attention shall be given to the structural design of the internal basket. The content of a shipping and/or storage cask requires a structural support to protect the spent fuel and maintain its integrity. In addition, since the basket has also a function in the assurance of

subcriticality, the structure must not exceed yield in any substantial portion. The yield stress limit ensures that the geometry analysed for criticality safety is maintained. Use of boron-stainless steel alloys can be acceptable if the material is allowed by an international standard such as the ASTM code.
Since the basket has also an important function in the process of decay heat transfer to the cask body and in reducing the spent fuel temperature, the basket design is a complex and iterative process and its optimisation one of the most challenging task.
There are mainly three design approaches for the basket (Fig. 13):

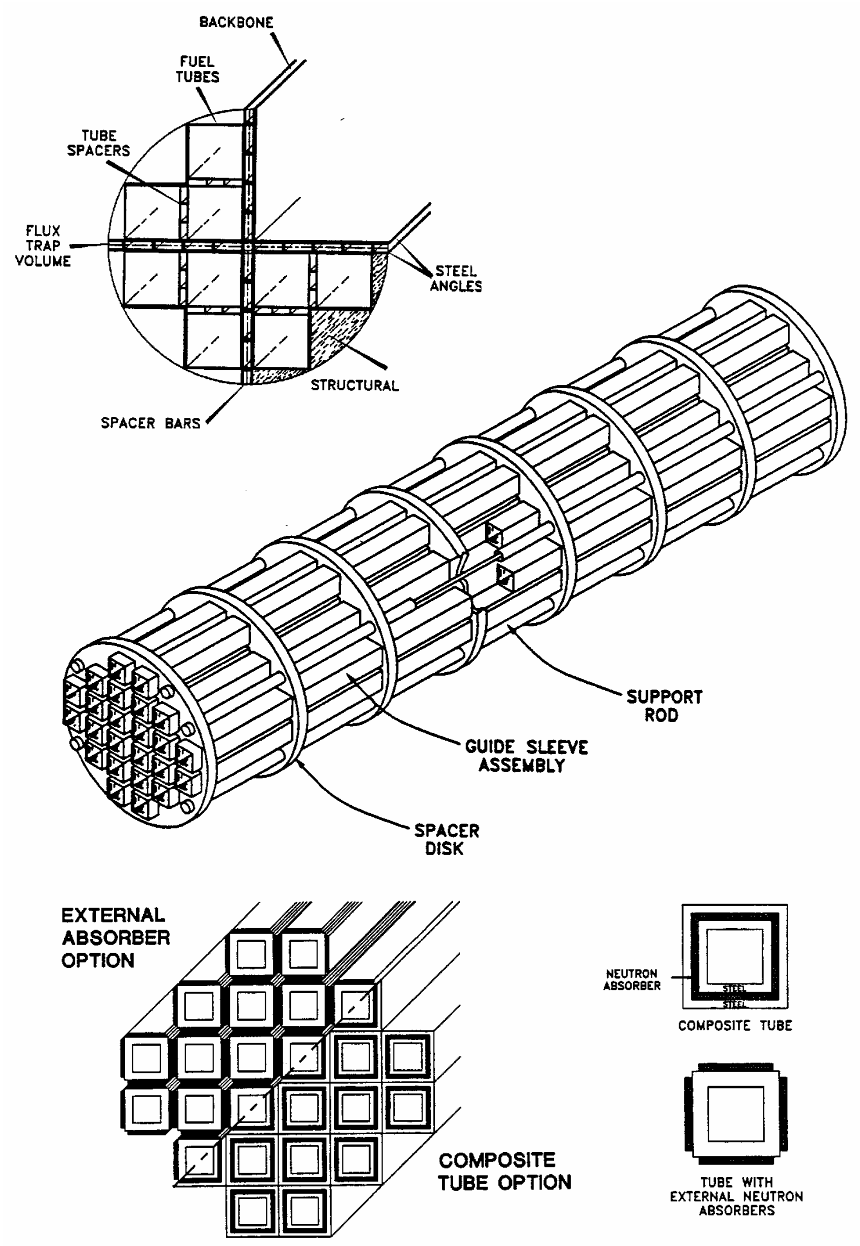
- tube style: full length independent tubes;
- support disk style: full length tubes supported by intermittent spacers;
- "backbone" style: full length backbone with full length fuel tubes.

The design of impact limiters is also a challenging goal. It must be based on a compromise between the need to a have a "soft" impact limiter, to reduce the accelerations to the cask and to the fuel, and the limitation existing for the volumes, and the weight, of the impact absorber. Impact limiters are generally present only in the transport configuration and not in the storage configuration. However, if accelerations are not acceptable in the latter condition, either exclusion of the drop shall be demonstrated through redundancies in the handling equipment, or special impact limiters shall be applied during cask handling.
Impact limiters can be fabricated by crushable materials (such as wood and honeycomb structures) or from materials with high elasticity such as foam. These materials are generally non code documented, and a documentation of tests is needed to establish their properties under all relevant conditions. If the material is not isotropic as wood, several angles shall be tested.
The impact energy must be absorbed in the most demanding case before any structurally rigid portion of the cask (such as trunnions) contacts the unyielding target, because such contact would generate very high localised stresses.

4.3 Materials

The materials selected for the fabrication of a cask are generally chosen not only for their structural properties, but also for their shielding properties and for their behavior in the long period against all challenges that can degrade them. Only qualified materials, accepted by international codes, should be used.
The mechanical properties of materials that are of primary interest to designers include stiffness, ductility, strength, toughness, resistance to fatigue, corrosion, wear and creep, expansion under thermal loads.
Other types of materials, which are normally used in the shipping packaging, are the energy-absorbing materials used as impact limiters to reduce damages to the packaging and to reduce the acceleration transmitted to their content.
The properties of materials are external measures of how the materials behave in the presence of applied loads and environmental conditions as internal or external pressure, dynamic and vibration loads such as free drops onto various structures, interactions with fluids, hot and cold temperatures and so forth. The objective is to use these properties to describe in the most complete way how packaging materials will react to any combination of applied loads and environmental conditions.
These properties are "continuous" and are based on the idealisation that the material behaviour may be considered isotropic, homogeneous and continuous over any length scale of interest. This is a useful and generally accurate assumption for the polycrystalline materials used in the size and quantities typical of a nuclear material shipping container.

Fig. 13 – Different options for basket design.



The first property to be considered is the modulus of elasticity and Poisson's ratio. When the applied loads to a structure are such that the resulting stresses and strains are small, the material behaviour is elastic and fully reversible. In polycrystalline metals, this elastic behaviour is typically linear, and the constant of proportionality between the tensile strength σ and the strain ε is the elastic, or Young's, modulus. The elastic modulus is important in describing the stiffness of a structure, and varies widely with different materials. The modulus of iron is 200 GPa, while that of aluminium is 69 GPa. The stiffness of plastic and organics as would be used in energy absorbers is less, i.e. in the range of 0,14 to 10 GPa. The elongation of a specimen in uniaxial tension may be accompanied by contractions in the perpendicular, transverse directions. Poisson's ratio is the ratio of induced transverse strains to axial strains. For elastic deformation, the Poisson's ratio of iron is 0,28, while for aluminium it is 0,34. For plastics, the value is typically 0,4. For plastic deformation in polycrystalline metals, where the material volume is nearly constant, Poisson's ratio is 0,5.

Ductility is the ability of a material to deform plastically prior to fracture. Ductility is important because of the unavoidable occurrence of stress concentrations associated with holes, notches, fillets and other discontinuities. In a ductile material, small plastic zones form at these stress concentration locations. Therefore a cask material should possess sufficient ductility to preclude, by a large margin of safety, fracture initiation at these stress concentrations. The question of what constitutes sufficient ductility is not easily answered. There is some consensus that above 20% elongation a material be considered ductile, while, below 5%, it would be characterised as brittle.

The next property to be considered is strength. The properties of metals relating to ductility and strength are functions of strain, strain rate and temperature. For many design situations, loads are of a magnitude and duration such that materials properties obtained by standard slow- and moderate-speed pull tests at room temperature are adequate. Several tests methods for measuring material strength and ductility at higher strain rates exist, but they are more difficult to standardise because of the complexity of the experimental apparatus and data taking requirements. However, data exist in the engineering literature. Standard practices do exist for measuring stress and ductility at elevated temperatures. It is worth mentioning that an impact limiter functions differently from the way other package components do, in that the impact limiter is expected to undergo large permanent deformations under impact conditions.

An important property is fracture toughness. All structures contain discontinuities or defects. These discontinuities may be inherent in the material as vacancies, dislocations or grain boundaries. They may be present in the material because of the fabrication processes, such as welding, rolling, or casting, which may produce slag inclusions, voids, or grain boundaries with excessive carbides.

When these defects are small and uniformly distributed, they determine the onset of plastic deformation in the material. However, when the defect is on the order of magnitude of the grain size or larger (on the order of 10 to 50 microns), and has a small radius, it may be termed as a crack. Because of stress concentration, plastic deformations tend to concentrate around this defect, and the net section-stress-level-to-failure may be less than the yield stress. This failure localisation phenomenon is described as a fracture, and it may be described as brittle or ductile based on the amount of deformation in the material directly adjacent to the defect or crack as it has grown. The fracture toughness is the ability of the material under stress to resist the self-similar growth of a crack to a larger defect. The fracture toughness of a material typically depends on the temperature, rate of loading and state of stress in the material near the crack tip. Its value is generally larger at higher temperatures because of the greater mobility of dislocations. The fracture toughness of structural steel with a ferritic microstructure is particularly sensitive to temperature. For dynamic loading rated, such a generalisation is not possible because the dynamic fracture toughness may be greater or less than the static value. Special casting processes can improve fracture toughness of materials such as cast iron, which is used also for casks.

The fatigue strength is the ability of the material, when loaded under cyclic stresses, to resist the growth of small defects to large defects. ASTM definition is: the process of progressive localised permanent structural change occurring in a material subjected to conditions that produce fluctuating stresses and strains at some point or points and that may culminate in

cracks or complete fracture after a sufficient number of fluctuations. The progression of small defects to large defects occurs as dislocations pile up to form slip bands, which move across grain boundaries to free surfaces to become extrusions, and the surface roughened by extrusions becomes the site for the formation of microscopic cracks, which become macroscopic cracks.
Corrosion may be defined as the surface degradation of metals resulting from chemical reactions with other materials in their environment. Corrosion degradation may take several forms, depending on the metal surface and the reactants. Forms of corrosion include:

- Uniform attack. The main remedy is to increase the thickness of the material based on the time of service and the degradation rate. For example, carbon steel immersed in sea water presents a typical corrosion rate of 130 micron per year.
- Pitting. The corrosive agent attacks some regions of the surface, preferentially. A hole or pit forms at the surface such that its width is comparable with its depth.
- Crevice attack. Attack concentrates at geometric discontinuities. The geometry of a crack tip region may accelerate or decelerate the diffusion of corrosive species to the crack tip region, but the large active and residual stresses, present there, accelerate the rate growth. In presence of corrosive media, normally ductile materials may fail in a brittle fashion at low loads.
- Galvanic coupling. When two metals, or dissimilar electrodes, are connected in an electrical conductive environment, an electrolyte, preferential attack occurs on the anodic material. The anodic material is the active or electronegative of the pair. The cathodic member of the couple will be protected from galvanic corrosion by its connection to the anode.
- Hydrogen embrittlement. Hydrogen embrittlement itself does not describe a single process, but rather a group of related processes during which hydrogen from various sources acts to degrade the ductility and fracture toughness of a metal structure.
- Corrosion fatigue. Cyclic stresses in a corrosive environment may cause premature failures.

Galling or adhesive wear is a common, unavoidable form of material surface degradation. It occurs when two bodies come into sliding contact and surface asperities contact each other. Large attractive forces may occur between the two surfaces, depending on the total area in actual contact. Adhesive wear may be greatly reduced by use of lubricants. Material selection or use of a sacrificial slip layer may also reduce the effect or the consequences.
Thermal expansion is a coefficient which gives the measure of volume modification of a material when its temperature is modified. For polycrystalline metal materials the assumption of isotropy is usually made. Thermal stresses arise when the expansion or contraction of a structural member is constrained.
Plastic deformation is a process that is affected by time, temperature and stress level at which the change of shape occurs. Plasticity in polycrystalline materials may be classified by several atomistic processes, which may occur. The material behaviour of creep is usually associated with plastic deformation phenomena which occur at low strain rates over long periods of time.
In general, degradation phenomena of materials associated with high neutron fluxes do not occur in metals. However these, together with gamma doses and dose rates, have to be carefully evaluated for organic materials.
Attention shall be given also to characteristics of the materials included in the packages. These may be spent fuel assemblies, or vitrified wastes, or cemented waste, or compacted wastes, etc. In many cases to assure the safety it is important to avoid degradation of these materials and in important cases (such as the case of spent fuel) there are also structural requirements, since they have to maintain their structural integrity after an accident involving severe stresses.

4.4 Thermal analysis

All packages that contain radioactive materials must be evaluated to determine their expected normal operating temperatures and their responses to the accident conditions specified in the regulations.
The major thermal design issue involves the conflict between passively removing the radioactive decay heat from a package while passively protecting the package from accidental external heat sources. Thus, a passive system (e.g. insulation) that might be used to protect the containment system from overheating in an accident involving a fire may exacerbate the difficulty of transferring the radionuclide decay heat from the containment vessel to the package surface during the normal conditions of transport.
An issue which may be relevant in some cases is the heat transferred into the package by effect of the external insulation. Typical values are reported in Table 6.
In the case in which the packaging has a thin external shell covering a thermal insulation, the package surface temperature should be based on the peak insulation flux. The interior temperature field should be determined by the application of the 12-h averaged heat flux applied to the external surface as a steady state temperature. A more accurate schematisation may be achieved by considering a 12-h constant heat flux given above and a 12-h without heat flux.
Conservatively the absorbed flux should be assumed equal to the insulation heat flux, i.e. abosorptivity equal to 1. Absorptivity of the surfaces may change with surface conditions (cleanliness, scratches, etc.), but a reduced absorptivity shall be proved before being accepted in licensing safety analysis.
A thermal load may cause pressure build-up in the containment, producing stresses. Thermal gradients in the package may also produce thermal stresses and other effects that may jeopardise the ability of a package to perform its functions.
Pressure build-up may be caused also by other possible phenomena. One is the loss of encapsulated gasses inside the package. This is the case of wastes, included in inner containments, which may rupture or the case of fuel rod gasses, inserted at pressure during fuel fabrication. Other phenomena include radiolysis and thermal decomposition of various materials, which may produce gases increasing the internal package pressure.
Issues related to thermal conditions are several and are briefly discussed below.
Thermal stresses are induced by thermal gradients through the material and the constraints on the possibility of the material to expand or contract. Another cause is related to different thermal expansion coefficients between two components that have an insufficient gap between them to accommodate the expansion difference without interference.
Temperature effects are in some cases combined to decrease the safety margins of the package. For example, an increase in the temperature may cause an increase in the internal pressure and therefore in stresses, while at the same time the mechanical resistance of the material may be reduced by the same increase in temperature.
Thermal effects may include also degradation of several safety functions of the package and of the contained radioactive materials.

Table 6 - Typical insulation values (NIS-SOGIN.

Form and location of the surface	*Average over 12-h (W/m^2)*	*Peak value (W/m^2)*
Flat surfaces transported horizontally		
Base	*None*	*None*
Other surfaces	*800*	*1260*
Flat surfaces not transported horizontally	*200*	*315*
Curved surfaces	*400*	*630*

The containment function may be degraded by increased temperature. Generally, the most delicate element is the sealing system, which may be in elastomeric or metallic materials. In both cases their sensitivity to temperature is higher than that of the package body, as they perform a critical function in assuring the containment function.
The shielding function may also be degraded, since shielding materials may be subject to volumetric dimensional changes or material phases change. In addition, in case of a thermal accident, such as fire, it may even melt or burn.
Criticality control is associated with the presence of neutron poisons, which shall remain in place and shall not deform, melt or disintegrate. Of course the same need of maintaining the geometrical integrity is associated with the fuel itself.
Evaluation of the heat sources shall include at least the following three:

- heat released from the radioactive content;
- insulation;
- heat received from fire during an hypothetical accident.

Typical temperature fields are presented in Fig. 14 and 15.

4.5 Containment

The regulatory provisions for containment vary somewhat by package type, but no loss or dispersal of content is permitted in normal conditions of transport. Under accident conditions the requirements vary much more widely. For some package types, loss of content is permitted while for others, such as Type B package, only a very small quantity of material is permitted to escape.
The main elements for the determination of the package containment performance requirements are:

- the quantity and isotopic makeup of the material in the package that could be available for release, should a leak occur between the cavity of the package and the environment;
- the nature, pressure and temperature of the medium that surrounds the contents within the cavity (typically air or inert gas that does not contribute to an oxidation or corrosion process);
- the components and configuration of the containment boundary and, in particular, the seals used in the boundary.

The term medium, as used herein, includes any gas or fluid which may not be itself radioactive, but which could carry radioactive material through a leak in the containment boundary.
The gases and particulate materials that could potentially escape from a package are typically determined from analysis of the contents, using computer codes such as ORIGEN or from direct sampling. In order to asses their potential to be released, the evaluation of their chemical and physical forms is needed. In some cases this is not easy and therefore conservative estimates shall be introduced.
In case of suspected failed fuel (fuel assemblies where one or more rods may present cracks or holes releasing fission products) characterisation of the fuel is very important in order to properly consider the source term also in normal conditions and in order to take appropriate measures for fuel dryout and for fuel containment (in such a case a metal inner container may be used in order to confine fuel pellets, should the failed assembly loose its geometrical integrity).
In order to have a release the primary driving force is a pressure difference between the cavity and the environment. For most of the casks the cavity is maintained in subatmospheric pressure and in any normal condition there is no escape at all to the environment.

Fig. 14 – Surface temperature field distribution for two CASTOR casks (SOGIN).

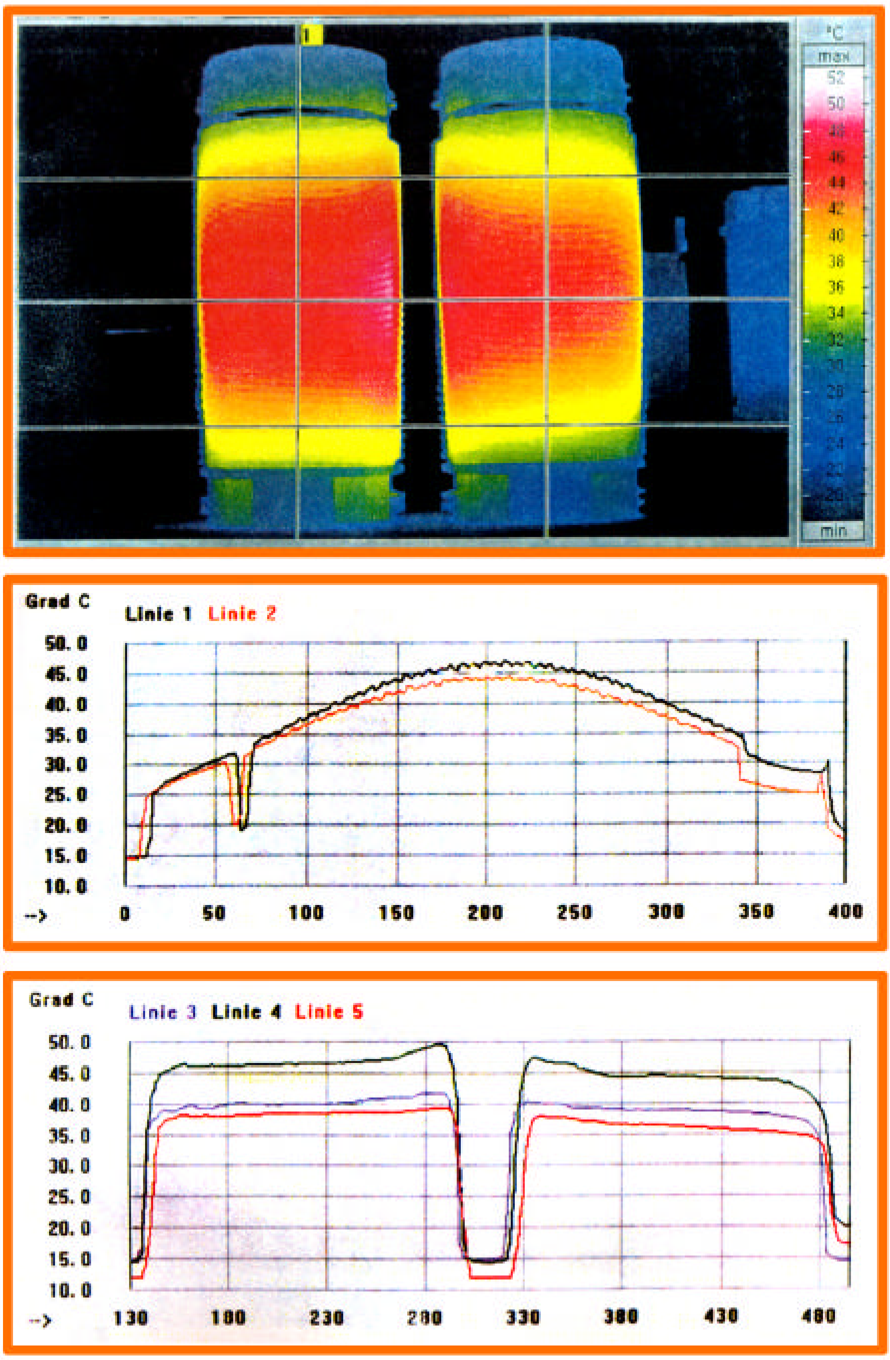
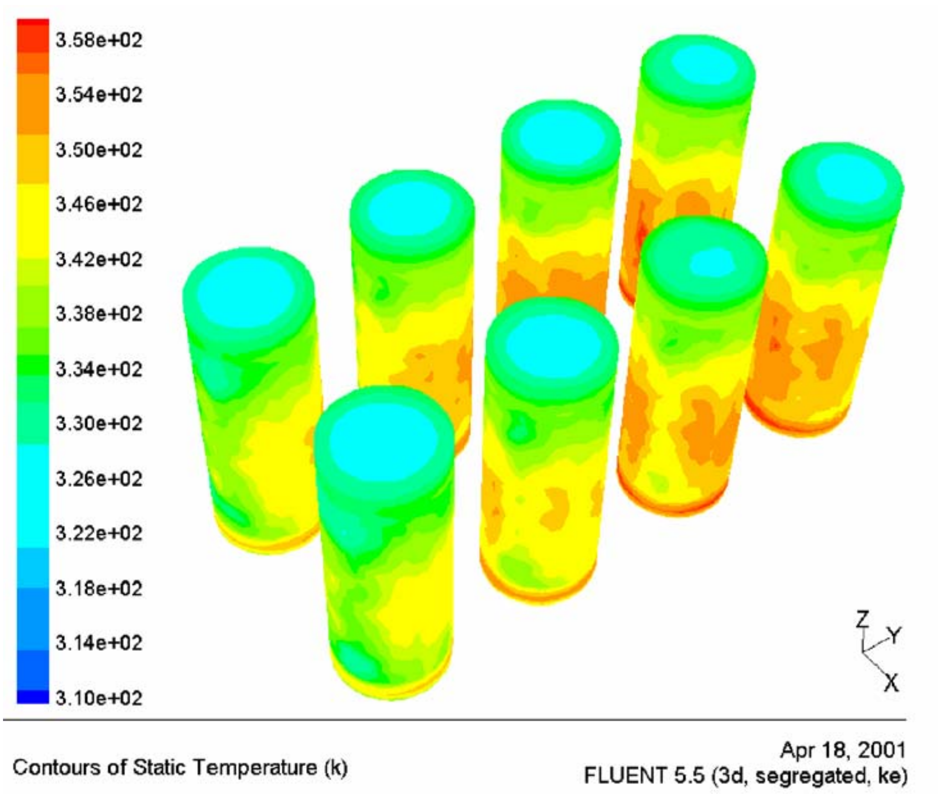


Fig. 15 – Surface temperature field for a two-rows casks array (SOGIN).



In addition, these casks at least for the storage conditions, shall have two seals and a pressurised gas is introduced between the seals in order to cause an inleakage into the cavity in case of a leak. Reduction in the pressure of the volume between the two seals will cause an alarm and a request for maintenance of the containment system.

In normal condition, however, at least one other mechanism should be investigated to calculate the potential for radioactive material release to the environment. It is known that a net flow of a specific material can be established where a difference in concentrations exists. In this case the environment and the volume between the seals have no or very low concentration of radioactive isotopes and chemical compounds which are present in the cavity of the package. A calculation can be performed for the net leak rate assuming a leak area in the seal that can be calculated on the basis of the tested leak rates for known differential pressures.

In cases of accidents during storage or transport, calculations of releases take into account several negative factors. The first is that in many cases it is assumed that only one seal remains reliable and at a reduced efficiency (while credit is given to the two seals in storage conditions). The second fact is that many accidents are related to fire and, therefore, to an increase in temperature and finally in pressure inside the cavity. This creates the driving force for material escape. A third factor is that the usual assumption is that 100% of the fuel rods fail and that the gases, accumulated in the fuel rod gap, are released inside the cavity.

It is difficult to evaluate the leakage of particulate. In general the size distribution of the particles is not known. It is clear that particles larger than the leaking orifice could not escape. The second assumption is related to the size and the shape of leak orifice, because usually may not be only one leaking path, but several ones of small dimensions and because they are not generally rounded holes with a precise diameter. Regarding the first problem there exist in the literature information about some theoretical and some experimentally measured distributions of particle sizes in different conditions. The second problem is solved assuming that all leaking paths are made equivalent to a single rounded leaking path.

Seals selection and use present some issues related to design configuration, operations and maintenance. A full range of elastomeric or metallic O-rings, seals, and gaskets are available. Elastomeric seals include any material that is not metal, like rubber, neoprene, viton, plastic, silicone, polymeric, teflon and similar materials. Metallic seals include both those that are solid and hollow and can be made of several materials such as aluminium and silver. Only metallic seals have credit as containment boundary for storage casks.

The containment boundary penetrations are generally dosed by bolted flanges, some of which, as the package closure, may be very large. The typical design for an O-ring seal incorporates a groove that permits the seal to deform until there is a metal-to-metal contact of the flanges.

Structure of a metallic seal may be complex and multi-material. Among other factors which shall be considered for seal selection, of specific importance is the sensitivity to radiations. Radiation resistance refers to the ability of the seal to resist significant degradation when exposed to gamma and neutron environment for long times. Such degradation typically results in changes in the mechanical properties of the seal. Elastomeric seal performance may be expected to degrade after about 10^4 Sv of radiation at about room temperature. Metal seals are largely unaffected by radiation.

In Fig. 16 a scheme of a CASTOR (GNB) containment monitoring system is presented.

4.6 Shielding

The shielding for a transportation package shall be designed to maintain radiation dose rates external to package surfaces below established regulatory limits under specified normal and accident conditions.

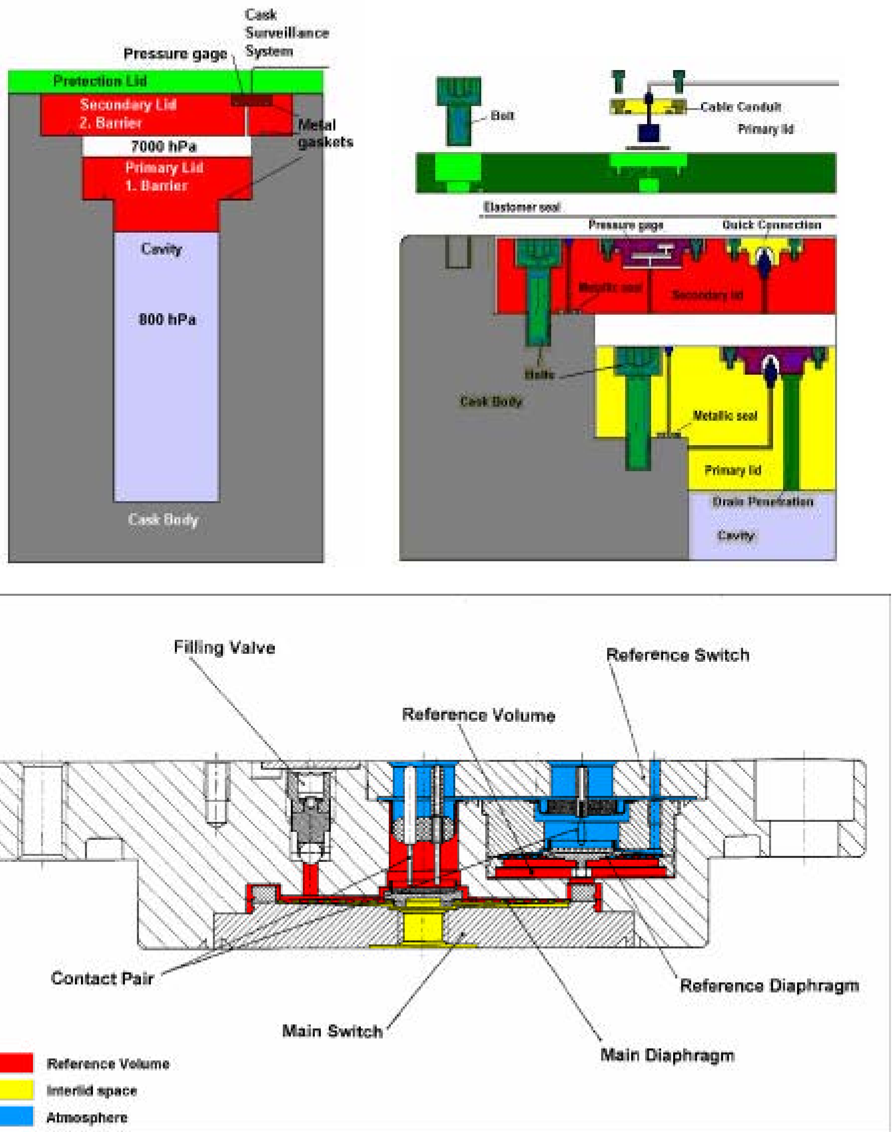
Two forms of radiation are of most concern in package design: gamma and neutrons. Gamma radiation requires dense materials for efficient shielding (e.g. lead, steel and depleted uranium). Neutrons require a light material, possibly containing significant quantities of hydrogen, in order to perform shielding. One of the most challenging shielding problems is the shield design for a spent fuel cask.

The cask shielding design process must interact with materials selection and structural analysis, since the design goal usually focus on the minimisation of cask weight, while maximising the payload. All these conditions, combined with other requirements, lead to contrasting design solutions and the final optimisation may require several iterations.

A difference shall be made between cask design for generic spent fuel, where only the boundary conditions for the spent fuel are determined (e.g. burnup and number of years of cooling in the plant pool) and custom made casks, where a contribution to design optimisation may be brought only by a detailed fuel characterisation and a loading strategy.

The first issue is the definition of the source term. Identification of the fuel fission products and actinides contents can be done with proper computer codes such as ORIGEN. More difficult, in some cases, is the identification of the gamma contributions from crud accumulated on the surfaces of the fuel assemblies and of the impurity activation of the fuel assembly structural material as well as of the fuel itself. In these cases operating experience, direct measures and fuel technical specifications may help to provide relevant data. In some cases these contributions may be significant also in comparison with that of the fuel itself.

Fig. 16 – Scheme of a CASTOR monitoring system and particular of the pressure gage (GNB).



The design goal for transportation as required by international regulations is that the dose rate does not exceed 2 mSv/h at any point on the external surface of the package. In addition, there are a number of exceptions and additional conditions to be considered to arrive to a final design goal.

4.7 Criticality safety

Criticality safety is the practice of ensuring that adequate protection is provided against an accidental self-sustaining or divergent fission chain reaction. For packages that transport fissile materials, this adequate protection is provided by using a design and safety assessment philosophy that effectively eliminates the possibility of a criticality event occurring under any credible scenario. It is usually required that a k_{eff} < 0,95 be demonstrated in all design conditions.
Use of favourable geometries, incorporation of neutron poison materials and moderator control are potential means of controlling the neutron balance. Credit to neutron poisons require special attention because their presence shall be assured under all conditions and because, when added to structural materials, they may change the mechanical properties of the host material.
Whatever the control mechanism, an adequate margin for subcriticality must be demonstrated for both the single package in isolation and for arrays of packages assembled together. In some cases, conservatively, it may be assumed that a perfect reflector be present around the cask, thus maximising the neutron economy and avoiding the need of considering situations such as cask external flooding and an array of several casks.
Undamaged and damaged packages shall be considered using credible fissile material configurations and moderator and reflector conditions that provide the maximum reactivity. Evaluations related to fuel integrity shall also be included in the considerations about the final geometry in case of accidents.
The traditional design basis has always been to use the isotopic composition of the fresh, unirradiated fuel in the criticality safety evaluations. This approach is straightforward, easy to defend and provides a conservative margin that precludes any concern about misleading events.
Since there is a clear trend in increasing spent fuel transportation and interim storage facilities, there is an incentive to reduce the number of packages also for safety reasons. At the same time there is a trend to increase the initial enrichment to increase the burnup and extend the time between refuelling. Therefore the concept of taking credit for the reduced reactivity caused by the irradiation or burnup of the spent fuel becomes an attractive design alternative. This concept is generally named "burnup credit". Depending on the burnup and initial enrichment, the use of the burnup credit assumption can provide a 20 to 40% decrease in the reactivity of spent PWR fuel compared with that of the same fresh fuel. The advantage for BWR would be lower, since criticality control can be achieved more easily without impacting the package capacity.
Although the fact that spent fuel has a decreased reactivity is not questioned, several issues must be addressed and resolved prior to using spent fuel isotopes in the design basis analysis. These issues include:

- validation of analysis tools and associated nuclear data to demonstrate their applicability in the area of burnup credit;
- specification of design basis analysis that ensure prediction of a bounding value of k_{eff};
- operational and administrative controls that ensure that the spent fuel loaded into the package has been verified to meet the assumptions used in the analysis.

Direct experimental methods of reactivity determination represent a potential source of excellent verification of criticality safety. The advantage of these methods is that they provide measured evidence of the safety of the actual system during operation. The measurement technique can be used during loading operation to measure the k_{eff} of the package as the loading proceeds. Potential measurement techniques are the multiplication technique, the Rossi-alpha technique, the ^{252}Cf noise analysis technique and the source-jerk technique. Each of these techniques has its own set of advantages and disadvantages. Problems of measurements (e.g. in a high gamma field) and interpretation of data (e.g. necessity to transfer from frequency domain and sensitivity of measurement to detector location) accompain each of the subcritical measurement techniques. The research continues to identify a proper, reliable technique.