

METHODS FOR EXPANDING THE CAPACITY OF SPENT FUEL STORAGE FACILITIES

PROCEEDINGS OF A TECHNICAL COMMITTEE MEETING
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD IN VIENNA, 12-15 JUNE 1989



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FOREWORD

At the beginning of 1989 more than 55 000 metric tonnes of heavy metal (MTHM) of spent Light Water Reactor (LWR) and Heavy Water Reactor (HWR) fuel had been discharged worldwide from nuclear power plants. Only a small fraction of this fuel has been reprocessed. The majority of the spent fuel assemblies are currently held at-reactor (AR) or away-from-reactor (AFR) in storage awaiting either chemical processing or final disposal depending on the fuel concept chosen by individual countries. Studies made by NEA and IAEA have projected that annual spent fuel arisings will reach about 10 000 t HM in the year 2000 and cumulative arisings will be more than 200 000 t HM.

Taking into account the large quantity of spent fuel discharged from NPP and that the first demonstrations of the direct disposal of spent fuel or HLW are expected only after the year 2020, long-term storage will be the primary option for management of spent fuel until well into the next century. There are several options to expand storage capacity: (1) to construct new away-from-reactor storage facilities, (2) to transport spent fuel from a full at-reactor pool to another site for storage in a pool that has sufficient space to accommodate it, (3) to expand the capacity of existing AR pools by using compact racks, double-tierce, rod consolidation and by increasing the dimensions of existing pools.

The purpose of the meeting was:

- to exchange new information on the international level on the subject connected with the expansion of storage capacities for spent fuel;
- to elaborate the state-of-the-art of this problem;
- to define the most important areas for future activity;
- on the basis of the above information to give recommendations to potential users for selection and application of the most suitable methods for expanding spent fuel facilities taking into account the relevant country's conditions.

The Agency wishes to thank all those who participated in the meeting. Special thanks are due to the Chairman of the meeting, Mr. P. Pattantyus, and the Chairman of the Panel discussion, Mr. K. Einfeld. The officer of IAEA responsible for the preparation of this document is Mr. F. Sokolov, Division of Nuclear Fuel Cycle and Waste Management.

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SPENT FUEL PIT AND HIGH DENSITY STORAGE RACKS DESIGN IN BELGIAN NUCLEAR POWER PLANTS

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Abstract

Belgium operates seven Pressurized Water Reactors. The fuel annually unloaded represents about 150 tons uranium oxide.

The need to increase the spent fuel storage capacity led to use compact storage racks.

This paper reviews the pools and the high density racks designs in the seven Belgian units.

1. INTRODUCTION

The Belgian nuclear power program comprises seven light water reactors. All units are pressurized water reactors and are in operation with a total electrical output of about 5.5 Gigawatt.

They are located at 2 different sites : 4 units at Doel and 3 units at Tihange.

One third of each reactor core is annually unloaded. This represents about 150 tons uranium oxide every year for the seven units.

The spent fuel is stored at the reactor site in storage pools before being sent to a reprocessing plant.

Approximately 600 tons of fuel unloaded from Belgian power plants will have been reprocessed up to this year by COGEMA.

TABLE I.

Unit	Electrical power (MWe)	1st criticality	FUEL ASSEMBLY		UO mass (t) ²
			Type	Quantity	
Doel 1	400	1974	14 x 14	121	31,74
Tihange 1	870	1975	15 x 15	157	67,84
Doel 2	400	1975	14 x 14	121	31,74
Doel 3	900	1982	17 x 17	157	72,44
Tihange 2	900	1982	17 x 17	157	72,44
Doel 4	1000	1985	17 x 17 XL	157	84,52
Tihange 3	1000	1985	17 x 17 XL	157	84,52

2. STORAGE POOLS DESIGN

The spent fuel elements are sent to the spent fuel pit and stored under water for a period of time before being eventually shipped to a reprocessing facility.

The storage pools are located in an auxiliary building connected to the reactor containment by the transfer canal.

Pools are stainless steel lined, concrete structures. The space between the stainless steel liner and the concrete walls is monitored for leaks detection and collection.

The concrete structures of the building and the pools are designated seismic category 1 and, on the last four units are airplane crash resistant.

The spent fuel pools cooling system consists in two seismic category 1 cooling systems each of which includes a pump and a heat exchanger.

The system capacity is designed to keep the spent fuel pool temperature below 50°C for all normal refuelings and below 70°C when a full core is offloaded in the spent fuel pool.

Filters, ion exchanger and skimming system are provided to clean pool water and to remove impurities and suspended particles from the surface of the pool water.

3. STORAGE RACKS DESIGN

- The function of the fuel racks is to store spent or new fuel in a subcritical array during all credible handling and storage conditions, and to insure that clad integrity is not damaged by mechanical or thermal conditions during the storage period.
- The design of the fuel racks is such that the effective multiplication factor (k_{eff}) does not exceed 0.95 in demineralized water and 0.98 with a highly moderating material such as water vapour.

The following are the conditions that are assumed in meeting this design basis :

- . The fuel assembly contains the highest enrichment authorized without any control rods.
- . No credit is taken for boron dissolved in the fuel pool water.
- . No credit has been taken for fuel assemblies burn up so far.

- The fuel racks are designed according to the American Safety rules 10 CFR 50 Appendix A, Standard Review Plans and Regulatory Guides of the USNRC, the construction requirements of the ASME Boiler and Pressure Vessel Code, Section III, division 1, subsection NF and ANSI Standards.

4. STORAGE CAPACITY DEVELOPMENT IN THE FIRST THREE UNITS

The capacity of the spent fuel pool was in the first units, four third times the core volume.

This capacity was required to have enough space available at all times to remove a full core of fuel elements from the reactor.

Very early, it appeared that the storage capacity had to be expanded for better flexibility of operating conditions and for reprocessing plant capacities.

a) Tihange - Unit 1

At the start of the study, the design capacity of the spent fuel pool was planned for the storage of 1.3 times the core volume.

The storage racks were made of 216 stainless steel cells with a 21" center-to-center spacing.

An increase in the allowable multiplication factor (Keff) in the spent fuel pool from 0.9 to 0.95 in ANSI N18.2 in 1973 permitted a reduction in spacing from previous 21" center-to-center spacing to 16" center-to-center spacing.

In 1975, before the first refueling, the pool capacity was extended to 324 storage cells which represents twice the core volume. The new spent fuel cells are made from stainless steel without neutron absorber.

The storage racks structures were subsequently modified in order to make them aseismic.

A future expansion of the spent fuel storage capacity is now limited by the strength capacity of the load-bearing structures of the pool. The stresses in concrete structures induced by a new increased load combined with the seismic reference spectrum of the site do not permit any overloading.

b) Units 1 and 2 at Doel

The units share the same storage facility (2 storage pools in the auxiliary building common to the 2 units).

The initial design capacity of the 2 storage pools was 211 spent fuel assemblies which represents five third of the reactor core volume.

The storage racks were made from a lattice network of stainless steel cells positioned on a 510 mm square pitch without neutron poison.

The need to extend the storage capacity led to the reracking of the two storage pools. The new spent fuel racks were made of stainless steel cells with a center-to-center spacing of 330 mm. This network brought the storage capacity to 479 fuel assemblies or 12/3 of the core volume.

In 1978, it was decided to increase again the capacity of one of the two storage pools by making use of neutron absorber.

One storage cell out of two was equipped with neutron poison : these cells have double-walled stainless steel casing containing carborundum. This neutron absorber allows the pitch to be reduced to 260 mm and pool storage capacity to be increased to 437 spent fuel cells or for the two pools 16 third of core volume.

To verify the behaviour of the carborundum under long term radiation exposure, samples of this material were periodically removed. In 1988, these in service controls have shown deterioration in the mechanical behaviour of the carborundum and the need for corrective action. At the same time, it was considered to increase the initial fuel enrichment up to 4.5 % to reach higher fuel burn-up. It was then quickly decided, with the agreement of the Belgian Safety Authorities, to remove all the carborundum and adapt the racks design to install Boraflex material. This replacement will be completed by the middle of june 1989.

5. STORAGE RACKS DESIGN IN THE LAST UNITS

In the last four units, the spent fuel strategy was to increase the fuel storage capacity :

- to set largest pools in the area reserved for fuel storage in the auxiliary building
- and to fill the pools with a maximum of storage cells.

This was achieved by compact storage with neutron absorber material.

Tihange 2 - Doel 3

The neutron absorber material chosen for storage racks in the units 2 at Tihange and 3 at Doel was boron stainless steel.

The structural part of the storage cells is made up of two types of steel corners which are fabricated from sheets of boron stainless steel. The two corners are welded together to form a box assembly. The individual cells are joined by welding to form a storage rack.

At Doel 3, the center-to-center spacing between fuel cells is 268 mm and the storage capacity is 672 fuel assemblies or 13 third times a full core.

At Tihange - Unit 2, the pitch is also 268 mm and the pools capacity is 706 fuel assemblies which represents 13.5 third of the core volume.

The racks for both units were designed for an initial fuel enrichment of 3.5 %, but allow for higher values up to 3.95 %.

Tihange 3 - Doel 4

Due to the fact that boron stainless steel was not economically attractive at the time of the plant construction, it was abandoned for a new neutron absorber material : Boraflex.

Each storage cell is constructed by welding two stainless steel corners together to form a can assembly. The external sides of the can are covered by a Boraflex sheet. The latter is supported by a 0.8 mm thick stainless-steel sheet tack-welded on the sides of the cell. The construction allows the shrinkage of the boraflex without restraint, the protective sheet does not permit the boraflex to fall down and the cells are provided with extra coverage of the fuel active length (to allow for boraflex shrinkage under irradiation).

At Doel 4, the storage cells are positionned with a lattice spacing of 268 mm considering an initial enrichment of 4,8 % UO_2 . The capacity of the storage pools is 628 fuel cells or 12² third of core volume.

At Tihange 3, the cells are positionned on a rectangular pitch 278 mm x 285 mm. The storage capacity is 820 fuel assemblies with an initial enrichment of 5 % UO_2 or 15.6 third times a full core.

EXPANDING SPENT FUEL STORAGE FACILITIES FOR CANDU REACTORS

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Abstract

ABSTRACT

This paper reviews the methods used to extend spent fuel storage facilities for CANDU reactors. AECL's experience has been focussed over a 15 year period on dry spent fuel storage in reinforced cylindrical concrete containers.

AECL commenced to develop the dry storage method as an alternative to wet storage at Whiteshell Nuclear Research Establishment (WNRE) in 1974.

During the subsequent years, concrete canisters were tested to store fuel from CANDU prototype and commercial reactors. Between 1984 and 1989 all spent fuel generated by AECL's prototype reactors (Gentilly-1, Douglas Point and NPD station) was safely and economically stored in concrete canisters, as part of the decommissioning activities.

The four AECL designed CANDU 6 stations built in Canada (2), Korea and Argentina are gradually reaching their pool storage capacity of 10 years of reactor operation. Point Lepreau station in Canada has already opted for dry spent fuel storage technology instead of constructing an additional storage pool.

The Ontario Hydro nuclear program relies on stations with multiple units which have common and large capacity spent fuel pools. The Pickering stations being the first units built will require supplementary spent fuel storage capacity in the early 1990's.

Ontario Hydro is considering (among several options) dry storage in single concrete containers, which could be used for transportation and possibly for eventual disposal of spent fuel.

AECL's dry storage technology is described as it evolved into its present state including the trend for development in the future. The factors influencing the evolution and the impact generated upon the design of AECL's future CANDU stations are highlighted.

Early Dry Storage Development Program

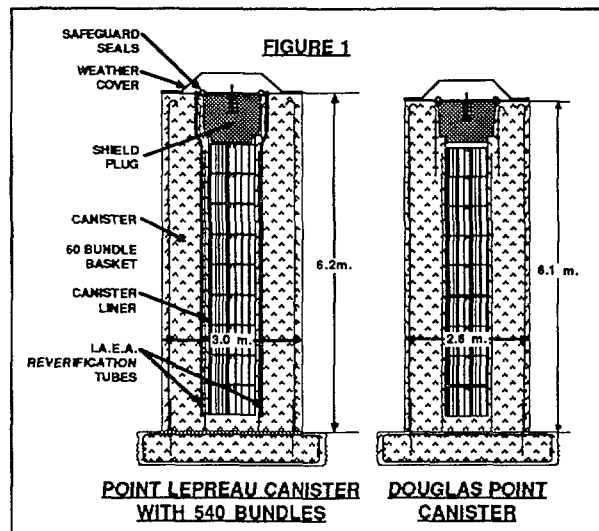
In the spring of 1974, Whiteshell Nuclear Research Establishment (WNRE) began development and demonstration of a dry storage concept, called the concrete canister, as a possible alternative to storage of irradiated CANDU fuel in water pools. The canister is a thick-walled concrete monolith, containing baskets of fuel in the dry state. The decay heat from the fuel is dissipated to the environment by natural heat transfer.

Four canisters were designed and constructed. Two canisters containing electric heaters were subjected to heat loads of 2.5 times the design, ramp heat-load cycling, and to simulated weathering tests. The other two canisters were loaded with irradiated fuel, one containing fuel bundles of uniform decay heat and the other containing bundles of non-uniform decay heat, in a non-symmetrical radial and axial array. The collected data were used to verify the analytical tools for prediction of effectiveness of heat transfer and radiation shielding and to verify the design of concrete structure.

The imposed design limits were:

- 250°C for hottest fuel element, in an inert gas atmosphere,
- 0.2 mm crack width on the external concrete surfaces,
- 100 micro Sievert/h exposure rate on the external concrete surfaces.

The current inventory in the WNRE Waste Management Area consists of 17 canisters. 10 of these canisters contain enriched fuel, one contains natural uranium oxide fuel and 6 are reserved for future storage and experiments.



Elements of AECL's Dry Storage Technology for CANDU Spent Fuel

CANDU Reactor Fuel. The typical CANDU fuel used in power reactors contains 37 fuel elements of natural uranium with Zirc-4 cladding. The elements are 13 mm diameter, by 495 mm length and are held together by end plates forming a 102 mm cross section diameter. The fuel bundle weighs about 24 kg and has an average burn up of less than 200 MWh/kgU.

Fuel Storage Basket. The stainless steel fuel basket consists of a bottom portion with a central pipe, and bundle guide plates welded together. The basket cover is placed over the basket (after inserting the fuel bundles) and seal-welded to the bottom as well as to the centre post.

The storage basket provides containment for the fuel during transportation to the canister site.

Concrete Canisters. The canister is basically a reinforced concrete cylinder with an inner liner, holding nine spent fuel baskets. Typically it provides a combined shielding of 0.86 m of concrete and 9.5 mm of steel (the Douglas Point Dry Fuel Storage program values). The opening at the top is rectangular and sized to accept the fuel transport flask during loading. The canister plug is seal-welded to the liner after the canister loading is complete.

The canisters are designed for deadweight, thermal, wind and seismic loads. They are supported on large reinforced concrete foundations (slabs), placed directly on bedrock, if possible.

The concrete mix is designed to provide high resistance to weathering with a 28-day design compressive strength of 30 MPa and with air entrainment to provide resistance to alternate freezing and thawing cycles.

The average contact dose rate on the canister is 25 micro Sievert/h.

Shielded Work Station. This is a shielded enclosure which facilitates drying the interior of a loaded fuel basket and welding the basket lid onto the bottom plate and the central pipe. Sufficient lead shielding is provided to keep the radiation field below 100 micro Sievert/h at the exterior of the station.

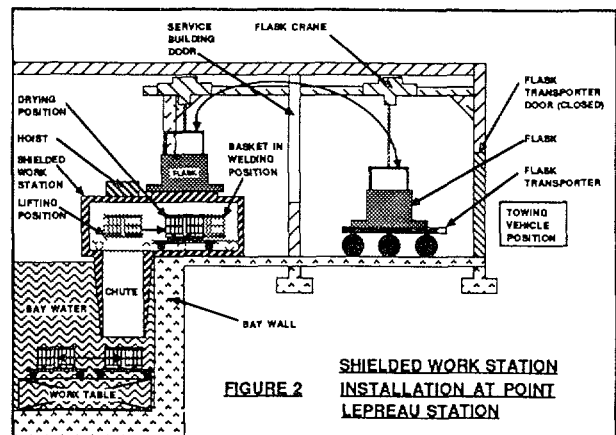
Prior to lifting a loaded full basket from the pool, the fuel transport flask must be positioned over the opening on the top of the shielding station.

Fuel Basket Transport Flask. The transport of the fuel baskets from the work station to the canister site is carried out using a road transport flask. This flask is equipped with hinged bottom shutters operated by cables, and a manual hoist for lifting or lowering the basket.

Radiation fields are kept below 200 micro Sievert/h on contact with the flask.

Fuel Handling Operations.

- (i) In the Storage Bay: This consists of the straightforward manual operation of transferring the fuel bundles from trays to fuel baskets. The work is carried out on an underwater work table, by operators using very long tools, from above.
- (ii) In the Shielded Station: The shielded station is mounted securely over the edge of the bay so that a loaded basket can be lifted into the station under the projecting shield of the station. After spray washing, the basket, the cover, and the bundles inside are thoroughly dried and then welded. Welding is semi-automatic, with the turntable rotated as directed by the control program. The lower weld and upper weld are completed in sequence, these operations are monitored by a TV camera. Acceptance of the weld is by visual inspection, and by the testing of weld quality control samples on a regular basis.
- (iii) At the Canister Site: After a welded basket has been verified acceptable, it is lifted into the transport flask which has been resting on top of the shielded station. The flask, securely closed, is then transported to the canister site on a truck bed. The canister plug is removed by a 3-ton hoist and the flask is positioned over the canister opening using a 15-ton hoist. The basket is then lowered into the canister using the manually operated flask hoist. After the canister has been loaded with 9 baskets, the canister plug is seal-welded and two IAEA safeguards seals installed (by the IAEA inspector).



The Ontario Hydro Dry Spent Fuel Storage Program

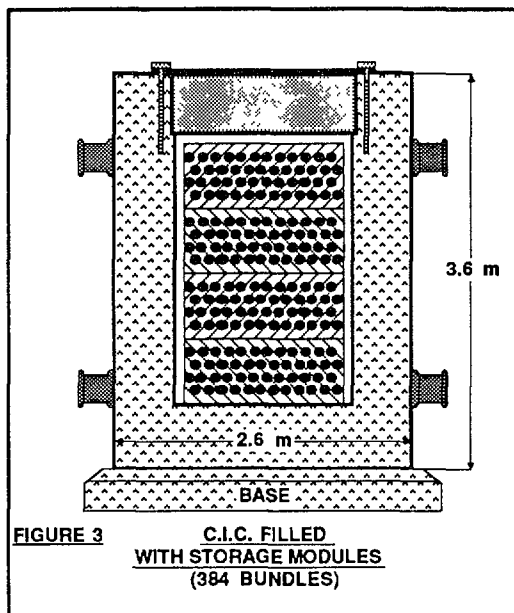
Ontario Hydro's method for alternate spent fuel storage in sealed concrete containers has entered the demonstration phase. The "Concrete Integrated

Container" (CIC) designation reflects the operational integration of storage, transportation and possibly eventual disposal of spent fuel, in a single container.

The CIC is cylindrical in shape measuring 2.6 m in diameter and 3.6 m in height, with a capacity to store four standard fuel modules (384 fuel bundles). Fuel is stored in some of Ontario Hydro's 16 CANDU nuclear power reactors in storage modules containing 96 bundles in horizontal position. The storage modules are stacked 8 high in the storage pool.

The CIC's payload chamber has a steel liner surrounded by a 460 mm thick, reinforced high density concrete wall and an outer steel liner. The closure system consists of lid seals and bolting arrangements. The lid plug is 500 mm thick. The gross weight of the container is about 70 Mg.

During the phase I of the demonstration program 2 prototype CIC's are being built and loaded with 10 year cooled fuel. To monitor the performance, the containers are instrumented so that data such as the thermal gradient, radiation levels and seal performance can be obtained.



Comparison Between Dry and Wet Spent Fuel Storage

The application of dry storage technology is very attractive to expand the fuel storage facilities of single unit stations. There is no large, up-front capital expenditure requirements (as it is in the case of constructing a new storage pool). Operating and maintenance costs are low and the final fuel disposal and station decommissioning activities are simplified.

A detailed technical and economical comparison between dry and wet fuel storage options has been carried out for Pt. Lepreau station in Canada.

The results indicated that the cumulative costs of the dry storage option could be 30 to 40 percent lower than for the wet storage option. The assessment included engineering, construction, operating and decommissioning expenses.

The Ontario Hydro nuclear program is based on four unit stations coupled with very large spent fuel storage pools of 3,000 to 6,500 MgU capacity. Due to the economy of scale, the construction cost of these storage pools is below \$4 (U.S.) per KgU. The wet storage has a distinct economic advantage over the dry storage method in this case.

Dry Spent Fuel Storage for Operating Reactors

The implementation of the AECL dry spent storage method by an operating station is a "back-fitting" process. The spent fuel pool usually has a canal extension built in for connection to a future storage pool. The building over the pool area has limited headroom. Any building modification requiring new foundations is restricted by the proximity of the pool support structure, which is very deep. Backfilling after construction of the pool is not necessarily suitable for receiving new foundations.

Integration of the dry storage technology with the pool storage of an operating station requires several modifications to the existing facility such as:

- building an extension to accommodate the permanently installed equipment (shielded work station, flask lifting crane) and provision for transport trailer loading space,
- provision for sufficient foundations to support the heavy equipment (shielded work station and crane),
- space allocation for fuel loading equipment in the storage pool,
- upgrading existing fuel handling crane (manbridge),
- expansion of the Waste Management Area to accommodate the canister site.

The operational aspects of the AECL dry storage method are not fully compatible with wet storage. The fuel bundles are stored horizontally in storage trays up to a stack height of 20 trays in AECL's CANDU 6 water pools. Fuel handling is done exclusively in the horizontal position. The AECL's dry storage concept on the other hand requires vertical positioning of the bundles in the storage baskets. Therefore, additional fuel handling steps need to be carried out to achieve the repositioning.

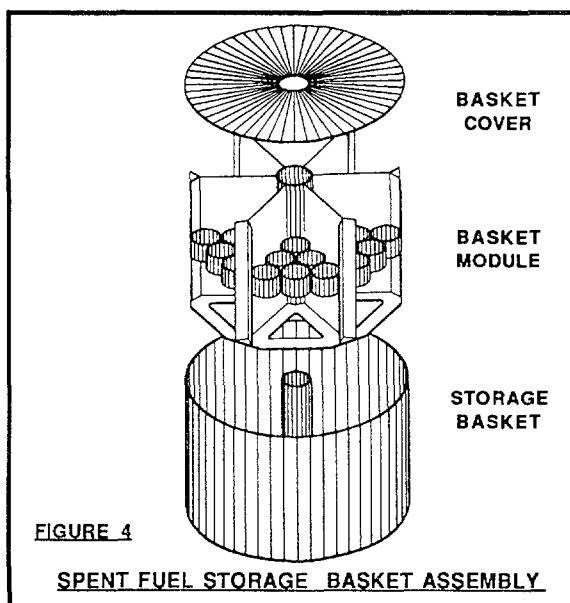
Evolution of AECL's Dry Spent Fuel Technology

The dry fuel storage concept has found widespread acceptance because of its cost effectiveness. Efforts are constantly being made to improve this feature of the technology, with emphasis on the most significant components in terms of expenditures. In order of importance of these components are the concrete canisters and storage baskets which must be built on a yearly basis.

The two limiting factors are the maximum acceptable temperature of the hottest fuel element, and that of the inside concrete surface of the canister. The natural direction leads to increased fuel storage density in the canisters and in the baskets. However, increased basket weight impacts on the lifting equipment and the safety aspects of the fuel handling. The evolution is therefore gradual.

Another significant factor is the cost of stainless steel for the baskets. Carbon steel as a straightforward replacement cannot be considered because of the effect of corrosion on the quality of the basket seal weld. (Corrosion would develop during the residence time of the baskets in the storage pool.)

It is however feasible to store the fuel in open stainless steel "basket modules" in the pool, which could be enclosed in a carbon steel shell following drying in the shielded work station. The basket modules could receive fuel directly from the reactor, would be stacked similarly to the trays and as such the fuel handling operations would be simplified.



Application to Decommissioned and Operating CANDU Stations

The first generation of concrete canisters has been successfully used to store the spent fuel from the decommissioned CANDU prototype reactors (i.e., WNRE, Gentilly-1, Douglas Point and NPD). The geometry of the design limited the storage capacity of each canister to 6.5 MgU of natural uranium oxide fuel.

The first commercial application to extend spent fuel storage facilities, relying on AECL's dry storage technology, is being implemented for the CANDU 6 Point Lepreau Station. An increase in the canister's storage capacity by 60% has been

achieved for fuel 6 years cooled. This resulted in better utilization of the canister's heat release capacity, while the economy of scale maintained a competitive edge over the wet storage option (i.e., the building of additional spent fuel pools). This size increase resulted in the need for the development of new fuel handling equipment, including a larger storage basket, transfer flask, and shielded work station.

The thermal limitations have not been reached either for the fuel or for the concrete. However, the upgrading of the lifting capacities for the fuel handling equipment has about reached the upper limit.

The capacity of the concrete canisters could be further increased by storing one or several additional baskets in each canister. The gain would be modest, while considerable requalification would be needed for safety reasons (drop accidents).

By grouping several canisters into one single concrete module, significant material savings can be achieved. The thermal constraints would require cooling by natural convection, similar to the concept of "ventilated concrete casks" for storing spent fuel of light water reactors.

Application to Future CANDU Stations

Future CANDU stations will benefit from the enhancement of the dry storage technology. The fuel handling facilities will be designed to facilitate execution of the dry storage. Therefore, modifications for implementation of the dry storage will be kept to a minimum.

The most significant cost savings can be realized if the future canister storage requirements are taken into consideration at the design stage of the station. Both up-front and operating costs can be significantly reduced.

The fuel can be stored in basket modules, at the time of discharge from the reactor, and be loaded directly into the basket shell, inside the shielded work station.

The concrete temperature limitation for stand-alone canisters is reached if fuel is stored with approximately 5 years of cooling. This translates into a smaller storage pool size than traditionally built which is 10 year storage capacity. Space for fuel in the storage pool is also reserved for one full reactor core discharge. This extra storage capacity can, however, be maintained indirectly, in prebuilt canisters, thus further reducing the spent fuel pool size.

Introducing grouped canisters cooled by air convection will not only save dry storage costs but will allow the storage of fuel with less than 5 years cooling, again allowing capacity reduction of the storage pool.

CONCLUSION

AECL has now virtually completed the dry fuel storage of all its decommissioned reactors in the "first generation" of concrete canisters.

The larger, 10 MgU capacity canisters will be used at operating CANDU 6 stations on a continuous basis.

The future CANDU stations, designed by AECL, will have reduced wet storage capacities and will rely on dry storage technology during most of their operating lifetimes.

Several improvements to AECL's dry storage technology are planned to safely and more economically meet future spent fuel storage requirements.

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TVO — EXPANDING THE SPENT FUEL STORAGE CAPACITY IN OLKILUOTO IN FINLAND

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Abstract

The Industrial Power Company ltd. (TVO) is a Finnish power company with two 710 MW ASEA ATOM type boiling water reactors. They have been in operation since 1978 on the west coast of Finland.

The original power plant construction included a three-year storing capacity for spent fuel.

During the construction phase it was realized that the storage pool constructions have to be changed, so that in 1980 it was possible to increase the storage capacity to eight years by high density storage racks.

During the years 1980 to 1982 TVO made preliminary studies on alternative methods for the intermediate storing of spent fuel.

The construction work for a separate intermediate storage, a water pool storage, was started in the beginning of 1984. The first fuel transports to the separate spent fuel storage (KPA-store) were done in September 1987.

In the KPA-store, the storage capacity is roughly 1200 t U, which represents about 30 years of production in the TVO I and II power plant units. However, the KPA-store's capacity can still be increased according to TVO's needs either by rod consolidation or by constructing more pools.

The KPA-store is located near the power plant units, operated by TVO I personnel, and it uses some services from TVO I. In this way, the unit cost for the intermediate storage of spent fuel in Olkiluoto is reasonably low. The investment costs for the KPA-store were less than 200 FIM/kg U in 1987.

1. GENERAL

Industrial Power Company ltd. (TVO) is a Finnish power company, which has two 710 MW ASEA ATOM type boiling water reactors in operation since 1978 on the west coast of Finland. TVO produces about 20 % of the electricity in Finland today. TVO has a personnel of approximately 500 persons.

TVO's total production of electricity so far is almost 100 TWh. The lifetime capacity factor for TVO I is 81 % and for TVO II 79 %. For the year 1988 the capacity factors are 92,9 for TVO I and 91,9 for TVO II.

The original net electrical power for TVO I and II was 660 MW per unit but it was upgraded to 710 MW in 1984.

The amount of fuel in TVO reactors is about 90 t U/reactor and the annual production of spent fuel from both units is over 40 t U.

2. THE SPENT FUEL STORAGE CAPACITY IN THE ORIGINAL POWER PLANT CONCEPT

The original idea in the early 1970's for TVO's spent fuel policy was that TVO's spent fuel will be reprocessed outside Finland. Therefore the TVO I and TVO II fuel pools were constructed for about three years' storage capacity of spent fuel. The total amount of fuel element storage positions in one unit was 800 consisting of 300 storage positions and 500 positions for unloading the whole reactor core.

However, the delays in the reprocessing capacity and the market price for reprocessing did increase to such an extent that TVO decided that 300 storage positions was not sufficient.

Already in the construction phase TVO made a decision to rearrange the fuel pool layout so that there was space for 4 more spent fuel racks in both units, also

taking into account the loads of high density storage racks. The original fuel pool layout and the rearranged layout are presented figures 1 and 2.

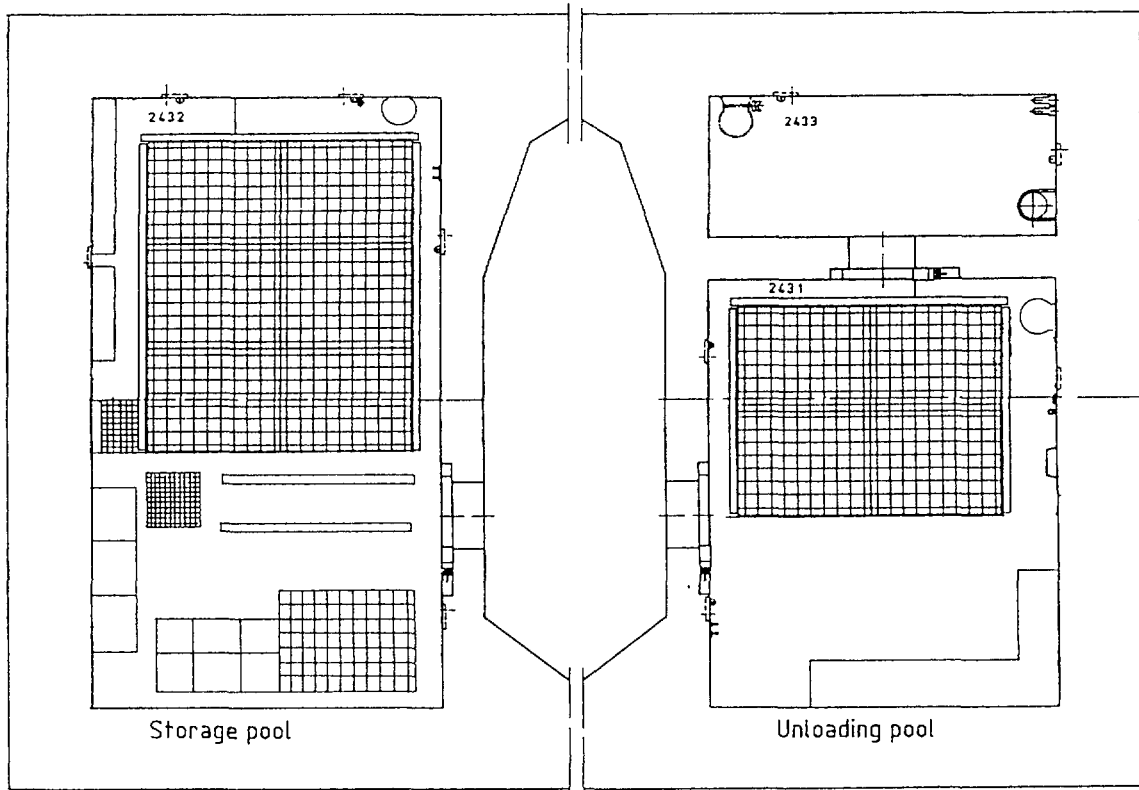


FIG.1. Original pool layout.

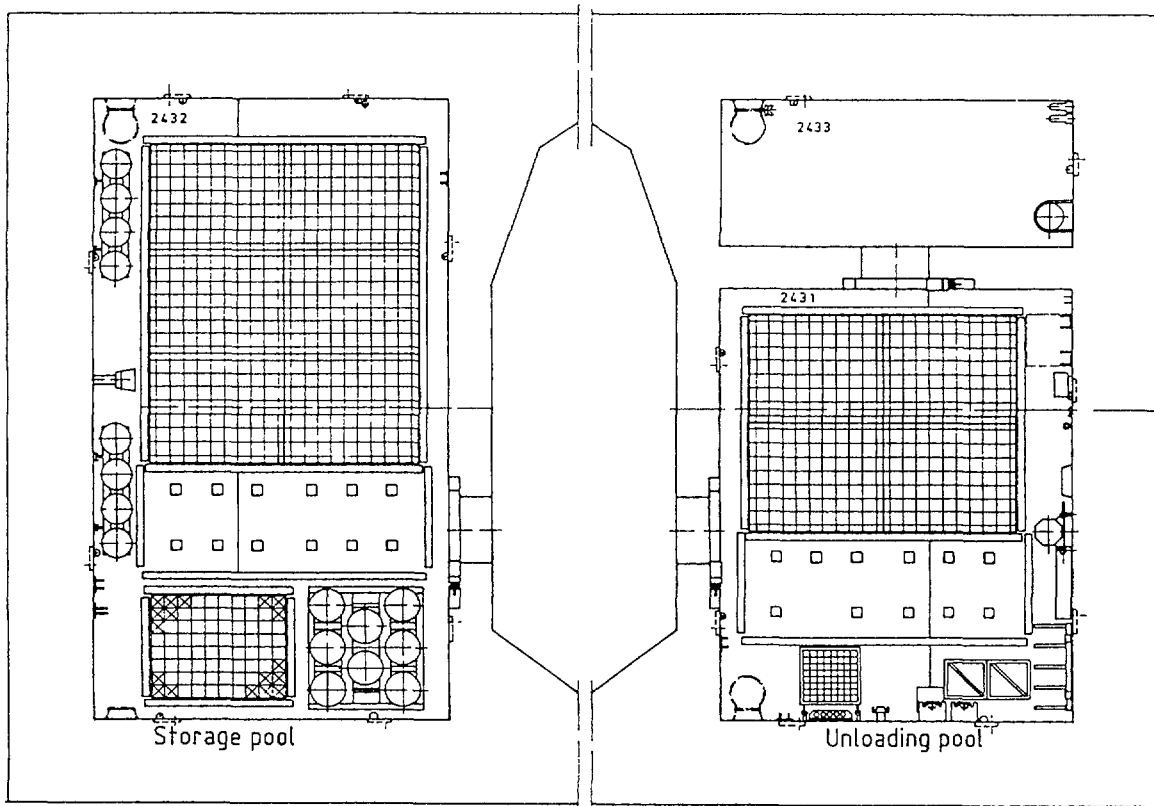


FIG.2. Rearranged pool layout.

3. THE AUTHORITY REQUIREMENTS DURING THE CONSTRUCTION PHASE OF THE POWER PLANTS

The main requirement during the construction phase of the power plant units was to present a way in which TVO will handle the spent fuel despite its lack of spent fuel reprocessing capacity.

To fulfill the above mentioned requirement, TVO made a preliminary plan together with Asea Atom for an intermediate water pool storage for spent fuel in Olkiluoto. This study was presented in the Seminar on the storage of spent fuel elements in Madrid 20.-22. June 1978 (figures 3 and 4).

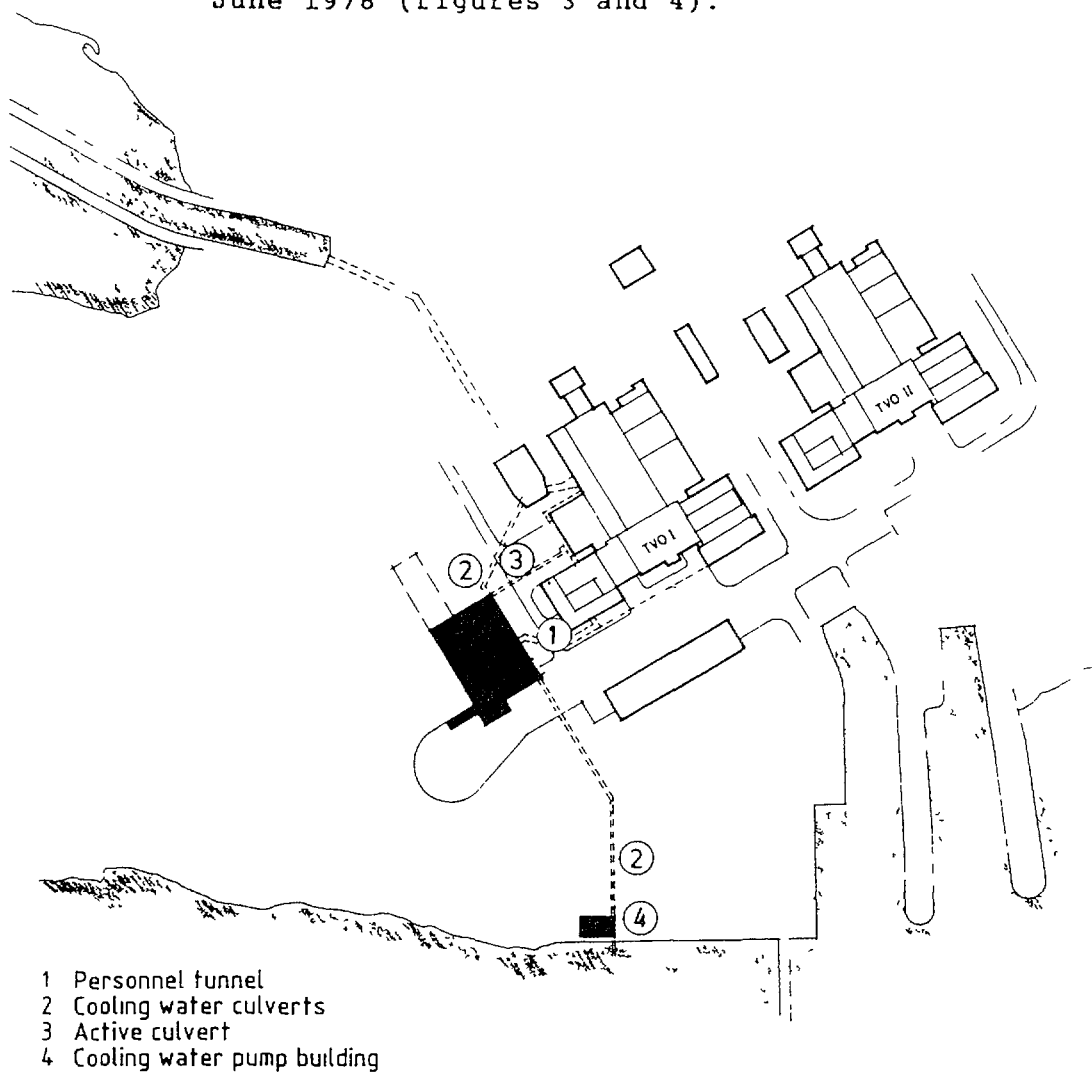


FIG.3 Location of the spent fuel storage facility

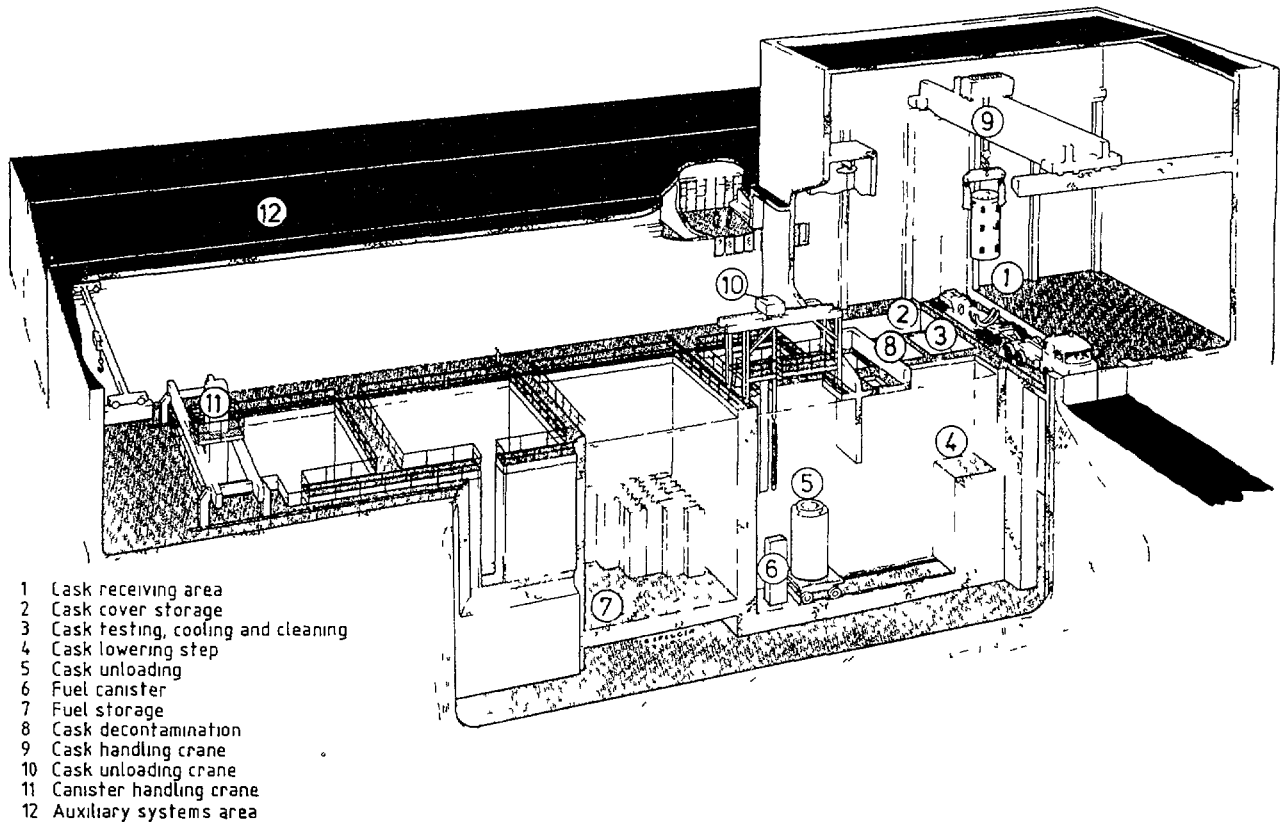


FIG.4. Spent fuel storage facility.

In this study it was shown that the storing can be realized in four years if necessary. During this phase the authorities also made an essential requirement for the capacity of spent fuel: "It must always be possible to unload any spent fuel pool into the other pools on the power plant site, and there must always be enough capacity to unload the reactor core. It is not necessary to fulfill the above mentioned requirements at the same time."

4. HIGH DENSITY SPENT FUEL STORAGE RACKS

Originally there were 10 aluminum storage racks in the power plant units. Each rack contains 80 storage positions. The center line distance between fuel elements is 250 mm.

In 1980 TVO ordered 10 KWU type compact storage racks to be installed in the TVO I and II power plant units. Each rack contains 180 storage positions. The center line distance is 165 mm and the outer measures are the same as in the original aluminum racks.

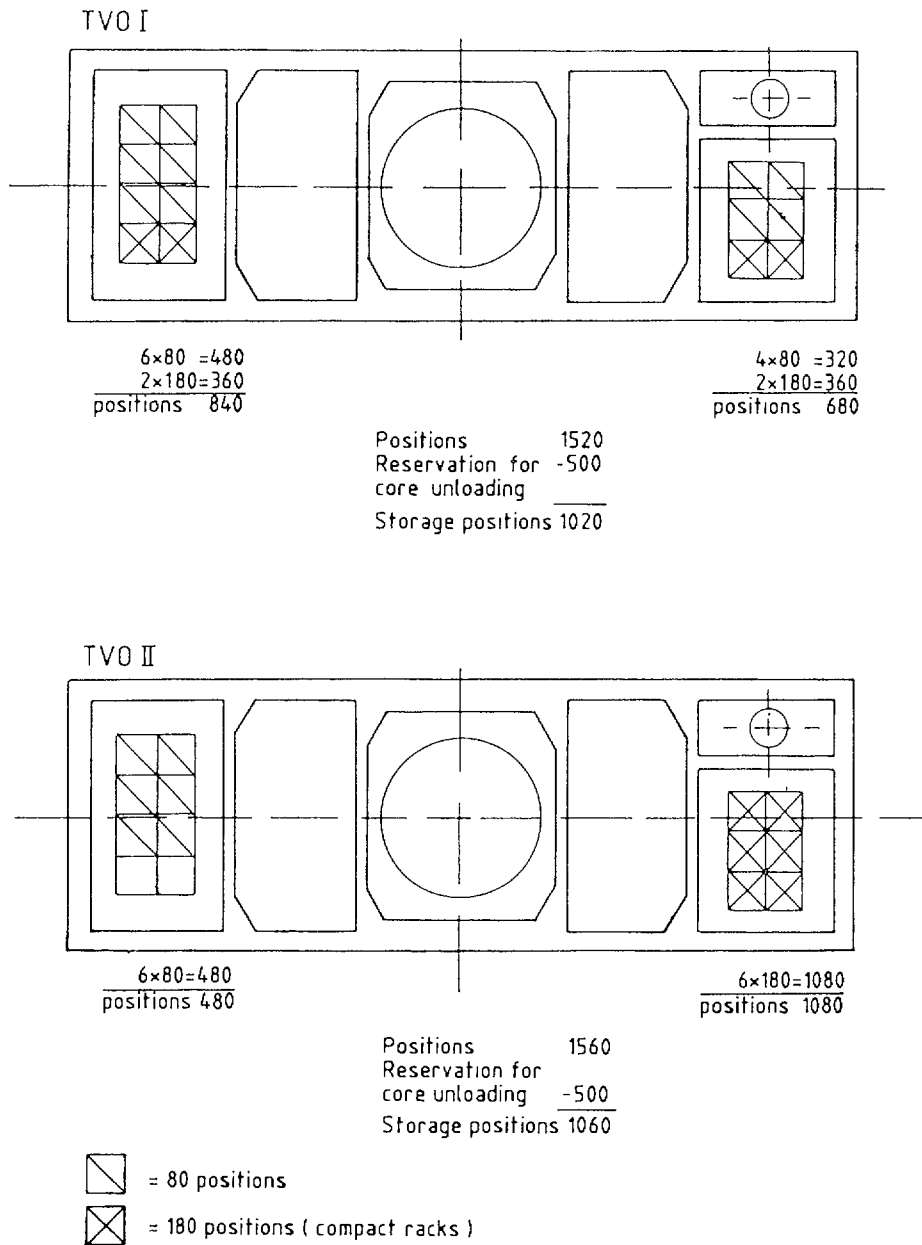


FIG.5. Spent fuel storage capacity in TVO I and TVO II.

The installations were realized according to figure 5. In TVO II 4 aluminum racks were replaced by 6 compact storage racks and in TVO I the empty positions were filled with compact racks. Thus there is still room for two more storage racks in TVO II.

As shown in figure 5, there are about 1000 storage positions and space for the whole core. Today, after the power upgrading with 8 %, this represents approximately 8 years' spent fuel production.

5. STUDIES FOR DIFFERENT TYPES OF INTERMEDIATE SPENT FUEL STORAGES IN OLKILUOTO

When TVO ordered compact racks for TVO I and II, it was also noticed that the enlarged storage capacity of the power plant units was not sufficient for lifetime intermediate storing of TVO I and II spent fuel. In 1980, a study was started on the different type of intermediate storing of spent fuel at the power plant site. This study was continued by a separate report in 1981, concentrating in more detail on the storing methods chosen in the first study.

In these studies the chosen methods were:

- water pool store of Finnish design
- dry vault store made by the English GEC-ESL (GEC = Energy Systems limited) for TVO
- cask store based on TN 1300 (TN = Trans Nuclear) cask

At that time (1982) the following results, among others, were found.

Technically, all three interim storage methods mentioned above are suitable for TVO. However, it can be stated that when the principle "proven design" for the spent fuel of light water reactors is followed, the order of preference is obviously the following:

1. water pool store
2. cask store
3. dry vault store.

When comparing the costs of the different alternatives economically, the dry vault store is the most favourable and the cask store clearly the most expensive solution. By computing real interest on the invested capital the cost differences of these three alternatives even out but the order will not be changed even with a +5 % interest. Although the cost estimates of dry stores are more unreliable than the

corresponding estimates of water pool stores, the dry vault store can be seen to be the most inexpensive alternative.

The water pool store is the only store which can positively be licenced. On the basis of the international development it seems that after a few years, also the cask store can be licenced, whereas the licenciability of the dry vault store can not be anticipated at the moment because the store type in question has not been realized at any other place or considered to be used for light water reactor fuel.

6. AN INTERMEDIATE STORE FOR SPENT FUEL IN OLKILUOTO (KPA-STORE)

On the basis of the studies presented in the chapter above, TVO decided to realize a water pool intermediate store (KPA-store) in Olkiluoto for spent fuel.

The project KPA-store was realized by TVO's own personnel by the end of 1987. The preliminary design work started in 1981, the construction work in 1984 and installations in 1985.

The storage capacity is about 1200 tons U and the realization costs were well below 50 M USD.

The store is presented in figures 6, 7 and 8.

6.1. GENERAL DESIGN PRINCIPLES FOR THE KPA-STORE

The KPA-store is designed to store all the spent fuel produced by TVO I and TVO II power plant units. Today's estimate is 1400 tons U. In the first phase the storage capacity is 400 tU/pool and there are three storage pools and one pool for fuel handling and the evacuation of the other pools. It is possible to enlarge the store by additional pools.

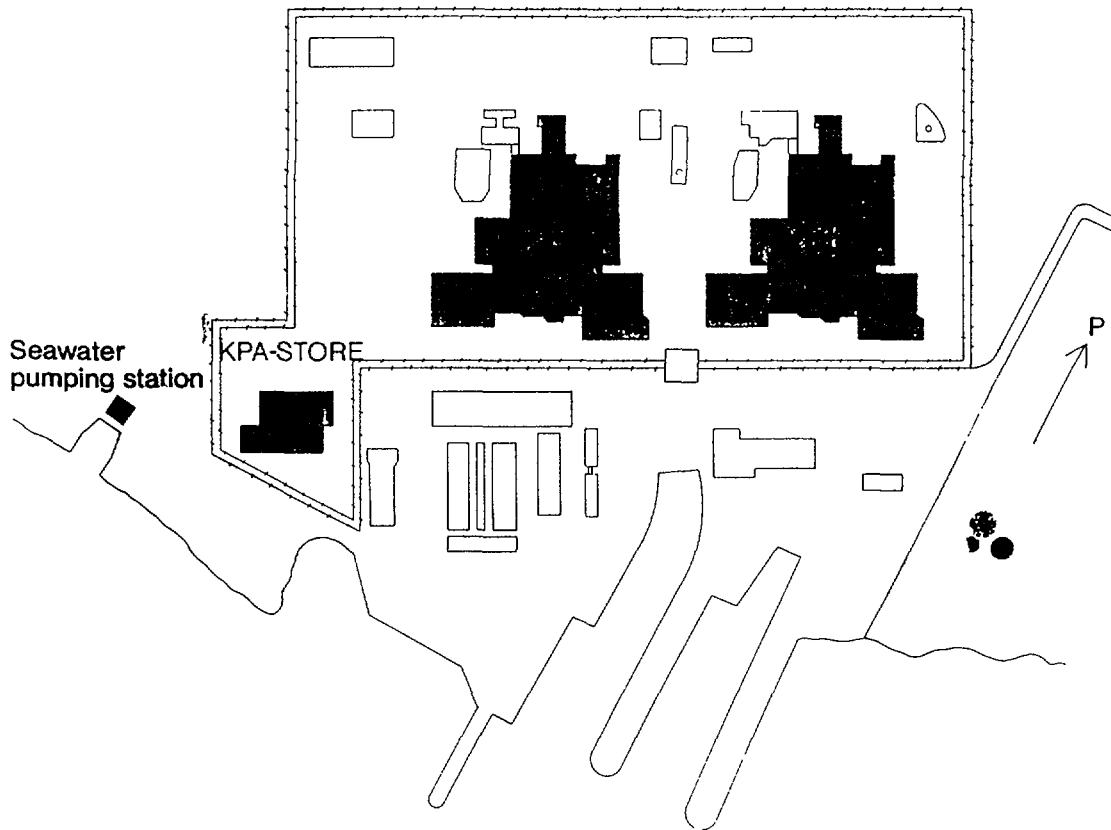


FIG.6. KPA store

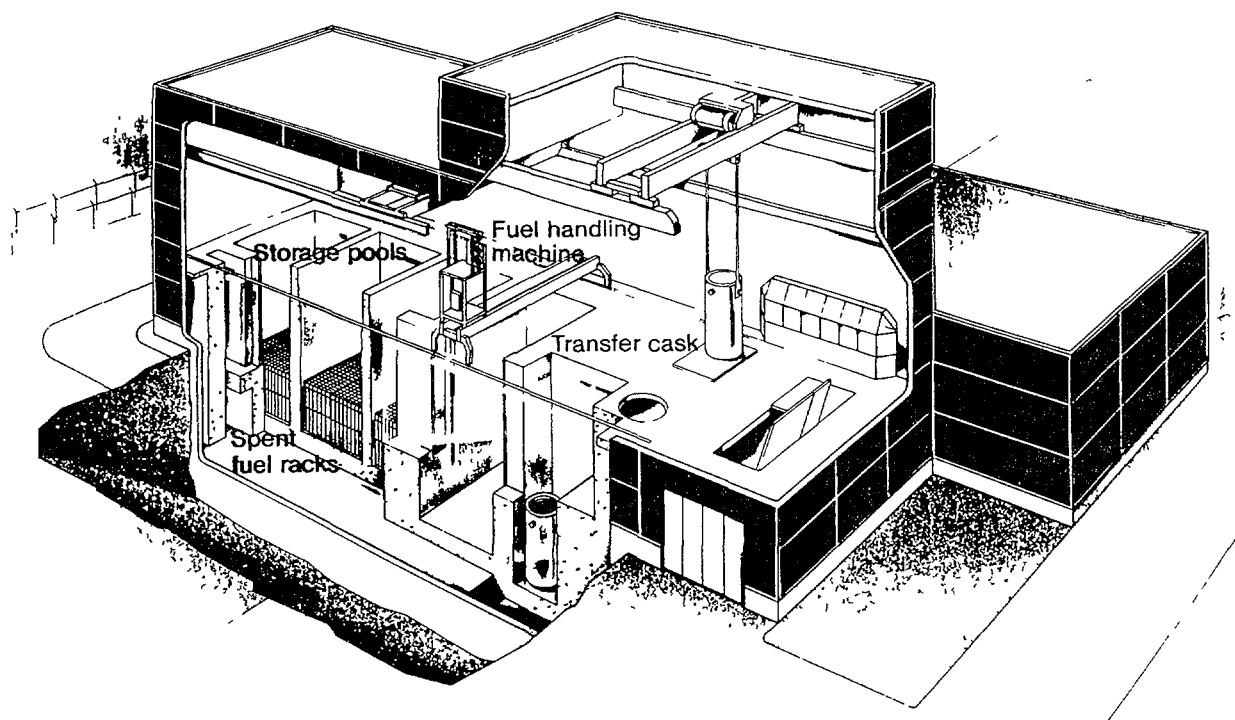


FIG.7 KPA store

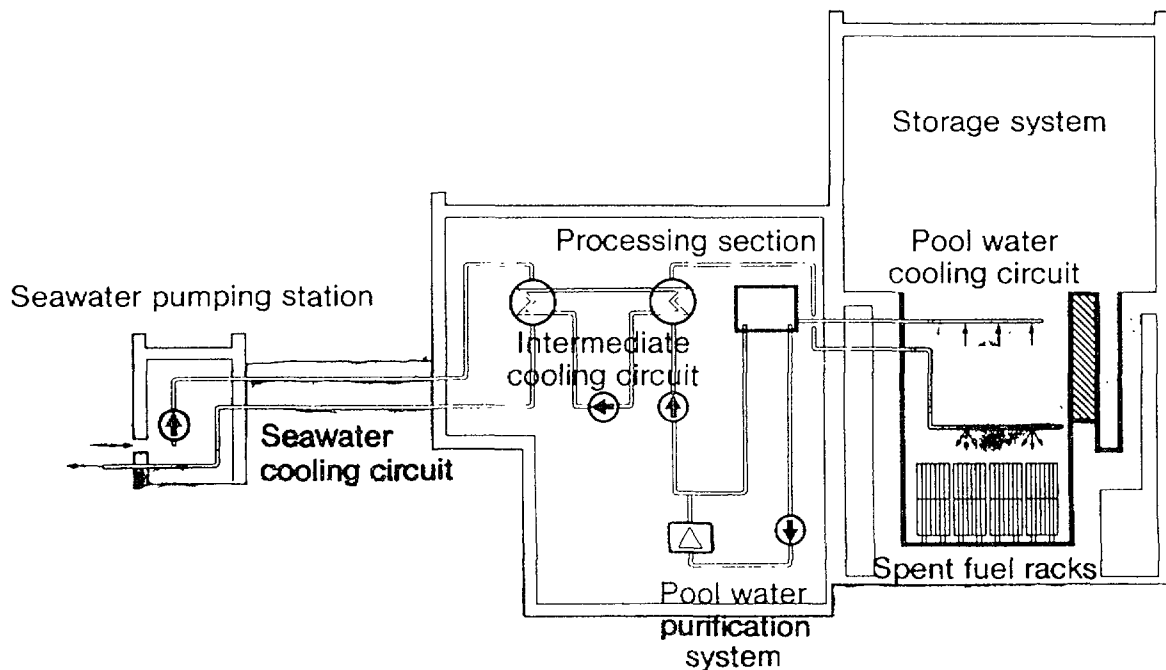


FIG.8. KPA store.

The designed lifetime of the store is 60 years. For an individual fuel element it means 40 years intermediate storing time.

After 40 years cooling time it is possible to dispose the spent fuel finally in the bedrock.

It is possible to enlarge the store so that other core components can also be stored, e.g. control rods, neutron detectors and so on.

It is possible to make wet fuel transport from TVO I and TVO II to KPA-store. Later it will be possible to construct an additional unloading line so that it is possible to transport fuel out wet or dry.

There is 2 x 100 % cooling capacity for the pools.

The KPA-store fulfills the Finnish rules, regulations and laws.

As applicable, also the international standards and regulations are fulfilled.

6.2. LAYOUT AND TECHNICAL SOLUTIONS

The KPA-store is situated on the nuclear plant site in Olkiluoto. The store is divided into reception, storage, process and control sections. The seawater pump station is located separately.

The reception section includes transport of the cask, cooling and purification position for the cask, cask unloading pool and decontamination pool. There is a transfer machine for unloading the cask and 125 t and 20 t cranes for cask handling.

The storage section consists of three storage pools and one spare pool where fuel handling and inspections can be done. The pools are 13,5 meters deep, of thick concrete with stainless steel cladding.

Spent fuel racks, where also consolidated fuel can be stored, have been installed on the pool bottom. Without rod consolidation the storage capacity is 400 t U/pool.

It is possible to construct additional storage pools.

The process section contains cooling circuits for the pools. The cooling circuit has 2 x 100 % capacity and there are pool, intermediate and seawater cooling circuits. For the pools there is also a separate system for purification. It consists of two precoat filters. There is a space reservation for extra filter and also for a totally new system for fuel crud handling.

All auxiliary systems, heating ventilation, drainage, makeup water system and so on are located in the process section.

The control section includes the control room and the electrical and instrumentation centres.

In the seawater pump station there are seawater pumps, which pump seawater to the heat exchangers in the process building.

There are also two tunnels from the KPA-store to TVO I power plant unit. One tunnel is a personnel tunnel from TVO I to KPA-store. There are pipelines, which feed the KPA-store with makeup water, pressurized air and firewater. KPA-store is heated by waste heat from TVO I. In the other pipe tunnel the liquid wastes, ion exchange resins, decontamination chemicals and drainwater are pumped into TVO I for treatment.

6.3. OPERATION OF THE STORAGE

In the first phase the storage is operated by TVO I personnel. This means that normally the storage is not occupied and that during daytime only routine maintenance and inspections are done.

It takes at least one week before the water temperature rises to 100°C, if there is no cooling. So there is no need for immediate actions after disturbances in the store. However, all noteworthy alarms also go to the TVO I control room.

When fuel is transported from TVO I or II to KPA-store there will be a special group to take care of fuel transport. This group will consist of about 10 persons.

In the second phase when TVO I and II power stations are no longer in operation, there will be a need for an operating personnel for the KPA-store. Today TVO estimates that this will not happen before 2010. At that stage TVO thinks that approximately 10 fulltime personnel will be needed, as well as a fuel handling group during the fuel transportation.

After about two years of operation it can be said that no major problems in the operation of the KPA-store have occurred, and the need for changes in the storage have been minimal.

7. TVO'S SPENT FUEL STORAGE CAPACITY TODAY AND IN THE FUTURE

TABLE I. TVO — STORAGE CAPACITY OF SPENT FUEL

	FUEL ELEMENTS	URANIUM	STORAGE CAPACITY
TVO I	1020 POSITIONS	182 TONS	8 YEARS/TVO I
TVO II	1060 POSITIONS	188 TONS	8 YEARS/TVO II
KPA-STORE	6804 POSITIONS	1210 TONS	27 YEARS
TOTAL	8884 POSITIONS	1580 TONS	35 YEARS

STORAGE CAPACITY UP TO YEAR 2010 WITHOUT ENLARGING THE STORAGE CAPACITY OF THE KPA-STORE

As we can see, the spent fuel storage capacity in Olkiluoto is about 1600 t U, which represents about 35 years' production of spent fuel from TVO I and II. Thus according to today's assumptions TVO has storage capacity up to the year 2015.

However, today we do not know if there will be more nuclear power in Olkiluoto or if the lifetime of TVO I and II will be longer than 30 years. Regardless of that, there is a possibility of enlarging the storage capacity if necessary. Technically we can today see two main choices:

- Rod consolidation in the KPA-store (pools and spent fuel racks are designed for rod consolidation).

- Increasing the number of storage pools
(process systems and layout are designed for
at least six storage pools).

Today we can see that we have achieved one of our targets, namely that we are flexible in our spent fuel intermediate storing and we can afford to wait for the final decision of the spent fuel handling (reprocessing or direct disposal) until the next millenium.

CONTRIBUTION OF SIEMENS TO SPENT FUEL MANAGEMENT TECHNIQUES

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Abstract

The Government of the Federal Republic of Germany is responsible for the final disposal facilities whereas the industry is responsible for all other facilities such as interim storage and reprocessing facilities. Sufficient storage capacity can be achieved by compact storage racks, rod consolidation, AR or AFR wet storage and dry storage. The development of storage racks from normal to compact racks accounting for burnup is pointed out. Rod consolidation as a further step to gain interim storage capacity is described. An additional at reactor wet storage facility was realized for the NPP Atucha I in Argentina. Finally a description is given of an at-reactor and away-from-reactor dry vault store with direct air-cooling by natural convection.

1 Introduction

The successful development of interim storage techniques permits to store spent fuel safely and economically for a long time. If interim storage is applied, all choices for disposal remain open and the decision between the two possibilities - reprocessing or final disposal of spent fuel assemblies - could be made in due course. Even in the case of reprocessing interim storage will be advantageous to reduce the activity inventory and the heat generation of the spent fuel. Fig. 1 shows that the activity inventory of the fuel assemblies after 7 years of interim storage decreases considerably by natural decay to about 3.6 % of the original inventory at discharge from the core. In case of reprocessing 77.5 % of the remaining inventory will go with the waste, mainly as vitrified high level waste. 21 % be feed back as recycled Uranium and Plutonium in the mixed oxide fuel fabrication and finally only 1.5 % is released as emission by air and water to the environment.

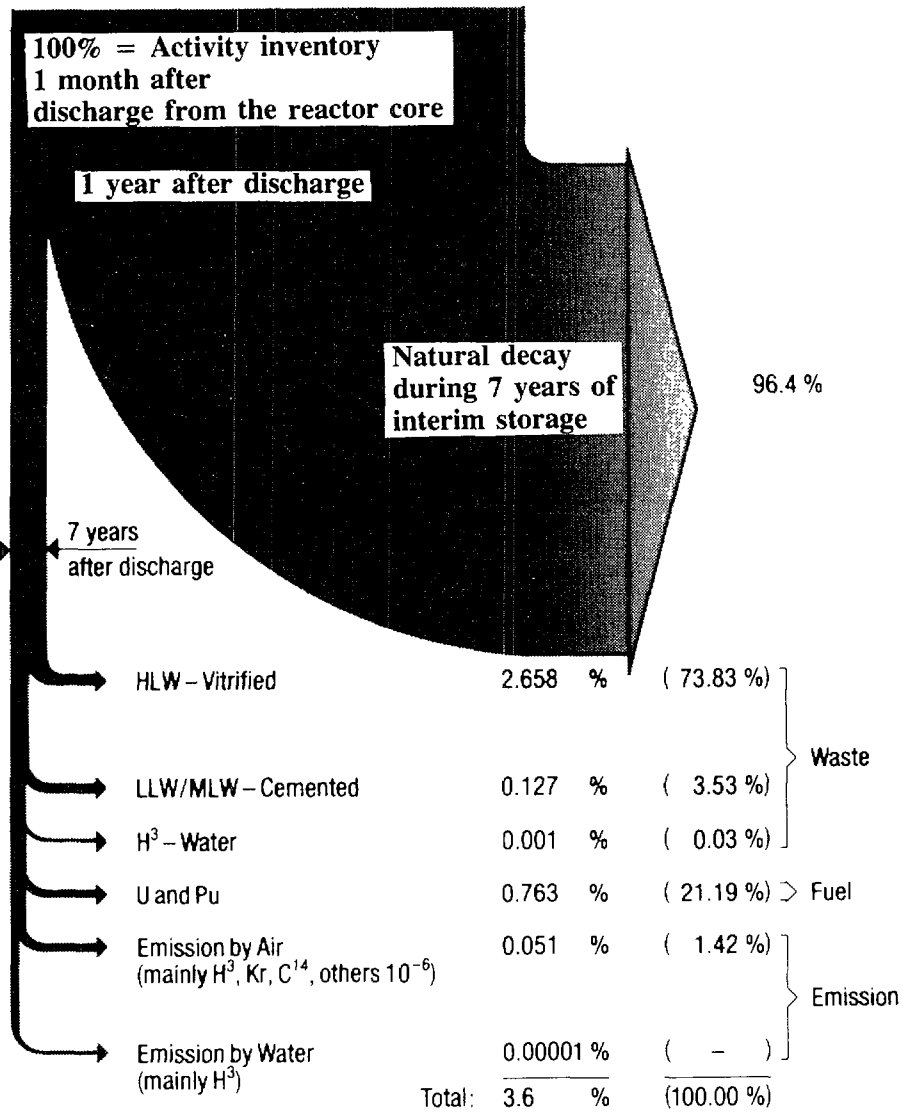


FIG.1. Activity inventory considering storage and preprocessing.

2 Back-End of Fuel Cycle in the FRG

According to the atomic energy law the reliable nuclear disposal is a prerequisite for the use of nuclear energy. Section 9a of the law gives priority to the re-use of valuable material, if re-use is technically feasible and economically justified, and it requires the disposal of not re-usable waste according to stringent safety requirements.

The federal government is responsible for the erection and operation of final disposal repositories (Fig. 2) whereas the industry is responsible for all other facilities such as

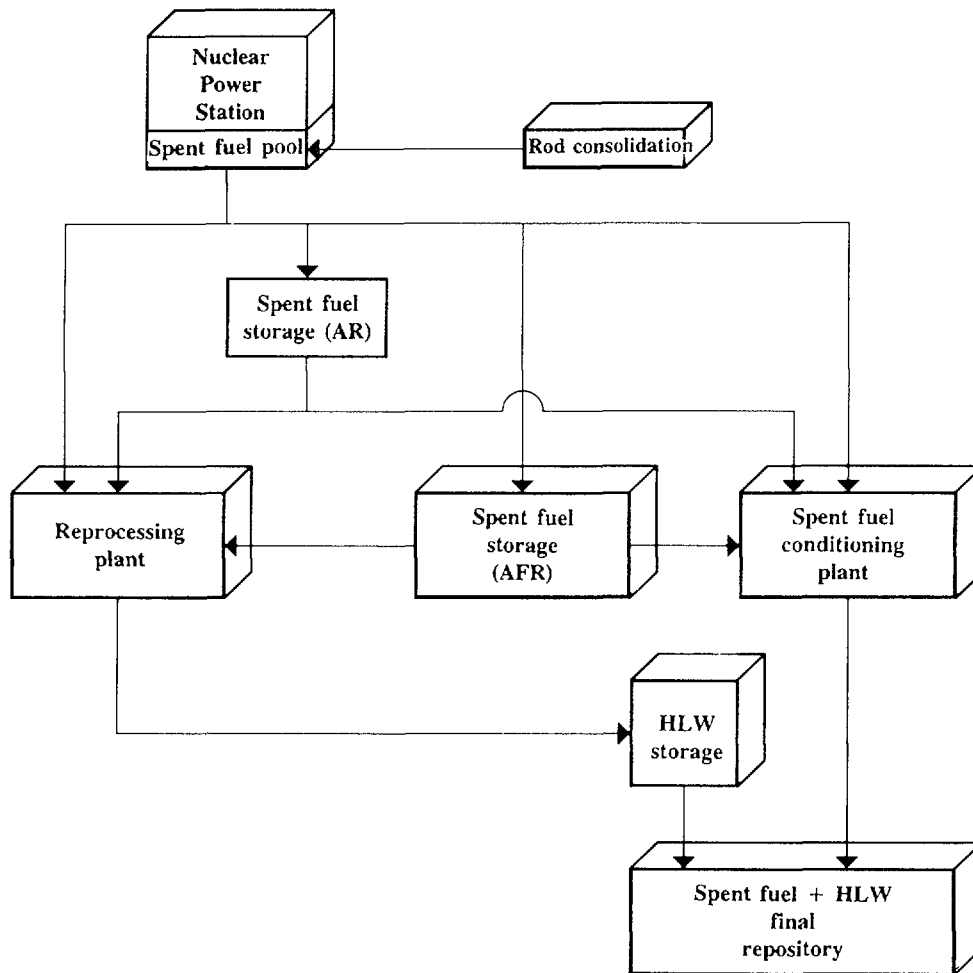


FIG.2. Spent fuel management.

- interim storage facilities for spent fuel assemblies at reactor (AR) or away from reactor (AFR)
- reprocessing plants for spent fuel assemblies including recycling of the fissile material
- conditioning facilities for radioactive waste
- interim storage facilities for radioactive waste
- conditioning and encapsulation plants for spent fuel assemblies if re-use is not technically advisable or economically justified.

The utilities have to give evidence to the authorities 6 years in advance how to handle the spent fuel. Generally this will be achieved by interim storage at reactor in high density racks.

Dry interim storage facilities away from reactor for spent fuel assemblies are in operation as in Gorleben or partly constructed as in Ahaus. The capacity of each facility is 1.500 t U.

The pilot reprocessing plant WAK in Karlsruhe with a capacity of about 40 t U/a is in operation since 1971. The trend of the last weeks indicates that the Wackersdorf plant will be waived and the German industry will probably participate in the La Hague - and/or Sellafield - plant.

Interim storage facilities for radioactive waste from nuclear power plants are in operation in Gorleben with a capacity of 40.000 drums since 1986.

The iron-ore mine "Konrad" will serve as final disposal repository for no heat generating waste. The capacity will be about 5.000.000 m³. The start-up is expected in the middle of the 1990's (1993).

The "Gorleben"-repository is foreseen for all categories of radioactive waste including high level active waste. A decision on its suitability will be expected in the late nineties (1995 - 1999). The operation is scheduled for the beginning of the next century (2008).

The feasibility of direct disposal techniques for spent fuel whose reprocessing is economically not justified will be demonstrated by means of a pilot conditioning and encapsulation plant. The plant shall be built at the site of Gorleben, adjacent to the spent fuel storage facility. The throughput will be 35 t U/a. The licensing procedure was initiated in May 1986. The start is scheduled for 1994.

The requirement for sufficient storage capacity can be fulfilled by different strategies (figure 3):

- Compact storage racks in nuclear power plants
- Rod consolidation
- Wet storage at reactor and away from reactor
- Dry storage concepts.

-
- Compact storage racks
in nuclear power plants
 - Rod consolidation
 - Wet storage at reactor (AR)
and away from reactor (AFR)
 - Dry storage concepts
-

FIG.3. Spent fuel storage strategies.

3 Compact Storage Racks in Nuclear Power Plants

For technical and economic reasons it has become an advantage to use the fuel pools in existing and planned nuclear power plants for extended interim storage of spent fuel assemblies. Siemens has therefore developed storage racks in which the spent fuel assemblies can be more densely packed than was previously possible. The required neutron absorption is no longer provided only by the pool water and the material of the rack structure but also by absorber channels manufactured from boron steel. In case of Germany, typically, the storage capacity of 5 thirds is increased to 12 thirds. The following figures 4 and 5 show compact storage racks installed and under assembling, respectively.

The relevant criteria for a proper design of compact storage racks are given in figure 6.

As an example the influence of some important parameters on subcriticality is shown in figure 7. For the storage of fuel assemblies from PWRs with 16 x 16 array, 230 x 230 mm width, and 3.5 w/o U-235 enrichment, the required wall thickness for the absorber channels is calculated to be 3 mm, the center-to-center spacing of the fuel assemblies to be 283 mm - compared to 380 mm previously - and the boron concentration in the absorber channel material is 1 %.

Several years ago the utilities changed their fuel management strategies to improve fuel economics and to reduce the amount of spent fuel. This was achieved by an increase of initial enrichment. Ini-

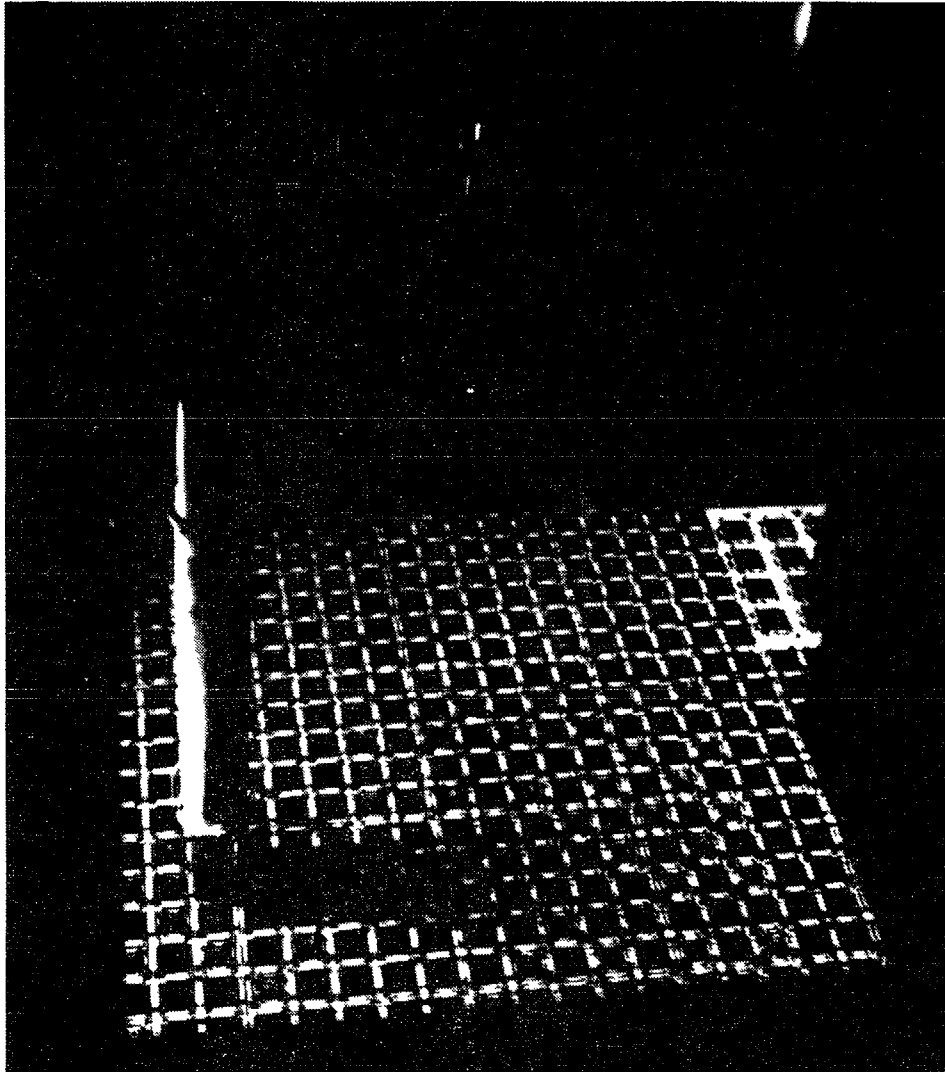


FIG 4 Spent fuel assembly storage in the nuclear power station

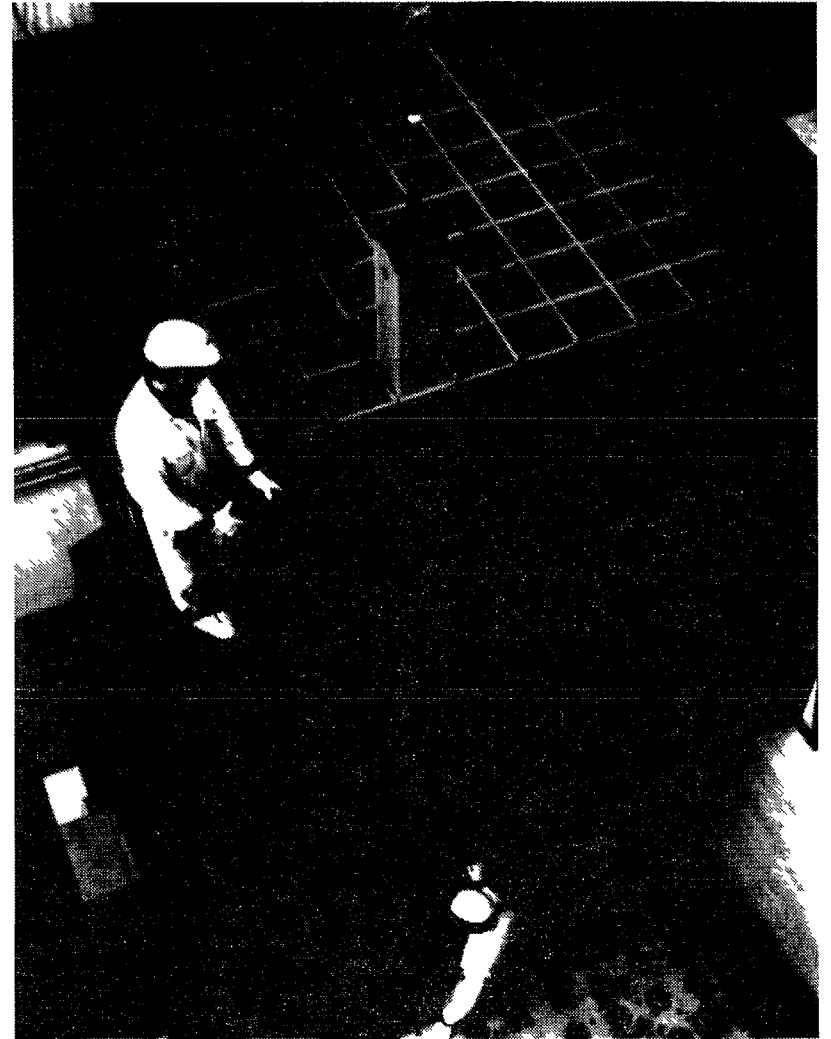


FIG 5 Spent fuel storage rack

-
- Subcriticality under normal and accident conditions
 - Mechanical design under normal and accident conditions, including static and dynamic analyses
 - Decay heat removal under normal and accident conditions
 - Maintenance of permissible dose rate
 - Adequate visibility during loading and unloading operations
-

Fig. 6. Storage rack design criteria.

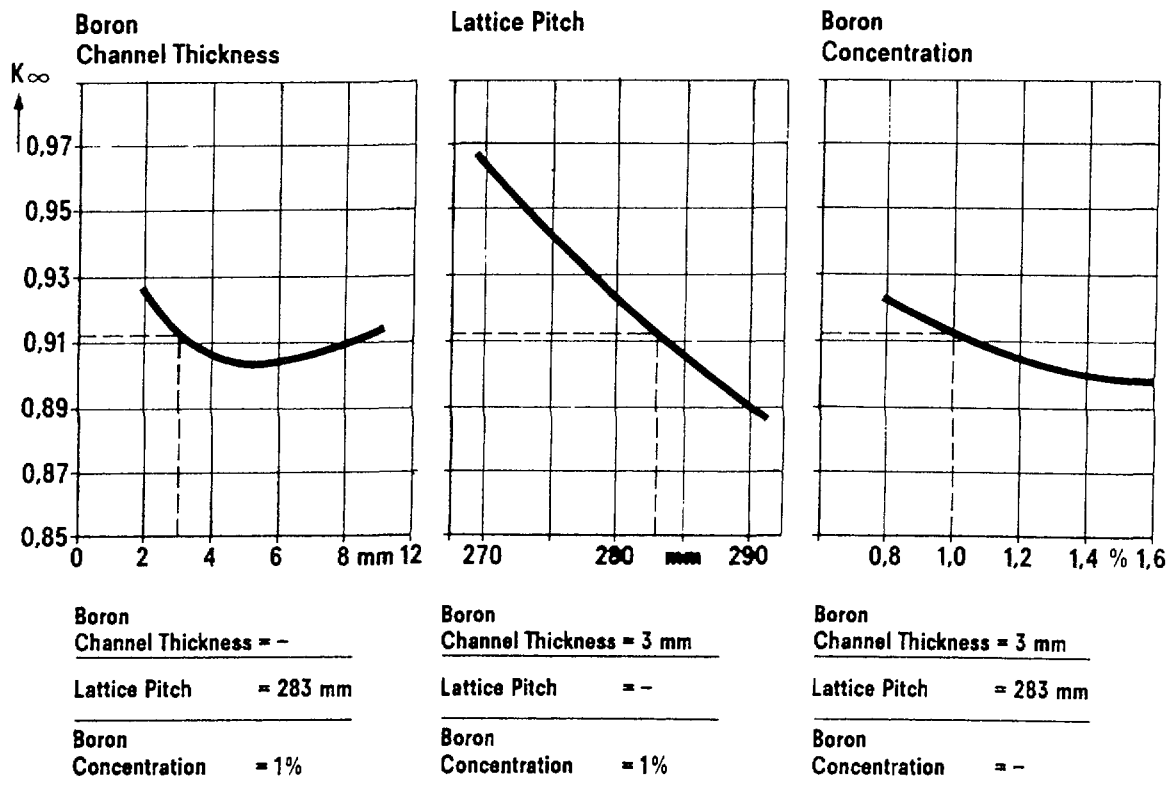


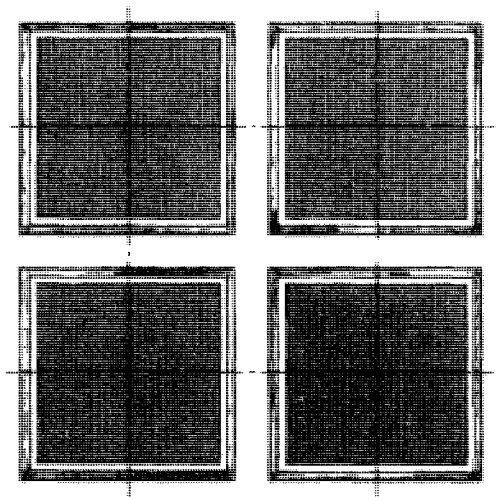
FIG. 7. Subcriticality of compact storage racks for 16 × 16 PWR fuel elements with 3.5% U-235.

tial enrichments of 3.0 to 3.5 w/o are attained and in future, even higher enrichments appear possible. The increase of initial enrichment results in the problem of economical storage of spent fuel assemblies in the spent fuel pool.

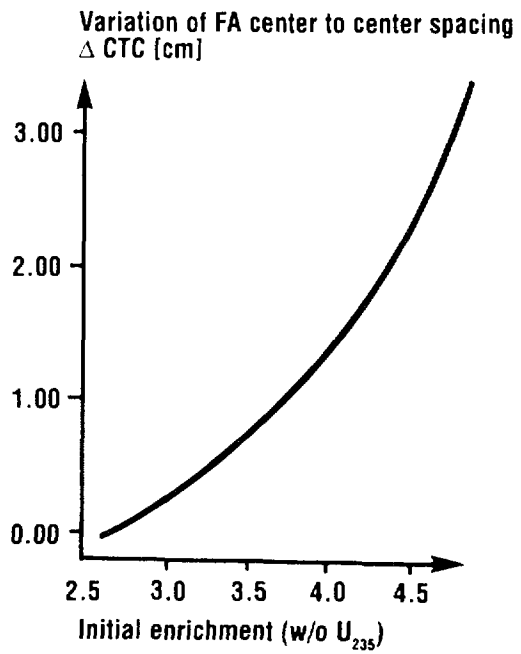
In the past, all spent fuel storage pools were licensed to store the highest enriched fresh fuel. There were no burnup restrictions on the fuel to be placed in the spent fuel storage, which was a restrictive simplifying assumption that severely reduced the spent fuel pool storage capacity and the enrichment limit even when high-density storage racks using neutronabsorbing materials were installed.

Consequently after determination of the requirements burnup credit was introduced in the United States in 1981 and till today some pressure water reactors are equipped with spent fuel storage racks licensed to account for burnup. Designing racks to take credit for burnup introduces several calculational complexities with respect to criticality that are not present in designing for unirradiated, fresh fuel storage. The criticality calculations shall consider the buildup and burnup of fissile materials, the buildup and postirradiation decay of fission products and the nonuniform axial burnup. To maintain a core off-load capability it is necessary to have a two-region pool. A small region must have racks capable of holding fuel of reactivity up to that of fresh fuel with the highest anticipated enrichment (figure 8). The rest of the pool contains racks that only accept fuel that has achieved a specified minimum exposure, typically chosen to be 80 % of the design discharge exposure (figure 9). For each region the criticality criterion of $K_{eff} \leq 0.95$ must be met with unborated water and must include uncertainties.

Criticality calculation in the case of accidents is similar to that of a single region pool with the exception that the effect of the inadvertent storage a fuel assembly with insufficient burnup in the pool region accounting for burnup must be considered. Credit may be taken for the presence of soluble boron in the pool water for this calculation. A final aspect of the two-region pool designs is assurance that only assemblies exceeding the required minimum burnup are stored in the region accounting for burnup. This can be achieved either by measuring the average burnup of each as-

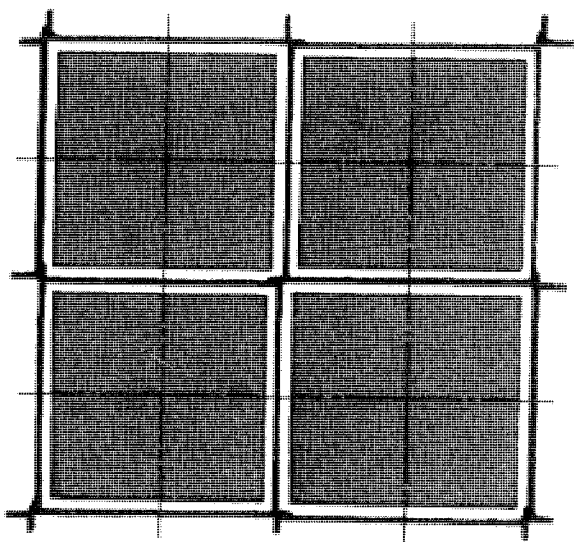


Fresh fuel rack with double neutron absorber region I design

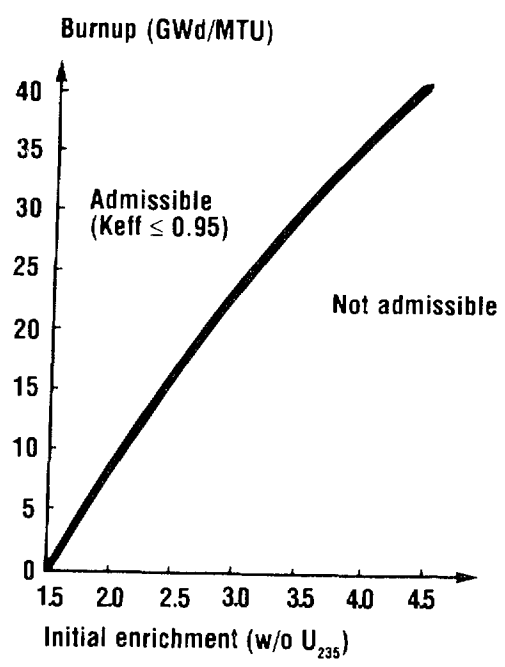


Equivalent reactivity diagram – region I (for illustrative purpose only)

FIG.8. High density spent fuel storage.



Spent fuel rack with single neutron absorber region II design



Equivalent reactivity diagram – region II (for illustrative purpose only)

FIG.9. High density storage of spent PWR fuel (burnup credit)

sembly or by measuring the average assembly reactivity in a standard configuration or by calculating the average assembly burnup by intergrating the exposure rate obtained from records of reactor power and assembly peaking factor as a function of time.

For on-site storage in the United States, the NRC has concluded that the calculation method is acceptable. To minimize the probability that a loading error will be made, the NRC requires strict administrative controls on spent fuel placement.

Till today, only one spent fuel storage pool is licensed to account for burnup in the European countries. The applicable standards allow burnup credit in those countries where the storage rack requirements are based on US-regulatory guides, US-NCR Regulatory Guide 1.13 "Spent Fuel Storage Facility Design Basis" and ANSI/ANS-57.2-1983 "Design Requirements for Light Water Reactor Spent Fuel Storage at Nuclear Power Plants". These standards are valid, e.g. in Finland, Switzerland and Spain and the Swiss and Spanish licensing authorities take a favourable view of burnup credit.

Even if in France and in the Federal Republic of Germany, until now, the criticality calculations have rested on unirradiated fuel assemblies, by all known odds burnup credit will be allowed in the future if requested due to use of higher enriched fuel assemblies or more economical storage of spent fuel assemblies. Whereas in Europe burnup credit is in its infancy it is well introduced in the United States where the first storage racks accounting for burnup were installed in 1984.

4 Rod Consolidation

Rod consolidation is an additional or alternative measure to gain interim storage capacity both at and away from reactor.

On top of that the transportation costs and frequencies can be reduced.

The canister housing the consolidated fuel rods should match in size the original assembly to be storable in existing racks of

the reactor pool. Within this limitation, consolidation of two assemblies into one canister has proven to be possible for practically all LWR's fuel assemblies. As an example, figure 10 shows the situation for two 16 x 16 PWR type assemblies.

After removing the upper and lower end fittings all the fuel rods are rearranged into a triangular lattice. As shown in figure 11, this is implemented by means of a funnel type interim container having its upper inlet openings arranged in a square lattice and its lower outlet openings arranged in a triangular lattice.

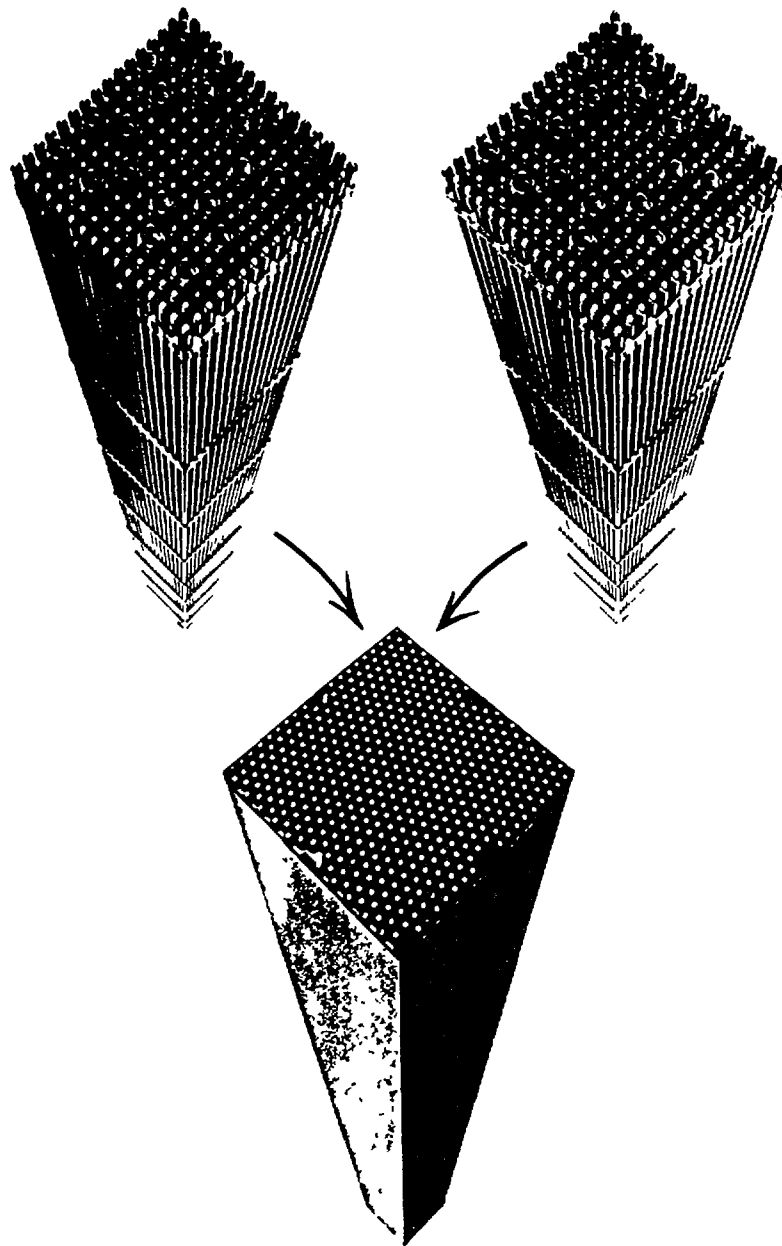


FIG 10 Rod consolidation

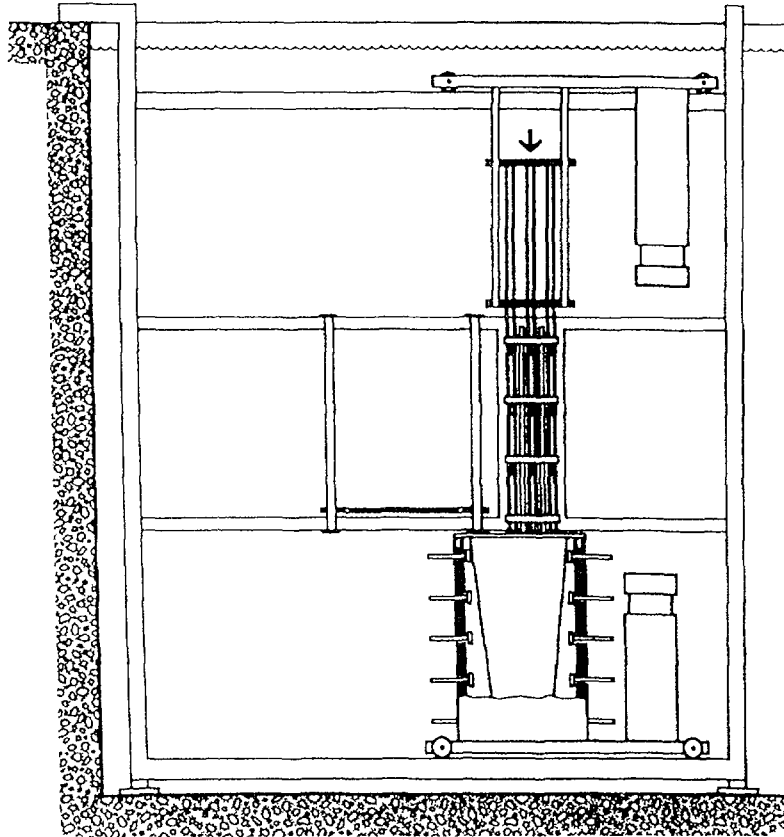


FIG 11 Rod consolidation at the reactor Pushing the fuel rods out of the assembly skeleton into the funnel

The height of that funnel corresponds to the length of the fuel rods. Every single rod is guided individually on a deflected line from its inlet to its outlet opening. This allows to insert the rods into the funnel: one by one, in groups, or all together.

Once the funnel has been filled with rods from two assemblies, the bundle is compacted in the second step, figure 12. To do so, the funnel is lifted leaving behind the rods which are pressed together successively by means of concentric clamps located at several elevations. After that operation the compacted bundle is held by the clamping mechanism in an outer shape that will fit into the consolidation canister.

In the last step, figure 13, the canister (already closed at its upper end) is lowered onto the compacted bundle while the clamps are successively removed. Finally, the lower end fitting of the canister - which had been placed below the funnel prior to its filling - is attached to the canister so that the consolidated unit is now ready for further handling.

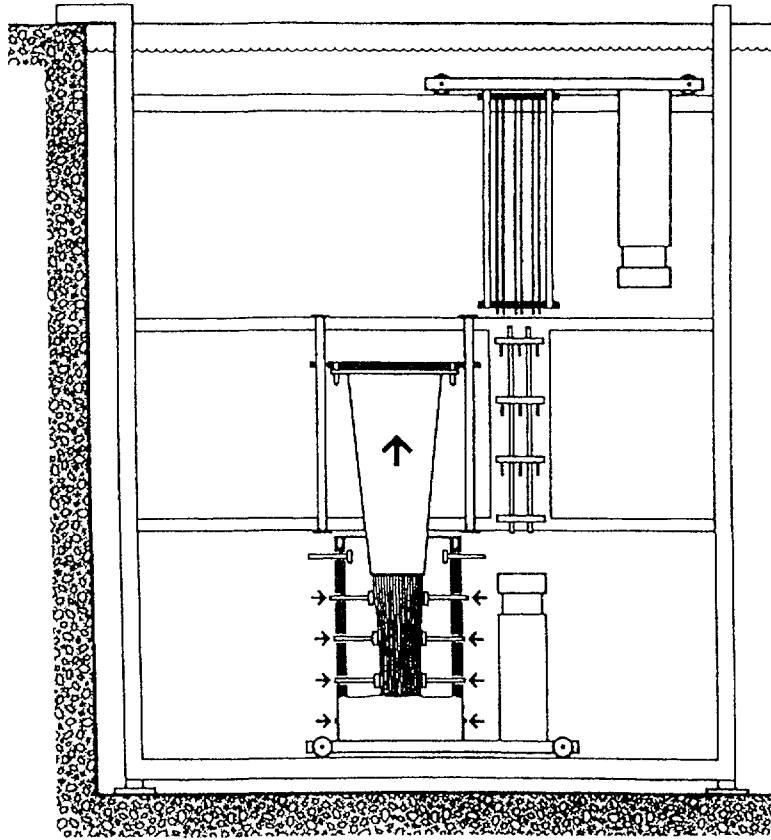


FIG 12 Rod consolidation at the reactor. Lifting the funnel and compacting the rod bundle

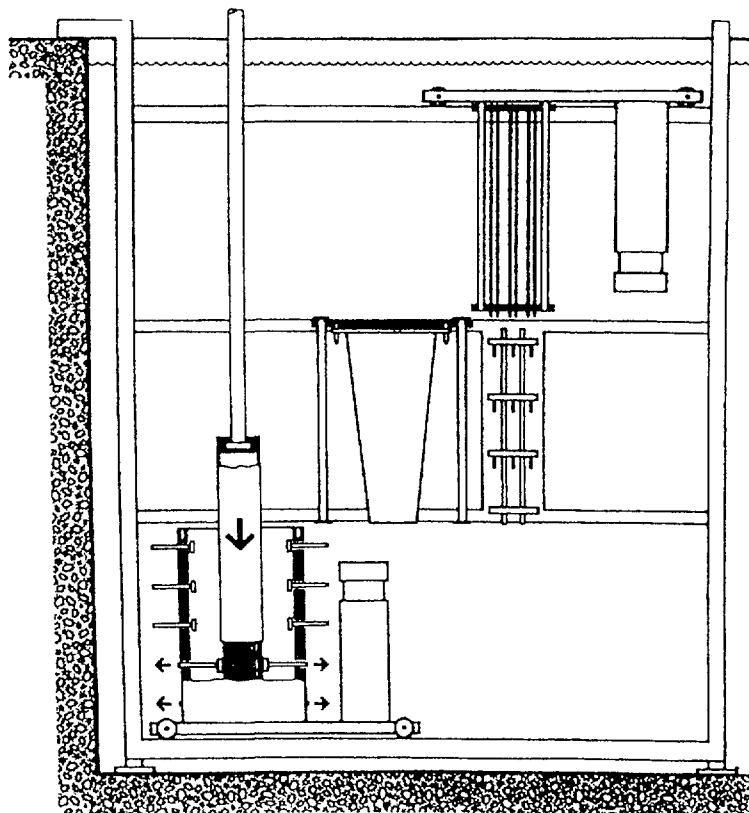


FIG 13 Rod consolidation at the reactor. Lowering the canister onto the compacted bundle

Assembly skeletons are compacted mechanically, stored and transported just like any other medium active waste.

Rod consolidation seems to be a fascinating method to enlarge the storage capacity of existing fuel pools but involves some significant disadvantages if performed in nuclear power plants such as

- time - consuming (it takes about 1 shift (10 hours) to consolidate a bundle),
- additional dose commitment of the operating personnel what may be considered to be inconsistent with the ALARA-principle, and last but not least
- the risk to destroy a fuel rod during consolidation and any difficulties arising therefrom.

5 Wet Storage Reactor and Away From Reactor

In the case that for all these efforts the reactor pool capacity has been exhausted an additional at reactor or away from reactor storage facility might be the next step in long term interim fuel storage.

An additional at reactor facility was realized by Siemens for the nuclear power plant Atucha I in Argentina, figure 14. The capacity (1.100 t U) of the second spent fuel store, which was ready for operation in 1981 is designed to last into the second half of the 90's.

The spent fuel assemblies are transported by a transport system from the reactor building through the first spent fuel pool into the second spent fuel storage facility which is provided with the same equipment as the first one, figure 15.

The spent fuel pool purification system has a mixed-bed filter for removing solid and dissolved impurities. The flow rate was dimensioned such that the entire water inventory of the spent fuel pool is recirculated once per day.

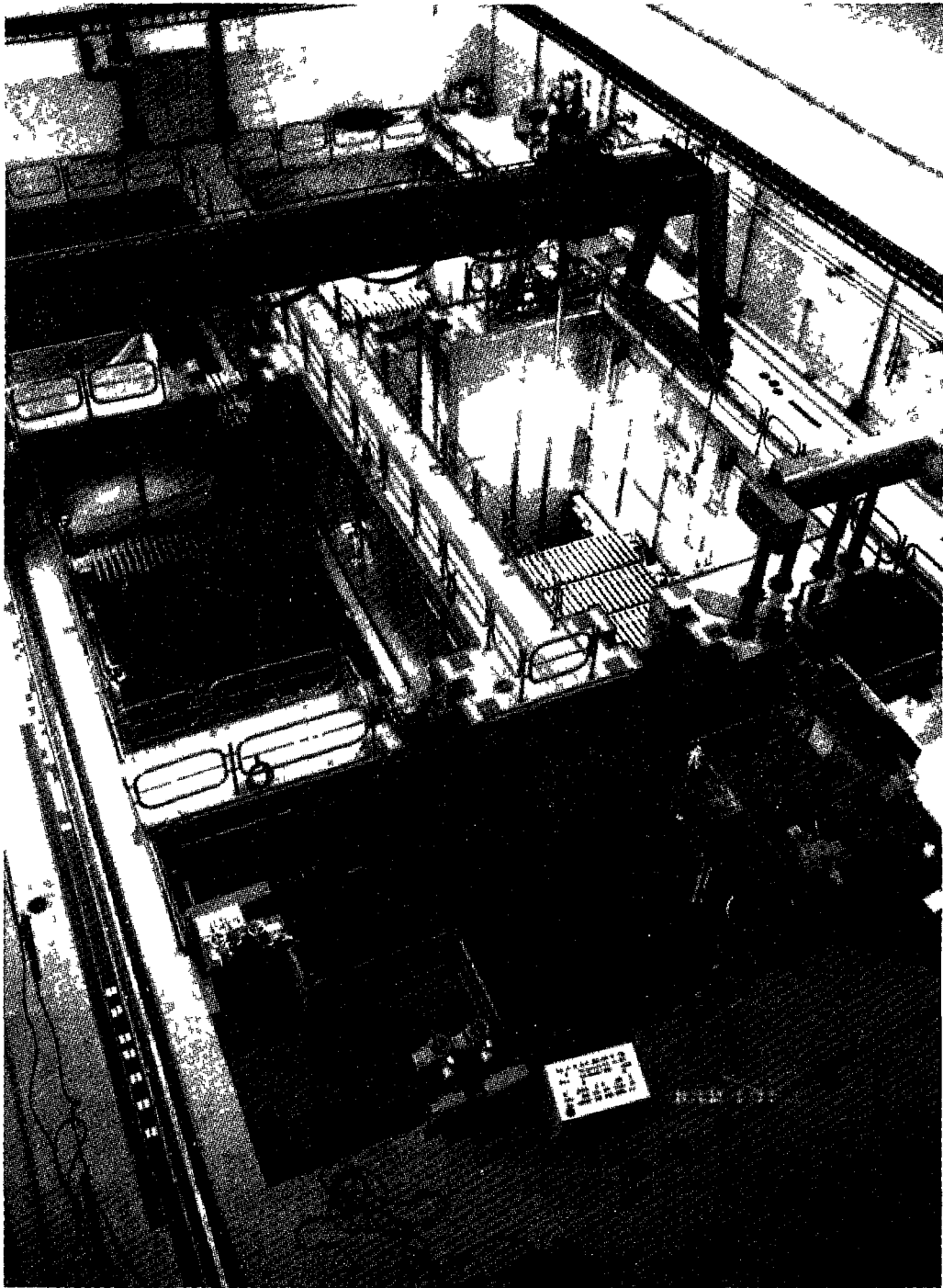


FIG 14. CN Atucha I Second spent fuel storage facility (1115 Mt U)

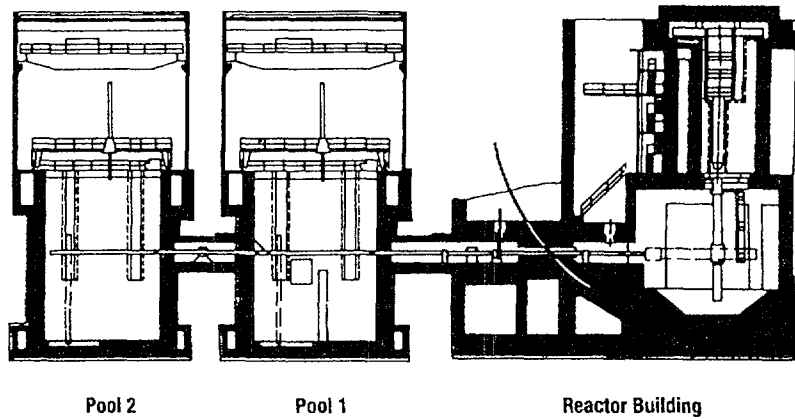


FIG.15. CN Atucha I. Wet storage facility-1 550 t U, wet storage facility-2, 1100 t U.

The ventilation system is provided with a recirculation cooling system and connected to that of the first fuel storage facility.

Supply and disposal (water, power, waste processing etc.) are effected by the corresponding facilities of the nuclear power plant that were partly extended for this purpose.

The experience gained in wet storage makes it possible to implement advanced designs as

- coolers and
- mixed bed filters incorporated in the fuel pool.

The advantage of the advanced designs is a reduction of investment costs, by avoiding the construction of an auxiliary building.

6 Dry Storage Concepts Away From Reactor

Wet storage is not the most economical solution because it requires permanent maintenance, permanent energy supply for active cooling system and permanent waste treatment over the whole storage period. R & D works started in the 70th to find alternative storage concepts, especially dry storage concepts.

A lot of different types particularly casks type, dry well type and vault type dry storage facilities have been developed, figure 16.

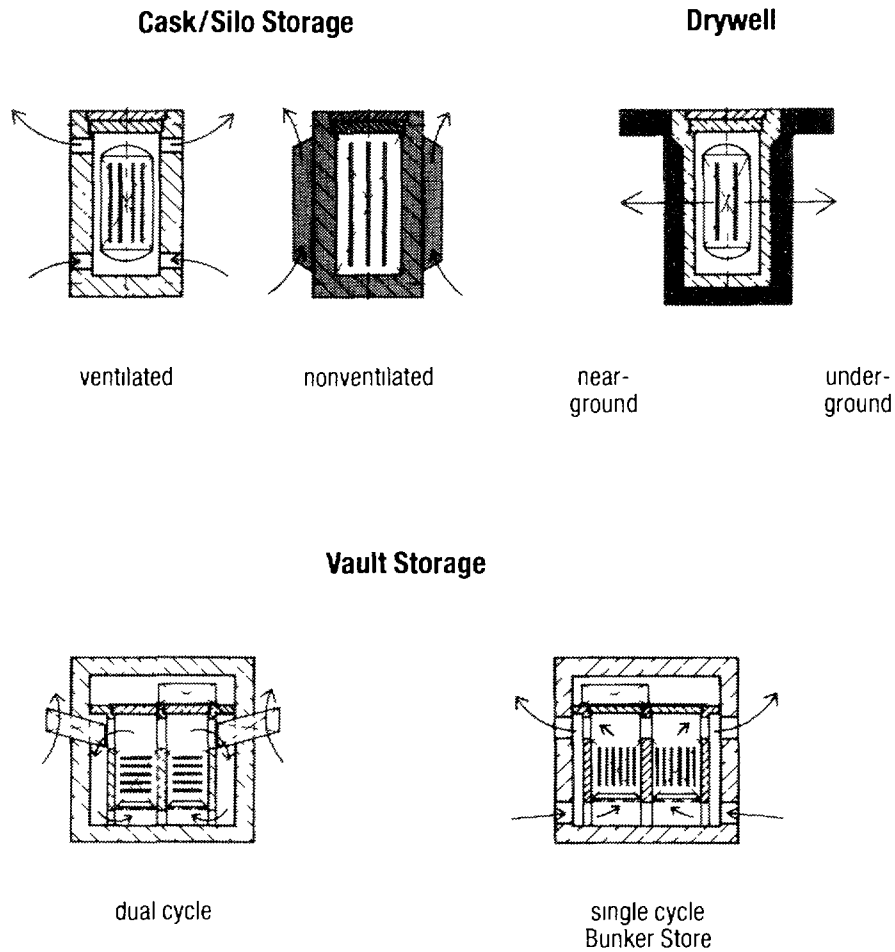


FIG.16. Basic principles of different dry storage options.

A safe and economic storage of spent fuel is possible by previous decay of spent fuel in the at-reactor pool and subsequently storage in dry storage facilities.

Cask Storage:

Combined transportation/storage metal casks are loaded with spent fuel assemblies at the reactor site and shipped to the storage facility where they can be placed on an outside storage pad or in a storage hall cooled by passive air circulation. With this type of system, the storage hall can be a simple warehouse, and most of the cost is for procuring the casks. But these can be build as needed, so the facility owners are spared the problems of a large initial capital spending for the storage structure. These casks can be stored horizontally or vertically. A cask storage facility has been built in the FRG near Gorleben and another one is under construction in Ahaus.

Silo:

Silos are cylindrical or box-shaped concrete structures that store irradiated fuel or HLW in sealed metal canisters. Two silo designs are in use: one involves heat removal by conduction through the concrete walls (non-ventilated) and the other involves natural convection passing between concrete and metal liners inside the silo (ventilated). Shielding is provided by the concrete walls and the top shield plug. The concept of using buried silos, near the earth's surface, has also been evaluated.

Dry Well:

Dry wells are cylindrical holes in the ground that are lined with concrete and/or metal. They may be placed just below the surface of the ground or in the floor of an underground rock tunnel. The fuel is stored inside sealed metal containers. Heat is removed by radiation to the liner and by conduction through the liner and the surrounding earth, while shielding is provided by shield plugs, the soil and the concrete structure. Dry well spacing depends on expected heat generation rates, thermal conductivity of the soil, criticality requirements and maximum allowable temperatures.

Vault or Bunker:

Vault storage involves a concrete bay, where the fuel or HLW is stored in sealed canisters or in baskets. Cooling may be by natural or forced air circulation, using air or an inert cover gas. In "closed-cycle" vaults, the primary coolant gas is recirculated and transmits its heat to an air heat exchanger: other operate in an once-through mode, using air cooling. Vault concepts are also being used for HLW glass packages in France, Belgium and India and are being designed for storage of defense HLW in the USA.

Siemens has designed an AR- and AFR-bunker storage facility dry vault type; figure 17 is showing the air flow and gives some preliminary temperature indications.

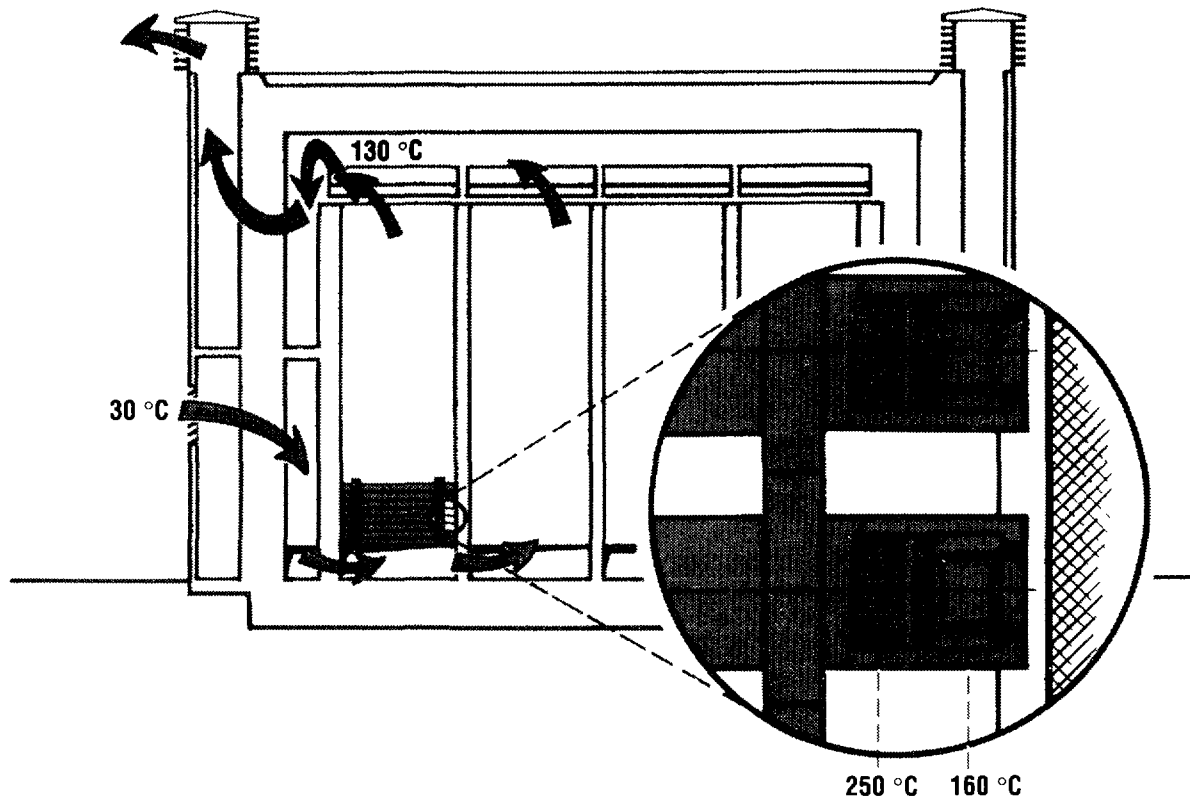


FIG.17. Dry spent fuel 'BUNKER' storage facility operating temperatures.

7 General Description of Dry Storage Facilities

7.1 Basic Description of an AFR-Storage Facility

The storage facility receives spent fuel assemblies from nuclear power plants in dry shipping casks. In the receiving area the shipping cask is lifted from the rail car or truck and placed on a carriage for subsequent mating with the unloading cell gate, figure 18. The spent fuel assemblies are unloaded, canned in steel canisters, which are sealed by welding and filled with helium in the service cell. The loaded canister is moved below the emplacement device, tilted into the horizontal position, and finally transferred to the storage location by means of the emplacement device, figure 19. The canisters are stacked horizontally above one another and provide besides the fuel rod cladding the physical barrier to prevent the release of radioactive particulate. Subcriticality is ensured by the chosen distance between the canisters, and in addition the spent fuel assemblies and canisters are always handled dry and flooding of the storage area is impossible. Shielding of the neutron and gamma radiation is ensured by the building

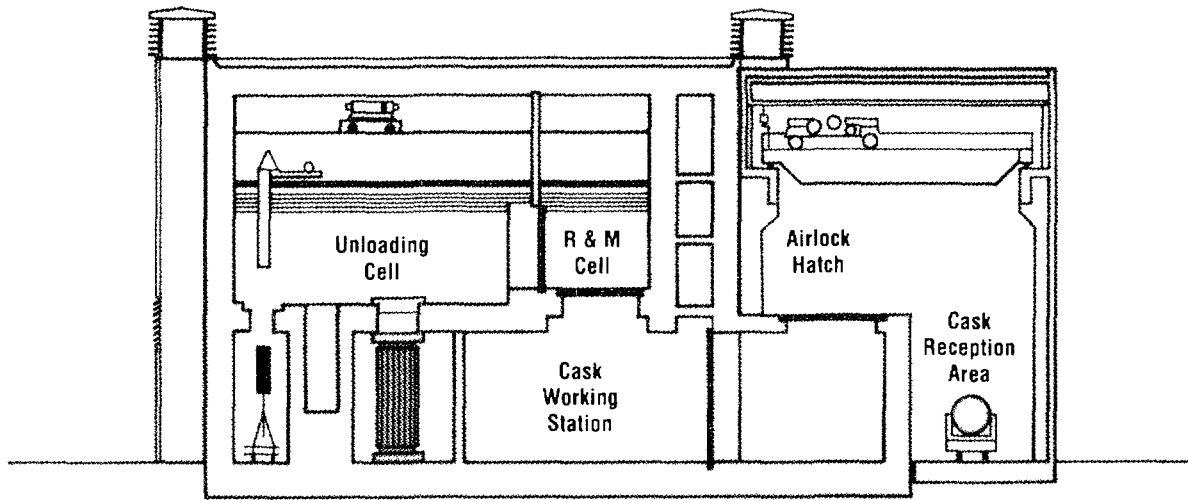


FIG.18. Independent dry spent fuel 'BUNKER' storage facility (3000 Mt U): section through cask handling area.

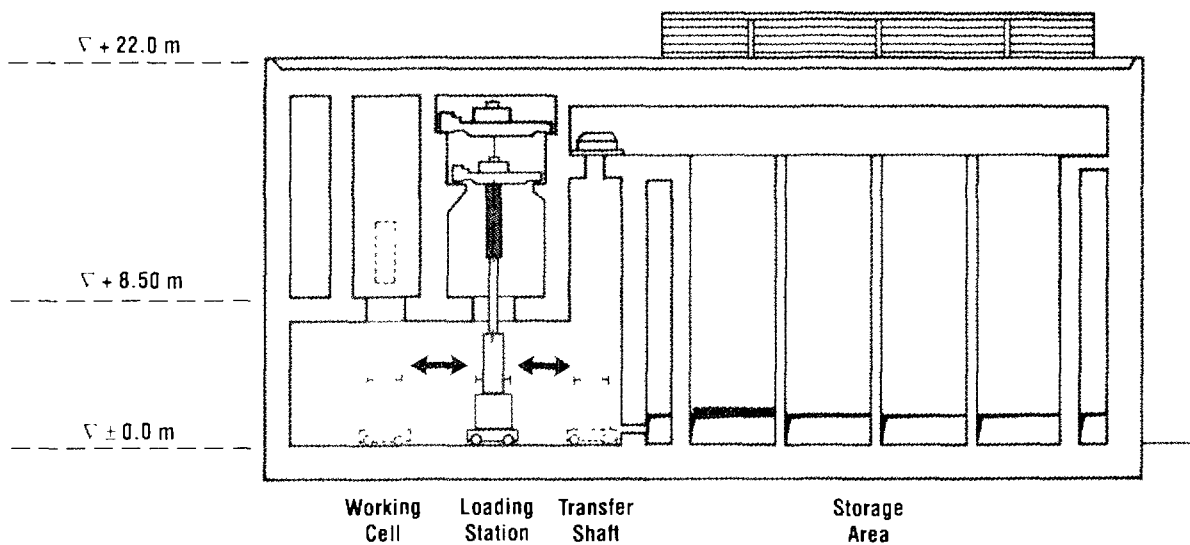


FIG.19. Independent dry spent fuel 'BUNKER' storage facility (3000 Mt U): longitudinal section.

enclosing the canisters. The heat removal system consists of a direct air cooling by natural convection with ventilation inlets and outlets spread out over the side-walls of the storage area building, and ensuring that fuel cladding and component temperatures are maintained below acceptable limits during any normal or off-normal condition. The auxiliary building (area) houses the supply and auxiliary systems.

7.1.1 Storage Facility Arrangement

Essentially, the storage facility consists of the following areas, figure 20:

- Shipping cask receiving area
- Loading area
- Storage area, and
- Auxiliary plant area

The loading and storage building is designed to withstand postulated accidents and house all systems and components important to safety and whose failure results in inadmissible release of radioactive particulate. All other systems and components are installed in the cask receiving respectively auxiliary area.

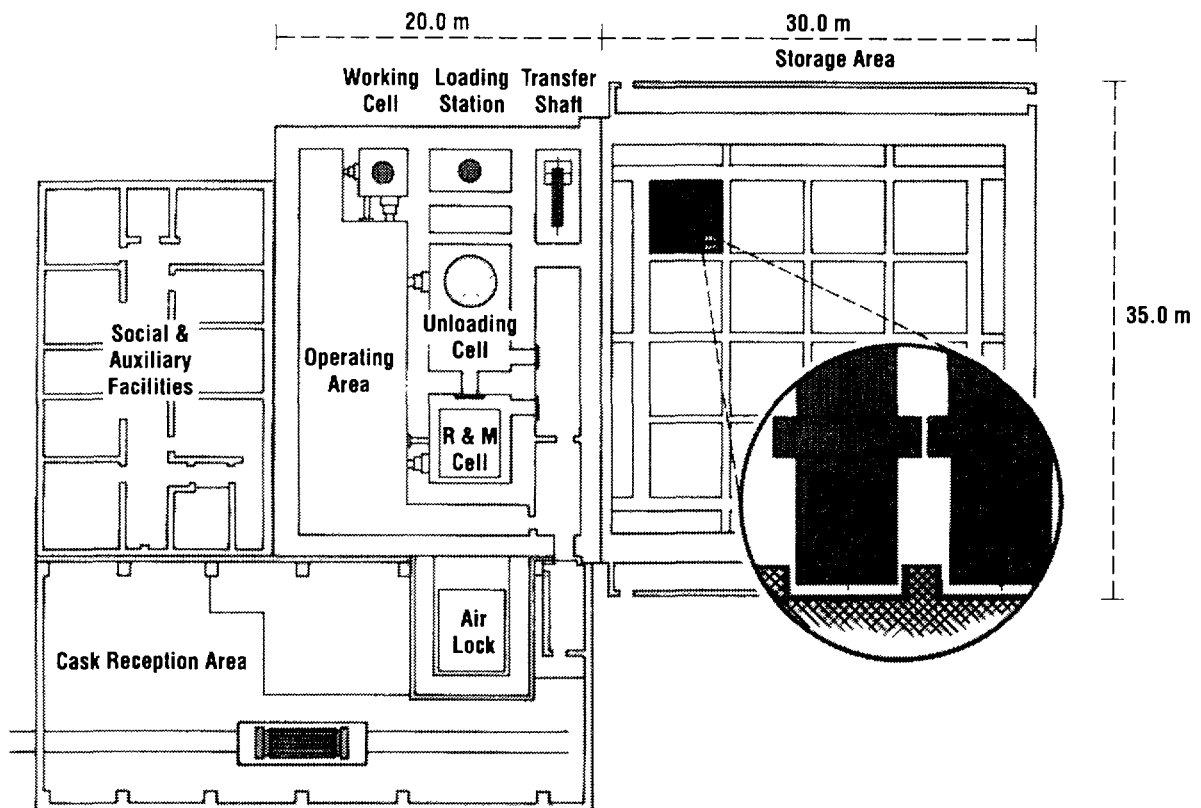


FIG.20. Independent dry spent fuel 'BUNKER' storage facility (3000 Mt U);
plan view + 8.50 m.

Shipping Cask Receiving Area

This area contains the systems and components necessary for the receipt and shipment of the transportation cask. In addition, this area provides temporary storage of casks and internal handling facilities and includes also the canister storage, mobile conditioning unit, and hot work shop.

Loading Area

Following facility components are arranged in the loading area:

- On site transportation and handling of shipping casks
- Spent fuel assembly unloading cell including lag storage, and operating floor
- Service cell and transfer aisle
- Systems and components required for spent fuel handling

Storage Area

The storage area is a reinforced concrete structure that is about 30 m long by 30 m wide. The structure is about 22 m above ground level and extends an additional 2 m below ground level. The enveloping building walls have a thickness of 2 m to guarantee the safe shielding of the expected maximum radiation as well as the necessary resistance against the assumed load cases. The side walls contain the air ventilation ducts, their cross sections are designed for safe natural convection. The internal store structure has the purpose of accomodating the fuel assembly canisters and serves as additional radiation shielding. It consists of a solid reinforced concrete structure which is only joined to the foundation slab of the storage building. U-steel profiles are imbedded and anchored in the reinforced concrete structure as guiding bars for the canisters. By means of the emplacement device the canisters are lowered and set down on the bottom supports. The load transmission is via the two spacer blocks mounted on the canister. Up to 34 canisters can be stacked on top of one another which makes it possible for 374 canisters to be stored in one quadrant. The storage area is separated from the traversing area

of the emplacement device by steel doors. When closed, the doors provide shielding from the storage area so that the traversing area can be entered and repair and maintenance work can be performed on the emplacement device.

Auxiliary Area

The auxiliary area contains the following systems, components and facilities:

- Heating, ventilation, and air conditioning
- Media supply
- Waste treatment
- Laboratory, work shop, and health physics office
- Washrooms
- Main control room
- Switchgear equipment
- Offices and recreation rooms
- Guard room

7.1.2 Cooling Concept

The decay heat of the spent fuel is removed to the environment by a passive heat removal system. Primary heat removal from the spent fuel to the sealed canister is by radiation and convection.

Secondary heat removal from the canister to the environment is by a self-regulating natural thermo-syphon buoyancy driven cooling flow using ambient air. The airflow path leads from the air inlet ducts, through the storage chamber and the canister array and out through the air outlet ducts. The air inlet and outlet ducts are provided with appropriate weather grills.

7.1.3 Shipping Cask and Fuel Assembly Handling Sequence

Once the shipping cask has arrived at the storage facility by rail or road, it is positioned under the over-head crane in the receiving area and after finishing the required receiving inspec-

tions the shock absorbers are removed. Following this, the cask is upended by means of a lift rig and either temporarily stored in a set down position or placed on a transfer carriage for internal transport through the airlock hatch to its handling position. After the required receiving inspections, the secondary lid is removed, and the cask carriage with the shipping cask is moved to its unloading position beneath the unloading cell for subsequent mating with the cell gate. By means of the cell crane the gate plug and primary cask lid are lifted off and placed beside. The fuel assemblies are individually unloaded and deposited in the lag store. The empty cask leaves the storage facility by reverse handling sequence. The fuel assembly is individually transferred from the lag store into the canister. The canister is moved in the transfer aisle beneath the service cell for subsequent mating with the service cell gate. The canister cover is welded to the canister body. The canister is evacuated and filled with inert gas and subsequent the helium valve is welded tight. The loaded canister is lowered, moved in the transfer aisle below the emplacement device, tilted into the horizontal position, and finally transferred to the respective storage position by means of the emplacement device.

7.1.4 Safety Regulations

All safety related systems and components are installed in the loading and storage building, which is designed to withstand postulated accidents. Following barriers prevent the activity release from the spent fuel to the environment:

- UO₂-matrix/fuel rod cladding
- Sealed canisters

The confinement air conditioning system in the spent fuel handling area is designed to move clean incoming air from areas of zero potential contamination to areas of low - then high potential contamination in order to maintain clean air in as great an area as possible in the event of leak of contamination. The waste air is discharged through filters.

Safety related active components for cooling of spent fuel do not exist, so that reliable decay heat removal in the storage area is ensured by natural convection even in the event of power outage.

The loading cell does not contain any system essential to safety because supposing power outage no inadmissible temperatures of the fuel cladding occur over a very long period of time, and the loading cell as a component remains intact as a whole, leaving more than three days time for taking measures to resume cooling. In the shipping cask receiving area, the decay heat of the spent fuel assemblies is removed via the shipping cask. The cask is designed in such a way that no inadmissible cladding temperatures occur. Thus, there are no engineered safety feature systems for fuel cooling provided in the entire vault store. The dry vault storage facility does not contain any other safety-related systems.

Standby power and direct-current system take over functions in the field of

- Occupational safety (emergency lighting, low-pressure maintenance)
- Fire protection (fire extinguishing system)
- Site security
- Activity monitoring (instrumentation)

Standby and DC system are built-up line-to-line owing to availability reasons. Temporary power failures here do not have any safety-related significance.

The ventilation system is designed in accordance with general specifications and regulations. Failures in the low-pressure system do not cause any inadmissible contamination.

The facilities for handling the shipping cask are designed in accordance with general specifications and guidelines.

The shipping cask itself is designed to withstand cask drop impact loading. The facilities for handling the spent fuel assemblies in the loading cell are also designed in accordance with general specifications and guidelines. Dropping of a fuel assembly or of the handling equipment does not result in any inadmissible release of activity.

The lag store for spent fuel assemblies in the unloading cell is designed to withstand dynamic loads such as those due to earthquakes.

The transport and tilting device in the transfer aisle is designed in accordance with general rules and guidelines. Dropping of a canister does not result in any inadmissible release of activity into the environment since, among other things, the lifting height is very low.

The emplacement device is designed in accordance with increased requirements, in order to prevent the canister from being dropped in the storage area since this would lead to internal restrictions.

7.2 Basic Description of an AR-Storage Facility

The storage facility receives spent fuel assemblies from the on site nuclear power plant in special dry shipping casks. The cask has a turning lock in the top section and a special fitment for unloading in the base. The shipping cask is transported by means of a trailer into the receiving area for subsequent mating with the unloading cell gate. The spent fuel assemblies are unloaded, canned in steel canisters, which are sealed by welding and filled with helium. The loaded canister is finally transferred to the storage location by means of the emplacement device. The storing system and heatremoval system is the same as in the AFR-storage facility and also subcriticality and radiation shielding are ensured in the same way as in the AFR-storage facility.

7.2.1 Storage Facility Arrangement

Essentially, the storage facility consists also of the following areas, figure 21:

- Shipping cask receiving area
- Loading area
- Storage area, and
- Auxiliary plant area

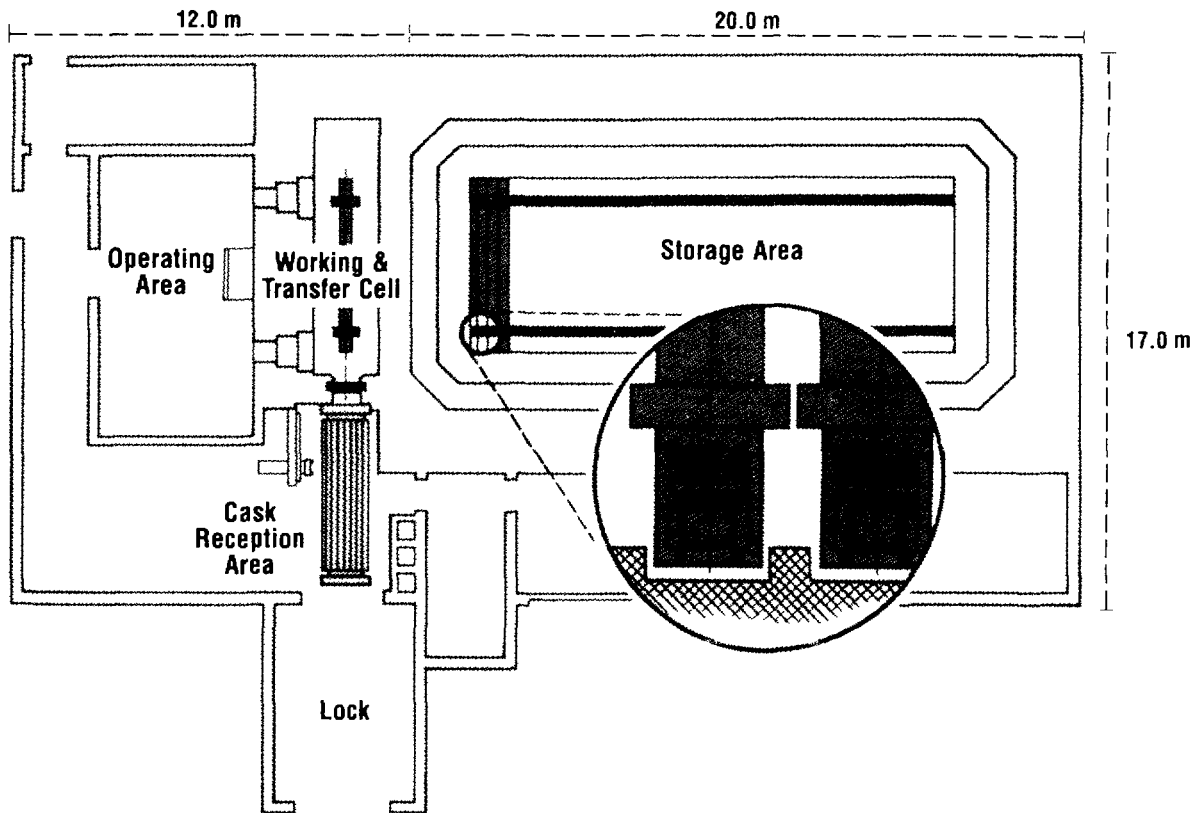


FIG.21. On site dry spent fuel 'BUNKER' storage facility (500 Mt U): plan view.

The main differences to an AFR-storage facility are:

- The smaller storage area adapted to the amount of spent fuel to be stored. In the showed figure 21 the storage area is about 20 m long and 17 m wide resulting in a storage capacity of 500 MTU.
- The simple cask reception and loading area due to the fact that the spent fuel assemblies will be transported one by one from the nuclear power plant to the storage facility by means of a special cask and
- the reduced auxiliary plant area as facilities of the onsite nuclear power plant may be used.

7.2.2 Shipping Cask and Fuel Assembly Handling Sequence

Once the shipping cask has arrived at the storage facility on a trailer, it is transported through the airlock to its handling position.

After the required receiving inspections, the shock absorbers and the lid are removed, and the cask trailer with the shipping cask is moved to its unloading position next to the unloading cell for subsequent mating with the cell gate. The steel-lead gate slide of the unloading cell and the turning lock of the cask head are opened and by means of the special fitment in the base of the cask the fuel assembly is moved into the canister placed in the unloading cell. After unloading, the gate slide and the turning lock are closed and the empty cask leaves the storage facility by reverse handling sequence. The processing equipment is swung in front of the gate slide and subsequent the gate slide itself is opened again. The canister cover is welded to the canister body. The canister is evacuated and filled with inert gas and subsequent the helium valve is welded tight. The loaded canister is moved below the emplacement device, coupled to the girder, and finally transferred to the respective storage position by means of the emplacement device.

7.2.3 Safety Regulations

The safety objectives regarding

- containment of radioactive materials
 - criticality
 - decay heat removal
 - shielding
 - limitation of the release of radioactive substances and of the exposure of the personnel to radiation, and
 - external events
- are met and are the same for AR- and AFR-storage.

8 Conclusion

So far, I could show you the concepts and strategies for increasing the storage capacity at reactor or away from the reactor presently being in use or under advanced design studies. Major emphasis has been put on research and development activities related to an important item namely the long term storage behaviour of the fuel rods.

It follows from the discussion of mechanisms posing a potential threat to fuel assembly integrity that no common-cause changes will occur in spent fuel assemblies with zircaloy cladding tubes at wet storage temperatures of around 40 °C. There is no difference in degree between storage periods of 30, 50 or 100 years. This statement is supported by experience to date with spent fuel assemblies in wet storage. 15 years of experience with wet storage of Zircaloy 4 fuel assemblies has now been attained (H. B. Robinson, 18 GWd/tU burnup). As part of numerous experimental programmes, interim inspections of selected intact and defective fuel rods in wet storage have been carried out, for example in the spent fuel pool at NPP Obrigheim, where 10 defective and 18 intact fuel rods with up to 39 GWd/tU burnup were observed for changes over a period of up to eight years. No evidence of any change was found in either the intact rods or the defective rods. For this reason, further programmes to ensure the safety of wet storage for periods up to 100 years are considered unnecessary.

In the case of dry storage, creep strain can become a threat to cladding tube integrity. In order to eliminate this possibility, an upper limit for the initial storage temperature must be set. The German interim fuel assembly storage facilities at Gorleben and Ahaus have been approved for a maximum initial storage temperature of 410 °C. It was demonstrated that creep strain comes to a halt after a few months. It is therefore of no consequence how long the period of dry storage lasts, whether 40 years or 100.

If the storage medium is assumed to be an oxidizing gas rather than an inert one, another discussion of the medium's effects on cladding corrosion becomes necessary. The temperature of the fuel rod during dry storage must be reconsidered for this purpose. Information provided in a safety analysis for a German storage facility for fuel transport casks show that after about three years, when the temperature falls below 250 °C, oxidation comes to a virtual halt: the annual increase in the oxide layer thickness at 250 °C is less than 0.1 µm. The length of the period of dry storage is thus of no importance as regards cladding corrosion either.

In another case, however, the duration of dry storage does have some effect on fuel assembly behaviour. Where, in some storage concepts, the ingress of air into defective fuel rods cannot be ruled

out, the oxidation of fuel necessitates a considerable reduction in the permissible initial storage temperature. Following the long decay time necessary for the fuel rods to reach such a temperature before storage, the further reduction in temperature which occurs after storage begins is relatively small. Since the corrosion process does not cease soon after storage begins, separate investigations of UO_2 oxidation for storage periods of 40 and 100 years seems to be necessary.

Summary

Contribution of Siemens to Spent Fuel Management Techniques

The federal government in the Federal Republic of Germany is responsible for the erection and operation of final disposal repositories whereas the industry is responsible for all other facilities such as

- interim storage facilities for spent fuel assemblies at reactor (AR) or away from reactor (AFR)
- reprocessing plants for spent fuel assemblies including recycling of the fissile material

The requirement for sufficient storage capacity can be fulfilled by different strategies:

- Compact storage racks and away from reactor
- Rod consolidation
- Wet storage at reactor and away from reactor
- Dry storage concepts.

For technical and economic reasons it has become an advantage to use the fuel pools in existing and planned nuclear power plants for extended interim storage of spent fuel assemblies.

Siemens has therefore developed storage racks in which the spent fuel assemblies can be more densely packed than was previously possible. The required neutron absorption is no longer provided only by the pool water and the material of the rack structure but also by absorber channels manufactured from boron steel.

Rod consolidation is an additional or alternative measure to gain interim storage capacity both at and away from reactor. The canister housing the consolidated fuel rods should match in size the original assembly to be storable in existing racks of the reactor pool. Within this limitation, consolidation of two assemblies into one canister has proven to be possible. A hot demonstration program will be performed by Siemens in the NPP Unterweser at the end of the year 1989.

In the case that for all these efforts the reactor pool capacity has been exhausted an additional at reactor or away from reactor dry or wet storage facility might be the next step in long term interim fuel storage.

An additional at reactor wet storage facility was realized by Siemens for the nuclear power plant Atucha 1 in Argentina.

Wet storage is not the most economical solution because it requires permanent maintenance, permanent energy supply for active cooling system and permanent waste treatment over the whole storage period. Therefore alternative storage concepts were investigated, especially dry storage concepts.

The at-reactor and away from reactor dry storage facility developed by Siemens receives spent fuel assemblies in dry shipping casks, which are transported by means of a trailer into the receiving area for subsequent mating with the unloading cell gate. The spent fuel assemblies are unloaded, canned in steel canisters and transferred to the storage location. The canisters are stacked horizontal one above another and provide the physical barrier to prevent the release of radioactive particulate. Sub-criticality is ensured by the chosen pitch and shielding of the radiation by the building. The heat removal system consists of direct air-cooling by natural convection.

RECENT ADVANCES IN SPENT FUEL CONSOLIDATION CONCEPTS

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Abstract

In France, at the present time, most of the nuclear spent fuel is stored under water, first in the reactor pools, then in the interim storage pools at La Hague, prior to reprocessing.

However, some fuel elements are presently stored in dry conditions :

- gas cooled/heavy water reactors
- fuel elements at the CASCAD facility at Cadarache
- breeder reactor fuel elements at Marcoule.

Among the various technologies available to increase the storage capacity of the water pools, fuel consolidation appears to be one of the most promising.

Different consolidation equipment have been developed in various countries. This paper presents two systems developed in France by SGN :

- the first one relates to a fixed facility with a large capacity dedicated to an interim storage,
- the second one a mobile compact unit which can be used in reactor pools. This system was developed with BABCOCK and WILCOX (Lynchburg, Virginia).

1 - DRY ROD CONSOLIDATION

An equipment was designed, manufactured and tested under inactive conditions in France.

It can consolidate 17 x 17 PWR fuel elements on dry conditions.

The annual capacity of this equipment is equal to 650 tU assuming two shifts per day.

The sequence of the main operations is as follows :

- (1) - The fuel element is fixed in a horizontal position,
- (2) - The central instrumentation tube plug is drilled out.
- (3) - The guide instrument tubes at the top of the element are cut from the inside out with a multiple blade cutting head and the top-end scrap hardware is removed to a container for Non Fuel Bearing Components (NFBC). The upper grid remains attached to the upper end-fitting which is removed with the portions of cut tubes : this provides greater access to the fuel rods.
- (4) - The fuel element is then translated in front of the rod consolidation equipment.
- (5) - Each fuel rod is gripped by means of a gripping jaw that is part of a gripping head and all the fuel rods are pulled simultaneously by retraction of the gripping head.
- (6) - While the fuel element is clamped, the gripping head is retracted and vertical and horizontal combs are placed throughout the fuel bundle in order to keep the fuel rods in an array identical to the array they constituted in the fuel element.
- (7) - Once rod removal have been completed, the horizontal combs are retracted step by step while the vertical combs remain stationary. At each step, the rods from a vertical row drop down into a tray beneath the fuel rod array. Then the rods which fell on the tray are pushed on the top of the previously reconfigured rods.
- (8) - When a complete new horizontal row of rods has been placed this set is lowered by one diameter rod pitch and a new row is formed above it.

(9) - When a complete set of rods has been consolidated, this set is pushed into a canister.

The multi-blade cutting head is driven by an electrical motor while the rod consolidation equipment itself is driven by an oil hydraulic system.

2 - IN-POOL ROD CONSOLIDATION

A very compact equipment has been designed, manufactured and tested by SGN and BABCOK and WILCOX to consolidate spent fuel elements inside existing storage pools. The floor space required to install and operate this equipment inside the pool is only 8 ft x 8 ft and only a few days will be required for equipment set-ups and equipment removal.

This equipment will not only consolidate fuel rods with a 2 : 1 ratio but will also reduce the volume of the Non Fuel Bearing Components (NFBC'S), with a compaction, ratio of 10 : 1. Consequently, the consolidation of spent fuel present in a pool will provide 40 % of free space.

The equipment will consolidate two fuel elements per shift.

The equipment consists of two main process units installed on a lightweight steel frame :

- The rod consolidation unit,
- The NFBC disposal unit

Associated with these are the requisite support subsystems such as cameras, lighting, specialized hand tools, and the filtration system.

a) Rod consolidation unit

Two fuel assemblies are placed in two modules. The upper end-fittings are removed by either precision drilling or use of an internal tube cutter.

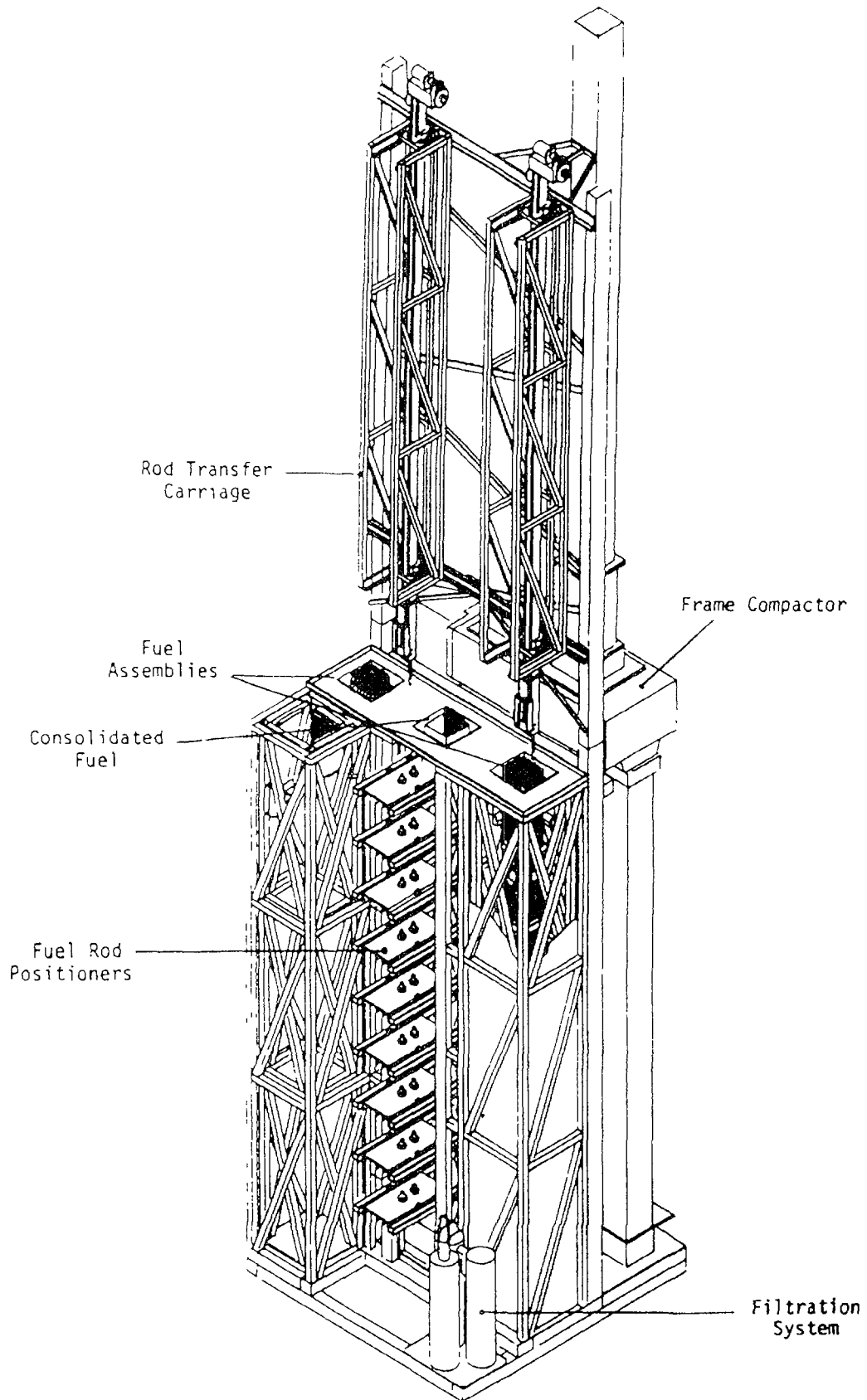


FIG 1. Rod consolidation unit.

By modifying the upper end-fittings, 6 out of 10 can be reused as caps for the consolidated fuel canisters, as well as for the NFBC canisters.

Fuel rod transfer is accomplished by two single rod pullers which are coupled with load cells to accommodate varying lengths within each assembly and sense excessive pull forces. The pullers are mounted on two carriages which move along the support structure.

As a rod is pulled from one assembly, a rod is inserted in the consolidation canister. This a rod is packed each half cycle of the individual puller. Transfer is fully automated and takes less than four hours to complete.

The fuel rods are repacked into a tight lattice directly in the consolidated fuel canister. Several rod positioners operate through windows in the canister wall.

Direct canister loading is much simpler than intermediate transfer devices such as funnels with the associated jamming risks.

At the end of the rod consolidation sequence, a cap is placed on the canister. The cap consists of two pieces : an adapter piece directly interfaces with the canister and the reconditioned upper end-fitting is then attached.

Then, the consolidated fuel canister is transferred to a storage location within the pool, and the two skeletons are moved to the NFBC disposal unit with an under water boom.

b) NFBC disposal unit

This unit includes five systems :

- a feeding stack with a pusher device,
- a clamp at the bottom of the feeding stack,
- the shear itself,

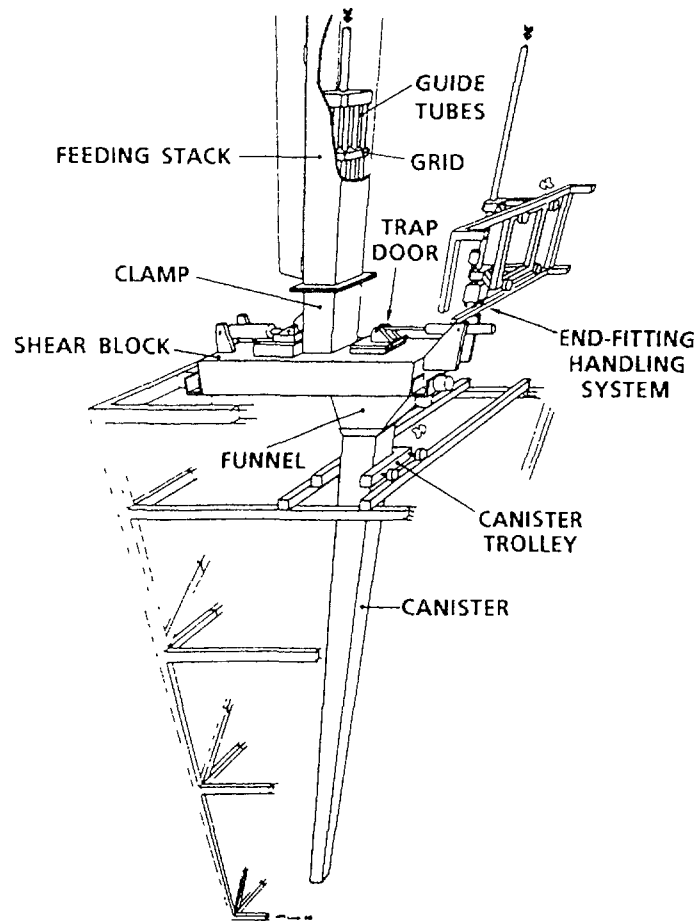


FIG.2. NFBC disposal unit.

- the canister trolley,
- the end-fitting handling system

The sequence of operations is as follows :

- (1) - a canister is placed on the trolley under the shear block
- (2) - a fuel assembly skeleton is introduced into the feeding stack
- (3) - the lower end-fitting is sheared
- (4) - the lower end-fitting is introduced into the canister with the handling system

- (5) - an upper end-fitting previously cut with the rod consolidation unit (see § 2.a) can also be introduced into the canister.
- (6) - The guide tubes and the grids are sheared into pieces with fall directly into the canister. About 150 shearing cycles are required for one fuel assembly skeleton.
- (7) - After several skeletons have been sheared, the canister is moved along the trolley and a cap is placed on the top of the canister. As for the consolidated fuel canister, this cap is made of two pieces : an adapter piece and a reconditioned upper end-fitting.

The snear and the clamp are operated with an hydraulic system using demineralized water. The pressure can reach 200 bars but 40 bars are sufficient for the shear and less than 100 bars for the clamp.

The NFBC disposal unit was designed and manufactured in France. Tests are being successfully performed. Several fuel assembly skeletons have been already sheared. The equipment will be shipped to Lynchburg, Virginia to be installed on the frame supporting the rod consolidation unit developed by BABCOK and WILCOX. Inactive tests of the two units will take place next fall. Then, an active test will be performed in a reactor pool.

PLANS FOR EXPANDING THE CAPACITY OF THE SWEDISH AWAY FROM REACTOR STORAGE FACILITY, CLAB

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Abstract

The Swedish Central Interim Storage Facility for Spent Nuclear Fuel, CLAB, is intended for storage of all the spent nuclear fuel from the Swedish nuclear programme. CLAB will be expanded in steps. In the first phase CLAB was built for 3000 tU. This capacity will cover the needs until 1996. Methods for expanding the capacity are now being studied.

In CLAB the fuel is stored in waterfilled pools in a rock cavern below grade.

Already during the construction of the first phase preparations for building new rock caverns parallel to the existing one was made.

As an alternative to building a new rock cavern with storage pools the possibility to extend the capacity of the existing pools are explored. Both introduction of neutron absorbers and taking credit for burnup are considered. Burnup credit will increase the flexibility with regard to acceptable initial enrichment, but will require close control of the fuel at reception.

A decision on the expansion of CLAB will be taken at the end of this year.

INTRODUCTION

Sweden has 12 nuclear reactors in operation, 9 BWRs and 3 PWRs, with a total electrical output of 9 900 MW and an annual production of about 66 TWh.

Each year about 250 tonnes of spent fuel is generated in the 12 reactors and must be disposed of. According to the Swedish policy all the fuel will be disposed of without reprocessing. Before final disposal the fuel will be stored in CLAB, the Central Interim Storage Facility, for about 40 years.

CLAB, which is located on the Simpevarp peninsula close to the Oskarshamn nuclear power station, was commissioned in 1985, and is licensed for totally 3000 tonnes (as uranium, tU) of spent fuel. In CLAB also some core components will be stored. At present about 1000 tU has been received for storage in CLAB.

The total Swedish nuclear programme will generate about 7800 tU of spent fuel up to the year 2010, including approximately 33 000 BWR fuel elements and 4 000 PWR fuel elements.

With the present situation CLAB will be full in 1996 and work has consequently started to prepare for an extension of the capacity. In this paper the plans for extension will be discussed.

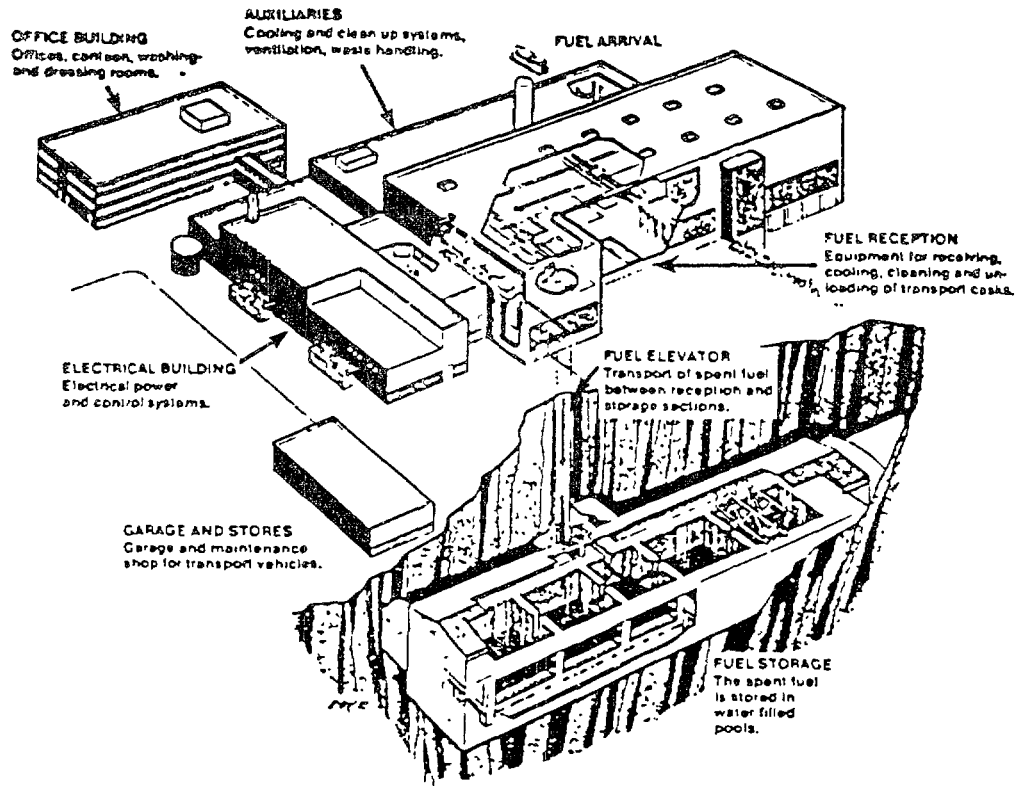


Figure 1: Cutaway view of CLAB

DESCRIPTION OF CLAB

In CLAB the spent fuel is stored in waterfilled pools. The facility consists of underground storage pools in a rock cavern and a receiving building and other interconnected buildings on the surface. In the receiving building the incoming spent fuel and core components are handled. Directly connected with the receiving building are buildings for auxiliary systems (cooling and purification of water, waste handling, ventilation etc), for service systems and for the electric power systems. The general features of the facility are shown in Figure 1 (Ref. 1).

The receiving building has three receiving pools, two of which are specially equipped for receiving the Swedish standard BWR and PWR fuel and core component cask TN 17/2. The third receiving pool can also be used to receive casks other than TN 17/2 although additional equipment specific to the actual cask must be provided.

This pool also accommodates the fuel leakage detection equipment. In other pools in the receiving section, empty fuel canisters can be stored as well as filled canisters before they are transferred to storage. The total receiving capacity is 300 tons per year or about 100 spent fuel shipping casks.

The spent fuel is transported from the receiving section to the storage section in a fuel elevator.

The actual storage section is located underground in a rock cavern, whose roof is located 25-30 metres below the surface. The rock cavern is 120 metres long, 21 metres wide and 27 metres high. It contains four storage pools and one smaller, central pool connected to a transport channel. Each storage pool contains about 3 000 m³ of water and can hold 750 tons of spent nuclear fuel with the storage method used at present. The storage pools now built can thus hold 3 000 tons of fuel, which will cover Swedish needs up to 1996.

The fuel elevator shaft is connected to the pools via a channel. The storage section is also connected to the surface buildings through a shaft containing a personnel elevator, ventilation ducts, electricity and water supply.

Present storage method

The spent fuel elements are stored in transportable canisters, containing 16 BWR or 5 PWR fuel elements. The loading of the canisters is made in the reception building and the canisters are then used as transport module for internal transports and for storage.

In the storage pools the canisters are stored in one layer. About three meters above the pool bottom the a grid network is installed. The purpose of the grid is to support the canisters and to guarantee a certain distance between the canisters also during a seismic event.

The canisters are made of stainless steel and have been designed to accommodate fresh fuel without risk of criticality. This is achieved for BWR fuel up to 3,6 w/o enrichment and for PWR fuel up to 3,75 w/o.

METHODS FOR EXPANDING THE STORAGE CAPACITY

A second rock cavern

Already during the construction of CLAB preparations were made for a future expansion of the facility with new storage caverns, parallel to the existing rock cavern. In total a maximum of three rock caverns can be accommodated, with up to 9000 tU capacity.

The straightforward way to expand the capacity is thus to build a new rock cavern with storage pools (See figure 2). For the Swedish programme one rock cavern with

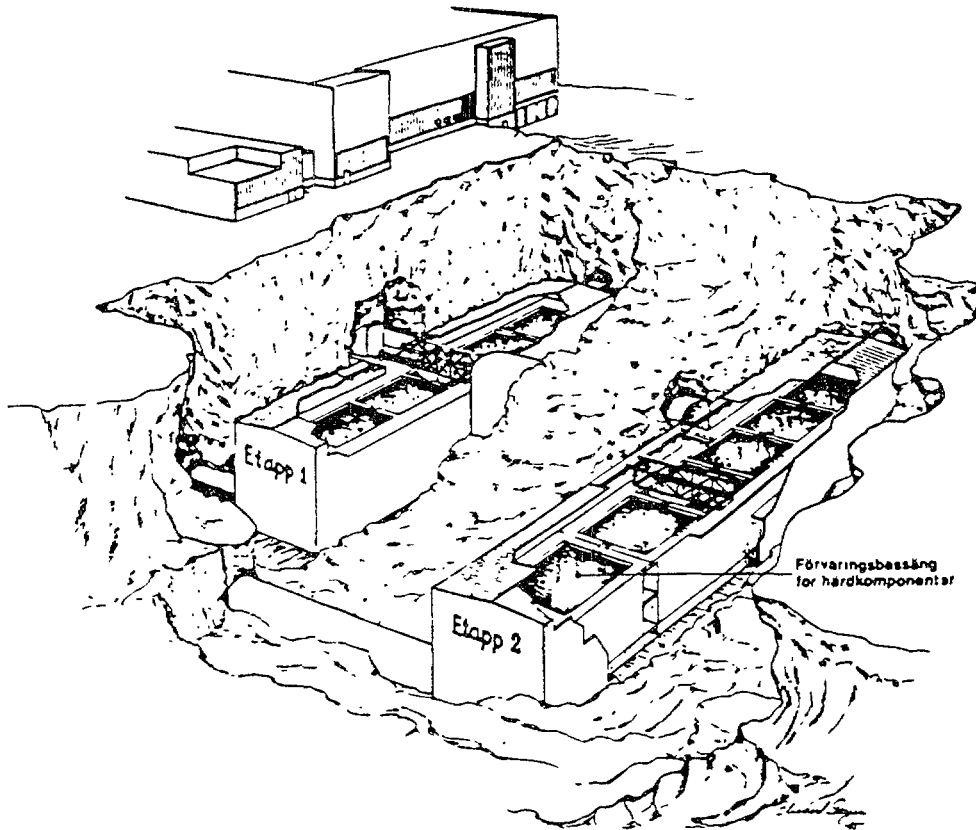


Figure 2: Expansion of CLAB by a new rock cavern

7 pools would be needed. The construction time for this has been estimated to 6 years.

Expanding the capacity of the existing pools

As an alternative to building a new rock cavern possible methods of expanding the capacity in the existing pools have been investigated. These include:

- more efficient use of the canisters
- rod consolidation
- double tier storage
- new canister dimensions
- installation of fixed storage racks

The last three of the alternatives were abandoned quickly as they would involve substantial changes in the facility and in the philosophy adopted for operation of the facility.

Rod consolidation

Techniques for rod consolidation are being developed by many manufacturers around the world and tests have been made in a few reactors in the US. The experience of these tests were collected and translated into the situation at CLAB. From this we concluded:

- rod consolidation has been demonstrated for PWR-fuel, but there will still be a few years until experience from routine operation will be available.
- For BWR-fuel the corresponding time is still further away.
- A compaction of 2:1 could only be achieved for PWR if the NFBC are not taken into account.
- For BWR-fuel the volume reduction achieved will be less because of the channels.
- The rod consolidation operation will increase the doses to the personnel.
- In order to achieve a compaction rate of 2 PWR assemblies per shift further development is needed.
- Space is available in CLAB for rod consolidation equipment.

From this it was decided not to pursue the rod consolidation route at present but to keep the option open for the future, at least for PWR-fuel.

More efficient use of existing canisters

The existing canisters have not been optimized for the number of fuel elements per canister, but for criticality reasons.

Geometrically it is possible to within the measures of the existing canisters accommodate 25 BWR fuel elements or 9 PWR fuel elements, that is an increase of 56 % or 80 % respectively for BWR or PWR (See fig. 3).

A closer packing of the fuel elements changes the conditions for the criticality analysis. To guarantee that all handling and storage of the fuel will remain with a safe margin to criticality two different methods could be applied:

- The canisters are equipped with neutron absorbers, eg borated steel or
- credit is taken for the fact that all fuel elements to be stored in CLAB have a certain minimum burnup, and are thus less reactive than fresh fuel.

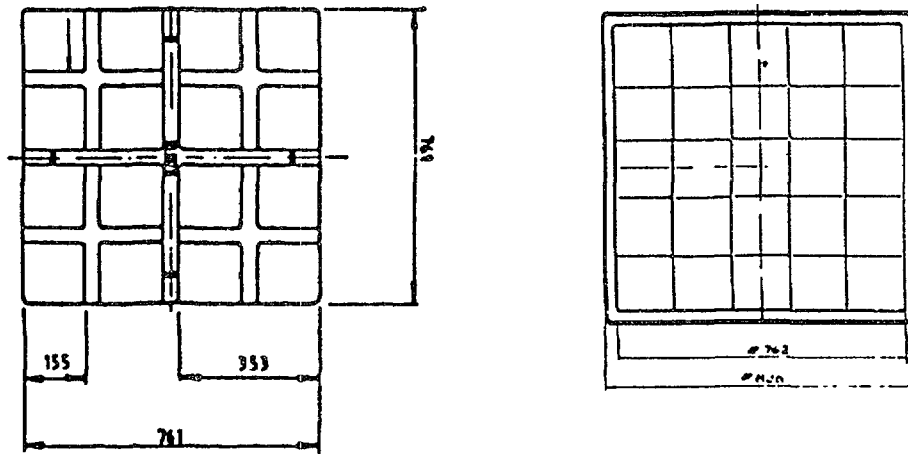


Figure 3: Cross section of existing (left) and new canister (right).

The first method has the advantage that the control of the fuel elements at reception can be alleviated as fresh fuel elements could be stored. It has however the drawback that the maximum allowable enrichment will be limited. According to our calculations the limit will be about 4,0 w/o for BWR and 4,2 w/o for PWR. This covers the present needs but gives no margins for further development.

In the case of burnup credit, which is approved in a number of US PWRs, an area of acceptance has to be established, which for different enrichments indicates the minimum burnup necessary. Before fuel will be accepted in CLAB a verification of the burnup of the individual fuel elements will be required, either by administrative means or by measurements.

Burnup credit has the advantage that the canisters could be made cheaper and that no enrichment limit will be needed, as the reactivity worth of the fully used fuel elements will be similar irrespective of initial enrichment.

A more efficient use of the canisters could raise the capacity of CLAB up to almost 5000 tU. A second rock cavern would still be needed, but about six years later.

In addition to criticality question also other impacts on the facility are being studied, such as the increased weight of the full canisters, the increased thermal load and other safety related factors. The preliminary result of these studies are that only minor changes will be needed in the existing systems.

DISCUSSION ABOUT BURNUP CREDIT

Background

With the tendency to go to higher enrichment levels in the nuclear fuel the efficiency of the fuel storage ponds are getting less, if all racks have to be designed to accommodate fresh fuel. This was first recognised in the US where some regions of the fuel ponds were licensed with burnup credit already in 1980, in Callaway 1 and Wolf Creek. Until now fifteen licensees have given for storage pools taking account of the burnup. So far only PWRs have been licensed as the boron content of the water is considered under some accident conditions. Credit for burnup is addressed in the draft Revision 2 of Regulatory Guide 1.13 (Ref. 2), which was published in 1981, and which has become the standard for the design basis of spent fuel storage facilities.

In the US a lot of work is continuously put into burnup credit consideration, eg for storage and transport casks. The US DOE is sponsoring a specific program on this issue.

In Europe so far no licenses for burnup credit has been given, but the issue is being assessed in some countries (Ref. 3).

Points of consideration

The introduction of burnup credit in a fuel storage pool introduces some new points to be considered in addition to the traditional criticality questions, both on the calculational and the administrative side. Examples of such new points are:

Calculations:

- Effect of burnup profile
- Effect of operation history
- Effect of void profile
- Accuracy of calculation methods, with regard to burnup, fission product and plutonium buildup etc
- Accuracy of calculation methods with regard to predicting criticality for spent fuel.

Verification of burnup:

- Accuracy of data acquired during operation of the power plants
- Measurements to verify burnup.

Area of acceptable burnup

If credit for burnup is to be taken an area of acceptable burnup as a function of initial enrichment has to be established. This area is limited by a line of equal

k_{eff} corresponding to $k_{eff} = 0.95$. An example of such a curve for BWR-fuel in CLAB is shown in figure 4. The curve should give $k_{eff} = 0.95$ for the most reactive fuel element, and taking into account uncertainties and possible handling mistakes or accidents.

The uncertainties could be divided into two groups physical uncertainties and uncertainties in the calculation methods.

The most important of the physical uncertainties are the axial burnupdistribution and for BWR the axial voidhistory. Both these factors have to be corrected for taking into account a realistically extreme distribution. Other physical uncertainties are tolerances in manufacturing and the position of the fuel within the storage canister.

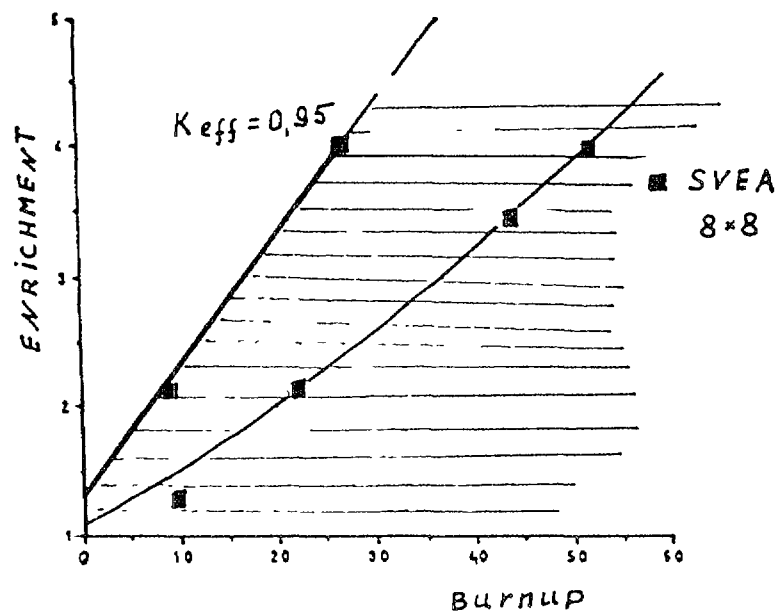


Figure 4: Area of acceptance as a function of enrichment and burnup.

The uncertainties in the calculation methods include capability of predicting criticality for a spent fuel system. Almost all criticality tests have been made with fresh fuel. Some of them though, eg the Swedish KRITZ-measurements have been made with MOX-fuel. A number of criticality tests are also made at the startup of a reactor after refueling, that give an indication of the capabilities of the calculation methods.

Special consideration should be given to the neutron absorbing fission products, that are not normally of interest for safety or shielding calculations.

When establishing the area of acceptable burnup also possible handling mistakes or accidents should be considered. Examples of such events in CLAB are dropping of a fuel element or canister, loss of water in the fuel elevator and close contact between canisters.

Verification of burnup

If burnup credit is introduced a verification of the burnup would be necessary. Different levels of verification could be considered, ranging from administrative verification to reactivity measurements.

As the burnup normally is well known from the operation of the reactors an appropriate level of verification to avoid non-acceptable fuel could be to verify by measurement that the burnup is close to what has been indicated from the reactor operator.

Different methods for this measurement are being considered. They include:

- Gamma spectrometry (eg Cs-137), or gross gamma
- Passive neutrons from Cm-242 and 244
- Reactivity measurement with active neutrons

CONCLUSIONS

The extension of the capacity of CLAB is needed until 1996. The preferred method is to increase the capacity of the existing pools by more efficient use of the storage canisters. By this the capacity could be increased to about 5000 tU. This will delay the need for pools in a new rock cavern until 2003.

In order to be able to store the fuel closer while still keeping the criticality safety two methods are considered, poisoned racks or burnup credit. The insertion of neutron absorbers in the canisters is quite straightforward, while burnup credit will involve a number of new studies and a somewhat changed philosophy. Burnup credit, however, has the advantage of greater flexibility concerning initial enrichment.

The decision on what route that will be finally pursued will be taken at the end of this year.

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THE CONCEPT AND DESIGN OF A LARGE SCALE STORAGE FACILITY FOR SPENT FUEL IN A REPROCESSING PLANT

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Abstract

The spent fuel receiving and storage facility of commercial reprocessing plant which will be constructed at Rokkasho-mura in Japan is now at the stage of basic design. In this paper, the outline and concept of this facility are described.

As a result, the spent fuel storage area is reduced effectively by adopting burnup credit.

This result seemed to be applicable to expanding the spent fuel storage facility which already exists and which will be newly constructed.

1. Introduction

A spent fuel storage facility in a reprocessing plant is designed for removing decay heat and decreasing radioactivity from spent fuel during the period between receipt of spent fuels at a reprocessing plant and reprocessing.

The concept and outline of the spent fuel storage pool in the reprocessing plant which will be constructed at Rokkasho-mura site in Japan is described.

Japan Nuclear Fuel Service Co., Ltd. (to be referred to as "JNFS" hereinafter) submitted the ADRB (Application for the Designation of Reprocessing Business) to the government agency

of Science and Technology Agency (to be referred to as "STA" hereinafter) on March 30th, 1989.

Now safety review by Nuclear Safety Bureau of STA has just been started and will be continued for a few years.

The construction of JNFS's reprocessing plant is planning to begin on 1991, and receiving of spent fuels to our plant and reprocessing on 1994, 1997 respectively.

(1) Concept of a large scale storage pool for spent fuel

The storage pool has been designed to take into account of adequate pool sizing and its layout. Especially for the rationalization of pool sizing, the burnup credit was adopted.

The consideration for safety of structural and seismic strength, economics and operability etc. leads to select 3 pool modules and their series arrangement.

(2) Design of the spent fuel storage facility

This facility consists of 3 main buildings, a cask temporary storage building, a spent fuel receiving and storage building, and a facility control building. (see Fig. 1 and Fig. 2)

A cask temporary storage building is designed for interim storage of loaded casks and unloaded casks, where loaded casks are stored at loaded cask storage area until fuels inside loaded

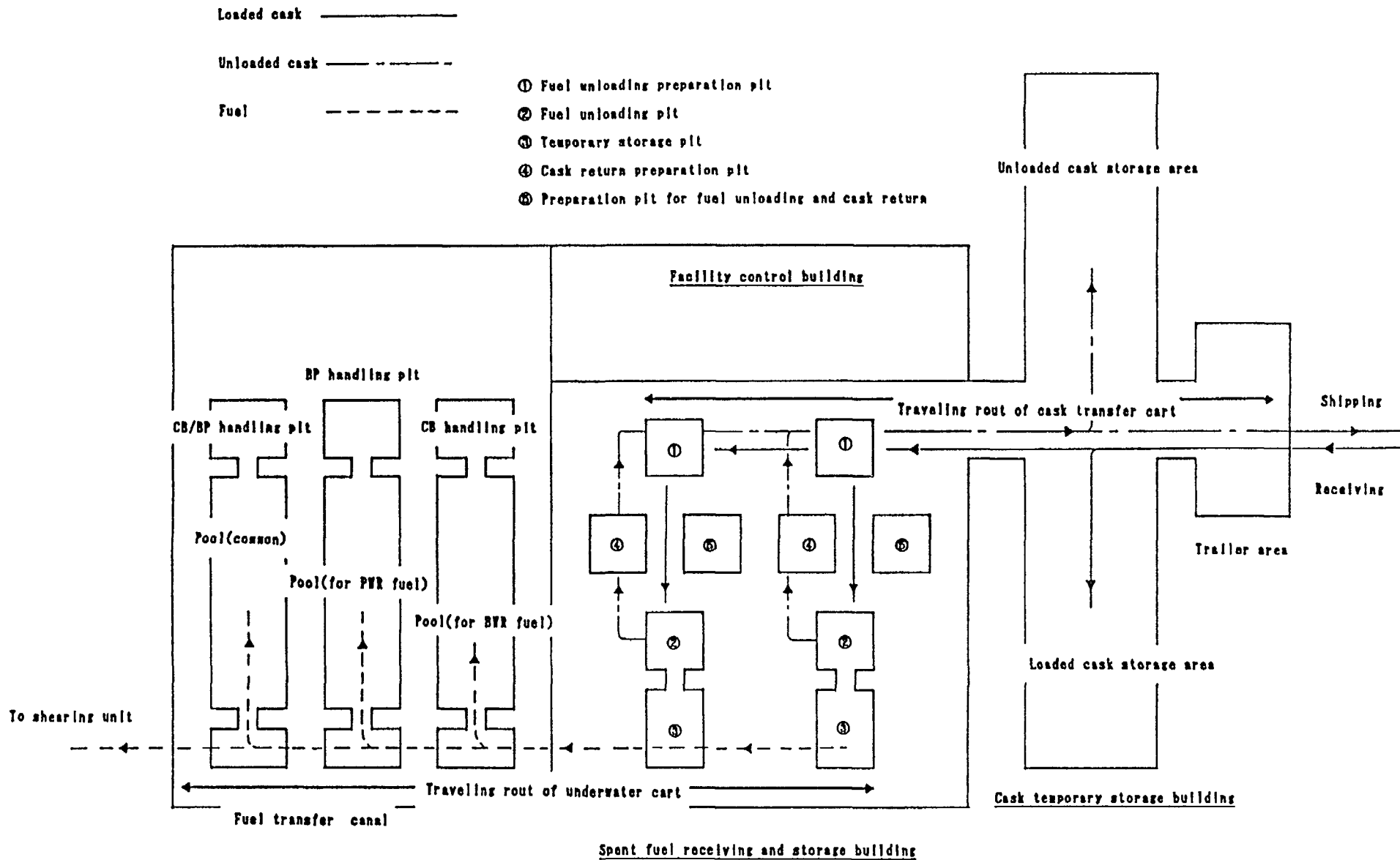


Fig.1 Main equipment lay-out of spent fuel receiving and storage facility

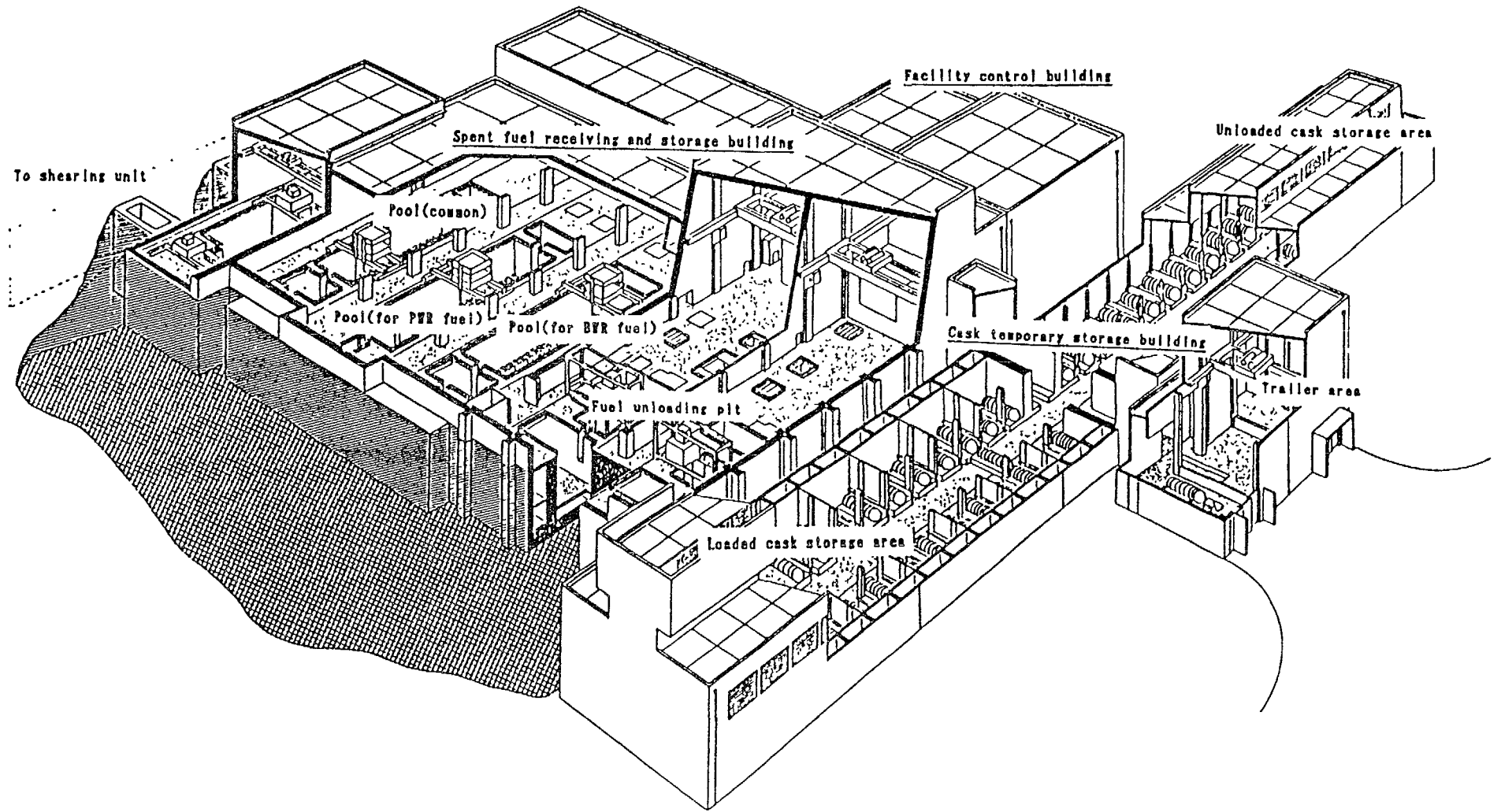


Fig.2 Bird's-eye view of spent fuel receiving and storage facility

casks are unloaded, and unloaded casks are stored at unloaded cask storage area in this building until they are returned to a reactor.

At a spent fuel receiving and storage building, spent fuels inside casks are unloaded and stored in pool.

The aim of a facility control building is: to provide various utilities to these buildings.

The main specification is shown as follows,

Specification of spent fuel

- a. Type : spent fuel of BWR and PWR
- b. Burnup : max 55000MWd/MTU
- c. Cooling time : more than one year
- d. Enrichment : max 3.5% (max 5.0% as initial enrichment)

Outline of this facility

- a. Receiving capacity : 800MTU/year (400MTU/year for BWR, 400MTU/year for PWR)
about 230 casks/year
- b. Storage capacity : 3000MTU (1000MTU/pool × 3 pools)
- c. Storage method : rack storage method (wet storage)

2. Concept of JNFS's storage facility

(1) Selection of 3 pool modules

Our reprocessing plant has a capacity of 800MTU/year (for PWR 400MTU/year, for BWR 400MTU/year).

3 years cooling time at storage pool is required for reprocessing. The storage capacity, therefore, becomes about 3000MTU (800MTU/year × 3 years = 2400MTU plus 600MTU as the allowance).

Our storage facility has 3 pools (one for PWR, one for BWR and one for common PWR and BWR, having capacity about 1000MTU of each pool). It is two main reasons why 3 pool modules were selected. The first is for continuous operation of receiving and storing spent fuels without any interruption. For example, in case of BWR fuels receiving and storing, if one pool is shut down, the other pool can be used. This is the same as for PWR fuels.

The second is economical reason.

(2) Rationalization of pool sizing based on burnup credit

In the near future, it is expected that the nuclear fuels with higher initial enrichment will be for longer continuous operation of a reactor. And as the interval between refueling

outage becomes longer, the fuel with higher burnup will come out of the reactor core and be transported to a reprocessing plant.

In the most case, a spent fuel storage pool is usually designed that the fuels with the highest enrichment, that is equal to the initial enrichment, can be stored for the sake of criticality. But it is considered that a wider space is needed for such fuels to be stored at a spent fuel storage pool based on this concept.

While spent fuels, even such fuels with higher initial enrichment, have low residual after having normally been burned at a reactor.

If calculations of k -effective are performed based on isotopics of the actual irradiated fuels, that is to say adopting burnup credit, rationalization of pool sizing can be possible.

The spent fuel storage pool of JNFS's reprocessing plant is designed to take into account this concept and its size is reduced about 20% compared with that determined by the conventional method based on the initial enrichment.

The sizing of one pool is also determined in consideration of the limitation of the length of fuel handling machine to move accurately and automatically as well as seismic consideration.

3. Facility arrangement

Spent fuel receiving and storage facility consists of 3 buildings, i.e., cask temporary storage building, spent fuel receiving and storage building and facility control building. The plane and bird's-eye view of these buildings is as shown in Fig. 1 and Fig. 2.

(1) Cask temporary storage building

There are cask receiving, storage, and return processes in this building. The dimension of this building is about 180m × 65m × 26m height.

(2) Spent fuel receiving and storage building

There are spent fuel receiving, storage, transfer, transport to shearing process, spent fuel storage pool cleanup and cooling process, radioactive waste treatment process, control room, etc., in this building. The dimension of this building is about 86m × 130m × 21m height.

(3) Facility control building

There are radioactive waste treatment process, ventilation equipment room, radioactive control room, analysis laboratory, entry/exit control system, etc., in this building. The dimension of this building is about 33m × 53m × 15m height.

4. Process

The whole process flow diagram in this facility is as shown in Fig.3 and mechanical flow diagram in Fig.4.

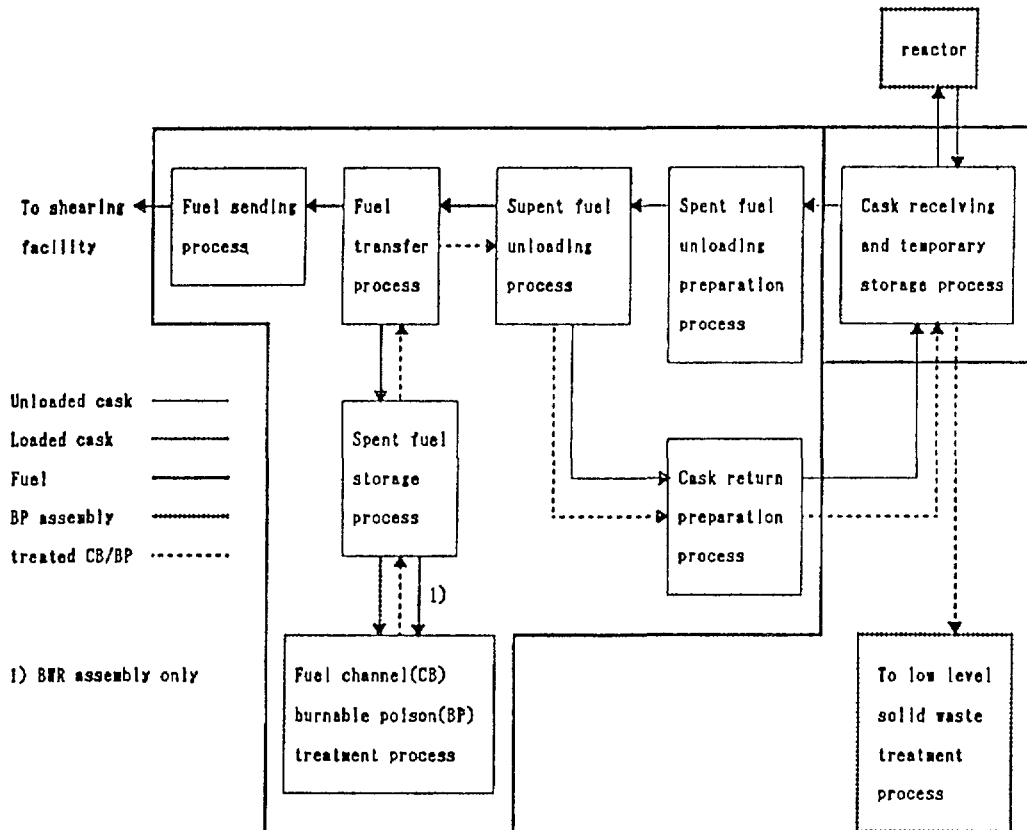


Fig.3 Process flow diagram in spent fuel receiving and storage facility

(1) Spent fuel receiving process

At first, loaded cask transported by trailer is transferred to cask transfer cart by overhead travelling crane at cask temporary storage building and then transferred to loaded cask storage area by cask transfer cart, and stored.

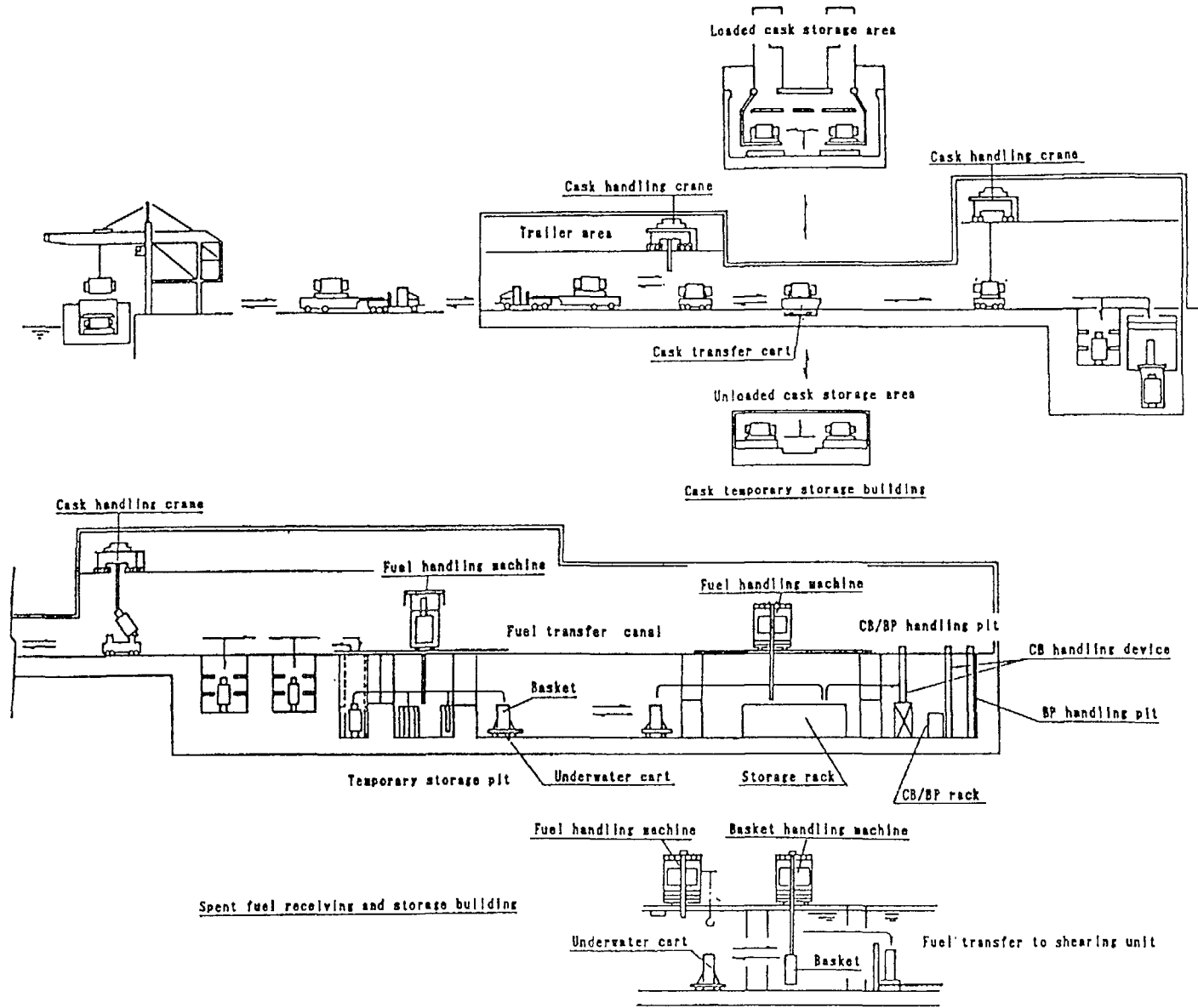


Fig.4 Mechanical flow diagram of spent fuel receiving and storage facility

As 2nd step, loaded cask in this building is transferred by cask transfer cart to spent fuel receiving and storage building, and then transferred by overhead travelling crane to spent fuel unloading preparation process, where cask is cooled and water inside cask is replaced.

As 3rd step, loaded cask is transferred by the crane to spent fuel unloading process, where prevention measures of cask outer surface from contamination are taken, cask is submerged into fuel unloading pit, and spent fuel is unloaded one by one by fuel handling machine and stored at temporary storage pit.

At temporary storage pit, burn-up of spent fuel is measured.

Then spent fuel is transferred one by one by fuel handling machine to the fuel basket set on the underwater cart in spent fuel transfer process.

In case of receiving the defective fuel, it is canned in the defective fuel container and loaded on the underwater cart.

Unloaded cask is transferred by the crane to cask return preparation process, then transferred to unloaded cask storage area for storage after decontamination of outer surface, drainage of water inside cask, clean-up of interior of the cask, leak test and inspection of contamination are performed.

After having completed all of the above, the cask is transferred to trailer area by cask transfer cart, returned to a reactor by trailer.

(2) Spent fuel storage process

Spent fuels loaded into basket are transferred by underwater cart through the fuel transfer canal to the spent fuel storage process, and removed from the basket to fuel storage rack in the spent fuel storage pool one by one by the fuel handling machine.

Defective fuel canned in the defective fuel container is transferred by the underwater cart, and stored in the exclusive fuel storage rack by the fuel handling machine.

Fuels are stored according to their residual enrichment. The fuel which has residual U-235 (to be referred to as "re." hereinafter) more than 2.0% is stored in the rack for the fuel of high residual enrichment, while the other is stored in the rack for the fuel of low residual enrichment.

Spent fuel is loaded in the basket one by one by fuel handling machine after having removed fuel channel (CB) for BWR and the burnable poison assembly (BP) for PWR fuel.

And spent fuel is transferred to spent fuel sending pit by underwater cart, and after temporary storage, spent fuels kept in the basket are set on the basket transfer cart by basket handling machine to be transferred to shearing process.

Clean-up and cooling process of spent fuel storage pool is installed to remove the decay heat of spent fuel by heat exchangers, from cooling water of spent fuel storage pool, spent fuel unloading pit and spent fuel sending pit, and to keep pureness and clearness of pool water by filter and demineralizer.

Auxiliary equipment cooling process, consisting of air-fin cooler, is installed to remove the heat from spent fuel storage pool clean-up and cooling process and from equipments in spent fuel receiving and storage facility.

(3) Auxiliary process

For operation and control of the spent fuel receiving and storage facility, auxiliary system, such as radioactive waste treatment process, ventilation system, analysis laboratory, electrical control system, and instrumentation and control system etc., are installed.

Radioactive waste treatment process unit, consisting of gas, liquid and solid treatment process unit, purifies,

separates, collects and treats the radioactive wastes generated in the spent fuel receiving and storage facility.

Radioactive gas waste treatment process unit is installed to release the radioactive gas waste through the stack after filtration unit.

Radioactive liquid waste treatment process unit is installed to treat the radioactive liquid waste by filter and demineralizer. Treated water is basically recycled and surplus water is released to ocean after the radioactivity is confirmed to be low enough.

Radioactive solid waste treatment process unit consists of CB/BP cutting process and spent resin storage process which stores resin generated from spent fuel storage pool clean-up and cooling process.

Other solid radioactive waste canned in drum, is stored in the storage vault.

5. Safety consideration in main system

(1) Spent fuel receiving system

a. Cask receiving and temporary storage process

The storage capacity is 30 for loaded casks and 32 for unloaded casks.

(a) Cask temporary storage building crane

This crane is overhead traveling type crane used for transshipping the cask between trailer and cask transfer cart. Its capacity is about 150t.

For prevention of fall-off of handling cask, double wire lifting system and hook with lock-up mechanism are adopted on this crane. Furthermore handling casks cannot be moved over storage fuels.

In case of the lack of electric power supply, the crane can leave the position of the lifted cask as it is.

The stopper for prevention of fall-off of the crane itself is also installed in case of earthquake.

(b) Cask transfer cart

This cart is lorry and dolly type and used for transferring the cask between cask temporary building and spent fuel receiving and storage building. Its capacity is about 150t.

For prevention of running over the normal operating range, stopper is installed.

b. Spent fuel unloading preparation process

(a) Cask cooling and clean-up system

c. Spent fuel unloading process

(a) Spent fuel receiving and storage building crane

This crane is overhead travelling type crane used for handling of casks at spent fuel receiving and storage building. Its capacity is about 150t.

For prevention of fall-off of handling cask, double wire lifting system and hook with lock-up mechanism are adopted on this crane. Furthermore handling casks cannot be moved over storage fuels.

In case of the lack of electric power supply, the crane can leave the position of the lifted cask as it is.

The stopper for prevention of fall-off of the crane itself is also installed in case of earthquake.

(b) Spent fuel unloading pit and temporary storage pit

Spent fuel unloading pit is used for spent fuel unloading. Temporary storage pit is used for temporary storage after unloading, inspection, and burnup measurement.

On these pits, leak detector is installed.

Stainless steel lining has enough thickness not to leak in the worst case of fall-off of fuel.

(c) Fuel handling machine

This machine is floor running bridge type and used for spent fuel transfer through spent fuel unloading pit, spent fuel temporary storage pit and spent fuel transfer canal.

For prevention of fall-off of handling fuel, double wire lifting system and hook with lock-up mechanism are adopted on this machine.

In case of the lack of electric power supply, the machine can leave the position of the lifted fuel as it is.

The stopper for prevention of fall-off of the machine itself is also installed in case of earthquake.

d. Cask return preparation process

Cask return preparation process consists of cask interior and cask lid cleaning equipments.

(2) Spent fuel storage system

a. Spent fuel storage process

(a) Spent fuel storage pool

Spent fuel storage pool consists of 3 storage pools (one for BWR fuel, one for PWR fuel, and one for common BWR and PWR fuel) and storage racks are installed at the bottom of storage pool.

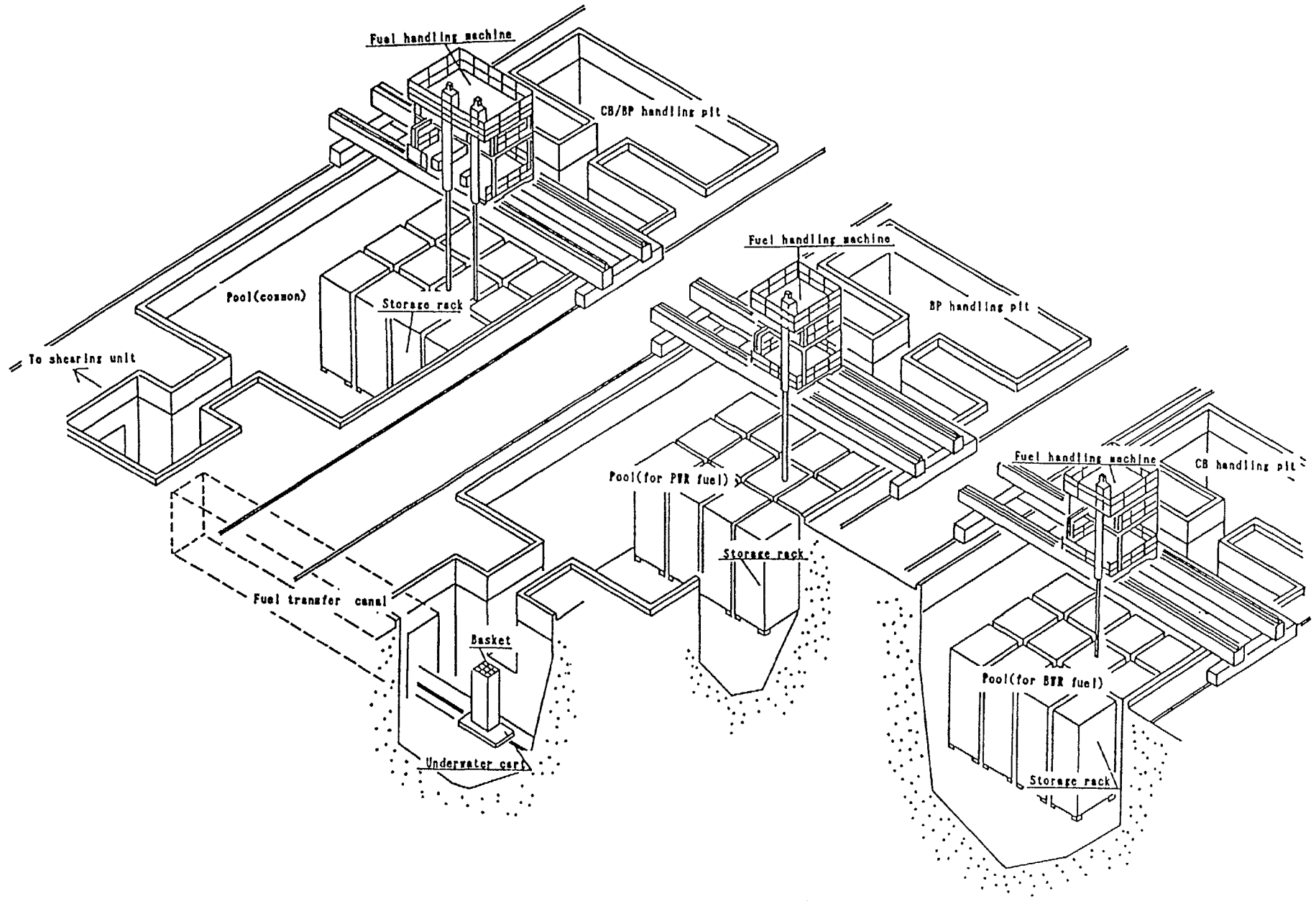


Fig.6 Conceptual drawing of spent fuel storage pool

Storage capacity of each pool is about 1000MTU. A conceptual drawing is shown in Fig. 5.

On these pits, leak detector is installed.

Stainless steel lining has enough thickness not to leak in the worst case of fall-off of fuel.

(b) Spent fuel storage rack

Spent fuels are stored vertically in this rack. There are high and low residual enrichment racks. The composition of the racks is as shown in Table 1.

Table 1. Composition of the racks

rack type Items	re. upto 3.5wt%		re. less than 2.0wt%	
	BWR	PWR	BWR	PWR
number of racks	2	3	60	63
number of stored fuels in one rack	30	20	143	56
main structural material	stainless steel			

Total storage capacity is about 3000MTU. (BWR: 1500MTU,PWR:1500MTU)

The rack is designed with such geometry that the establishment of a critical array is impossible even if the fuels were all of the maximum reactivity. Then effective multiplication factor is less than 0.95. Furthermore the rack has a strength to maintain the safe-spacing in criticality under any outer impact, for example, fall-off of fuel or earthquake.

(c) Defective fuel container

The container is cylindrical can type.

b. Spent fuel transport to shearing process

(a) Spent fuel sending pit

On this pit, leak detector is installed.

Stainless steel lining has enough thickness not to leak in the worst case of fall-off of fuel.

(b) Basket temporary storage rack

This rack is installed in the spent fuel sending pit and used for temporary storage of both loaded and unloaded baskets.

The rack is designed with such geometry that the establishment of a critical array is impossible even if the fuels in the baskets were all of the maximum reactivity. Then effective multiplication factor is less than 0.95. Furthermore the rack has a strength to maintain the safe-spacing in criticality under any outer impact, for example, earthquake.

(c) Basket

The basket for BWR fuel and PWR fuel store 9 fuel assemblies for BWR, 4 for PWR.

The basket is designed with such geometry that the establishment of a critical array is impossible even if the fuels were all of the maximum reactivity. Then effective multiplication factor is less than 0.95. Furthermore the basket has a strength to maintain the safe-spacing in criticality under any outer impact, for example, earthquake.

(d) Basket handling machine

This machine is floor running bridge type.

For prevention of fall-off basket, double wire lifting system and hook with lock-up mechanism are adopted on this machine.

In case of the lack of electric power supply, the machine can leave the position of the lifted basket as it is.

The stopper for prevention of fall-off of the machine itself is also installed in case of earthquake.

c. Spent fuel transfer process

This process consists of transfer canal which connects spent fuel unloading process, spent fuel storage process and transport to shearing process.

(a) Underwater cart

This cart moves horizontally on rails set at the transfer canal bottom and has support structure for holding the basket.

For prevention of running over the normal operating range, stopper is installed.

(b) Fuel handling machine

This machine is floor running bridge type and provided for transferring spent fuel between spent fuel storage pool and transfer canal.

For prevention of fall-off of handling fuel, double wire lifting system and hook with lock-up mechanism are adopted on this machine.

In case of the lack of electric power supply, the machine can leave the position of the lifted fuel as it is.

The stopper for prevention of fall-off of the machine itself is also installed in case of earthquake.

(c) Spent fuel transfer canal

Stainless steel lining has enough thickness not to leak in the worst case of fall-off of fuel.

d. Clean-up and cooling process of spent fuel storage pool

This process consists of cooling unit and clean-up unit.

(a) Cooling system of spent fuel storage pool

This system has two line, and each line has a capacity of heat removal to keep the temperature of pool water at 65 °C.

In case of the lack of electrical power supply, the system can be connected to emergency electric generator.

6. Safety for criticality control

The criticality control and main systems for its control of the spent fuel pool of JNFS's reprocessing plant are described.

(1) The composition of storage racks

The most spent fuels which have normally been burned at a reactor have re. less than 2.0wt%.

So the most racks are designed to be able to store spent fuel with re. 2.0wt% or less. And as a margin, a small amount of racks, which are designed to be able to store spent fuels with re. 3.5wt% or less, are installed in the pool. The value of re. 3.5 wt% is determined as fuels are burned for the period of one continuous operating cycle of interval between refueling outages at a reactor. For example, if fuels with initial enrichment 5.0 wt% are normally burned for one continuous operating cycle, they have about re. 3.5 wt%.

Thus the design philosophy to use two different pitched fuel storage racks in a pool has been introduced to reduce the pool sizing rationally.

The two types of racks are one for fuels with higher re. upto 3.5wt% and the other for those with lower re. less than 2.0wt%.

The composition of these racks in the pool is as shown in Table 1; with less weight on higher residual.

(2) Criticality control based on burnup credit

While the fuel is in a reactor, the behavior of its chain reaction can be precisely known by calculation code which has already been proven to be correct and practical.

And the burnup control is highly trustworthy at a reactor in following two reasons. One is that the calculation code is established. And two is that the management of calculation data is exact by using computer. So this fact leads to the result that there is no need to reconfirmation of isotopics of spent fuel for criticality control at the spent fuel storage pool even designed based on burnup credit.

And this furthermore indicates that the check of the agreement between the identification number on the data, which includes the amount of isotopics in actual irradiated fuel

(U-235 depleted and plutonium built-up etc.) and is attached with each fuel from reactor, and that on each fuel assemblies is only needed.

But although the calculation data are reliable enough to identify them with each fuel assemblies, it is recommended to check actual burnup of each fuel when it is accepted at the pool. By doing this, the amount of isotopes in the spent fuel are measured. That is to say, the confirmation of the data with each fuel from reactor is performed by measuring burnup of each fuel supplementarily. This procedure reinforces the criticality control at the spent fuel storage pool.

The receiving procedures are as follows (see Fig. 6).

(a) 1st step (check)

The agreement between the identification number on the data with each fuel from reactor and that on each fuel assembly is checked and only fuel which has re. less than 3.5wt% is received into the pool.

(b) 2nd step (measurement)

First, the measurement of gamma spectrometry, gross gamma counting and passive neutron counting are performed simultaneously.

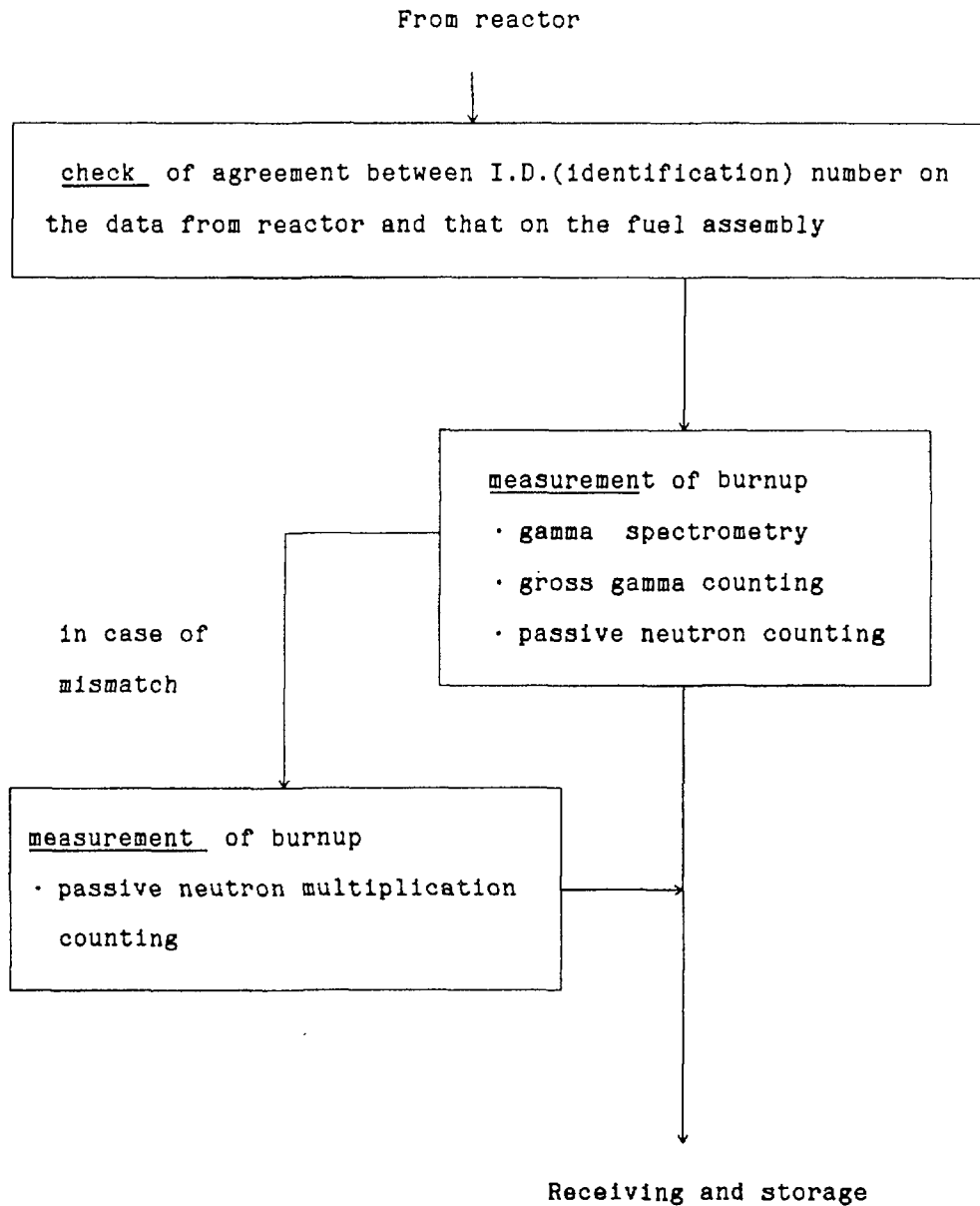


Fig. 6 Block diagram of spent fuel receiving flow

If this measurement discloses discrepancy from reactor data, the another measured value shall be governed in the next step. The measurement of passive neutron multiplication counting is performed in this step.

These procedures, check and measurement, give redundancy for confirming the irradiated fuel isotopics. An internal correlation between the results obtained by these procedures guarantees the correct operating of the criticality control at the spent fuel storage pool based on burnup credit.

7. Conclusion

In this paper, the concept and design of JNFS's spent fuel storage pool are described.

Especially the consideration concerned with adopting burnup credit seemed to be applicable to expanding spent fuel storage facility which already exists.

APPENDIX

RATIONALIZATION OF POOL SIZING BY ADOPTING BURNUP CREDIT

1. Introduction

In the near future, it is expected that the nuclear fuels with higher initial enrichment will be used for longer continuous operation of a reactor. And as the interval between refueling outage becomes longer, the fuel with higher burnup will come out of the reactor core and be transported to a reprocessing plant.

In most case, a spent fuel storage pool is usually designed that the fuels with the highest enrichment, that is equal to the initial enrichment, can be stored for the sake of criticality. But it is considered that a wider space is needed for such fuels to be stored at a spent fuel storage pool based on this concept.

While spent fuels, even such fuels with higher initial enrichment, have low residual after having normally been burned at a reactor.

If calculations of k-effective are performed based on isotopics of the actual irradiated fuels, that is to say adopting burnup credit, rationalization of pool sizing can be possible.

The spent fuel storage pool of Japan Nuclear Fuel Service Co., Ltd. (to be referred to as "JNFS" hereinafter) reprocessing plant is designed to take into account this concept and its

size is reduced about 20% compared with that determined by the conventional method based on the initial enrichment.

This concept is also seemed to be applicable to expanding spent fuel storage facility which already exists.

In this appendix, the criticality control and main systems for its control of the spent fuel pool of JNFS's reprocessing plant which will be constructed at Rokkasho-site are described.

2. Composition of storage racks

The most spent fuels which have normally been burned at a reactor have residual U-235 (to be referred to as "re." hereinafter) less than 2.0wt%.

So the most racks are designed to be able to store spent fuels with re. 2.0wt% or less. And as a margin, a small amount of racks, which are designed to be able to store spent fuels with re. 3.5wt% or less, are installed in the pool. The value of re. 3.5wt% is determined as fuels are burned for the period of one continuous operating cycle of interval between refueling outages at a reactor. For example, if fuels with initial enrichment 5.0wt%, are normally burned for one continuous operating cycle, they have about re. 3.5wt%.

Thus the design philosophy to use two different pitched fuel storage racks in a pool has been introduced to reduce the pool sizing rationally.

The two types of racks are one for fuels with higher re. upto 3.5wt% and the other for those with lower re. less than 2.0wt%.

The composition of these racks in the pool is as shown in Table 1; with less weight on higher residual.

Table 1. Composition of the racks

rack type Items	re. upto 3.5wt%		re. less than 2.0wt%	
	BWR	PWR	BWR	PWR
number of racks	2	3	60	63
number of stored fuels in one rack	30	20	143	56
main structural material	stainless steel			

Total storage capacity is about 3000MTU. (BWR: 1500MTU,PWR:1500MTU)

3. Criticality control based on burnup credit

While the fuel is in a reactor, the behavior of its chain reaction can be precisely known by calculation code which has already been proven to be correct and practical.

And the burnup control is highly trustworthy at a reactor in following two reasons. One is that the calculation code is established. And two is that the management of calculation data is exact by using computer. So this fact leads to the result that there is no need to reconfirmation of isotopics of spent fuel for criticality control at the spent fuel storage pool even designed based on burnup credit.

And this furthermore indicates that the check of the agreement between the identification number on the data, which includes the amount of isotopics in actual irradiated fuel (U-235 depleted and plutonium built-up etc.) and is attached with each fuel from reactor, and that on each fuel assemblies is only needed.

But although the calculation data are reliable enough to identify them with each fuel assemblies, it is recommended to check actual burnup of each fuel when it is accepted at the pool. By doing this, the amount of isotopics in the spent fuel are measured. That is to say, the confirmation of the data with each

fuel from reactor is performed by measuring burnup of each fuel supplementarily. This procedure reinforces the criticality control at the spent fuel storage pool.

The receiving procedures are as follows (see Fig. 1).

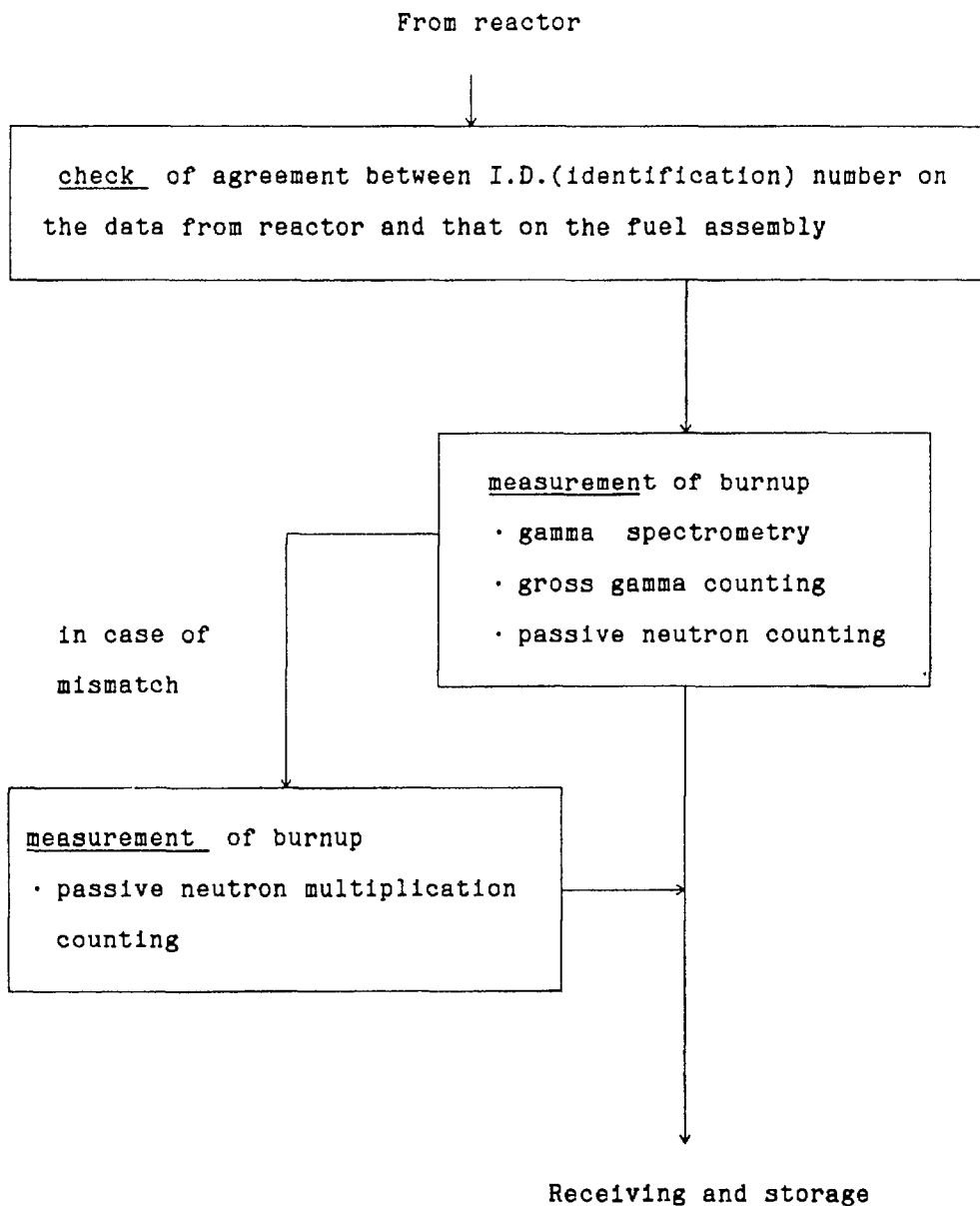


Fig. 1 Block diagram of spent fuel receiving flow

(a) 1st step (check)

The agreement between the identification number on the data with each fuel from reactor and that on each fuel assembly is checked and only fuel which has re. less than 3.5wt% is received into the pool.

(b) 2nd step (measurement)

First, the measurement of gamma spectrometry, gross gamma counting and passive neutron counting are performed simultaneously.

If this measurement discloses discrepancy from reactor data, the another measured value shall be governed in the next step.

The measurement of passive neutron multiplication counting is performed in this step.

These procedures, check and measurement, give redundancy for confirming the irradiated fuel isotopics. An internal correlation between the results obtained by these procedures guarantees the correct operating of the criticality control at the spent fuel storage pool based on burnup credit.

4. Conclusion

The result of adopting burnup credit is effective to reduce the spent fuel storage area about 20% compared to that of adopting the conventional method, i.e. the storage pitch of spent fuel assemblies is determined based on initial enrichment.

CRITICALITY REANALYSIS OF NPP KRSKO SPENT FUEL RACKS

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Abstract

The Nuclear Power Plant Krsko (NPP Krsko), Yugoslavia, started with 9 spent fuel racks with full capacity of 180 fuel assemblies in the first operating cycle (1982/1983). The half of spent fuel pool has been occupied with racks. The first analyses have shown that spent fuel pool could sustain load of about 800 assemblies. Thermal-hydraulic and criticality study for new fuel racks (9x7 and 9x8) and 3.5% enrichment has been done. The modification and installation of 12 new racks has been made since 1983 and the first spent fuel loaded during the first outage. Since that time NPP Krsko has been trying to improve fuel economy. Low leakage pattern, enriched fuel up to 4.3% with axial blankets and extended burnup capability were accepted. It was necessary to reanalyse criticality of spent fuel pool. Comparison between past and current valid analyze and methodology is discussed. Exact modeling of fuel geometry with KENO-IV code shown that enrichment limit is 4.2%. Reactivity equivalence methodology was performed by PHOENIX code to establish fuel assembly minimum burnup vs. initial U-235 enrichment for storage in fuel racks. Also checkboard pattern (three of four assembly loading scheme) was analyzed to establish maximum fuel enrichment. Reanalysis preserves spent fuel storage capabilities.

INTRODUCTION

The Nuclear Power Plant Krsko, Yugoslavia, has been operating for 7 years and since 1982 two different approaches have been used to increase spent fuel pool capability.

1. increasing fuel storage capability by inserting new more dense fuel racks
2. reanalyse criticality conditions for existing racks design

The first operating cycle (1982/1983) was started with 9 spent fuel racks standard Westinghouse design in the pool with full capacity of 180 fuel assemblies. The half of spent fuel pool has been occupied with racks and total capacity was estimated for 6 reloads. Because this fact NPP Krsko decided to increase fuel storage capacity, encouraged by the fact that spent fuel pool could sustain load of more than 800 assemblies. New racks installed are composed of individual vertical cells made by austenitic stainless steel which are fastened together to form modules. These modules are vertical supported on the floor of spent fuel pit via module base plates. Two types of racks are used: 9 times 8 and 9 times 7 storage cells. Total 12 modules are used with 828 cells (8 times 72 and 4 times 63 cells).

Thermal-hydraulic and criticality study for new fuel racks and 3.5 w/o U235 enrichment has been done. The modification and installation of 12 new racks was made in 1983 and the first spent fuel loaded during the first outage.

Since that time NPP Krsko has been trying to improve fuel economy. Low leakage pattern, enriched fuel up to 4.3 w/o U235 with axial blankets and extended burnup capability were accepted. It was necessary to reanalyse criticality of spent fuel pool for higher enrichments. Two separate rack regions or arrays were analyzed.

It has been shown that adequate modeling with licensed computer programs without losing conservative margin could preserve spent fuel storage capability.

CRITICALITY ANALYSIS OF KRSKO FUEL RACKS
RELATED TO SPENT FUEL STORAGE CAPABILITY

Description of methodology

The criticality calculation method and cross section values are verified by comparison with critical experiment data for assemblies similar to those for which racks are designed. The benchmarking data is sufficiently diverse to establish that method bias and uncertainty will apply to rack conditions which include strong neutron absorbers, large water gaps and low moderator densities.

The design method which insures the criticality safety of fuel assemblies in the spent fuel storage racks uses AMPX system of codes for cross section generation and KENO IV three dimensional Monte Carlo code for reactivity determination. The 227 energy group cross section library that is the common starting point for all cross section used for the benchmarks and the storage racks is generated from ENDF/B-V data.

Additionally reactivity equivalence curve was generated by using PHOENIX two dimensional multigroup transport theory computer code.

Design basis

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor of the fuel assembly array will be less than 0.95. as recommended in ANSI 57.2-1983, ANSI 57-3 1983.

Assumptions:

1. Fuel assembly contains the highest enrichment authorized, is at its most reactive point in life and no credit is taken for any burnable poison in the fuel rods

2. All fuel rods contain uranium dioxide at highest enrichment over the infinite length of each rod
3. No credit is taken for any U234 and U236 in the fuel, nor is any credit taken for build up fission product poison material
4. The moderator is pure water at temperature 293 K and density of 1.0kg/m³
5. No credit is taken for any spacer grids or spacer sleeves
6. The array is infinite in lateral and axial extent which precludes any neutron leakage from array.

Discussion

Previous analyze for the same spent fuel storage (tab 1, figure 2) had shown the limit of 3.5 w/o U235 enrichment. It was reasonably expected that there was some limitations for fuel with higher enrichments.

So two different racks region were analyzed. The first was four of four storage where all fuel locations were occupied to determine probably the highest possible enrichment and the other, three of four storage (figure 3). It is obvious that in the second case storage capability is decreased.

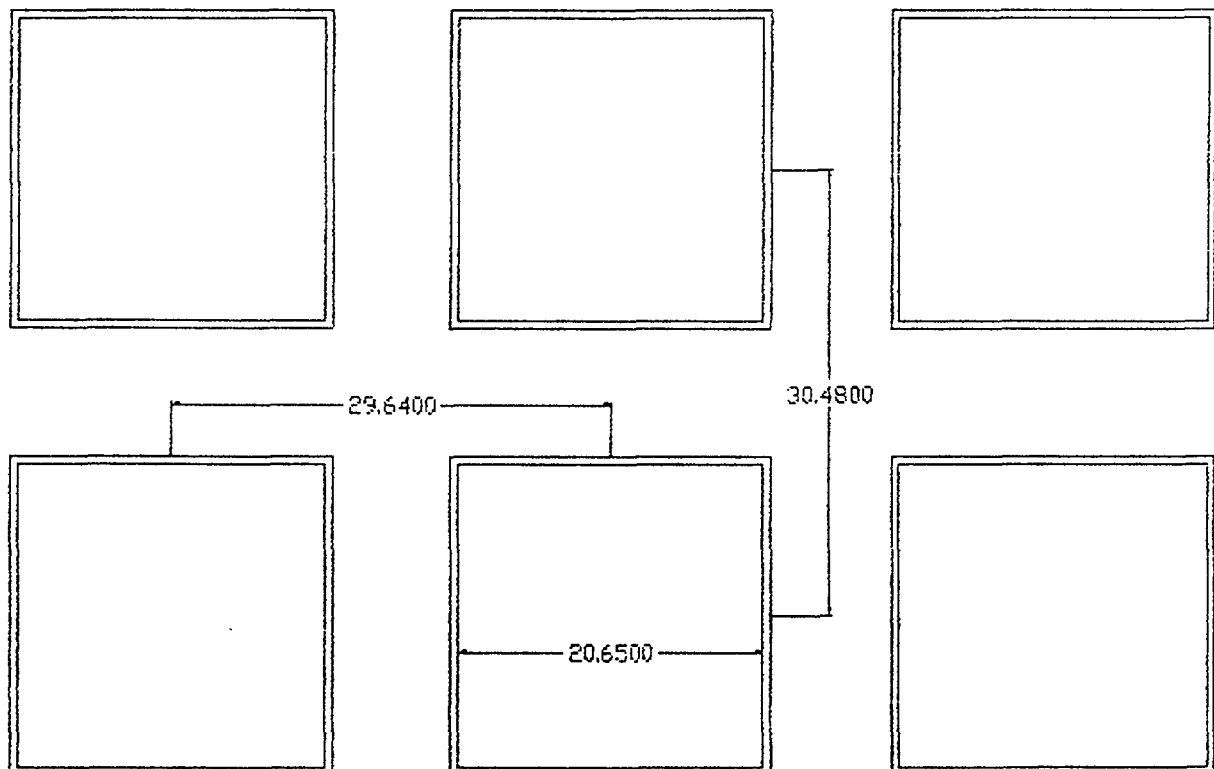


FIG.1. Krsko fuel storage, nominal dimensions (cm)

TABLE I. SPENT FUEL PIT COOLING SYSTEM DESIGN PARAMETERS

1. Spent fuel pit storage capacity, cells	828	
2. Spent fuel pit water volume	430,463 gal	1629 m ³
3. Nominal boron concentration of the spent fuel pit water, ppm	2000	2000
4. Storage		
a. Design case 40% of a core unload with 16 yrs. storage from previous refuelings		
Decay heat production	9.3 x 10 ⁶ BTU/HR	2727 KW
Spent fuel pit water temperature (SFPAHSF 1 Only)	≤ 130 ^o F	≤ 54.6 ^o C
(SFPAHSF 2 Only)	≤ 120 ^o F	≤ 49 ^o C
Spent fuel pit water heatup rate, assuming loss of cooling	2.59 ^o F/HR	1.44 ^o C/HR
b. Maximum heat load case - 828 elements stored (full rack plus complete unloading of the core)		
Decay heat production	23.16 x 10 ⁶ BTU/HR	6785 KW
Spent fuel pit water temperature (SFPAHSF 1 Only)	≤ 183 ^o F	≤ 83.9 ^o C
(SFPAHSF 2 Only)	≤ 150 ^o F	≤ 65.5 ^o C
Spent fuel pit water heatup rate, assuming loss of cooling	6.48 ^o F/HR	3.6 ^o C/HR

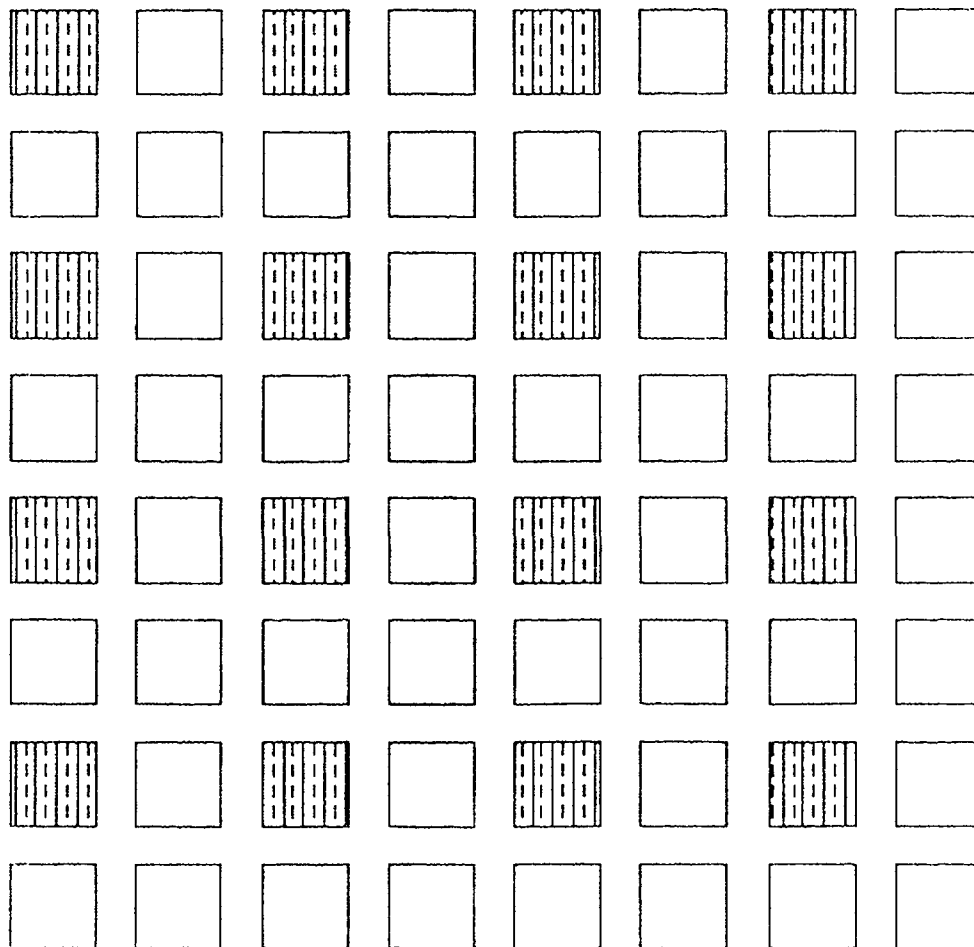


FIG.2. Three of four fuel assembly loading scheme.

We were able to shift the limiting enrichment for four of four storage to 4.2 w/o U235 with the same methodology. The main reason for that relatively large change lays in the fact that fuel assembly was explicitly modeled in KENO-IV code rather than homogenized as was seen in past analyze. Homogenization of clad, gap, water and fuel was in that case to conservative because it led to overmoderation.

The next step was to take into consideration the changes in fuel and fission product inventory resulting from depletion in the reactor core up to enrichment 5.0 w/o. The fuel was depleted in the conditions similar to what the assemblies would have seen in the core for different amount of time and reinserted to spent fuel racks to find multiplication factor.

The KENO-IV computer code was used to calculate the storage rack multiplication factor with an equivalent fresh fuel enrichment of 4.2 w/o. Combinations of fuel enrichment and discharge burnup yielding the same rack multiplication factor as at the zero burnup intercept were determined with PHOENIX computer code (figure 4). The burnup credit curve was developed from that data using burnup dependent reactivity penalty to conservatively take credit for reactivity loss due to being burned in the core (figure 5). Thus from figure 5 every assembly with enrichment 5.0 w/o and burnup higher then 4000 MWD/MTU can be stored in each cell locations.

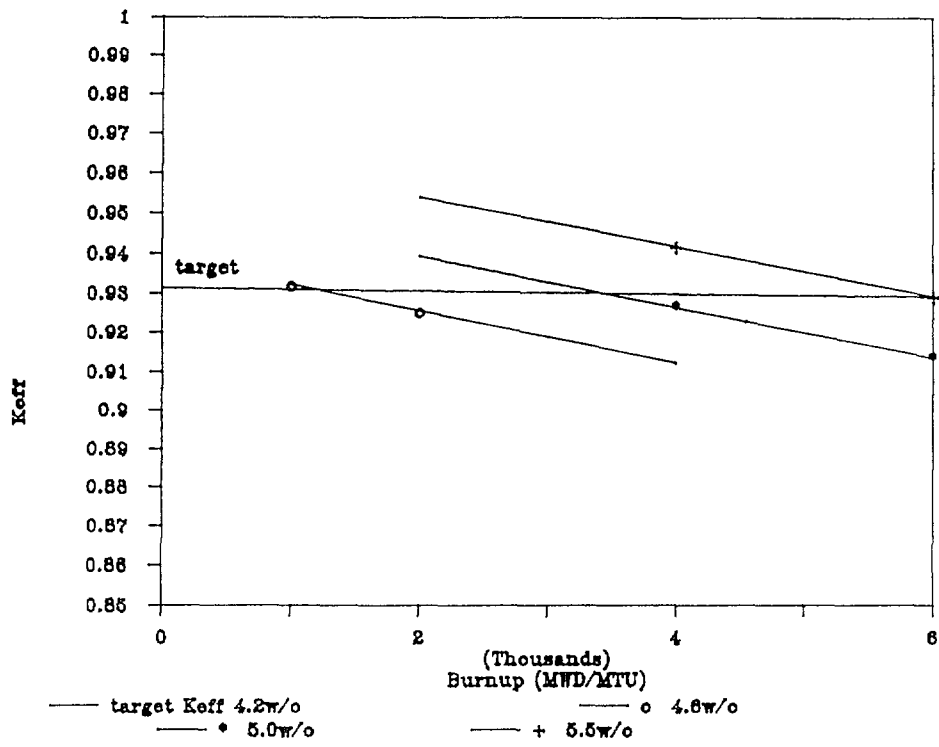


FIG.3. Keff versus burnup and enrichment.

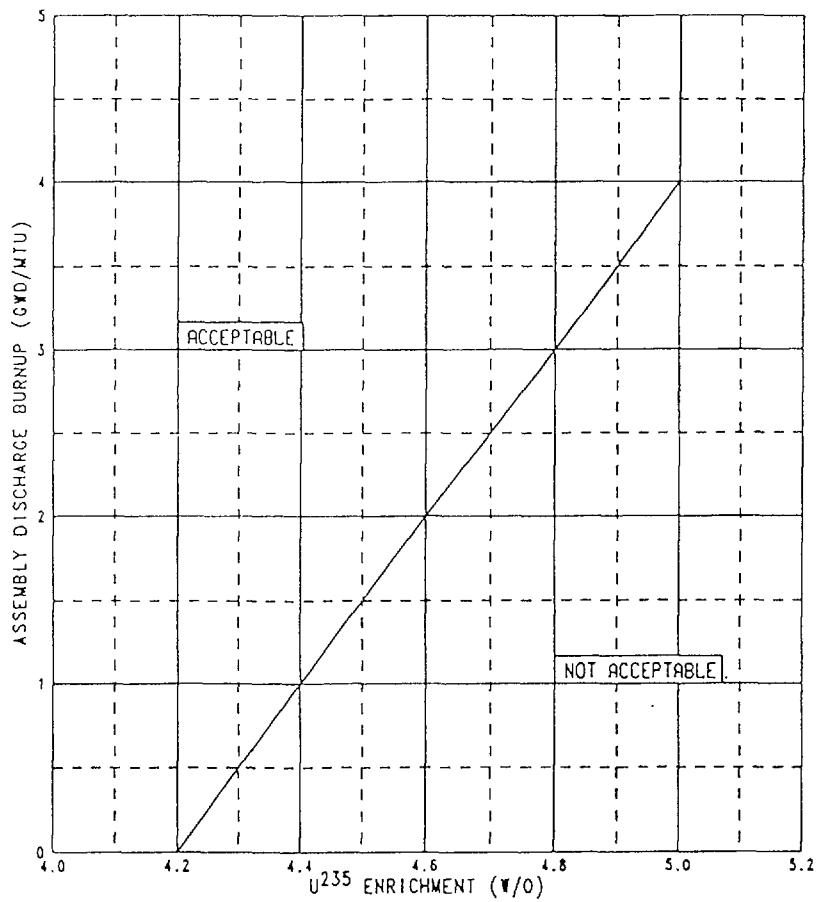


FIG.4. Krsko fuel assembly minimum burnup versus initial U²³⁵ enrichment for four of four storage.

TABLE II. FUEL PARAMETERS EMPLOYED IN CRITICALITY ANALYSIS

Parameter	W 16x16 STD & V5
Number of Fuel Rods per Assembly	235
Rod Zirc-4 Clad O.D. (inch)	0.374
Clad Thickness (inch)	0.0225
Fuel Pellet O.D. (inch)	0.3225
Fuel Pellet Density (% of Theoretical)	96
Fuel Pellet Dishing Factor	0.0
Rod Pitch (inch)	0.485
Number of Zirc-4 Guide Tubes	20
Guide Tube O.D. (inch)	0.471
Guide Tube Thickness (inch)	0.018
Number of Instrument Tubes	1
Instrument Tube O.D. (inch)	0.471
Instrument Tube Thickness (inch)	0.018

The analyze of three out of four pattern (figure 3) showed that 5.0 w/o fuel could be stored in the rack in such configuration. There were some difficulties to exact model the 16 by 16 fuel (figure 7) due to KENO asymmetry limitations.

Conclusion

Exact modeling of fuel and rack geometry with KENO-IV code, possibly extended to three dimensions with evaluating all absorbing materials shows that reanalysis could be the way to temporary extend fuel storage capability if transition to higher enrichment is necessary. Burnup credit curve or reactivity equivalent curve is very useful to achieve better spent fuel performances. But at the end it is necessary to point that other aspects of the problem should be carefully analyzed.

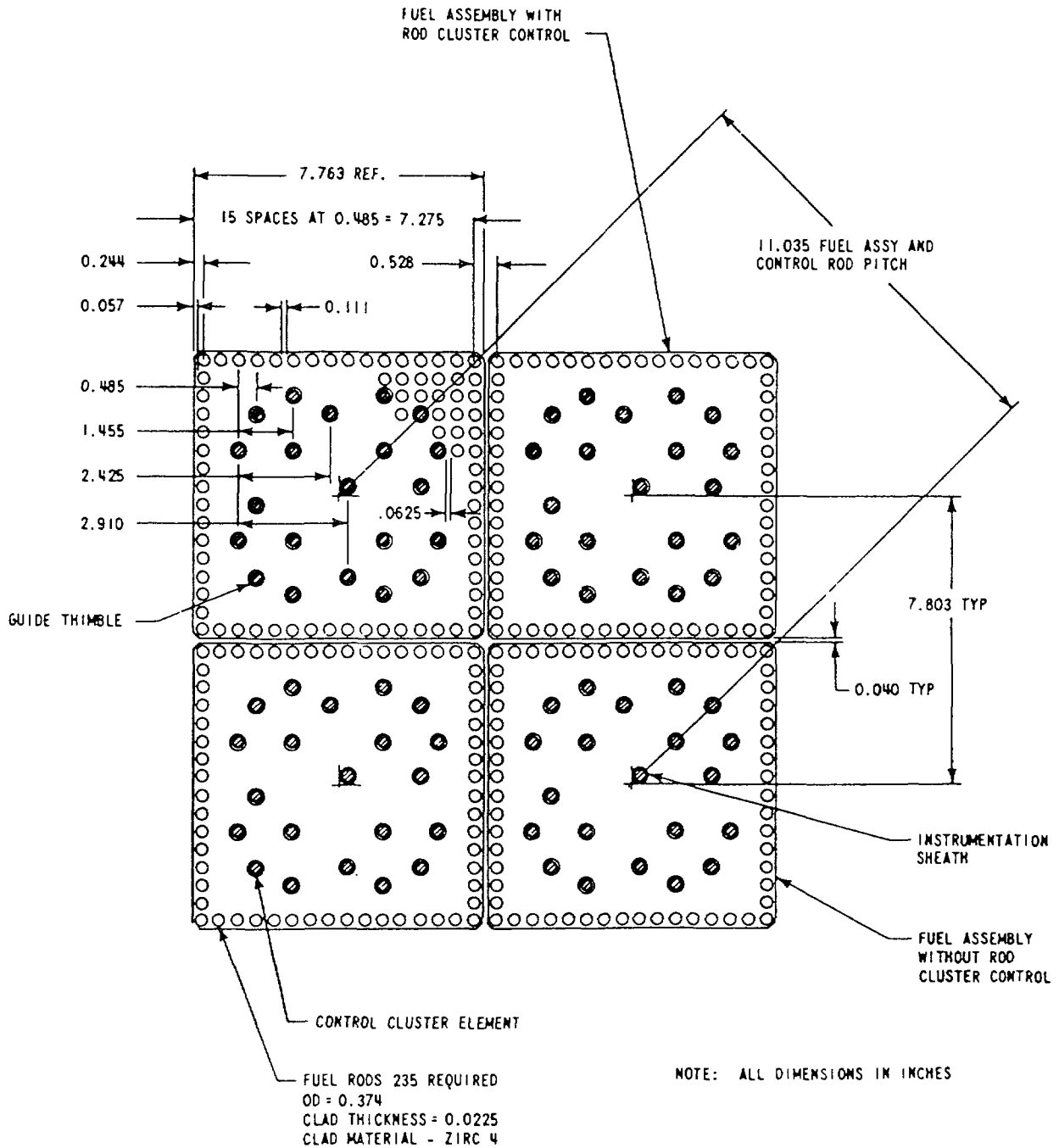


FIG.5. Fuel assembly cross section 16 x 16 (conceptual)

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**PRINCIPLES OF VVER AND RBMK
SPENT FUEL STORAGE AND METHODS
OF HIGH DENSITY STORAGE**

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Abstract

The principles of storing spent fuel of reactor type VVER and RBMK are considered. The order of management with fuel during storage in at-reactor and away-from-reactor storage are discussed. The methods of high-density storage spent fuel by means of arrangement of assemblies into special basket-absorbers made of boron steel are proposed. The possibility of extending the storage capacity is shown.

**1. Types and amount of spent nuclear fuel and its management
in the USSR**

In the USSR spent nuclear fuel is at first cooled in the reactor pool during 3-5 years and then it is transported to an intermediate storage facility or directly to a reprocessing plant.

In 1988 42 power units were operated at the USSR's nuclear power plant (NPP) with total power of 35000 MW (e) /1/. The structure of nuclear power generating capacities is the following:

- NPP with water-cooled and water-moderated reactors (VVER),
GW 18,00
- NPP with channel water-cooled graphite-moderated reactors
(RBMK), GW 16,00
- NPP with fast-neutron reactors (BN), GW. 0,75

This power production was achieved with 10 units of VVER-440 type, 1 unit of VVER-365 type, 13 units of VVER-1000 type, 15 units of RBMK-1000 and -1500 types and 2 units of BN

type (BN-350 and BN-600) /2/. Based on mean annual discharge the total accumulated quantity of spent nuclear fuel for storage in at-reactor pools is about 1000 tons.

In accordance with the USSR energy program up to 2000, the increase of NPP energy production should be realized by preferential construction of NPP with VVER-1000 reactors. At present transport vehicles have been created for delivery of VVER-1000 fuel assemblies to storage facilities, and a reprocessing plant for this fuel is designed. This reprocessing plant will comprise several facilities which will be gradually put into operation with arising of the demand for plutonium used in mixed fuel production. The plant will reprocess fuel assemblies from all NPP with VVER-1000 including those under construction abroad. The reprocessing plant fuel storage capacity will be 3000 tons /3/. Its construction, analogous to those existing at NPP, is described in the report presented to IAEA conference in 1982.

As regards the VVER-440 and RBMK fuels, it was reported /3,4/ that at the first stage of nuclear power development pilot plants were constructed and operated to elaborate fuel reprocessing technology for VVER-440 reactors.

2. The requirements on the spent fuel storage facilities

In the USSR "Basic safety regulations for design, construction and operation of nuclear power plants" are in force /5/. This document extends to storage facilities as constituent parts of nuclear power plants.

From the viewpoint of safety the following technological requirements are imposed upon storage facilities:

- fuel arrangement in the pool providing for nuclear safety during storage and handling operations;
- radiation protection of personnel during storage facility service according to current specifications;
- control of fuel storage, ensuring of its safeguarding and preservation in storage facility;
- pool water purification from radioactive substances released in water by damaged fuel elements and by assemblies with surface contamination as well as from corrosion products in order to achieve water transparency necessary to perform remote underwater fuel handling operations;

- elimination of radioactive releases from storage facility ventilation system with the use of cleaning filters;
- prevention of water leakage into environment using the storage facility design which eliminates the water leakage into soil during facility service and localizing local leakages;
- pool water cooling to the temperature not exceeding 50 °C with residual heat removal from the fuel;
- rational organization of fuel transportation from the storage facility;
- possibility of storing assemblies with damaged fuel elements.

To meet the above-mentioned requirements the following technological and constructive solutions are used when designing NPP fuel storage pools:

- the density of fuel assemblies arrangement in the pool, the water layer above the active part of fuel assemblies and the thickness of pool walls meet the requirements imposed by nuclear and radiation safety (maximum design $K_{eff} = 0,95$);
- the use of light dismountable ceiling over the pool provides for stable ventilation conditions in the above-water space of the pool;
- internal surfaces of the pool are lined with stainless steel;
- leakages from under pool lining are collected and controlled to exclude the penetration of pool water in adjacent rooms or in soil;
- the pools are equipped with the systems for water cooling and purification, maintenance of required water level, filling and emptying of pools, special ventilation of above-water space, technological and radiation control;
- fuel assemblies with damaged pins are stored in sealed cans;
- storage of fuel assemblies in water pools is performed according to two schemes depending on the design of assemblies: upper storage when long assemblies are hung on the metallic ceiling of pool (for RBMK reactor), and lower storage when assemblies are stored on the bottom (for VVER and BN reactors).

3. Storage of different types of fuel

Fuel is cooled in at-reactor pool but its intermediate storage and preparation for conveying to reprocessing plant are performed in a special storage building common for all nuclear power plant.

3.1. Spent fuel storage at NPP with RBMK reactors

3.1.1. At-reactor pools

At-reactor pools for spent fuel cooling are situated in the central room of reactor unit. Fuel assemblies are suspended on the beams of pool ceiling. Storage spacing of spent assemblies 160x250 mm is defined by fuel element design and meets the requirements of nuclear safety.

The equipment for spent fuel handling comprises: transfer machine, 50/10-ton bridge crane, on-floor 1-ton crane jib, cans for spent fuel assemblies, loading mechanism for the transfer of assemblies from handling machine in cans.

For transportation of spent fuel from at-reactor pool to storage building a special container-car and transport baskets holding 9 assemblies are used.

Spent fuel is transferred from reactor channels into cooling pools using handling machine without reactor shutdown. The assemblies with suspensions are unloaded from the machine in cooling pools.

The arrangement of assemblies in the pool is performed under protecting water layer using on-floor crane-jib.

After three-year cooling the spent fuel assemblies should be dispatched to intermediate storage facility.

3.1.2. Special storage building

Long-term intermediate storage of spent fuel and its preparation for dispatching to reprocessing plant are performed in the storage building located at the nuclear power plant site. The storage building is designed for reception of spent fuel from four reactors during 30 years. Storage capacity is 1800 tons (storage density 3,0 t/m²) /6/.

The storage building design is defined by the following engineering solutions:

- storage of spent nuclear fuel assemblies in cans to avoid accidental damage of zirconium fuel cladding;
- remote transfer by air of transport baskets with spent fuel from container-car to reception compartment of pool and backwards;
- the use of protecting water layer when transferring spent fuel from nine-assembly transport basket in cans during its movement in pools;

- the separation of an assembly into two parts in storage building;
- the transportation of spent fuel assemblies to reprocessing plant in large shipping casks.

3.2. Spent fuel storage at NPP with VVER reactors

Irradiated fuel of VVER reactors is initially stored in at-reactor pools for 3 years. Fuel assemblies discharged from reactor are placed on racks standing on the floor of the pool. Fuel handling is performed during reactor shutdown; for this purpose there is a special unloading bassin operated only at reactor shutdown and connecting the reactor with the storage pool. Before beginning the scheduled discharge this bassin is filled with borated water and connected with the reactor and the storage pool. Fuel assemblies of VVER reactors remain in water during all fuel handling operations with shielding water layer being minimum 3 m at fuel discharge.

The storage capacity is 600 tons (storage density - $1,1 \text{ t/m}^2$) /6/. The fuel is stored in baskets designed for 30 sealed or 18 leaking fuel assemblies. They are placed on the pool floor and protected by three-meter water layer.

4. Spent fuel storage experience.

The operation of VVER-400 reactors allowed us to gain the experience on spent fuel storage and its dispatching from NPP with reactors of this type. Our experience shows that the fuel claddings fabricated from Zr + 1 % Nb alloy are in satisfactory condition during long-term residence in at-reactor pools.

The experience on VVER-1000 fuel storage and dispatching is scanty due to short-time operation of NPP with reactors of this type.

Provision is made for short-time storage (not more than 1 year) of VVER-1000 fuel in shipping casks after three-year cooling in reactor pools pending subsequent transportation of this fuel.

We have got good results of reactor fuel storage in water-filled pools: the worsening of fuel condition due to storage was not observed (for the fuel with mean burn-up). Purification systems allow to maintain water quality within prescribed limits. Fission product release in pool water is insignificant. The water radioactivity is mainly caused by cesium, cobalt and manganese radionuclides. The most contribution to total water activity

connected with fission products is made by cesium isotopes, evidently released in water from defective fuel elements.

5. Nuclear safety of high-density fuel storage.

The problem of increasing the time for spent fuel storage at NPP imposes additional requirements on storage facility design. In USSR the increase of storage capacities is mainly achieved in two ways:

- by densification of fuel arrangement in at-reactor cooling pools;
- by construction of additional away-from-reactor storage facilities.

Densification of fuel storage in reactor pools at NPP with RBMK reactors can be achieved by using more dense hanging of cans with fuel assemblies and at NPP with VVER reactors by using racks made of boron steel.

Away-from-reactor storage facilities for VVER-1000 spent fuel are put into operation at Novovoronezh NPP and those for RBMK spent fuel at Leningrad, Kursk and Chernobyl NPP. An autonomous storage facility for VVER fuel is built in GDR, away-from-reactor storages are under construction in Bulgaria and Czechoslovakia.

High-density fuel storage in at-reactor cooling pools requires additional expenses to ensure nuclear safety, but in the case of its realization there is no need for building of extra cooling bassin whose cost is evaluated to be about 10 % of that of power unit, and the volume of transport and technological operations with spent fuel can be substantially reduced.

At present VVER spent fuel assemblies are stored in aqueous medium, and subcriticality necessary for fuel storage safety is reached by selection of assemblies spacing. Nuclear safety regulations used in the USSR require the subcriticality not less than 0,05 ($K \leq 0,95$) on filling the fuel storage with cold (20 °C) water without dissolved absorber for the fuel with maximum enrichment used in this type of reactors. The increase of spacing between fuel assemblies causes the decrease of K in storage pools.

To ensure the required subcriticality (0,05) it is necessary to place 4,4 %-enriched VVER-1000 fuel assemblies with spacing not less than 40 cm and 3,6 %-enriched VVER-400 fuel assemblies with spacing not less than 21,5 cm. The density of fuel assemblies arrangement descreases compared to that in reactor by a

factor of $\sim 2,8$ for VVER-1000 fuel and by a factor of 2,2 for VVER-440 fuel.

From the point of view of storage capacity the most effective method allowing fuel assemblies to be placed with minimum spacing consists in siting every fuel assembly in a hexagonal basket made of neutron-absorbing material with a minimum gap necessary for its location. With the use of method, thermal neutrons released from assemblies are absorbed by baskets and fast neutrons are moderated in the gap between the hexagonal baskets and subsequently absorbed on them (Fig. 1).

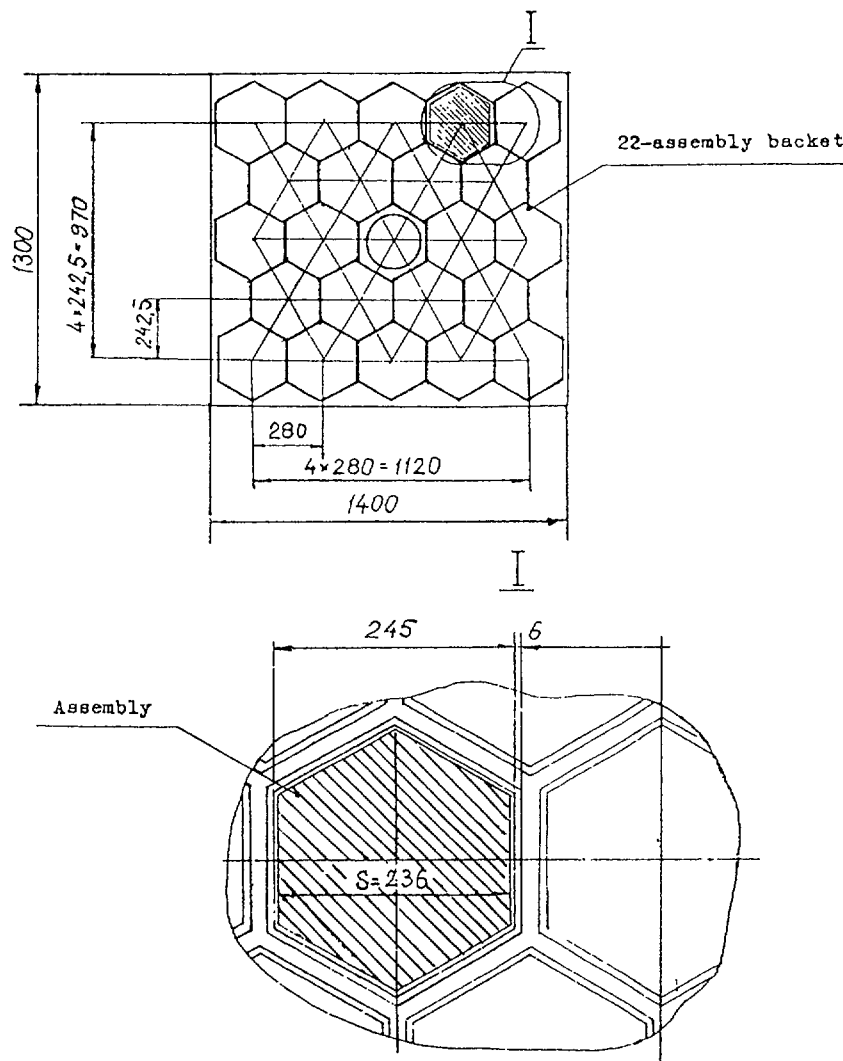


Fig. 1. Arrangement of VVER-1000 fuel assemblies in the hexagonal baskets.

The increase in pool capacity relative to undensified version grows with the fuel enrichment. This is due to a more weak dependence of fuel spacing on its enrichment for high-density cooling pool compared to undensified one.

The considered method of high-density fuel storage become more and more advantageous as it makes possible the increase of the pool capacity with the growth of fuel enrichment.

It should be noted that the location of VVER-1000 and VVER-440 fuel assemblies in hexagonal baskets-absorbers made of boron steel with 1 % - boron content providing for required subcriticality permits to increase by a factor of two the density of fuel assemblies in the cooling pool and to ensure nuclear safety in emergency conditions when the density of coolant decreases.

Calculations and experiments should be continued in order to evaluate the nuclear safety of high-density fuel storage and to study the possibility of nuclear accidents in low-density cooling pools for development of measures ensuring the safety of fuel storage.

6. Main trends of investigations.

In the USSR wet storage is the basic method and, evidently, it will predominate during next 20-30 years. At present it is based on a more developed technology ensuring safe and fairly economic storage of spent nuclear fuel /9/.

As the necessity of long-term storage of spent nuclear fuel became evident, extensive studies of different methods of intermediate storage were initiated in the USSR. The most prompt solution of this problem consists in locating additional amount of spent fuel in at-reactor pools and away-from-reactor storages. The capacity of cooling pools can be increased by 30-50 % due to the growth of fuel storage density by filling unused space; so, the density of spent fuel in intermediate storage facility at RBMK reactor is now 3 t/m³, and the works are conducted to increase the density to 4-5 t/m² by siting additional amount of cans with spent assemblies /7/.

The optimization of fuel use technology allows us to reduce the demand for capacity of at-reactor pools by 10 %. At a number of NPP with VVER reactors the construction of storage racks has been changed, and the second tier is used for short-term storage. Additional increase of fuel storage density in water-filled pools

can be achieved by the disassembly of fuel bundles and by the storage of RBMK assemblies in the form of consolidated matrices. Research works on fuel storage densification include also the investigation of storage density dependence on the fuel burn-up. At VVER reactor units the racks made of boron-containing materials absorbing neutrons are used /10/.

The possibility of prolonged fuel storage stimulates further research directed to the improvement of technology and to the reduction of both capital and operational costs.

The objectives of this research are:

- to study spent fuel behaviour during the storage time more than 10 years, especially for the fuel with damaged cladding;
- to improve water quality specifications and to refine the means for maintaining high water quality;
- to study the sludge migration for the development of recommendations on improvement of radiation conditions in fuel storages;
- to continue the study of cooling pool construction material behaviour including the problem of replacement of the existing materials by cheaper ones.

The investigation on dry storage technology will be carried out in order to show the efficiency and the economy of this method as well as the possibilities of its use for long-term storage of RBMK spent fuel as its reprocessing is now inexpedient.

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DESIGN AND CONSTRUCTION OF AFR SPENT FUEL STORAGE FACILITY AND AUGMENTING EXISTING POOL CAPACITIES IN INDIA

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Abstract

Tarapur Atomic Power Station fuel pools were designed to store only 528 spent fuel assemblies. It was envisaged to reprocess these assemblies after 100 days of cooling. As reprocessing could not be taken up storage capacity of the pools was increased by about three times by replacing old aluminium racks with high density stainless steel racks. Now an AFR Storage Facility is being built to store the entire fuel assemblies that will be discharged during the 30 years operations of the two reactors at Tarapur.

At Rajasthan Atomic Power Station the storage capacity of the pool has been increased by storing more spent fuel trays at each location and reducing space between two adjacent stacks of trays.

1. INTRODUCTION

Fig.1 enclosed shows the INDIAN Nuclear Power Programme. This paper deals mainly with TAPS & RAPS fuel storage facilities which are subject to IAEA inspection.

PART-A

2. TAPS POOL

Tarapur Atomic Power Station (TAPS) was built with two dual General Electric cycle BWRs, each of 660 MWt (210 MWe) capacity and came into commercial operation in October 1969. Each of the reactor are loaded with 284 fuel bundles. Twin fuel pools in between the two reactors were initially designed to store 528 fuel assemblies. The pools were sufficient to provide space for unloading of the whole of one reactor core in case of any exigency, and space for storage of a few batches of spent fuel bundles discharged from the reactors during annual refuelling outages. It was envisaged that

NUCLEAR POWER PLANTS

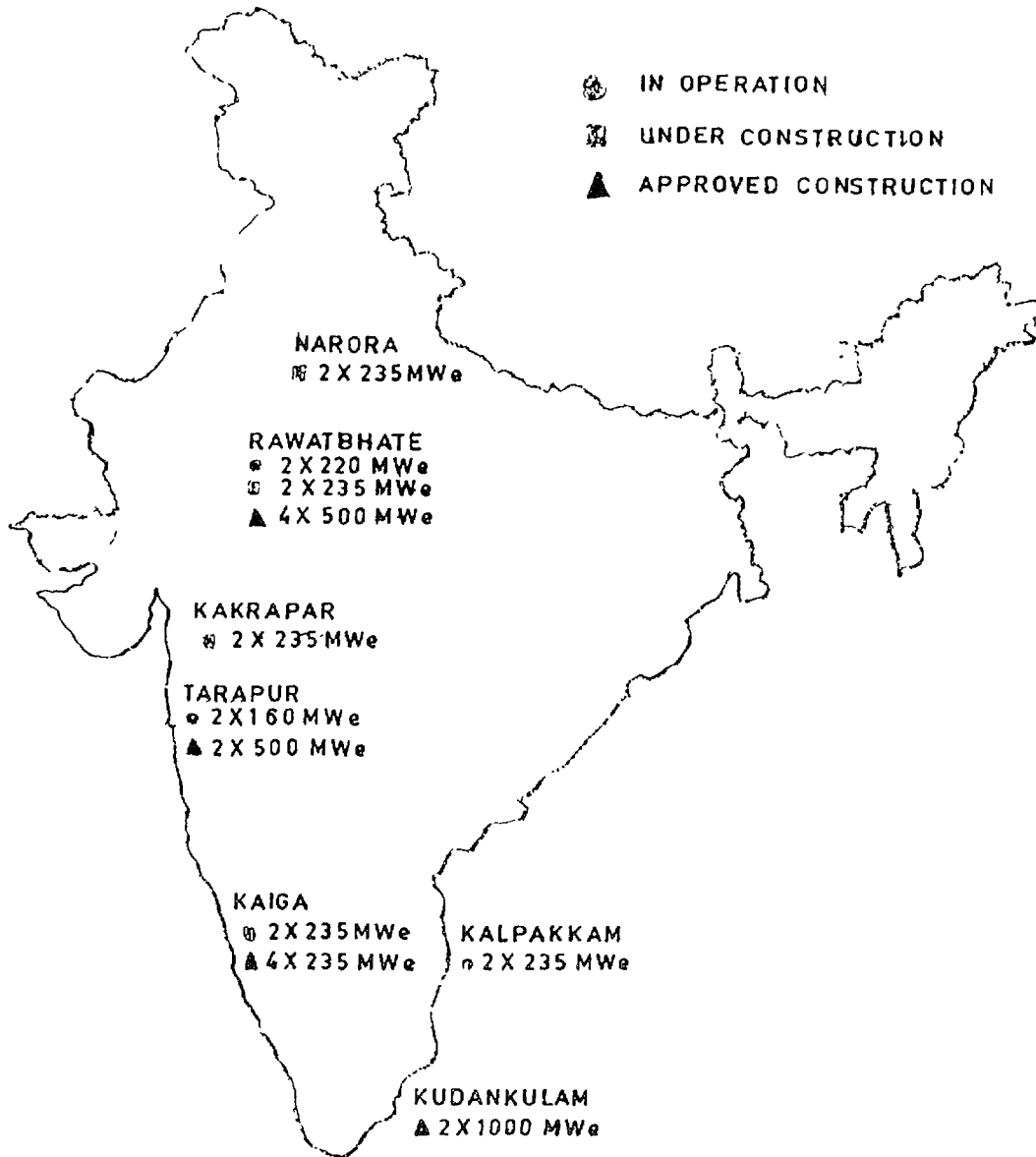


FIG.1. The Indian nuclear power programme.

the discharged bundles would be shipped to neighbouring reprocessing plant after 100 days of cooling in the fuel pools. But reprocessing of the irradiated fuel bundles could not be taken up as envisaged. On the other hand the pools were getting cluttered up with poison curtains removed after first refuelling fuel channels, fuel support plugs, incore monitors etc. Hence a number of remote operated tools were obtained from IAEA on loan to cut the irradiated core components mentioned above, underwater into small pieces and dispose them off in 200 litre drums. These items were occupying lot of floor space in the pool.

A detailed evaluation was made later to determine the extent to which storage capacity of the existing pools could be augmented by using different materials of construction for the storage racks. The study indicated

that a maximum storage capacity of 2156 spent fuel bundles can be created by installation of eleven Boral/SS with boron (poison) racks of 196 modules. High costs involved in importing boral racks/sheets made it uneconomical to go in for such racks. Further, fuel pool floor loading with poison racks was also found to exceed the permissible/safe load of the pools. The next best feasible alternative was determined to be installation of SS racks of 144 modules which could provide interim storage space for about 1500 fuel bundles. This scheme was adopted, vide figure 2 & 3. The capacity so augmented provided storage space for discharged fuel bundles upto the end of 1987.

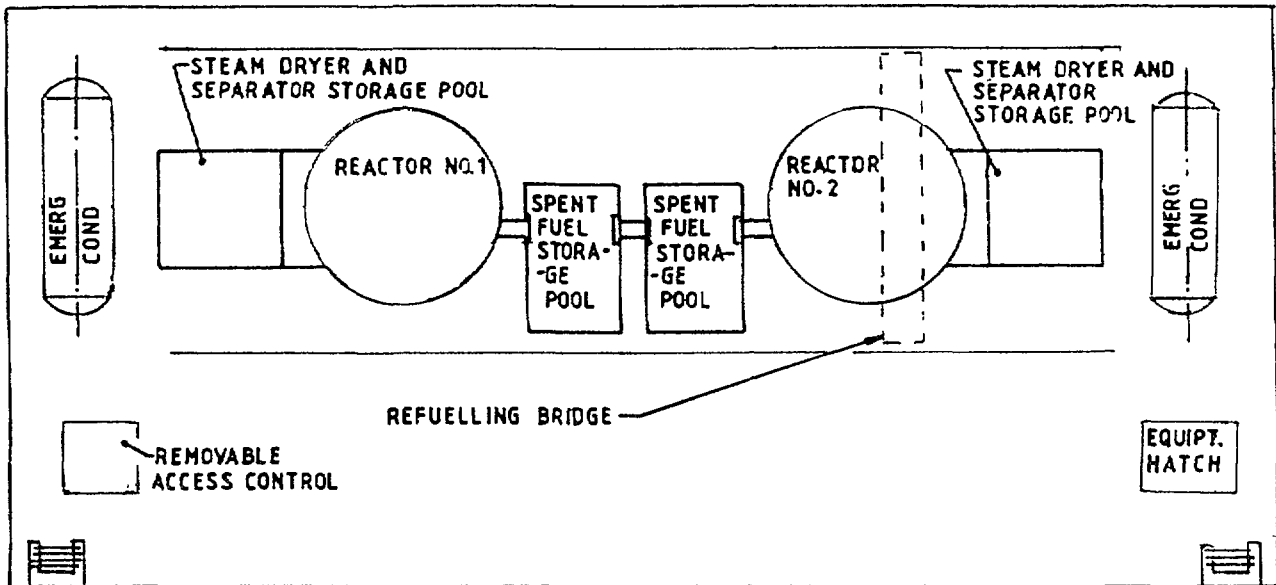


FIG.2. Layout of two reactors with fuel pool at TAPS (seen from 200' elevation).

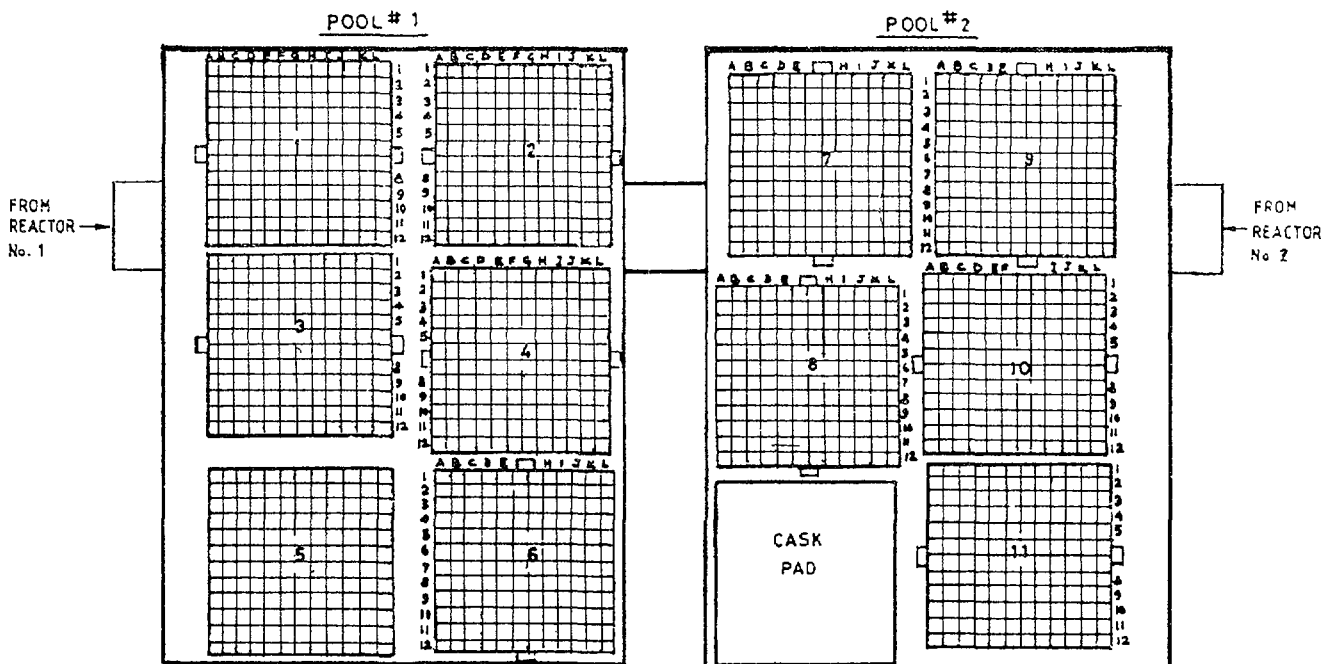


FIG.3. Tarapur atomic power station: fuel pool racks arrangement.

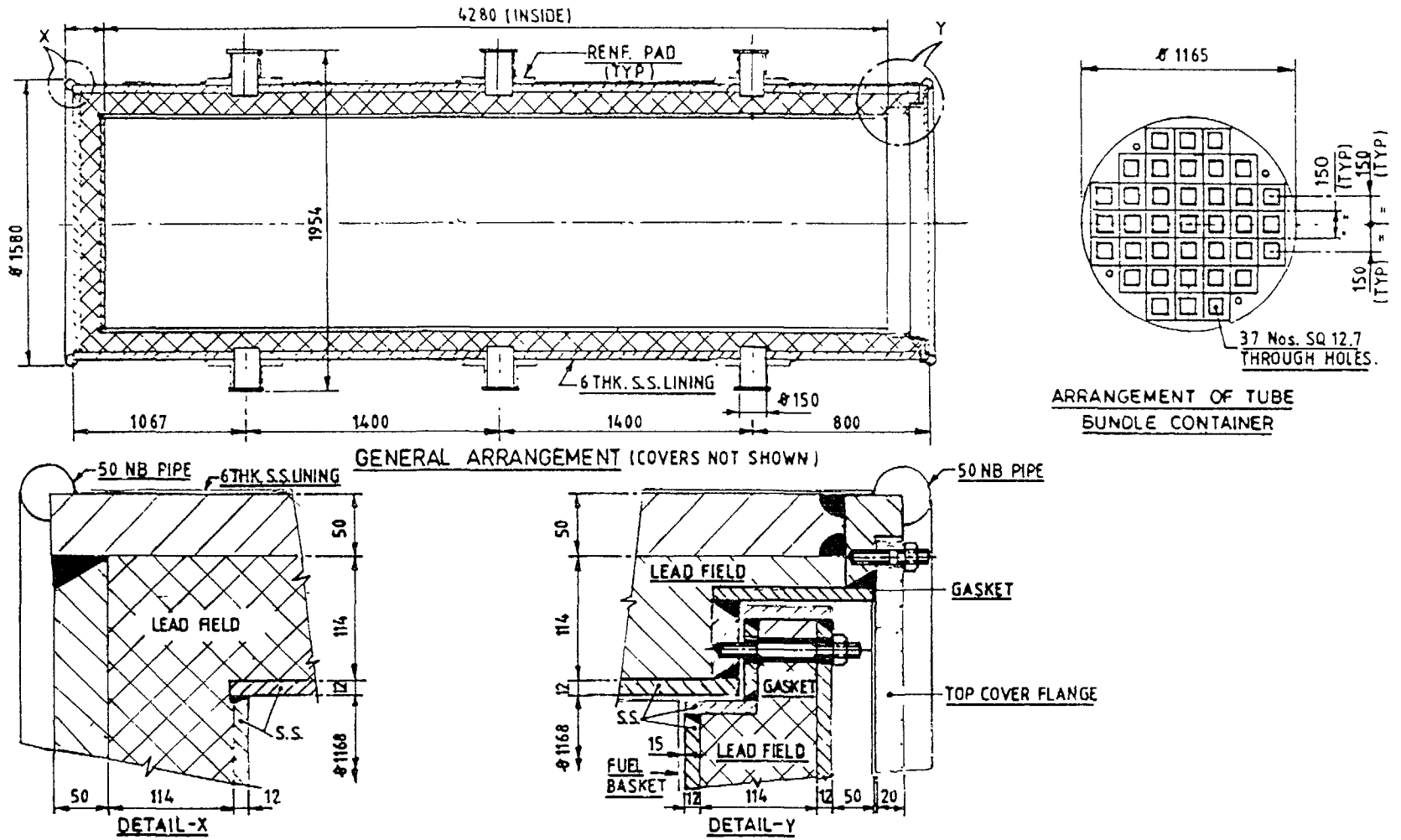


FIG.4. Spent fuel storage and handling cask for taps.

Later four dry storage casks, each capable of storing 37 fuel bundles were procured to store bundles, which have been cooled in the spent fuel pool for over ten years. The design and fabrication of these casks shown in Figure 4 were done indigenously. With these casks and some vacant locations in the pool, we will be able to manage refuelling operations till 1989. The high discharge rate of fuel is due to defective bundles received in early years of operation and also due to floatting guide tubes damaging the bundles which in turn was due to insufficient design of hold down devices of peripheral guide tubes, which were subsequently rectified.

3. FEASIBILITY STUDY

A feasibility study for construction of an additional storage facility was undertaken in 1983-84. A total number of 3312 spent fuel assemblies would be discharged from the two reactors at TAPS during their 30 years of operation. The two options of water-cooled and air-cooled, for spent fuel storage were examined. Even though theoretically vault type dry storage mode offered advantages such as economy, modularity, minimum environmental impact, safety, low man-sievert dose commitments, very little staffing and almost zero radioactive waste, it was decided to go for a wet storage facility because of the following reasons:-

- Concept of drystorage of BWR fuel is new & problems that might be encountered in future are not known.
- Wet storage has proven technology
- Drystorage needs a complicated remote operated fuel handling machine which would have to be imported.

A survey was made to find a suitable location for AFR facility. It was finally located at a place which is about half kilometre to TAPS, close to a solid waste management facility and the approach road to TAPS. In view of close proximity to TAPS the location is suitable from the point of view of supply of water and power as well as security and surveillance.

4. DESIGN FEATURES OF THE STORAGE FACILITY

Civil works and structures.

The AFR spent fuel storage facility has following civil structure and is shown in Fig. 5 & 6.

- i. Fuel pool & spent fuel building.
- ii. Service building.

- iii. Radioactive waste management building.
- iv. Foundations for acid/alkali storage tanks, transformer, cooling tower & other miscellaneous structures.

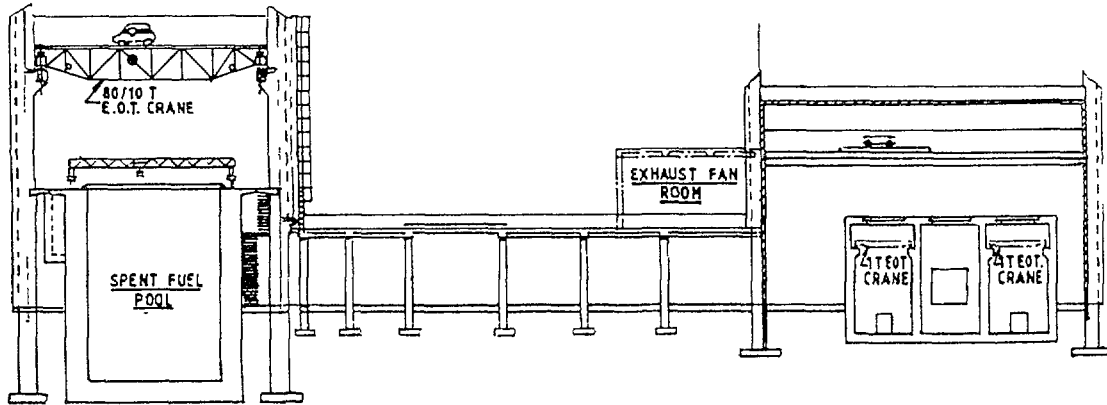


FIG.5. AFR spent fuel storage facility for Tarapur atomic power project: general arrangement.

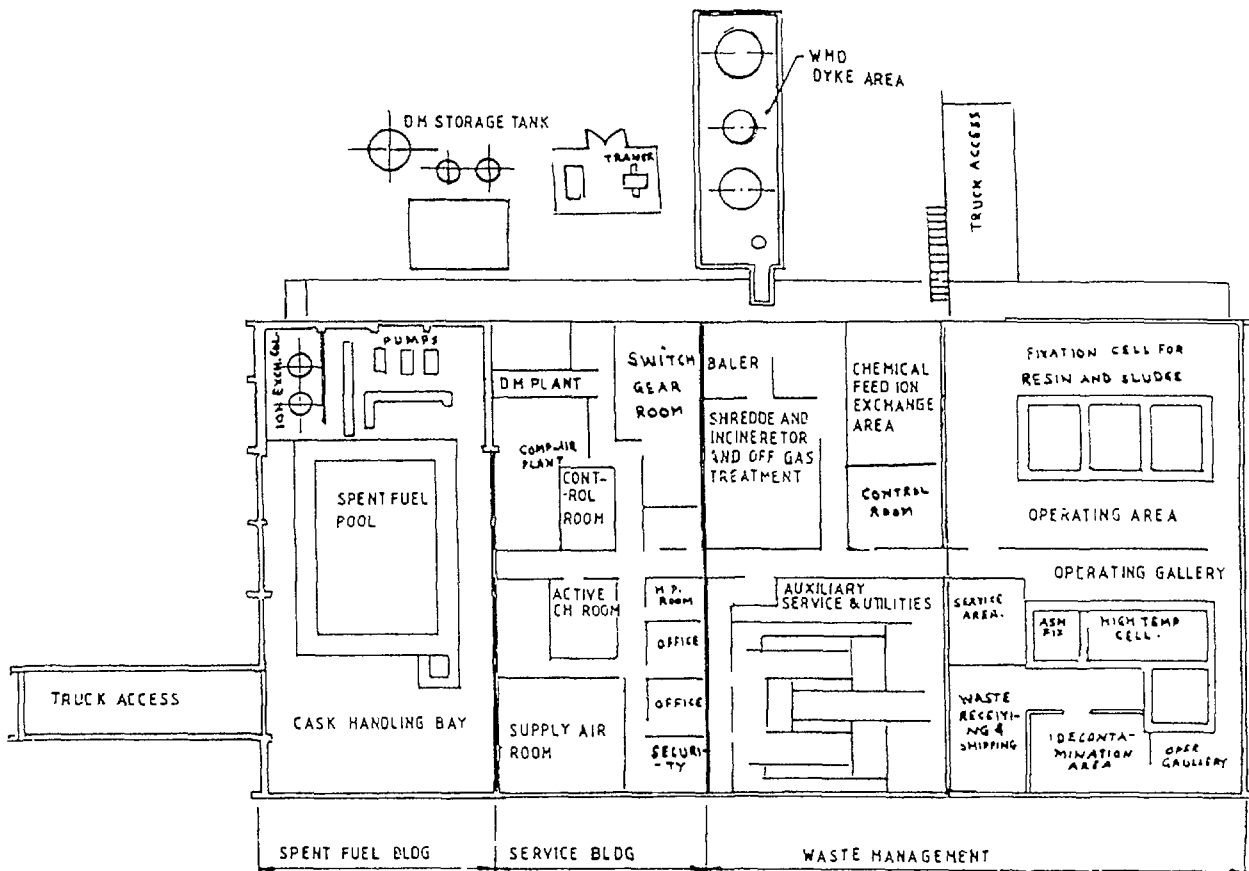


FIG.6. General arrangement (plan)

The safety principles that was considered in the design of the AFR facility are as follows:

- i) To prevent release of radioactivity beyond acceptable limits to the environment during normal operating and fault situations.
- ii) To protect the enclosed facilities and systems from hazardous external loading.
- iii) To ensure that the expected direct radiation dose rate from stored fuel assemblies does not exceed acceptable limit.
- iv) To provide necessary biological shielding.
- v) To house, segregate and protect the fuel assemblies, their storage facility, handling equipment and other enclosed plants in a suitable environment: in conjunction with ventilation and cooling systems.
- vi) To provide access for the receipt and despatch of Spent Fuel.
- vii) To prevent spread of fire from localized potential sources.

5. STRUCTURAL AND CIVIL DESIGN

The structural design of the building took into consideration following loads:

i) Live Loads: The general loading at ground floor is 5T/sq. mtrs. The vehicular approach area and the air lock in ground floor are designed to take tractor trailer load with cask. Provision to store 2-3 casks on the ground floor has also been made. The cask storage area, cask decontamination area, fuel pool & air lock area designed for loading of 50 T per square meter. The floors of the service building where a heavy or moving equipment has been installed are designed for loading 5T/sq.M. The other area in the service building are designed for 1T/sq.M.

ii) Seismic Design: The facility falls in an area which has maximum horizontal ground acceleration of 0.2g for safe shutdown earthquake (SSE) and 0.1g for operating basis earthquake. Depending upon the safety requirements, the buildings have been classified into three categories i.e. SE-1, SE-2 & general categories. Spent fuel building has been put in SE-1, dynamic analysis for which has been carried out for OBE level of ground spectra with 5% damping and also SSE level of ground spectra of 10% damping as exciting motions in three orthogonal direction. Spent fuel pool has been put in SE-2 classifications. Dynamic analysis of this

building was undertaken for SSE level of ground spectra with 10% damping, as exciting motion in three orthogonal directions.

The rest of the building have been designed as per provisions of Indian Standard code with Importance Factor of 1.5. As part of the fuel pool structure is 5 metres below the ground and fuel pool wall thickness is as high as 1.5 metres, the spent fuel assemblies remain submerged under water in the adverse circumstances even if water drains from above ground level.

iii) Wind Pressure: Design wind speed for AFR Storage Facility site is 200 KM per hour at an elevation of 30 M from the ground level. The corresponding wind pressure, 178 kg/sq. meter has been considered in design.

iv) Flooding: The protection against the flood has been achieved by raising the level above the highest sea water level. The protection against the flood due to precipitation of rain water has been achieved by providing storm sewer of adequate size.

6. FUEL POOL AND RACKS

The storage pool is 9 M wide x 13 M long x 13 M deep. It provides space for storage of 3312 bundles that will be discharged during 30 years operating life of Tarapur Atomic Power Station (TAPS). The depth of the pool is 13 meter which accounts for 4.4 M for fuel height, 5.0 M for the cask height, 2.6 M for the shielding and 1 M free board. The pool is designed to take load of a cask weighing 70 tons.

The pool is lined with stainless steel conforming to ASTM-167, type 304L. The sheets are annealed & surface cleaned on both sides. The stainless steel sheet thickness at the cask loading area is 25 mm thick. Stainless steel liner is supported from mild steel tee bar embedded in concrete. All welding on the liner is done by TIG method using SS filler metal. All the weld joints are inspected by dye penetrant and vacuum box checks. There are weep holes to monitor leakage past the liner when the fuel pool is filled up with water.

Before lining the pool the concrete structure was checked for leakage by initially filling water outside the pool upto ground level. This was done to ensure that no ground water leaks in the pool. Later testing was done by filling the pool fully. Many wet patches and leaks were noticed on outer surface of the walls. The cement grouting initially at a distance of one meter was done on almost all the construction joints. Later this distance was reduced to 0.5 meter for the locations where the leaks persisted. Polyurathine was also used along with the cement grout. As some wet patches on the wall still persists, assistance of a company in UK

having experience to stop leaks through the concrete structures such as, biological shield and fuel pool are being taken to fix the leaks.

The fuel racks of design similar to that of TAPS are to be utilised in the pool. Each rack can accommodate 144 spent fuel assemblies and has stainless steel as construction material. (Fig.7.)

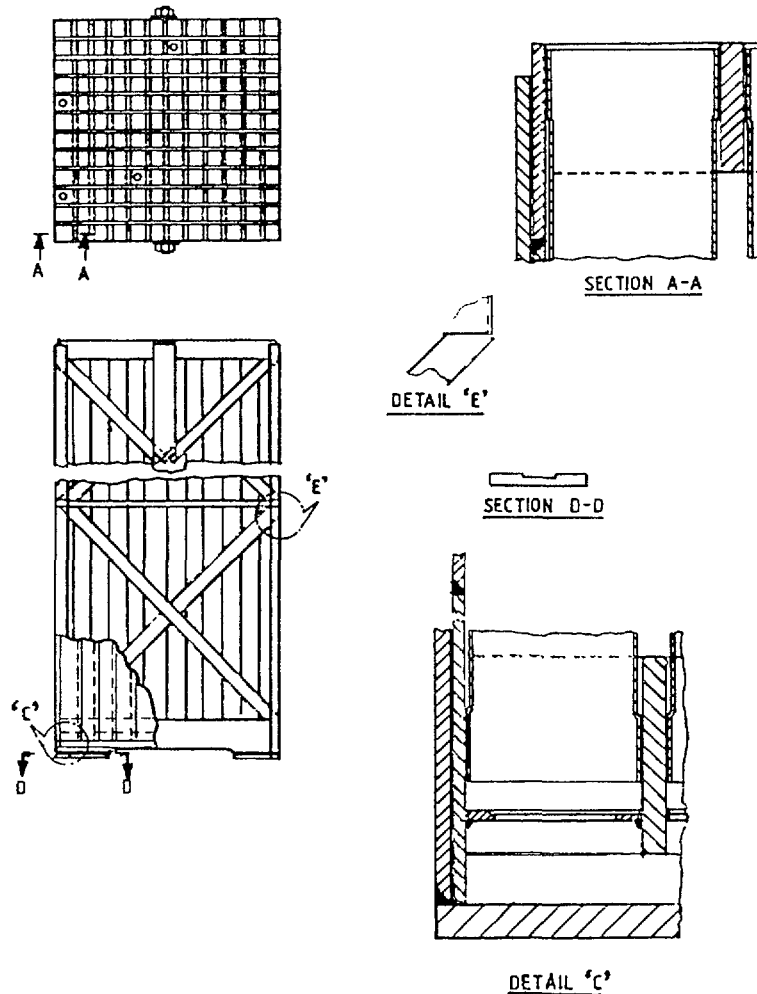


FIG.7. 12 x 12 rack outline used at TAPS.

Criticality analysis calculation for the racks have been performed by GE, USA with MERIT computer programme, a Monte Carlo programme which solves the neutron transport equation as an Eigen value or a fixed source problem, including effects of neutron shielding. This programme is specially written for analysis of fuel lattices in thermal nuclear reactors. The maximum calculated K for normal conditions is less than 0.89. It is expected that abnormal conditions such as fall of a fresh fuel from 8 meters or of an irradiated fuel from 4.5 meters will not increase K to a value greater than the limiting value of 0.95.

7. FUEL HANDLING SYSTEM

The transportation of spent fuel bundles from TAPS to the new storage facility is done in a lead cask of dimension 1945 mm x 1405 mm x 5143 mm weighing about 70 tons. The cask is transported to AFR storage pool by means of a tractor trailer. An EOT crane of 80/10 tons capacity has been procured to handle cask in the AFR facility. The crane serves the following functional requirements.

- to lift the 70 Te cask horizontally from trailer and locating over trunion, which is placed on cask loading/unloading bay.
- to revolve the cask by 90 degree on the trunion and make it vertical.
- to lift the cask and locate it inside the pool over raft of the storage pond in the space earmarked for cask placement (cask pad).
- to remove the lids underwater. After removal of the lid, the fuel assemblies are removed one by one by a special purpose grapple mounted on a fuelling bridge and the fuel assemblies placed in predetermined position in SS racks in the pool.
- to put back the lid on the cask, remove cask from the pool and putting them on trunion to turn the cask from vertical position to horizontal position and loading the cask on tractor trailer.
- to handle heat exchangers, exhaust fan components, steel gratings, removable slabs, cation and mixed bed exchanger for purpose of maintenance.

As the main hoist of the crane handles 80 Te load containing irradiated fuel, the hoist has been provided with (a) normal hoisting/lowering speed, creep speed and inching speed. This has been done to avoid cask crash against the pool bottom because of high speed and cause damage to the liners.

Single failure proof features have been included in the crane so that any credible failure of a single component does not result in loss of capacity of the crane to stop and hold the load. Rope reeving system, brakes, sheave pins for the hook block, drum supports, gear sets between hoist motor brakes and drum have been duplicated.

Structural design of the crane has been checked for dynamic conditions corresponding to OBE level of earthquake. Some of the critical components such as crane rails, rail stops, brakes and panel of the crane are designed to meet more severe seismic, i.e. SSE requirements.

8. IMPORTANT AUXILIARY SYSTEMS

8.1. FUEL POOL COOLING SYSTEM (WIDE FLOW SHEETS)

A table giving the decay heat of the discharged bundles at 20000 MWd/ton exposure is given in the Table-1. The decay heats for a given condition has been evaluated from the predictions using computer code "ORIGIN". The decay heat after one year of cooling to 9 years cooling varies from about 1 KW/bundle to 0.1KW/bundle. The design of the cooling system

TABLE I. FUEL HEAT EVOLUTION FROM DISCHARGED BUNDLES AT 20 000 MWd/t EXPOSURE

Time from discharged	Heat content Btu/hr/Te	KW/bundle
9 days	264×10^5	1088.29
34 days	1.4×10^5	5.77
184 days	46056	1.89
1 year	26666	1.09
1.5 year	17889	0.73
2 years	13056	0.54
2.5 years	9816	0.40
3 years	7811	0.32
3.5 years	6256	0.25
4 years	5033	0.20
4.5 years	4411	0.18
5 years	3844	0.16
5.5 years	3511	0.14
6 years	3222	0.13
6.5 years	3050	0.12
7 years	2900	0.12
7.5 years	2867	0.12
8 years	2722	0.11
8.5 years	2640	0.11
9 years	2562	0.10
9.5 years	2444	0.10
10 years	2367	0.097
10.5 + years	2250	0.093

envisages storing of spent fuel bundles, which have already been cooled for 9 years in the pool in the station. The fuel pool cooling system is initially designed to cool 2000 bundles stored in the pool with provision to augment it at a later stage to cool 3312 bundles.

The design envisages to maintain fuel pool water temperature at 42 degree C under normal operating condition. Pool temperature is likely to rise 0.5 degree C per hour in event of loss of pump due to power loss. A detailed calculation, taking into consideration of heat dissipation of pool water through conduction, convection and radiation indicated that temperature of the pool would not exceed 60 degree C if the power was not available for seven days. However, this temperature is likely to increase if power is not restored after seven days. The pool is likely to see the worst temperature potential across the pool of 36 degree C due to loss of power for seven days. Therefore, the design temperature differential across the pool has been taken as 40 degree C.

8.2. FUEL POOL WATER POLISHING SYSTEM (WIDE FLOW SHEETS)

The rate of radioactive buildup in the fuel pools at TAPS is around 0.5 curies per day. The activity release in the new pool is expected to be around 2 Ci/day taking into consideration the decreased heat load and fission product inventory as a result of radioactive decay. The polishing plant basically cleanup the spent fuel storage pool water. It is proposed to have a flow of 30 cubic metre/hr. i.e., same as fuel pool cooling system. Hence, both fuel pool cooling and polishing system are in series. There are no separate pumps for them.

Under normal operating conditions of the spent fuel pool, the ionic impurities will be mainly from dissolution of atmospheric gases. Possible contamination may also occur during cask and fuel handling operation. During normal storage conditions of spent fuel pool, the turbidity is considered to be 50 ppb comprising mainly of insoluble iron (Fe) which may increase to about 500 ppb during fuel movement periods. Fuel pool water cooling & polishing system has been shown in Fig.8.

8.3. COMPRESSED AIR SYSTEM

It has two compressors, each of 100 cfm capacity and develop 7 kg/cm.sq. pressure. The system additionally consists of after coolers, air receiving tanks, moisture separators and a duplex air drying unit with automatic change over facility. The system supplies both service and instrument air in the plant.

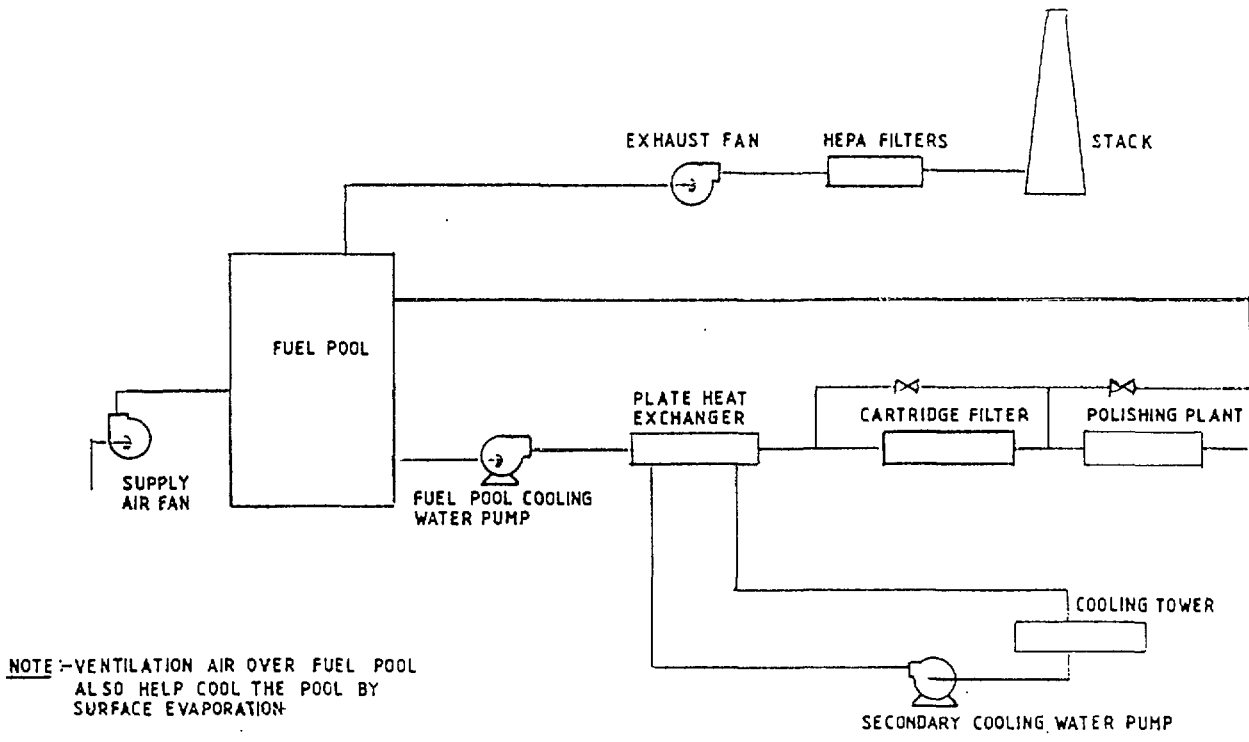


FIG.8. Line diagram of fuel pool cooling and polishing plant.

8.4. VENTILATION SYSTEM (WIDE FLOW SHEETS)

Ventilation system in the plant has been divided into two types, one for the active areas and the other for inactive areas. The active areas comprise of waste management facility, spent fuel building and active change rooms in the service building. All other areas in the service building, housing DM plant, compressed air plant, inactive maintenance room, switch gear, battery room, inactive change room and toilet are considered inactive area.

The ventilation system for the active areas is so designed that the air exhausted is more than the air supplied there by eliminating any possibility of leakage of active air outside the building, and in inactive area, air supplied is more than the exhausted thereby eliminating the possibility of ingress of dust and active air inside the inactive areas. The above philosophy has been adopted to ensure that flow of air from the active zone to inactive zone is completely eliminated. (Fig.9 enclosed gives the flow sheet).

8.5. ELECTRICAL SYSTEM

Total power requirement of the AFR facility is 534.9 KVA. This includes maximum demand of 227.3KVA for waste management facility and 75 KVA of illumination

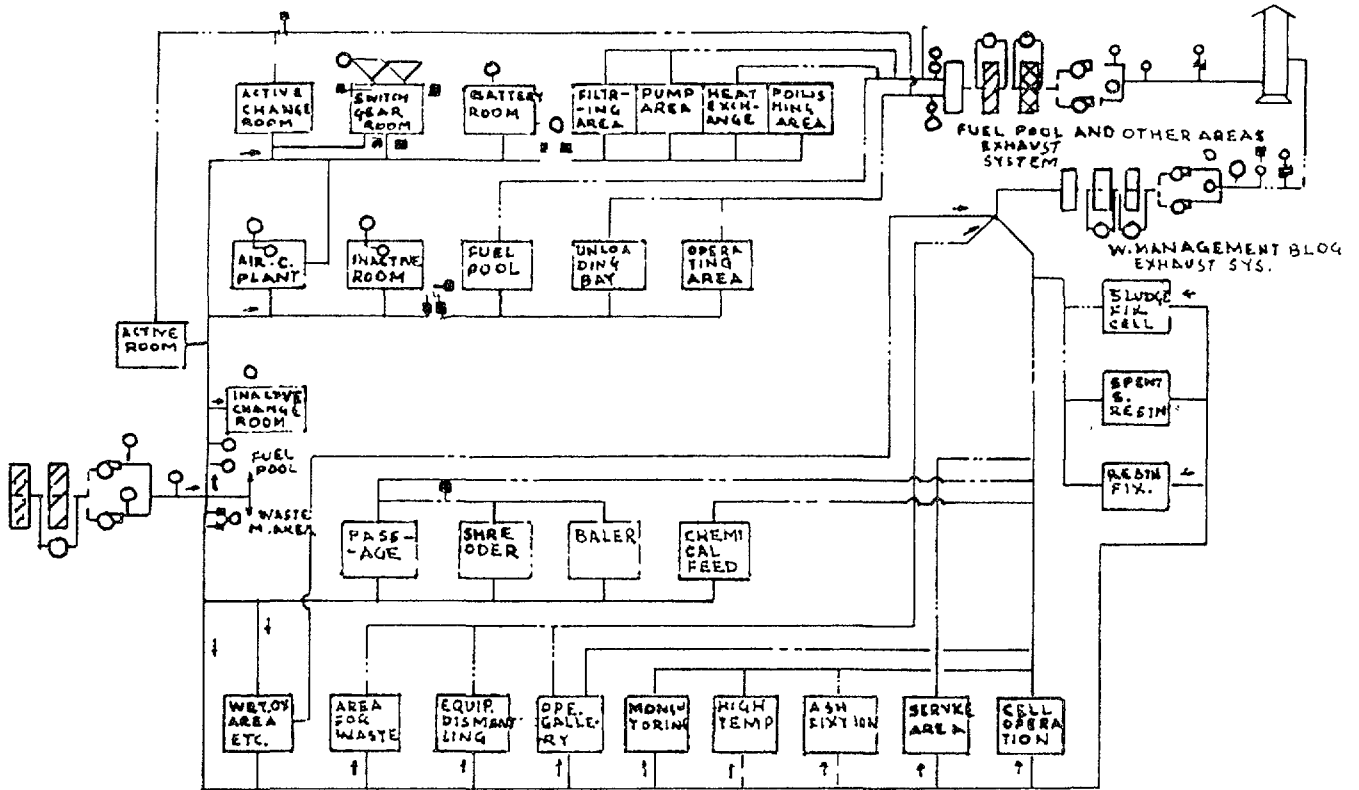


FIG.9. AFR spent fuel storage facility for Tarapur atomic power project single line flow diagram ventilation system.

load. Power supply at 3300 volt, 3 phase, 3 wire is received at AFR facility through an underground cable and is transformed to 415 volt, 3 phase, 4 wire for distribution in the AFR facility. A second cable from a different bus has also been laid to supply emergency power upto 60 KVA.

8.6. FIRE PROTECTION SYSTEM

Adequate number of fire extinguishers have been provided all over the plant for protection against fire. In ventilation system fire dampers have been provided both in the main supply and exhaust duct. Fire dampers provided in the supply duct are of fusible link type followed by motorised type. The motorised fire dampers are operated with the sense of rate of rise of temperature detectors and/or smoke detectors, located at various strategic locations inside the spent fuel storage facility.

8.7. WASTE MANAGEMENT FACILITY

It is fully equipped to process both radio-active liquid and solid wastes generated by the AFR facility. It is also equipped to process the resins and sludge discharged from TAPS. The equipment/facilities installed in the WMF are Resin Fixation Cell. Resin

Storage Cell Sludge Fixation Cell, Material Handling System, Ash Fixation Plant, Chemical Feed Ion-exchange Plant, Baler, Incinerator, off gas system Wet Oxidation Plant, High temperature volume reduction, Densification, Decontamination Centre, etc.

PART-B

Augmentation of storage capacity of PHWR Spent Fuel Storage Pool at RAJASTHAN ATOMIC POWER STATION (RAPS)

Station at RAPS consists of two CANDU type 220 MWe PHWRs. The pool was designed to store bundles with idea that the bundles will be sent to processing plant near TAPS after a few years, of cooling. This facility got delayed and priorities got altered to process MAPS fuel. Acute shortage of space for storage of spent fuel bundles was experienced at RAPS. Initial arrangement envisaged storage of 10 racks, each with 11 fuel bundles at one location. After calculating the minimum water shielding requirement for the spent fuel bundles, loading capability of the pool raft and checking for the possible criticality, the stack height of the racks are being increased in steps to 30 racks. This has helped to increase the storage capacity by about 3 times. The space between racks are also envisaged to be reduced. This will further increase the storage capacity of the spent fuel pool by about 1.2 times. Thus, the capacity of the pool has been increased from about 10000 to 32200. The old and new arrangements of storage have been shown in Figures 10 & 11. The process of shuffling

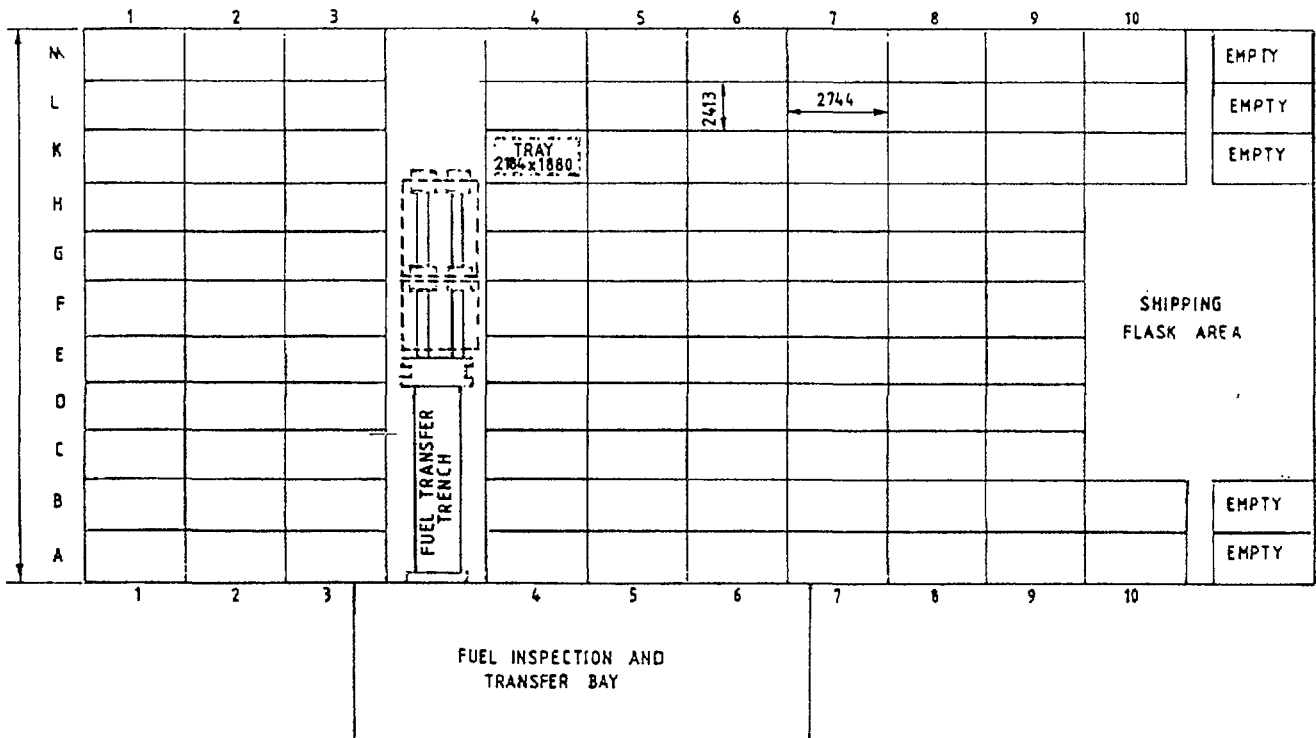


FIG.10. RAPS fuel storage bay grid arrangement (present).

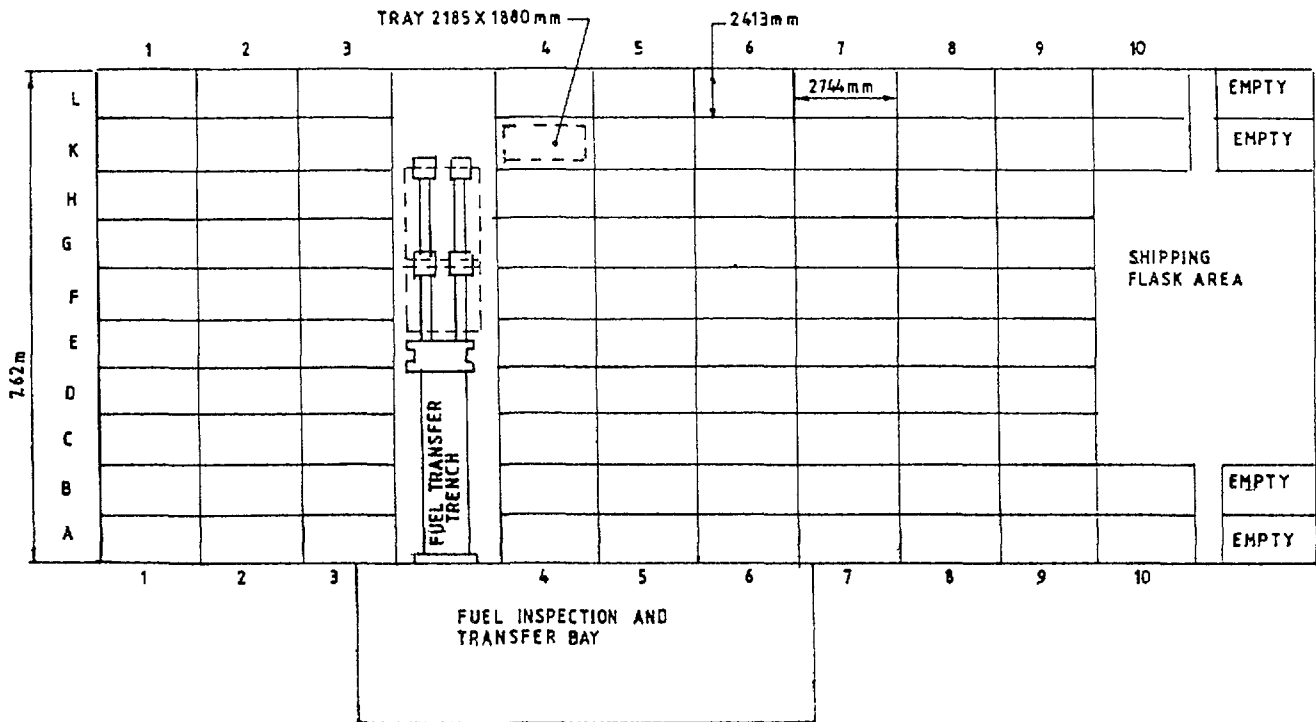


FIG.11. RAPS fuel storage bay grid arrangement (old).

of spent fuel racks is completed 75% and by August 89 we expect to complete entire shuffling. The storage position which had become very acute has now considerably been eased for the next several years because of these rearrangements.

Provision exists to store RAPS spent fuel bundles at the AFR spent fuel storage facility at Tarapur in case of exigency. The shipping casks for spent fuel bundles of RAPS are rectangular in shape and have larger bearing area than TAPS spent fuel casks which are cylindrical and kept vertically. The cask pad has been designed in a manner that it can take both TAPS and RAPS casks, one at a time. The arrangements of storage of TAPS fuel alone and TAPS& RAPS fuel bundles are shown in Figures 12 & 13.

Other stations: With the experience gained and delays in putting additional spent fuel reprocessing plants all station from MAPS onwards the storage capacity of the pools is increased to 12-15 years of discharge from the twin reactor station. A fuel reprocessing plant near TAPS Kalppakkam is fast getting completed and expected to be ready by 1990 to process fuel from MAPS.

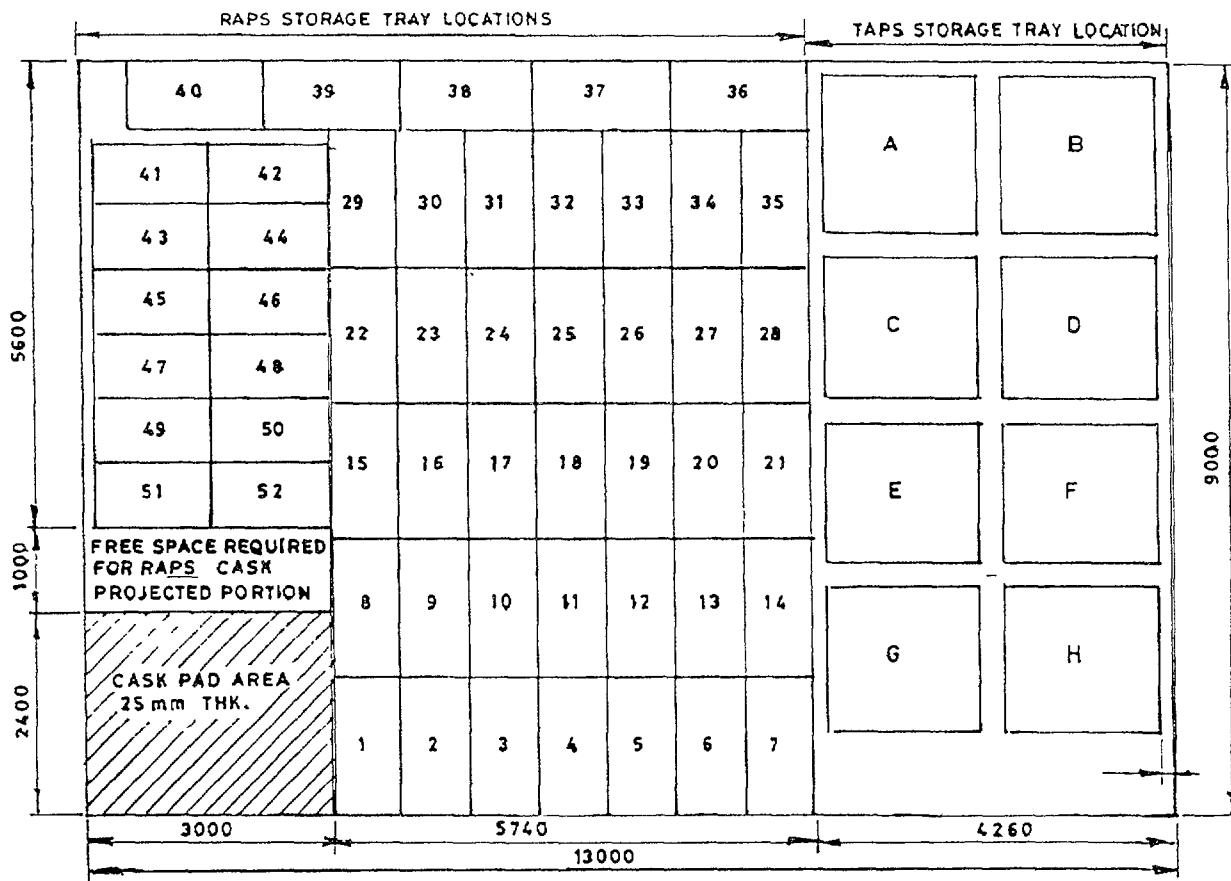


FIG 14. TAPS AFR spent fuel storage facility. Storage scheme for TAPS and RAPS fuel bundles.

SUMMARY OF THE MEETING

The meeting agreed that in view of the growing amount of spent fuel, technologies for storage of spent fuel prior to reprocessing or final disposal, are of high importance.

It is recognized that several technical solutions exist but depending upon the individual circumstances, e.g. licensing conditions, an optimum solution can be selected. Therefore, it is of importance that different technologies are available to the users for making an optimum choice.

The state of the art in the following areas has been established:

- rod consolidation;
- high density racks;
- fuel handling;
- metal transport/storage casks;
- casks for long-term storage;
- silos;
- vaults.

From this it appears that work in the following areas should be continued or initiated for optimizing spent fuel storage systems:

- Dry and wet rod consolidation techniques, particularly with respect to augmenting storage capacity for spent fuel and conditioning of "Non Fuel Bearing Components" (NFBC).

Rod consolidation, when fully demonstrated, could be a valuable technique for expanding existing or setting up new spent fuel storage facilities and for final disposal programs.

High density racks including burnup credit permit a significant increase of the capacity of a storage pool without changing neither the pool geometry nor the fuel assembly geometry. The use of burnup credit is of great interest and could be promoted by improving criticality calculations for burnup credit and continuing work on burnup verification techniques.

- Optimization of fuel handling operations. The objective is to minimize the risk of damage of fuel and components and release of radioactivity.

- Continuation of metal storage cask development should be directed toward the following objectives:

. increasing the capacity of storage casks by various means including the use of burnup credit;

. qualifying metal storage casks, (which have not been previously licensed) for use as dual purpose transport/storage casks provided that this does not impair the economics for use of the cask for storage.

- Development of transport/storage casks also being suitable for final disposal preferentially using consolidated fuel and taking into account repository material requirements. This technology will allow the introduction of long-term storage.

- Optimization of the dry spent fuel storage technology at reactor to meet particular needs of potential users is recommended.

- Dry spent fuel storage in silos at reactor has found wide application for CANDU fuel and is in its early stage of implementation for Light Water Reactor fuel.

- Wider acceptance could follow for LWRs, once experience is gained from ongoing applications and new concept demonstrates their effectiveness both technically and economically.

1) Rod Consolidation

Rod consolidation programs both on dry and wet conditions are under progress in the United States, France and the Federal Republic of Germany (FRG).

A dry rod consolidation equipment is being tested with spent fuel for the US Department of Energy in Idaho. Six in-pool rod consolidation demonstrations with spent fuel have also been conducted. Rod consolidation ratios up to 2:1 have been attained with irradiated spent fuel. As there is very limited experience in treating, packaging, and

storing the Non Fuel Bearing Components (NFBC) waste resulting from the consolidation process, some concern about the impact on plant operation has been expressed.

Dry and wet rod consolidation prototypes are also being developed in other countries, including FRG and France (in association with a US Company), and hot demonstrations anticipated in the near future. These prototypes incorporate systems for NFBC reduction of volume, either by mechanical compaction in FRG, or by shearing in France.

2) High density racks

The installation of high density storage racks is normally the first choice when an expansion of the storage capacity of existing storage pools is considered since it is a well established technology.

The high density is normally accomplished by the introduction of neutron absorbers in the racks. Different proven materials are available, such as borated stainless steel, boraflex and cadminox, using boron or cadmium as the neutron absorber. The choice of neutron absorbing material depends upon the circumstances; but due to the wide experience already gained, the economic factor is probably the most important consideration. Other factors such as the foreseen operating life and the monitoring measurements also have to be considered.

Traditionally, storage pools have been designed to accommodate fresh fuel, which puts a limit on the acceptable degree of initial enrichment. For spent fuel storage pools in AFRs and in ARs this imposes an unnecessary restriction and limits the capacity of the storage.

It is thus considered appropriate to take credit for the fact that the spent fuel is less reactive. Burnup credit has been licenced in a number of PWRs in the US and for one PWR in Yugoslavia. Burnup credit is now considered also in Japan, Spain and Sweden. The need for burnup credit will increase with the projected higher fuel enrichment in future.

The criticality calculations for spent fuel is straightforward, but involves certain new aspects to be considered, e.g. for BWR fuel.

The burnup of spent fuel is normally well known and documented, and can be controlled by administrative means.

However, for an AFR or a reprocessing plant, receiving fuel from a number of different reactors an extra safety margin is obtained by verification of the burnup by measurement. Different burnup measurement methods are available but certain development is needed to obtain fast and accurate readings.

3. Fuel handling

The purpose of the various fuel storage expansion methods is to store additional fuel assemblies. This necessitates an increase of fuel handling operations such as:

- fuel movements in the storage pools
- casks and fuel baskets loading and unloading
- fuel assembly inspections.

Specific and new fuel handling methods may be necessary. For example, two handling equipment were presented at this meeting:

- fuel elevator used at the CLAB facility for fuel transfer between the reception pool and the underground storage pool;
- under water basket equipped carriage at the storage facility in Japan used to transfer the fuel assemblies between
 - the cask unloading pool
 - the storage pools
 - and the shearing cell.

For operation related to rod consolidation such as fuel rod handling, rod canister manipulation, handling and gripping tools have been developed but some have to be hot tested in wet or dry condition.

Several other fuel handling devices have been in use for many years. The equipment are both known and available.

4. Metal Storage Casks

Designs of metal casks for use in spent fuel storage have been in existence since the late 1970's. The cask designs have different capacities, ranging from 4-26 PWR fuel assemblies to 10-60 BWR assemblies. Different designs have different gamma shielding materials

(nodular cast iron, forged steel, or Pb/SS), neutron shielding materials (polyethylene, polyester resin, glycol, polypropylene or boron resin 3) and basket construction (borated SS, aluminium-boron, SS-boron or stainless steel). The casks are generally equipped with a double-lid system to insure safe containment of contents. These casks have been subjected to a variety of tests and demonstrations since the early 1980's using both intact and consolidated fuel, and a number of different designs have been approved for use by different national regulatory authorities. The casks are being used, or planned for use, to supplement storage capacity needs in both AR and AFR facilities.

Results of demonstration activities have shown the following:

- radiation and thermal levels resulting from metal cask storage have been acceptable;

- no fuel rod failure has resulted during demonstration storage;
- no secondary wastes result from the storage operation.

Thus, metal cask storage can be considered to constitute a mature and approved technology that is available to meet future storage needs. It has the advantage of being flexible inasmuch increments of capacity can be added as needed. On the other hand it suffers somewhat from the standpoint of cost, although this situation may improve by the economies of higher production rates in cask fabrication. Nevertheless, country-specific considerations can significantly affect these costs.

Metal storage casks are also being considered for combined use for spent fuel storage and transport. Some of the existing storage cask designs may be used without modification, while other designs may have to be modified.

5. Long Term Storage Casks

Different concepts for final disposal of spent fuel are being investigated in several countries.

One of the concepts under investigation is the so-called tunnel storage concept, in which final disposal casks which are self-shielded and meet the IAEA transport requirements are placed into tunnels. As the casks have to be sealed for final disposal, they also can be used for long term storage in above ground installations and subsequent disposal in deep geologic formations. Two type of cask systems are at present under development:

1. The "Concrete Integrated Container" (CIC) is cylindrical in shape (2,6 m Ø) and has a height of 3,6 m. The fuel bundles are placed into a cylindrical steel chamber, surrounded by a 460 mm thick, reinforced high density concrete wall and an outer steel liner. The gross weight is about 70 Mg.

Loading of the CIC is performed under water in the storage pool and its outside surfaces are subsequently decontaminated and the stored fuel bundles dried. 2 prototypes of the CIC have been manufactured.

2. The "Pollux Cask" is designed according to the double shell principle, the inner steel shell providing temporary gas tightness by means of a screwed-in primary lid and long-term containment after welding in a secondary lid. The outer shell is fabricated in ductile cast iron assuring the necessary shielding under transport, storage and accidental conditions including type B (U) requirements.

The Pollux cask can store up to 8 consolidated PWR or 4 unconsolidated PWR fuel elements. The gross weight is 65 Mg. A prototype cask is in the manufacturing process and nuclear licence applications for transport and storage will be filed in the summer of this year. The welding process for the final, sealing of the secondary lid has been developed.

Loading of the cask is to be performed in a special conditioning facility comprising of a hot cell, which is in the licensing stage. The issue of the construction license is expected at the end of 1989.

In conclusion it can be stated that the development of final disposal casks as independent packages has reached such a state that the long term storage of spent fuel in above ground storage facilities can be foreseen.

6. Silos (concrete casks)

- Spent fuel storage technology in concrete casks at reactor (AR) found application in Canada and USA and is also under development in other countries.

WEPCO's Point Beach and Consumer Power's Palisades plants selected ventilated concrete casks - designated VSC-24 to expand their storage facilities. The VSC-24 containing 24 PWR fuel assemblies is mobile and cooled by natural convection.

B & W is developing similar concrete casks which are cooled by heat pipes. CONSTAR 125 will store 16, while CONSTAR 180 is being designed for 28 PWR fuel assemblies.

- A dry storage method for CANDU spent fuel in concrete canisters has been developed. Spent fuel is contained in welded stainless steel storage baskets which are placed into the cavity of vertical, cylindrical, stationary concrete canisters. Cooling is achieved by natural conduction to the atmosphere through the concrete.

Recently Point Lepreau station in Canada has opted for this method to expand its spent fuel storage facility designed for 10 years of reactor operation. The future CANDU stations designed by AECL are likely to have reduced wet spent fuel storage capacity (approximately 5 years of reactor operation). The fuel handling methodology for dry storage requirements are to be considered at the design stage.

7. Vault Storage

- Vault storage involves a concrete bay, where the fuel is stored in sealed canisters or in baskets. Cooling may be by natural or forced air circulation. In closed-cycle vaults the heat is transmitted to a heat exchanger, other operate in a once-through mode.

Vault concepts are in operation for HLW glass packages in France, Belgium and India, for defense HLW in the USA and for gas-cooled reactor fuel in the UK and USA for several years, e.g. the Marcoule storage facility started its operation in 1978 and the Wylfa facility in 1979. So long experience with dry vault system has been gained and today dry vault systems are available to provide additional at-reactor and away-from-reactor storage for LWR and HWR spent fuel. Methods which are approved by US NRC are the horizontal concrete modules concept and the modular vaults concept. The horizontal concrete modules concept was realized for H.B. Robinson Nuclear Power Plant in South Carolina, USA.

The license was approved in August 1986 and the first modules were loaded in May 1989. Other dry vault storage concepts are in the stage of license application and are ready for utilization. The estimated low costs for vault storage makes it attractive.

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