OPERATING EXPERIENCE IN WET AND DRY STORAGE

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RESULTS ON TECHNICAL AND CONSULTANTS SERVICE MEETINGS ON LESSONS LEARNED FROM OPERATING EXPERIENCE IN WET AND DRY SPENT FUEL STORAGE

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Abstract

Spent fuel storage has been and will continue to be a vital portion of the nuclear fuel cycle, regardless of whether a member state has an open or closed nuclear fuel cycle. After removal from the reactor core, spent fuel cools in the spent fuel pool, prior to placement in dry storage or offsite transport for disposal or reprocessing. Additionally, the inventory of spent fuel at many reactors worldwide has or will reach the storage capacity of the spent fuel pool; some facilities are alleviating their need for additional storage capacity by utilizing dry cask storage. While there are numerous differences between wet and dry storage; when done properly both are safe and secure. The nuclear community shares lessons learned worldwide to gain knowledge from one another's good practices as well as events. Sharing these experiences should minimize the number of incidents worldwide and increase public confidence in the nuclear industry. Over the past 60 years, there have been numerous experiences storing spent fuel, in both wet and dry mediums, that when shared effectively would improve operations and minimize events. These lessons learned will also serve to inform countries, who are new entrants into the nuclear power community, on designs and operations to avoid and include as best practices. The International Atomic Energy Agency convened a technical and several consultants' meetings to gather these experiences and produce a technical document (TECDOC) to share spent fuel storage lessons learned among member states. This paper will discuss the status of the TECDOC and briefly discuss some lessons learned contained therein.

1. INTRODUCTION

While there have been approximately 60 years of operating experience in wet and dry spent fuel storage without any incidents; it worth noting that Winston Churchill said "those that fail to learn from history are doomed to repeat it."

This quote brings up three questions:

- What defines our history?
- How does the nuclear community gather our history?
- What should be learned from our history?

Taking them in order, the first question is easily answered if you look at events like Chernobyl and Three Mile Island, which, clearly, are an important part of our history and tell us many things about how human action can shape the events at a nuclear facility. There are also numerous other actions, many of which are successes and best practices, from which a great deal can be learned. As the anonymous quote so eloquently states, "The smart man learns from his mistakes. The wise man learns from the mistakes of others".

Due to the large number of Member states represented at the International Atomic Energy Agency (IAEA), and the method of gathering information required by the IAEA, collecting our history can be a tedious task. While there are sources such as the Nuclear Energy Events Web Based System at the IAEA, not to mention individual country and utility websites, most of the information available on these platforms is related to accidents and incidents. There is not a significant amount of information available on best practices from which others can glean useful methods for licensing and operating spent fuel storage facilities. The only way to ensure the successes and best practices are available is to gather and write them in a document.

Given that an event, no matter how small, makes headlines in the area, if not the entire country, where it occurs, there should be a significant effort by the nuclear community to share best practices and lessons learned.

2. DOCUMENT INITIATION

In December 2007, the IAEA convened a consultants' meeting, "*Spent Fuel Storage Facility Operations: Lessons Learned,*" to discuss the best methods to obtain information for a technical document on lessons learned for operations of spent fuel storage facilities. The consultants developed a questionnaire that was formally sent to Member States that store spent fuel. The questionnaire requested experiences on all aspects of wet and dry spent fuel storage.

In looking back at the original questionnaire, one might wonder if it was too ambitious. The Terms of Reference for the consultancy to develop the questionnaire included: "one means of support is to review storage facility operations and to share lessons learned from practical experience among IAEA Member States, highlighting practices to emulate (and to avoid)". The key words here are "facility operations" and "practical experience". It is possible that the original questionnaire was broadened beyond the Terms of Reference in that it included items such as planning, licensing and constructing new dry storage facilities; and implementing modifications for both wet and dry storage facilities. In a very broad sense, these could be viewed as operations since they may be needed for continued operations, and might very well be useful for implementing these facilities, but when re-reading the original Terms of Reference could the questionnaire have been too ambitious?

Ten months later, a technical meeting was attended by 16 participants from 10 countries. Attendees provided either a presentation or verbal discussion of their experiences and perspectives (i.e., regulators vice licensees) in storing spent fuel. As with most lessons learned documents, these experiences were predominantly instances where regulations were not followed, or failure of fuel or equipment, which either led to or could have resulted in an exposure. These experiences were generically incorporated into an annotated outline.

Two subsequent consultants meetings were held in May 2009, and March 2010 to form these experiences into a technical document. After reviewing the information in the document while preparing for the second consultants meeting, it became apparent to some that there was a dearth of information. Without more lessons learned the document would have holes and not be as useful as other similar documents produced by the IAEA.

The first step in shaping a useful document is determining the audience, since the information included in the document should be determined by the intended audience. Regardless of the audience more lessons learned are needed to complete the document.

3. AUDIENCE

What constitutes a well-written lessons learned document? First the document provides best practices and lessons learned useful to the audience in sufficient detail for the readers to evaluate whether the information applies to their facility, and if so, apply it appropriately. The intent of this technical document, to cover both wet and dry storage, indicates that the audience would be both current reactor facilities and new entrants into nuclear power, as well as operating nuclear power plants that will need or have dry storage facilities to ensure sufficient spent fuel pool capacity for their reactor lifetime.

Finally, there are a number of power reactors worldwide that are shut down and in various stages of decommissioning that may desire to decommission their spent fuel pool and transition their spent fuel to dry storage facilities.

4. DOCUMENT CONSTRUCTION

The first consultancy was aimed at taking the experiences from the presentations given at the consultants meeting and forming a cogent document. Basic statements were extracted from the presentations and placed in the appropriate sections of the document. Based on experience, each consultant started writing a different section of the document. Since some of the experiences were negative, specifics such as country and facility names were omitted.

In preparation for the second consultant's meeting, it became clear to some that the document needed extra experiences and best practices to both complete the document and ensure that it is useful to readers. Additionally, some of the information included needs more explanation so that readers can apply it to their facility. With over 400 nuclear power reactors and 100 dry storage facilities worldwide, sufficient information exists to create a very useful document for both current nuclear programs and new entrants into the nuclear power community. The only question remains the best method for obtaining the best practices and lessons learned.

Two options were discussed, a second questionnaire and data mining. With the upcoming 2010 International Conference on Management of Spent Fuel from Power Reactors, the consultancy developed a second, simpler questionnaire narrowing the document scope to a general discussion focused on best practices. This second questionnaire would be sent to Member States and distributed at the conference, with the hopes of obtaining more information to place in the document. Additionally, it was simplified to request users to reference information that may be publicly obtained for inclusion in the document to minimize the amount of work for any responder.

In the event that the second questionnaire does not produce sufficient data to complete the document, two options were discussed for the third consultant's meeting: data mining and abandonment of the effort. The latter option would clearly be a last resort. Depending on the number of holes, data mining via the World Wide Web would be the preferred option.

5. LESSONS LEARNED AND BEST PRACTICES

While there are a number of items in the document that could be improved, there are best practices and lessons learned for both wet and dry storage operating experiences that would be beneficial to share throughout the nuclear industry. Below are some items for gleaned from the current document and when necessary, some clarifying discussion on needs for the item.

5.1. Wet Storage

Even with all the worldwide experience with SFM misloads still occur a few times each year, although not as frequently as in the past. There are a number of experiences related to misload events and some best practices on methods that have shown to work to minimize misload events:

- High initial enrichments and burnup credit coupled with high-density storage racks tend to increase the probability of a misload event. Extra care should be taken when moving spent fuel in this situation;
- Certain positions within the spent fuel pool that have been prohibited for storing spent fuel should be blocked by mechanical means to minimize the probability of loading a fuel assembly in one of these locations;
- Poor administrative burnup control coupled with burnup credit may lead to violations of pool design basis;
- High density racks due to design of the funnel require much more operator involvement and experience when loading the fuel assemblies into racks. Some explanation on the funnel design and how it affects high density storage racks might be appropriate;
- Every fuel assembly needs to have a specified storage location, and the history of the fuel assembly must be available in plant records and clearly specified for each location.

Some fuel designs are susceptible to inter-granular stress corrosion cracking and inter-granular corrosion cracking in the presence of aggressive contaminants such as chloride, fluoride, and sulphur compounds, which may be present in the spent fuel pool. The separation of the top nozzle occurred in several states due to a fracture at the bulge joint area of the sleeve. Detailed hot-cell examination revealed that the sleeve failed due to inter-granular stress corrosion cracking. The fuel assembly was handled with a special designed tool. There is discussion on pool water chemistry, its importance and suggested values for some parameters such as pH, conductivity, and contaminants (chlorides, sulphates, sodium, Caesium, etc.) for a variety of fuel designs.

Similar to the lessons learned in the U.S. during dry storage operations, one facility's experience lead to ensuring that staff is adequately trained in movement of spent fuel prior to re-racking operations.

Ensure proper precautions are taken to minimize the amount of foreign material that may enter the spent fuel pool. For example, during grinding of replacement doors facility operations failed to use sufficient protection measures such that debris from grinding entered the spent fuel pit. This resulted in more than 20 fuel assemblies that can no longer be inserted into the core due to damage.

There has been significant history of degradation of Boraflex and Carborundum, both of which are included in the document. While the degradation of these two materials is briefly described, the root causes are not provided.

A facility has found that their high burnup fuel tends to exhibit mechanical distortion such as bowing and S-shaped fuel, along with excessive build-up of crud. It is not clear whether this is a design flaw (material or structural), operational issue or normal operations for a particular reactor design. More information on this particular issue would benefit other facilities in determining the root cause.

Another facility experienced problems with spacer designs that are prone to damage during handling which could potentially lead to handling problems. There have been instances in the U.S. and other countries of fuel assembly problems with spacers during operations, but without additional information, it is not clear what might be the cause of these issues.

5.2. Dry storage

The most important aspect of planning for dry storage is timing. Early planning and communications with both the regulators and other members of industry are crucial to the success of any loading campaign, especially the first one. Facility upgrades, such as cranes or floor supports, will only add to the minimum amount of time needed to prepare for the first loading. Benefits could be gained in selecting already approved technologies. Given the potentially large list of facility upgrades and regulatory approvals, and that most licensees will know when their spent fuel pool will fill up, it is prudent to begin planning for an ISFSI a decade prior to the actual need date.

Generally speaking, spent fuel storage cask designers do not provide services for nuclear power plants, such as steam generators, cranes, and pumps, which may lead to situations where the storage cask designers, while very familiar with their designs, may not be sufficiently familiar with the plant's needs. Early discussions between plant personnel familiar and the storage cask vendor should occur so that the plant personnel become familiar with the storage cask design and its requirements and the storage cask vendor become familiar with the plants requirements and facility.

One facility indicated that dry storage is preferred over wet storage if the storage term is expected to be long (e.g., more than 10 years). While this may be true for some Member States; those who have sufficient storage space in their spent fuel pool may not want to bear the cost of implementing dry storage until plant decommissioning, if a solution to the spent fuel accumulations has not been decided.

One of the items suggested at the technical meeting was "Relevant US regulations were considered as authoritative." An important lesson to learn from this statement is that there are a number of member states and utilities that have significant experience in different aspects of the nuclear industry. Consulting with other Member States and organizations often proves beneficial in learning from past mistakes and may yield quicker improvements. An example of where the U.S. is implementing this philosophy is development of its knowledge base in reprocessing. The U.S. Nuclear Regulatory Commission will be participating in technical exchange meetings, facility tours, and conferences; and conducting information exchange activities with the U.S. Department of Energy, international regulators and operators of reprocessing facilities to expand its knowledge base using others who have more experience in regulating and operating reprocessing facilities.

A facility's detailed review of the failures of the systems associated with processing spent fuel for dry storage showed that the typical difficulties were associated with:

- Lack of availability of the seismic restraints of the fuel handling machine;
- Lack of availability of the inflatable seal in the fuel drying process;
- Blockage of the filters in the liquid waste system, and
- Problems with fan belt in the ventilation system.

These types of instances are beneficial and could be expanded to provide even more information to organizations instituting dry storage.

6. CONCLUSION

So where does this leave the document? The challenge before us is to work together to complete the document. Two actions need to be taken: refine available information and gather additional best practices and lessons learned, to ensure each topical area is adequately covered.

Starting with the current available information, a request for more information should be sent to the Member State that submitted the information to gather clarifying information. Each item discussed should include problem solutions with sufficient details to allow readers to determine whether it applies to their facility and if so, implement a recommended solution.

I encourage each attendee at the conference to take the revised questionnaire back to their respective countries, make copies and distribute them to others who may have experience or a vested interest in this document and encourage them to complete it and return it to the IAEA in a timely manner. Both of these steps should be taken in parallel to yield the most amount of available information as possible for the next consultants meeting.

It is apparent from the current state of the document that significant time and effort will be needed to revise the current draft into a workable document. Unfortunately, most of the consultants also have other duties that keep us gainfully employed. A suggestion would be for the next two consultants meetings to be a weeklong each to maximize the amount of time available for writing the document. Prior to these meetings, the consultants “do our homework” via email to evaluate and take responsibility for the sections in our area of expertise.

As the Irish writer and poet Oscar Wilde said “Anybody can make history. Only a great man can write it”. Now is the time for us to write our history to minimize the risk for future operations.

INDIAN EXPERIENCE IN COMMISSIONING, OPERATION AND SAFETY OF WET TYPE SPENT FUEL STORAGE FACILITIES

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Abstract

The Indian Nuclear Power Programme is heavily based on PHWR. The spent fuels generated in Indian PHWRs are stored in water pools At Reactor and Fuel storage Facilities for interim periods before reprocessing. The technology of design, construction, commissioning and operation of these fuel storage facilities has been mastered by BARC. Some of the innovative features and operating experience of these facilities are described in this paper.

1. INTRODUCTION

India has opted for closed fuel cycle to achieve the three stages nuclear power programme. Demand on back end of fuel cycle is increasing because of increase in requirement of fissile material from Spent Fuel as an input fuel to second stage of nuclear power programme consisting of number of fast breeder reactors (FBRs). The spent fuel from the pressurized heavy water power reactors and research reactors are stored in water pools for interim periods ranging from 3 year to 10 years before transfer to reprocessing plants. The reprocessing plants also have in- built water pool, for fuel inventory of 6–12 months of operation. These in-built storage facilities generally have common services etc. power, compressed air, DM water etc. fed from the main reprocessing plant. Two large size wet type spent fuel storage facilities have been built recently which are located closer to the reprocessing plants and have independent service support systems. Figs 1 and 2 give layout of new facilities.

2. DESIGN CONSIDERATIONS

The new fuel storage facilities have been designed with highest standards of safety and have a number of innovative safety features. Operation feedback of more than 40 years of operating experience of fuel storage facilities has gone into design of these facilities [1]. Some of new and innovative features provided in these storage facilities are given below,

- An infiltration trench around the pool to maintain water table below the raft of the pool has been provided along with automatic water pumping facility. The pool raft and walls are not subjected to external hydrostatic load, which might uplift the pool structure during construction or in case of emptying of the pool;
- Improved pool lining design with an elaborate leak detection and leak collection system has been provided. In case of failure of lining, the water from inside pool or outside areas will pass through the pipes connected to the grid channels and finally get collected in a sump. The identification of location of the leak and suitable repair methods has been made available;

- A pull-pull type ventilation scheme has been incorporated for flow of contaminated air from pool surrounding through the exhaust duct provided at top of the pool above water surface. This also helps in making an air curtain over the water surface;
- A single failure proof 75 Te EOT crane for handling the shipping cask has been provided with limited reach in the pool for in-built safety during handling of heavy cask. The crane has been designed to withstand design basis earthquake and has safety features like load cell for automatic tripping of hoisting in case of failure of any of the load path components, VFD driven smooth motion control and anti toppling devices, etc.;
- A seismically qualified pool bridge with automation features for under water handling of fuel storage trays and fuel bundle has been provided with precise motion control so that 75 Te crane is not subjected to small loads;
- The pool floor has provision of locators for seismic stability of stack of trays under the design basis earthquake [4]. These have added advantage in X-Y positioning of the pool bridge for handling of trays;
- VFD drives for ventilation fans have been provided to save energy during non-peak hours where air changes required can be drastically reduced. The fans are run at 25% or 50% speeds in off line time. There is an- built safety design in the control logic of supply and exhaust fans e.g. the supply fans have automatic tripping if exhaust fans fail.

3. CONSTRUCTION

These facilities have been constructed, commissioned and made operational within a very short span of 2 years due to innovative methods used in standardization of design of equipment, construction and commissioning procedures. High performance concrete (HPC) with micro-silica has been used for high impermeability for pool raft and walls, which is an innovative feature in these fuel storage facilities in addition to well-proven box type waterproofing done for the external surface of walls. Parallel construction and installation of equipment methods were adopted for reducing the construction time.

4. REGULATORY AUTHORISATIONS

The facilities have gone through three tier regulatory reviews at different stages of clearances for design, construction; commissioning and operation .Various regulatory inspection teams (RIT) were deployed by regulatory authorities to audit the quality assurance compliance during construction and commissioning phases of the facility. Various safety related recommendations of these RITs were implemented before start of operation. The documents for commissioning and regular operation have been thoroughly scrutinized by the regulatory body before granting the clearance for commissioning and regular operation. The document list includes plant technical specification for operation, commissioning reports, final safety analysis report, operation and maintenance manuals, emergency operating procedures, training manuals, radiation protection manual and fire manual. A post accident management plan has been developed to mitigate the consequences of any beyond design basis accident based on a radiological safety analysis carried out for the facility for such a postulated accident.

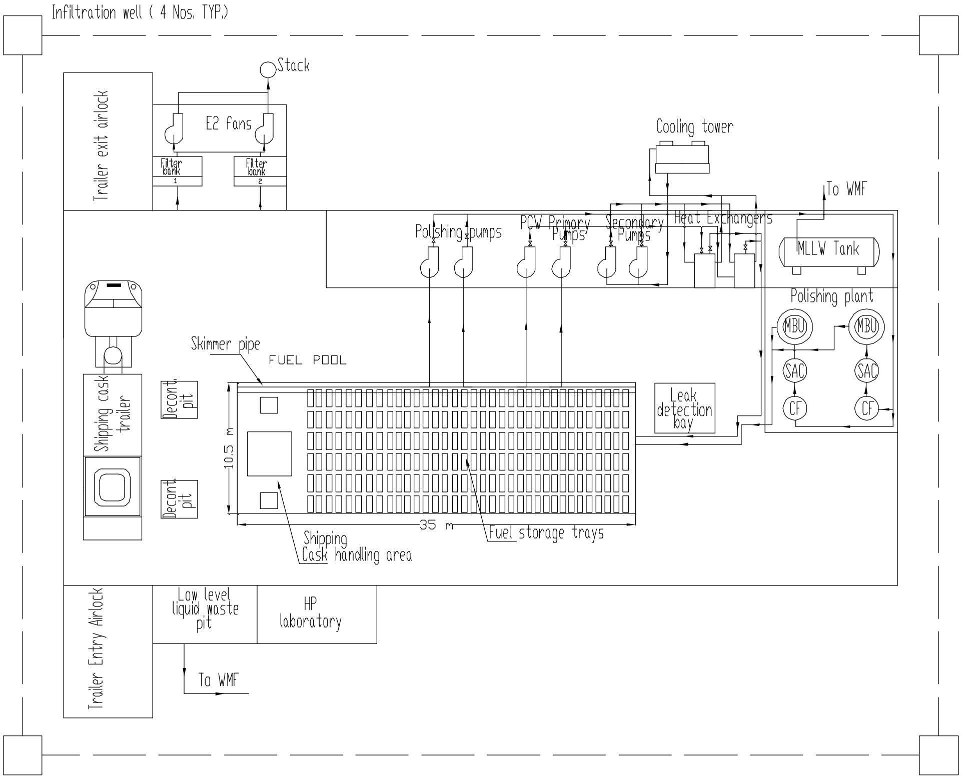

FIG. 1. Lay out of the new SF storage facilities.

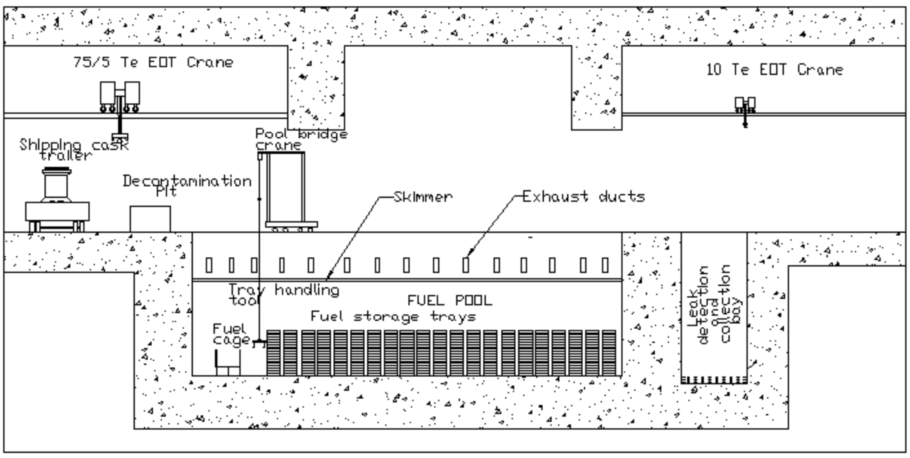

FIG. 2. Cross section of the new SF storage facilities.

5. COMMISSIONING

For the commissioning of these facilities, step-by-step commissioning procedures have been followed. The fuel pool leak tight integrity has been checked before and after lining. The leak detection and collection headers, SS liners have been 100% checked for weld integrity by dye-penetrant test and vacuum box tests. The fuel handling equipment like 75 tonne and 10 tonne EOT cranes and Pool Bridge has been load tested after installation. The fuel handling devices and tools are tested under water with simulated loads to meet design intent. The ventilation system is checked for flow balancing and filters bank efficiency. The low level and intermediate level liquid lines for waste transfer are tested with hydrostatic pressure and flow checks.

All the safety equipment e.g. Area gamma monitors, continuous air monitors, fire detectors and fire alarms, fire water system, infiltration water piping system, pool water level and temperature instruments and control room instruments are calibrated and tested in the field condition prior to start of operation. The shipping cask with tractor-trailer, waste transfer casks and fuel bundle handling operation trials are carried out to ascertain the crane reaches and handling equipment compatibility. The class –IV power failures and automatic switch over to class-III power mode are also tested with main loads. The training of operators has been carried out through classroom lectures and field training during construction and commissioning phase.

Some of problems experienced during the commissioning phase were (i) early replacement of HEPA filters due to dust accumulation in construction phase (ii) teething problems with brakes setting and VFD tripping of 75 tonne EOT crane in first 10 consignments (iii) failure of pool bridge shaft due to misalignment of rails during first few consignments of shipping casks (iv) Stuck up of cask lid with body in water and it's remote retrieval.

6. OPERATION

The spent fuel bundles are transported from reactor pools (At-Reactor) through heavy lead shielded casks, weighing around 62 tonnes, which are qualified as type B (M) packages [2]. Both road and rail mode of transport are used. The transport of spent fuel is done under the regulatory requirement laid down by Atomic Energy Regulatory Board (AERB) [3]. The fuel storage facilities have trailer receiving airlocks and decontamination pits for cask. The casks are handled in pool for unloading and spent fuel bundles are stored in fuel storage racks inside the fuel pool in a stacking of 20 trays. The fuel pools are provided with a 75 tonnes crane to handle the casks, a pool bridge along with fuel handling tools for under water handling of trays and fuel bundles and under water lighting.

The pools have water clean up system (Fig. 3), cooling system, ventilation and waste handling provisions (Fig. 4). The pool water chemistry control, pool water level and temperature control, water activity, ventilation control and waste management are the main operating activities for safe operation of the spent fuel storage facilities. These facilities are run as per plant technical specifications approved by regulatory authorities and any violation of these specifications requires the filing of safety related incident report for review by the regulatory authority. In cases of violation of safety limits, the facility has to be shut down till further review and remedial actions.

The facilities are in operation for last 3-4 years and all the safety systems have performed well. The important periodic activities like emergency drill, fire drill, load testing of cranes, surveillance of instruments, infiltration well monitoring, monitoring of leak detection, monitoring of stack release etc are carried out.

Some of the important Operation and Maintenance activities in Spent Fuel Storage Facility:

- Receipt, handling and storage of Spent fuel from PHWRs as per the approved check list;
- Monitoring of the fuel pool water level, temperature and recording in log book;
- Maintaining Pool water level;

- Maintaining pool water temperature below 45º C by operation of Fuel pool cooling system and recording various parameters of primary cooling water pumps, secondary cooling water pumps, heat exchangers and cooling water;
- Maintaining pool water activity below 185 Bq/ml by operation of Pool water polishing system and recording of various parameters including Pump discharge pressure, water inlet/out flow rate, DP across Cartridge filter, Cation cartridge and MBU, etc.;
- Operation of Muck cleaning unit to clean the fuel pool floor;
- Regular sampling and analysis of Pool water make up water, Infiltration well water.
- Physical inspection inside leak detection and collection bay and ensuring no water collection inside;
- Local verification of alarm status in Area gamma monitors and Continuous air monitors. Spot air sample analysis around fuel pool. Changing filter papers for all the Continuous air monitors. Swipe survey taken at various floor areas and checking of contamination level. Radiation survey carried out at various areas;
- Monitoring and recording of supply and Exhaust fans status including Speed, Bearing temperatures, current, Pressure drop across HEPA and Prefilter banks, etc.;
- Monitoring and recording of low level and medium level liquid waste levels in the tanks;
- Collection, segregation, packing and transportation of CAT-I (Mops, Shoe covers, hand gloves etc) , CAT II / CAT III (Cartridge filter & Cation cartridge of Pool water polishing unit, HEPA filters of E2 system and muck cleaning filters) solid wastes to Waste management’s facility for safe disposal;
- Pumping and disposal of liquid wastes CAT-I and II arising from active floor drains, cask decontamination wastes, CAT III (MBU regenerant waste of Polishing unit) through waste transfer pipelines to waste management facility;
- Operation of DM water plant for fuel pool make up and also for cask decontamination unit;
- Operation of Infiltration well pump and recording the hour meter readings;
- Surveillance of Various systems including Health Physics monitors, Pool water level and temperature instruments, EOT Cranes, DG sets, Battery banks, Public address system, Exhaust filter bank efficiency, Air receiver pressure testing, fire fighting system and functional checks of logic of ventilation system and infiltration system as per the technical specification.

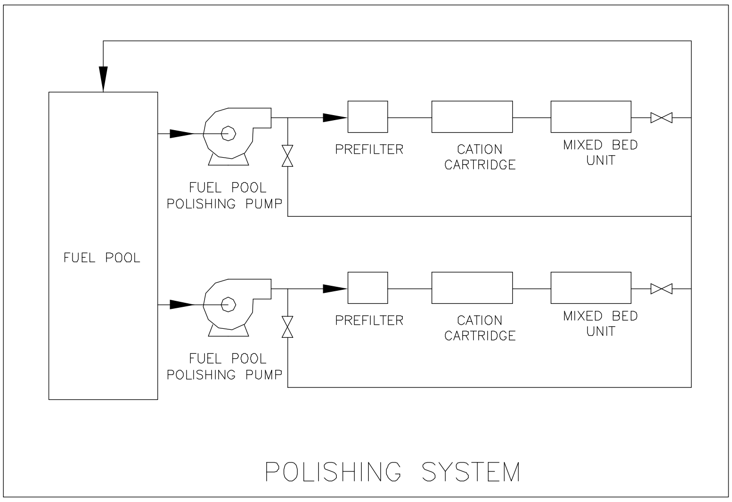

FIG. 3. Spent fuel pool clean up system.

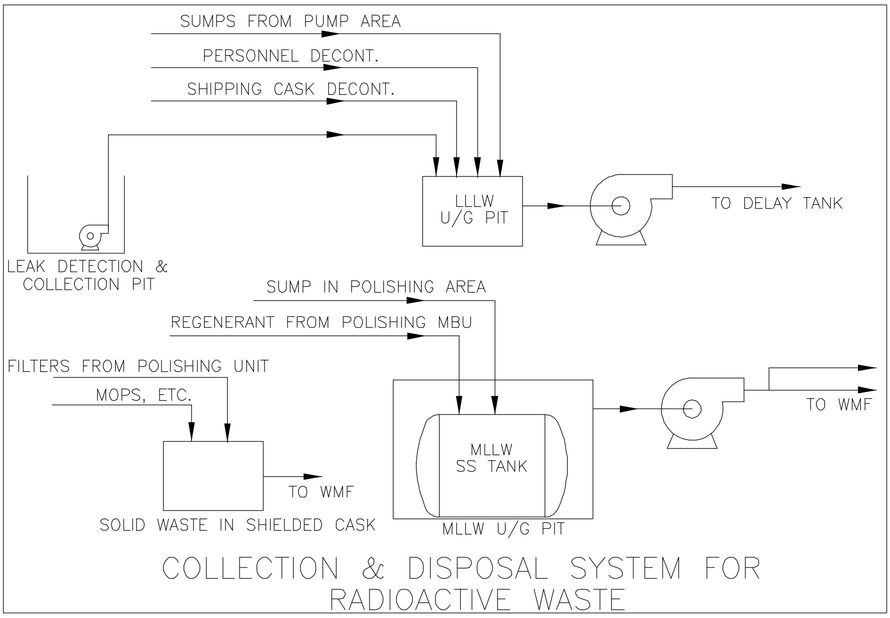

FIG. 4. Spent fuel pool ventilation and waste handling system.

7. CONCLUSION

The design, construction, commissioning, and operation of wet type fuel storage facilities have been carried out within the requirement of regulatory frame work. The necessary operational requirements are to be followed strictly for the safe management of these fuel storage facilities.

ACKNOWLEDGEMENTS

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OPERATING EXPERIENCE AND CONDITION ASSESSMENT OF SPENT FUEL DRY STORAGE SILOS AND SPENT FUEL POOL AT EMBALSE NPP

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Abstract

At Embalse Nuclear Power Plant (NPP), spent fuel removed from the reactor core is placed in a carbon steel basket before it is removed from the Plant spent fuel storage pool. Then, baskets are carried in a shielding container to a storage silo, where they remain until their final disposal. The silo system consists in a concrete cylinder of 2.80 m external diameter and 0.85 m thick, internally lined with a carbon steel cylinder of 9.5 mm thick. This structure is supported by a 0.60 m thick concrete slab. This work reviews the Condition Assessment of Embalse Spent Fuel Dry Storage Silos and was performed following the procedures implemented in the Embalse Refurbishment Project. A review of nondestructive and destructive methods is presented so as to assess the condition of concrete and carbon steel of this structure. Future tasks to be performed in the Spent Fuel Pool is presented.

1. INTRODUCTION

At Embalse NPP, the spent fuel removed from the reactor core is stored in a spent fuel pool, whose demineralised water simultaneously acts as a radiological shield and coolant to remove the heat from fuel decay. When the fuel decay heat is reasonably low, fuel is moved to a temporary site (dry silos) where it remains until final disposal.

216 silos have been built at Embalse NPP so far. The first 80 were built in 1993, 40 were built in 2001; another 64 in 2005 and the last 32 silos were finished in 2009. This implies a current storage capacity of 116640 fuel bundles, which corresponds to the amount of fuel produced by Plant operation until the end of 2011. A Spent Fuel Dry Storage Silos battery is shown in Figure 1.

During Phase I of Embalse NPP Life Extension Project, a number of Life Assessments and Condition Assessment of different Systems, Structures and Components were carried out, including spent fuel storage silos.

The Condition Assessment of Dry Fuel Storage Silos was performed by Embalse NPP, which includes the review of the design, operating experience and maintenance of these silos, the aging assessment of components and subsequent conclusions and recommendations.

FIG.1. Spent fuel storage silos battery at embalse NPP.

2. SYSTEM DESCRIPTION

The silos system consists in a concrete cylinder of 2.8 m external diameter and 0.85 m thick, internally lined with carbon steel of 9.5 mm thick. This structure is supported by a concrete slab of 0.6 m thick. Drawing of structure is shown in Fig. 2.

The internal carbon steel liner keeps the silo leak-tightness intact over time. Apart from its structural function, concrete provides radiation shielding from the interior. The following table summarizes the main characteristics of the materials.

TABLE 1. MAIN CHARACTERISTICS OF MATERIALS

SUBCOMPONENT	DESIGN DESCRIPTION	PERFORMANCE EXPECTATIONS
Concrete	Type II Portland Cement f´c = 21 MPa Density = 2400 kg/m^3	Compressive forces resistance according to structural analysis. Radiation shield. Environmental condition: 60°C outer surface and 120°C inner surface.
Reinforcing steel	Fy = 420 MPa (grade 60)	Tensile forces resistance according to structural analysis.
Steel liner	ASTM A-36	Spent fuel baskets containment, radiation shield.

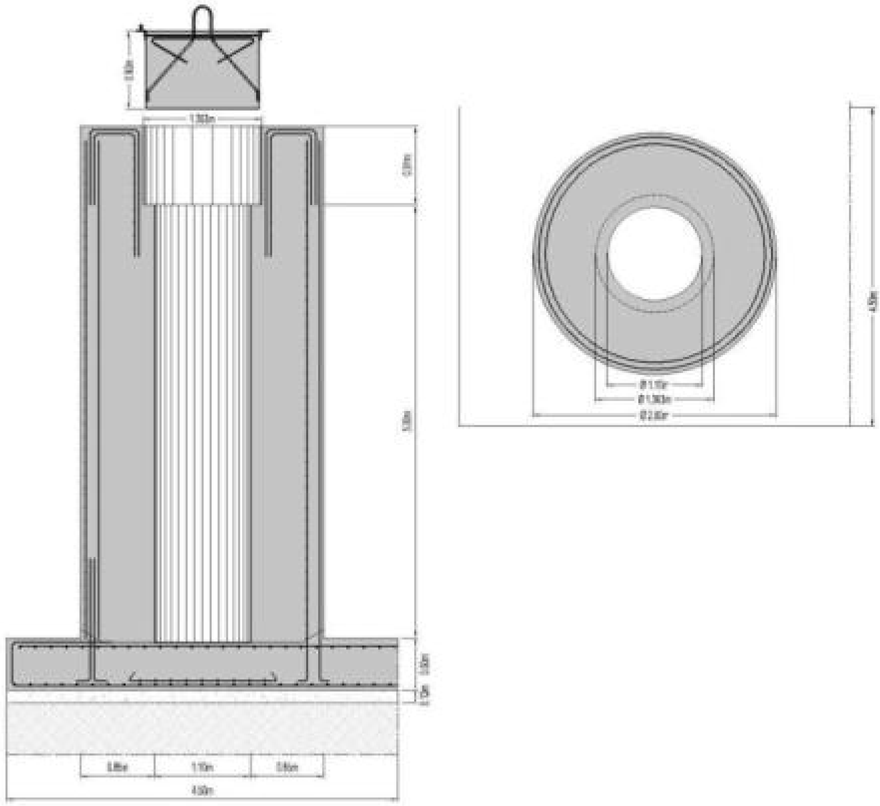

FIG. 2. Spent fuel silo-structure.

3. CURRENT MAINTENANCE AND INSPECTION TASKS

According to the Plant Maintenance and Inspection Program, there are two types of control for storage silos: internal and external. Internal Controls include the following activities:

- Monitoring of silos **Error! Reference source not found.**: some silos are monitored to measure the internal moisture content and are later purged with compressed air. These activities are done every six months in order to remove the moisture and mitigate the effects of corrosion in the steel liner and baskets.Establishing a limit value of 30% RH, above which the internal air must be swept to values of 20% RH or lower. Since the beginning of this essay in 2003, the average moisture is about 20%. Air sweeping due to excessive moisture in the air needs to be performed in only 10% of the silos reviewed. It is assumed that moisture penetrates through the inspection piping of the liner. This work is complemented with the measurement of noble gases in the interior;
- Containment barrier integrity verification **Error! Reference source not found.**: a monitoring of aerosols and noble gases in the internal atmosphere is performed every six months to verify the integrity of the SFB and baskets. The presence of Cs-137 and Kr-85 is controlled by using a noble gases analyser and an aerosol analyser which has a filter paper and an activated carbon filter. There is a loss of containment every time the concentration limits of radionuclides Cs137 and Kr 85 are greater than 1 DAC. Several tests have been conducted, yet they are unable to prove the presence of noble gases and aerosols in values higher than those recommended; hence, the containment provided by fuel bundles and baskets is considered to be in good condition.

External controls include the following activities:

- Annual External Radiological Monitoring **Error! Reference source not found.**: it is carried out by using TLD dosimeters to check any changes in radiological conditions. In order to quantitatively assess the evolution of annual doses, that is to say, if they increase or decrease to values that need to be researched, the average annual dose and standard deviation are calculated. If the value measured in the year falls between two standard deviations (higher or lower), the value is considered to be acceptable. If it falls between the ranges of two or three standard deviations, the causes are investigated but, if the value is greater than three standard deviations, some corrective actions should be taken in this regard. A table summarizing the results obtained and a drawing with the position of detectors are attached.

TABLE 2. ANNUAL DOSE RATE AND AVERAGE VALUE OF EXTERNAL RADIOLOGICAL MONITORING **Error! Reference source not found.**

Site	Annual dose rate mR/year 2009	Average dose rate mR/year 1998–2008
1	307,5	300,9
2	270,5	322,1
3	214,8	255,8
4	263,3	291,0
5	346,2	395,3
6	474,7	389,6
7	155,9	217,9
8	638,9	176,1

From time to time, an external radiological monitoring is performed by measuring the doses in contact with the silos. The contact dose rate at one meter above the floor varies between 1.2 and 9.1 μSv/h, according to the position of the silo and the spent fuel content characteristics. The dose rate at one meter from the silo and at one meter above the floor varies between 0.9 and 8.2 μSv/h. In conclusion, it is possible to state that dose rate values higher than expected have not been detected to date.

A radiological test is performed in the silos every six months by placing X-ray films on the outer surface of those areas where cracks are observed. The shield capacity of the structure is verified by checking that there are no radiation leaks through those imperfections. The formwork system has been improved by controlling the appearance of such cracks.

On the other hand, in order to control the effects of different degradation mechanisms that could affect the integrity of these structures, the Plant has a prototype silo with two baskets and metallic cores within. Every two years, Embalse NPP performs in the silos the following controls:

- A helium leakage test to verify losses of leak-tightness in the baskets;
- Visual inspection and thickness measurement of the basket and internal liner of the silos;
- Corrosion analysis in metallic samples.

The above mentioned tests have not shown any evidence of unexpected degradation mechanisms, and corrosion values do not exceed the estimated values.

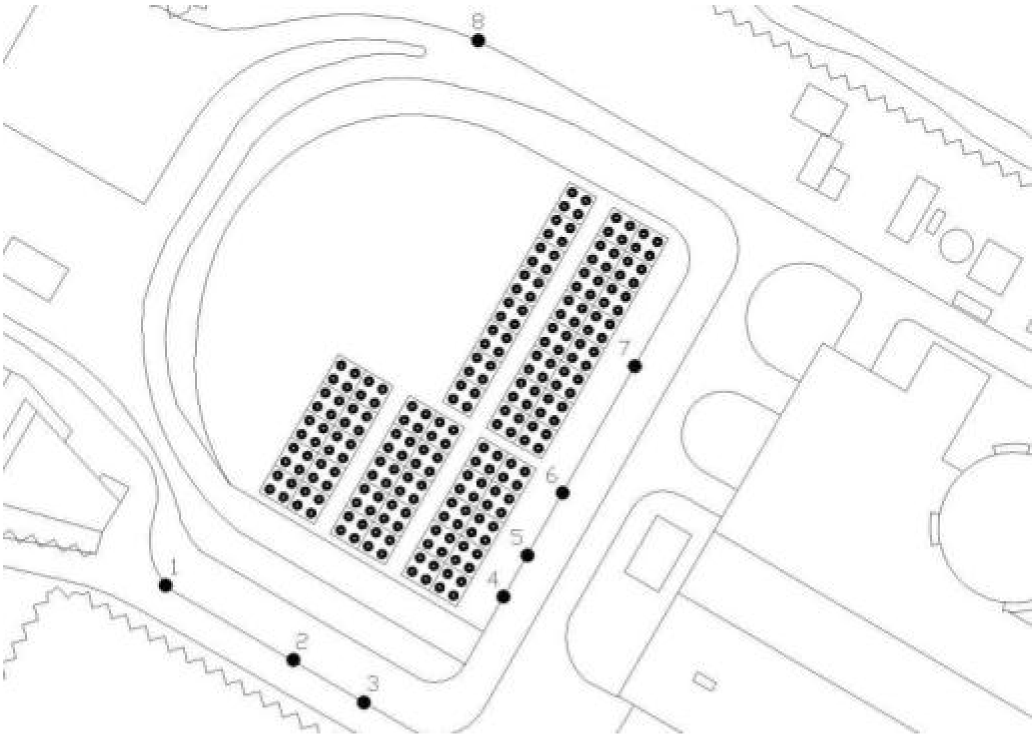

FIG.3. Approximate position of TLD dosimeters.

4. CONDITION ASSESSMENT OF SPENT FUEL DRY STORAGE SYSTEM

4.1. Degradation mechanisms

Due to the condition assessment study of spent fuel dry storage silos, a series of ageing related degradation mechanisms (ARDM) were identified for each one of the components (concrete, reinforcing steel and steel liner). Such degradation mechanisms could affect silos, taking as a reference the characteristics of the materials and the environmental conditions to which they are subjected.Table 3 shows potential ARDM`s for spent fuel silos.

These degradation mechanisms were quantified according to their detection and the existence of detection and mitigation strategies for such a mechanism. For example, water and soil chemical tests showed that sulphate content in water is lower than the minimum required to damage concrete, so sulphate attack is low.

4.2. Recommended assessment procedures

After identifying those mechanisms that could affect silos, relevant evaluation and/or detection procedures were established, as well as the necessary actions that need to be taken to prevent the propagation of these mechanisms on the structure. These detection mechanisms were established according to normal practices and the feasibility of implementing these types of methods in these structures. The following table summarizes the Evaluation Procedures established for each ARDM.

TABLE 3. POTENTIAL ARDMS FOR SPENT FUEL DRY STORAGE SILOS

ARDM	Concrete	Reinforcing steel	Metallic liner
Sulphate attack	*		
Acid/base attack			
Alkali — aggregate reactions	*		
Carbonation	*		
Freeze-thaw attack			
Elevated temperature/thermal cycling	*		
Irradiation	*		
Fatigue/vibration			
Settlement			
Creep/shrinkage	*		
General corrosion		*	*
Weathering			
Wear			

Only three of the aforementioned procedures (Visual inspection, Electrical Resistance and Permeability Test) are tests that do not involve concrete coring extract. For the rest of them, the possibility of moving spent fuel from old silos to allow the extraction of some cores from a representative silo is under consideration.

5. CONCLUSIONS

The Spent Fuel Dry Storage Silos are included in the Preventive Maintenance Program of the Plant. These inspections are primarily designed to control the corrosion of the steel internal liner. No problems were found regarding the action of such mechanism.

The concrete structure does not show any relevant evidence of surface degradation. The first silos that were built show some surface micro cracks presumably caused by concrete shrinkage, formwork and filling methods and environmental action. Formwork system and works were improved, significantly reducing the appearance of cracks due to these works.

At present, Embalse NPP is developing a Maintenance, Monitoring and Inspection Program which is suitable for assessing the silo concrete structure to detect and mitigate the degradation effects on this structure.

Due to the content of silos and their unavailability, Embalse NPP should determine the feasibility and possibility of implementing the above mentioned evaluation methods.

TABLE 4: EVALUATION PROCEDURES

	EVALUATION PROCEDURES					
ARDM	Visual Inspection	Petrographic Analysis	Permeability Test	Chemica Analysis	Carbonation Depth Measurement	Electrical Resistance – Resistivity Measurement
Sulphate Attack	*	*		*		
Acid/Base Attack						
Alkali-Aggregate Reactions	*	*				
Carbonation	*	*	*		*	
Freeze-Thaw Attack						
Abrasion/Erosion/Cavitation						
Elevated Temperature /Thermal Cycling	*					
Irradiation		*				
Fatigue/Vibration						
Settlement						
Creep/Shrinkage	*					
General Corrosion						
Weathering Wear						*

6. SPENT FUEL POOL

The Spent Fuel Pool at Embalse NPP has 19 m long, 12 m width and 8 m depth. The Spent Fuel bundles are placed in the pool for 6 years. After those 6 years, fuel bundles are moved to silos, were it remains until final disposal.

The liner system of the pool is constituted by 5 layers of epoxy coating and fiber-glass cloth with epoxy, forming a total layer of 3 mm thick.

A visual inspection was performed by Embalse NPP. This inspection showed:

- There is no evidence of leakage in external surfaces;
- There is no evidence of leakage in the Spent Fuel Pool Under drainage System;
- There are several bubbles in the last layer caused by temperature and irradiation action.

For this reason, Embalse NPP is making some arrangements for underwater coating using Bio-Dur 561 epoxy paint as preventive measure.

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REGIONAL STRATEGIES CONCERNING NUCLEAR FUEL CYCLE AND HLRW IN CENTRAL AND EASTERN EUROPEAN COUNTRIES

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Abstract

In March 2009 a regional meeting on national strategies concerning nuclear fuel cycle and high level radioactive waste (HLRW) was held in Budapest with the participation of Central and Eastern European countries, Russia and France. Following the meeting a Task Force was set-up with fuel cycle experts from different countries in order to analyse the possible fuel cycle strategies in the region. The Task Force produced an Opinion Paper in spring 2010 on the Regional Strategies Concerning Nuclear Fuel Cycle and HLRW in Central and Eastern European Countries with several recommendations. The Opinion Paper emphasizes that the countries in the Central and Eastern European region are small, and they have modest NPP capacities compared to large nuclear countries. Spent fuel reprocessing facilities are not available in the region, but Russia and France offer such services for these countries. Deep geological repositories are not in operation in any of these countries, and in some of the countries the geological conditions do not allow to design such facilities. For these reasons the countries of the region may need special services and a regional approach could produce common benefit for waste management.

1. INTRODUCTION

In March 2009, a regional meeting on national strategies concerning nuclear fuel cycle and high level radioactive waste was held in Budapest with the participation of Czech Republic, Lithuania, Ukraine, Bulgaria, Romania, Slovakia, Slovenia, France, Russia and Hungary [1–14]. The meeting was organised in the framework of IAEA RER/3/008 project. Following the meeting a Task Force was set-up with fuel cycle experts from different countries in order to analyse the possible fuel cycle strategies in the region. The Task Force completed the following actions:

- Potential scenarios of nuclear energy development in the region have been analysed on the basis of economical and electricity consumption data. Prediction of future electricity consumption and installed capacity of NPPs has been calculated using three different scenarios;
- International development of fuel back end has been reviewed. The advantages and disadvantages of open fuel cycle have been identified. The current reprocessing options

offered by French and Russian companies have been compared. The transition to closed fuel cycle with fast breeder reactors (FBR) has been analysed from the point of view of uranium resources, maturity of FBR technology and availability of advanced reprocessing technology;

- The role of interim storage for spent fuel (SF) and for HLRW emerging from fuel reprocessing, from normal operation and decommissioning of NPPs has been evaluated;
- An attempt has been made to carry out technical and financial comparison of ultimate waste management options on the basis of today available French and Russian reprocessing technologies;
- Services and market development needs have been analysed for different fuel cycle strategies and mass flows in different back end scenarios has been simulated;
- Institutional issues to organize pilot market conditions for fuel services have been investigated considering the existing situation in the European Union and taking into account the potential use of MOX fuel for LWR and FBR reactors;
- The multinational co-operation and R&D for flexible fuel back end strategies has been summarised.

The Task Force produced an Opinion Paper in spring 2010 on the Regional Strategies Concerning Nuclear Fuel Cycle and HLRW in Central and Eastern European Countries.

2. SPECIFIC FEATURES OF THE REGION

Central and Eastern European countries have been operating nuclear power plants (NPPs) for several years (some of them for more than two decades) and a few of them have launched a program to complete the construction of NPPs which had to be suspended after the historical changes in political and economical systems. Besides that, considering their future energy needs, the issue of energy supply security, the age of their electricity production plants, the majority of the countries of the region have started a process which should result in building of new NPPs. Keeping in mind that for a nuclear power plant about 10 years from the decision are needed to be connected to the grid, the first of these NPPs may start operation around 2020 and the others will follow up to about 2030. Moreover, this ten-year period can strongly and dramatically increase if public is not supporting the project.

Today in the region, except Austria, public acceptance is quite high (generally about 60%, in some countries up to 70%). Nevertheless, because of the possible consequences far beyond the borders in case of an accident, and because nuclear energy is also an objective which stiffened the fight among various opponents, there is no doubt that, for each project, there will be a need of spreading full and wide information regarding three main key issues:

- Nuclear safety;
- Non-proliferation;
- Radioactive waste and SFM.

Generation-2 reactors have shown very good safety records and were upgraded to comply with EU safety harmonization. Generation-3, the new generation of NPPs with enhanced safety being currently proposed by industry are presently under construction in some countries.

Safety and non-proliferation risk are managed in accordance with the international rules issued both by IAEA and EURATOM in the EU. In the region, all of the countries have signed the corresponding agreements and the majority of them have created the legal structure (Nuclear Safety Authority, or Committee or Office).

As regards radioactive wastes, particularly high level wastes and spent fuel, most of the countries have long term policies. The establishment of new nuclear units and the nuclear technology developments offer new perspectives, which may need reconsideration of fuel cycle policies and to open more active regional and Pan-European co-operation.

The aim of this Opinion Paper is to give an overview of the present nuclear fuel cycle possibilities and then to propose some tracks towards a sustainable and economically viable strategy. This work has been done considering two specific features of the region and a third which is widely shared in Europe:

- Presently, countries of the region are mostly using Russian fuel in their NPPs;
- There is no spent fuel reprocessing facility in the region;
- For electricity production, the main technologies of this century will be Generation-3 Light Water Reactors (LWR) with time life extension of existing Generation-2 LWRs.

Moreover, because a lot of the countries of this region are members of the EU, possible strategies have been examined keeping in mind the European energy policy for which the "Strategic Energy Technology Plan" (SET-Plan) defines the necessary technology development, the work in progress of the European Nuclear Energy Forum (ENEF) and the likely evolution of nuclear energy in Europe as it is summarized in the "Vision document" of the "Sustainable Nuclear Energy Technology Platform" (SNETP).

It is important to mention that in the SET-Plan, nuclear fission is identified, together with other low carbon technologies such as renewables and Carbon Capture and Storage (CCS) technology, as one of the key contributors to meet the 2020 challenges: "By maintaining competitiveness in fission technologies, together with long term radioactive waste management solutions, fission energy will continue to be leading low carbon energy technology in Europe". Beyond the 2020 objectives, the SET-Plan also identifies fission energy as one of the contributors to the 2050 objectives of a low carbon energy mix, relying on the Generation-3 reactors, closed fuel cycle, and start of implementation of Generation-4 reactors making nuclear energy sustainable.

Thus, regarding spent fuel and waste management, European strategy aims at developing long term solutions for the closure of fuel cycle. This strategy is one of the objectives of Sustainable Nuclear Energy Technology Platform which is supported by about 70 European organizations involved in nuclear energy, amongst them most of R&D organizations but also industry (AREVA, Ansaldo, Empresarios), the TSO (technical supports of safety authorities) and the most important utilities (EDF, E-ON, RWE, ENEL, CEZ, GDF-Suez, TVO).

3. EXISTING AND FUTURE FUEL BACK END TECHNOLOGIES

Today, available technologies used in the management of HLRW and SF are the followings:

- <u>Interim storage</u>, which means the storage of SF assemblies or HLRW after specific conditioning. Interim storage facilities are generally located near the reactor, either in a form of wet or a dry tank or a building equipped with heat removal system to avoid accidental situation which would be difficult to manage (risk of releasing gaseous fission products which are in the SF assembly). It is a well and mature technology either for SF or for vitrified HLRW. There is now a large, worldwide experience feedback and the progress is much more a matter of design optimization and engineering methods than an R&D topic. Interim storage installations were originally designed and built for 30-50 years. Today, it is generally accepted that there is no major difficulty to design an interim storage for more than 50 years, maybe for about 100-150 years but with some developments to master storage materials, safety systems and I&C ageing phenomena;

- Transportation of SF inside the country or to another country, which means that all the possible obstacles (political, legal, organizational, etc.) are overcome and the whole process has been agreed upon, including the final solution in case of transport to another country;
- Conventional reprocessing, which means that a specific chemical treatment of SF is made in order to separate components which might be further utilized (U and Pu, or a part thereof) as a resource in a new manufactured fuel such as the MOX (Mixed Oxide Fuel) already used in several NPPs in Western Europe. Current Generation-2 and even Generation-3 reactors can use only a small part of the uranium energy content (less than 2%) of the fuel. Residual materials resulting of this recycling process are then appropriately conditioned (e.g. vitrified), capsulated for interim storage and final disposal. Mature reprocessing for LWR spent fuel is a pre-requisite for fast breeder reactor deployment which is necessary to reach long term sustainable nuclear energy. In the same time, MOX recycling in LWRs will reduce the natural uranium requirements and the amount of high level waste.

For the future, two advanced technologies are under development:

- Advanced reprocessing, which is based on partitioning and transmutation of SF in order to increase the efficiency of fissile material use, and also to decrease the radioactivity and radiotoxicity of residual wastes to be disposed. Integral recycling of actinides should result in ultimate wastes whose potential radiotoxicity decays to that of the uranium ore (from which the initial fuel was manufactured) in about three centuries, compared to 100 000 years of a direct disposal of SF. Combined with ultimate waste conditioning (vitrification for example), this process is believed to improve long-term safety. However, this process needs, for transmutation, Generation-4 systems which will be not deployed at industrial level before 2050. If Generation-4 systems are Fast Breeder Reactors (FBR), the energy production from a given amount of uranium can be multiplied by at least a factor of 50 compared to current reactors and consequently ensure resources for thousands of years and this may solve the issue of energy supply security of the EU;
- Geological repository: After a while in interim storage, HLWR and/or SF will be specifically conditioned and put into a final disposal. Deep geological repositories (several hundreds of meters below the surface) are the only realistic option for isolating the highly radioactive materials from the biosphere for very long time. Geological repository is considered as definitive but in some countries it is now asked for reversible geological repository for different reasons. One of the main issues about geological repository is the content and conditioning of wastes which will be disposed. Direct disposal, i.e. without reprocessing, is considered as the reference management strategy of SF in many countries of the region today, but no deep geological repository is in operation in any country. In the USA, the Obama administration has recently cut most of the Yucca Mountain geological repository project’s 2010 funding and asked an expert commission (“The Blue Ribbon Commission on America's Nuclear Future”) to make recommendations for developing a new plan for the back-end of the fuel cycle.

However, it has to be noticed that geological repository and advanced reprocessing are not opposite solutions; as it will be shown below, in some management strategies they are complementary.

4. FUEL BACK END STRATEGIES

Basically, from the above-mentioned various technological solutions and political choices, five fuel cycle back-end strategies can be deduced depending on their combinations:

- Open fuel cycle (direct disposal or once-through option) in which, after a temporary storage, SF assemblies will be placed in special containers and moved into deep geological repositories. The technology for producing such containers and for excavation of underground system of tunnels exists today;
- The main advantage of open fuel cycle is its apparent simplicity. Today it seems to be a cheaper solution compared to closed fuel cycle. Moreover, it is likely that most of the NPP operating countries could find potential geological formations for domestic repository;
- The main disadvantage of open fuel cycle is that the fissile materials, which could be used in future reactors, remains embedded in the waste. Furthermore, SF has much higher radiotoxicity and larger volume than those of wastes from reprocessing. For countries with a very small reactor fleet it will be difficult to receive public acceptance regarding the need of a site for which specific long-term safety measures should have to be taken;
- Conventional closed fuel cycle (reprocessing and vitrification) in which SF is reprocessed in order to recycle the part (U and Pu) which can be re-used in a new fuel, while the surplus materials are vitrified and conditioned in a specific canister to furthermore be disposed into deep geological repositories;
- This strategy presents three important advantages: 1) a vitrified matrix for high level waste which is no more comparable to containers of SF (as in direct disposal) regarding long-term safety; 2) less amount of some Pu isotopes, which shall be disposed of after reprocessing; and 3) a lower volume due to the lower heat content and the compaction resulting from vitrification;
- The main disadvantage of this strategy is the cost of reprocessing which depends in fact on the number of reactors and the cost of uranium ore. Two other drawbacks are the need of more transport of radioactive materials and that some spent fuel still remains to be disposed of. However, if Generation-4 reactors are expected to be built, this final disadvantage is limited and this strategy can be considered as a bridge between Generation-3 and Generation-4 reactors;
- Advanced closed fuel cycle (reprocessing with partitioning, transmutation and vitrification): this strategy is rather similar to the conventional closed fuel cycle, but as described above, uranium utilization is optimized and ultimate wastes are composed only of fission products whose volume and radiotoxicity are strongly reduced. Like conventional closed fuel cycle, ultimate wastes are vitrified and conditioned in a specific canister to be disposed into deep geological repositories. No spent fuel remains for disposal;
- This strategy can really be considered as a sustainable way of using nuclear fission to produce energy. However, its main problem is the need of R & D which hinders its industrial development utilization today. Nevertheless, in long term, this strategy can be viewed as an extension of a strategy based on conventional closed fuel cycle, if not chosen from the very beginning;

- Postponing strategy (wait and see), which means extension of the use of interim storage for several decades in order to wait for demonstration of one or more of the above mentioned three strategies before final decision;
- Of course, this strategy is the cheapest but it does not really prepare the future and does not answer to people's fears. Moreover, because of the increase of reactors in operation and consequently the large amount of SF which will accumulate in interim storage, such a strategy can burden the transition and potential cost when a solution will be chosen;
- Return strategy. Depending on availability, some countries may also choose to return spent fuel to the country of origin, risking long-term dependence on fuel availability.

Because there is not yet any final geological repository in operation, none of these strategies is fully available today. Nevertheless, except the final stage, the different stages of the two first strategies (open fuel cycle and conventional closed fuel cycle) presently exist at industrial level. Of course, it is not easy to choose the proper long-term strategy today, since certain attractive strategies are not available commercially at present. In order to demonstrate the permanent positive and active attitude of the responsible organizations avoiding a premature final decision ("do and see"), public should be continuously informed on the potential solutions and investigations aiming at the construction of a deep geological repository which should start at the proper time. The final decision on the management of SF could be made when the potential new technologies (e.g. FBR, partitioning and transmutation) will be commercially available and the countries of the region will establish better cooperation in this field.

Today, most of the countries of the region are following a kind of mixed strategy between the open fuel cycle and postponing strategies. Compared to the work performed by Finland or Sweden, with the exception of the Czech Republic, their involvement in the construction of a geological repository is rather limited in spite of that, as we have seen, such a disposal will finally be needed. Probably, one of the reasons is the cost of research but also current international discussions, from which regional strategy and waste disposal could emerge.

The following current activities are, in particular, worth to be mentioned:

- The IAEA promotes strategies for SFM in order to improve the capability of interested countries to plan and implement improved strategies by identifying problems and fostering collaboration. The agency also provides technical guidance on good practices for long–term management of spent fuel to improve the capability of interested countries to develop individually, or through international co-operation, methods for long term management of spent fuel. Representatives of the agency pointed out that multinational repositories could enhance global safety and security by making timely disposal options available to a wide range of countries;
- The European Nuclear Energy Forum (ENEF), a forum created in 2007 for a broad discussion on transparency issues as well as the opportunities and risks of nuclear energy, which gathers all stakeholders of the EU in the nuclear field (governments, European Parliament, European Economic and Social Committee, nuclear industry, electricity consumers and the civil society), is working on recommendations which could further be adopted in the form of an EU directive, in particular if these recommendations are endorsed by the ENSREG (European Nuclear Safety Regulatories Group) whose missions have recently been extended to waste management;
- The recently established EU technology platform dedicated to radioactive waste management (Implementing Geological Disposal of Radioactive Waste Technology Platform - IGD TP -) claims in its vision document that by 2025, "the first geological

disposal facilities for spent fuel, high level waste, and other long-lived radioactive waste will be operating safely in Europe".

All these acting groups could lead the EU countries to adopt a binding national programme for radioactive waste. Each country would assess different technologies or solutions regarding spent fuel and high level wastes. Recognising that several Member States pursue the closed fuel cycle and others would prefer open fuel cycle, the EU could require that each member state define a clear strategy regarding deep geological disposal with appropriate licensing requirements and procedures, including site selection and safety criteria, as well as the responsibilities and rights of the actors involved.

5. ASPECTS OF SELECTION AND IMPLEMENTATION OF SF AND HLRW MANAGEMENT STRATEGY

The choice of a strategy for SF and HLRW is a long process as it involves several stakeholders in the decision making process and is often presented as a question of time and money.

Indeed, comparison between the two main options, which are roughly open cycle and closed cycle with FBR, may differ from one country to another as advanced technological solutions are not yet mastered and final disposal conditions are not yet clearly defined in most cases. Encapsulation process, transportation of used fuel after long storage, volume and shape of ultimate waste to be disposed, localisation and cost of implementation of the final repository itself and other parameters are difficult to estimate precisely as of today. Moreover, such a comparison must not be limited to final disposal conditions but must include all services (mining, enrichment, fuel fabrication, interim storage, reprocessing, conditioning, transportation, safety measures, etc.) for both options.

Clearly, the choice and the implementation of a HLRW and SF management strategy must definitely take into account several aspects:

- The costs and constraints of the two main technical options. Because of the complexity of making thorough studies and comparisons, a close cooperation of countries having the same nuclear history, similar fleets of reactors and comparable political situation could favour its accomplishment and strengthen its conclusions;
- The advantages and drawbacks of deep geological disposal based on one country or a regional approach. That topic, which should not be limited to technical aspects but should include political and social ones, needs also a close cooperation between the countries which could be involved in a common regional repository;
- The needs of the introduction of FBR in the second part of this century in connection with the issue of availability of uranium resources. Because the introduction of FBR cannot really be envisaged in a small reactor fleet, a regional approach should be promoted in parallel with cooperation with countries having experience on that kind of reactors;
- The social issue of very long-term waste management for which Generation-4 concepts should offer new options by 2050. However, to be able to assess these new options, cooperation of countries developing these concepts is absolutely needed;
- The guarantee of available services by industry for both fuel supply and SFM;
- The question of independence which is related to the security of fuel supply and the strategy chosen for SF, particularly if it is based on the return to the country of origin;

- Safety, security and non-proliferation provisions related to the specific fuel back-end strategies.

Considering the complexity of these questions, all countries of the region would benefit from a regional synergy. A regional cooperation could enable the sharing of cost studies and, if a common strategy is adopted, could be beneficial to all countries involved due to the scale effect.

6. CONCLUSIONS AND RECOMMENDATIONS

In conclusion, after having pointed out the stakes and presented the advantages and drawbacks of the different strategies for spent fuel and high level waste management, we would like to make some recommendations which could make an easier choice and give the opportunity to set up a more visible strategy in order to favour the deployment of future nuclear reactors:

- “Wait and see” strategy must be banished because it clearly transfers the responsibility to future generation;
- Optimum fuel back-end strategy should be considered under existing environment and local conditions with potential modification of existing ones;
- Regional and Pan-European cooperation should be developed and strengthen in order to:
  - Help to make a consistent choice of a national HLRW and SF management strategy and demonstrate its viability;
  - Guarantee long term fuel cycle services and enable their optimization;
  - Enable a real development of deep geological repositories and reduce the cost of investment and operation per country;
  - Have the critical size to develop Generation-4 reactors for long term sustainability of nuclear energy;
  - Be a credible partner for countries which are the leaders in the development of fbrs,
  - Allow the needed flexibility in fuel and Pu management particularly in case of implementation of FBRs;
- Appropriate legal and regulatory international framework for the safety and security of fuel services and deep geological repositories should be considered at the regional level with respect to EURATOM and IAEA rules;
- Harmonization of strategies and rules at a regional level should also:
  - Create favourable economic conditions for discussions with investors and companies specialized in fuel services;
  - Help to convince population particularly if the selected strategy leads to a sustainable nuclear energy.

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OPERATING EXPERIENCE IN SPENT FUEL STORAGE CASKS

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Abstract

A safe storage of spent fuels has been considered as one of the inevitable tasks for TEPCO for the last few decades. In order to increase flexibility for the fuel storage measures, TEPCO has been storing spent fuels in an on-site dry storage facility at Fukushima-Daiichi Nuclear Power Station. Since 1995, more than 400 fuel assemblies have been safely store. Integrity of storage casks and fuels were carefully checked by periodical inspections, which were conducted in 2000 and 2005. The next investigation will be held within a few years in order to verify the safety conditions even after a 15-year storage. These series of inspections will give plenty of useful data for the design and operation of the Mutsu facility, which will be the first off-site interim spent fuel storage facility away from any reactor site in Japan.

1. INTRODUCTION

In Japan, 55 light water reactors are in operation. The amount of spent fuels generated in Japan reaches as much as 1000 tU every year. Japanese utilities have made great effort to start up a national reprocessing facility in Rokkasho, Aomori in order to meet our country's policy of reprocessing spent fuels. Now the reprocessing facility is in the final phase of it's test operation. When the reprocessing plant starts its full-power operation, it is expected to contribute to complete our country's nuclear fuel cycle. However, since the Rokkasho reprocessing facility has a capacity of 800 tU/year, the amount of spent fuel waiting for reprocessing is considered to increase in the long term.

Under such situation, Tokyo Electric Power Company (TEPCO), which operates 17 BWR plants, is continuously coping with these increasing spent fuels. We have been trying to keep flexibility for the storage of spent fuels. Among many measures, there are on-going projects to replace the current large spent fuel racks into those of more compact designs, construction of an on-site interim storage facility and a common spent fuel storage pool at Fukushima-Daiichi NPS. These facilities have been contributing to our company's tactics for the management of spent fuels. In addition to these measures, TEPCO decided to construct an off-site interim spent fuel storage facility away from our reactor sites as a joint facility with JAPC (Japan Atomic Power Company) at Mutsu, Aomori [1]. The joint company, Recyclable-Fuel Storage Company (RFS), applied for its application for its establishment permit in 2007 aiming to begin operation in 2012.

This Mutsu project was planned as the first off-site interim spent fuel storage project away from any reactor site in our country. We plan to store 4,000 tU of spent fuels for the period of 50 years. This amount was calculated according to the following consideration:

- About 500 tU of spent fuels is generated every year from TEPCO's NPPs;
- Rokkasho reprocessing facility will be able to reprocess about 60% of the above quantity;

- The remaining 40%, that is, 200 tU/year, will be stored under our responsibility;
- TEPCO has therefore decided to build an off-site interim storage facility. The size of the facility has been designed to meet this remaining amount which will be generated over the next 20 years.

At present, TEPCO is preparing for the Mutsu project along with the effort to accumulate experience and knowledge in storing fuels in dry storage casks in Fukushima-Daiichi NPS. In the future, both knowledge attained from the Mutsu project and the on-site dry storage project at Fukushma-Daiichi NPS will be used for planning the further measures for the storage of spent fuels.

The next section gives a brief overview of our on-site fuel storage capacity at Fukushima-Daiichi NPS, which is our oldest NPS. Then in the following sections, our experience in storing spent fuels in dry casks is introduced .

2. OUTLINE OF SPENT FUEL STORAGE AT FUKUSHIMA-DAIICHI NUCLEAR POWER STATION

TEPCO operates 6 BWR plants at Fukushima Daiichi NPS, that consists of 1 BWR-3, 4 BWR-4, and 1 BWR-5. As a result of their operation, about 30 billion kWh of the electricity and about 700 fuel assemblies are generated every year. In the past, more than 9,000 fuel assemblies were transported to overseas or to domestic reprocessing facilities. The other spent fuels have been stored in the site. At present, total storage capacity at the site is 15,558 fuel assemblies that are composed of reactor pool (8,310 assemblies: unit1–6), common spent fuel storage pool (6,840 assemblies: unit 1–6) and dry storage casks (408 assemblies: unit 4, 6). This amount is nearly equivalent to 450% of the total core capacity of the six plants.

3. FEATURES OF DRY STORAGE CASK FACILITY

Fukushima Daiichi NPS got a licence to store spent fuels using 20 dry storage casks in 1994. It began its operation in the following year. Up to now, 9 dry casks are situated in the cask storage building; 4 of which are for unit 4 (37 assemblies in a cask) and others are for unit 6 (52 assemblies in a cask). Remaining 11 casks are being planned for installation in several years.

In the storage building, all cylindrical forged carbon-steel casks are stored safely in lateral position. Major specifications of the casks are listed in Table 1 and the overview of the cask storage building and the cask are shown in Figs 1 and 2, respectively.

Following is the main concept of the storage cask design:

3.1. Heat transfer

Residual heat produced inside the cask is removed through its outer shell. Then the heat is carried outside the building by a natural air-cooling system without any dynamic equipment.

3.2. Shielding

Gamma-ray is shielded by a forged carbon steel layer of 26- to 30-cm thickness.
Neutron is shielded by a boron- added resin layer, which is 14~17 cm in thickness.

3.3. Sub-criticality

In order to remain sub-criticality and realize an efficient storage, internal basket is made of B-10 added aluminum alloy.

3.4. Containment

Dual lids with metal gaskets are installed in order to realize an excellent containment.

Fuel specifications permitted for storage is tabulated in Table 2. Burnup of each fuel assembly must not exceed 40GWd/t and initial enrichiment of each fuel assembly should be 3.0 wt% or less. In addition, fuel assemblies installed in a single cask should be selected so that the average burnup of fuel assemblies in the cask be 33 GWd/t or less. Also, more than four-year cooling-off period is required to all the fuel assemblies before inserting into the cask.

TABLE 1. CASK SPECIFICATIONS

	Large type (for Unit 6)	Medium type (for Unit 4)
Number of cask	5	4
Weight	115 tons	96 tons
Length	5.6m	5.6m
Diameter	2.4m	2.2m
Number of assemblies loaded material	52	37
Body	forged carbon steel	
Neutron shield	boron-added silicon resin	
Primary(inner) lid	forged carbon steel	
Secondary(outer) lid	forged stainless steel	
Basket	borated aluminum alloy	
Cavity gas	helium	
Sealing system	dual lid system with metal gaskets	

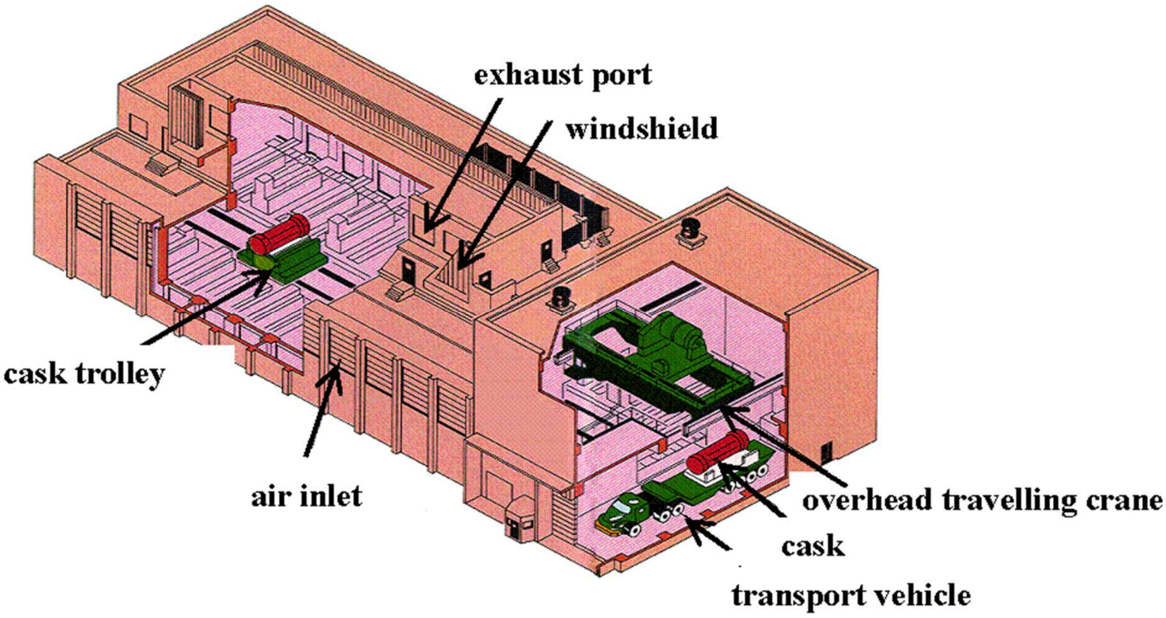


FIG. 1. Overview of the cask storage building.

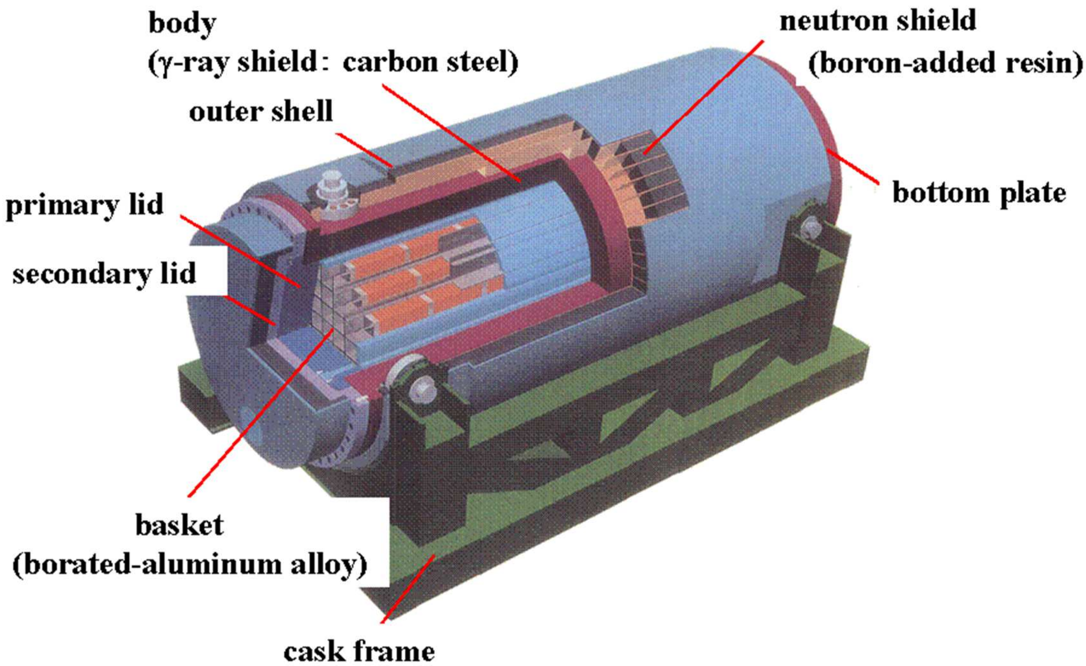


FIG. 2. Overview of the storage cask.

TABLE 2. .FUEL SPECIFICATIONS IN CASKS

	Fuel Specifications
Fuel type	8×8
Initial enrichment (%)	≤about 3.0
Average burnup in a cask (MWD/T)	≤33000
Cooling-off period (years)	≥4

4. MONITORING SYSTEM FOR EARLY DETECTION OF ABNORMAL EVENT

During the fuel installation process, helium gas is enclosed before closing the double lids. As shown in Fig. 3, pressure in the cask is kept under atmospheric pressure (0.8 atm. pressure). Space between the dual lids is kept in positive pressure of about 4 atmospheres. Two pressure censors are mounted in the secondary lid in order to mesure pressure of the space between two lids. The measured value is constantly monitored at the unit-5 main control room. In the case if the sealing performance decreases and the pressure between the lid falls to the alarm level, an alarm is set off in the control room and the defect can be detected in the early stage of an abnormal event. In addition to the pressure between two lids, several values are always monitored at the control room. The monitored values are cask surface temperatures, area radiation levels in the cask storage building, an air outlet temperature of the building, and an air temperature difference between the inlet and outlet of the building. According to this monitoring system, the safety of dry cask storage system can be guaranteed.

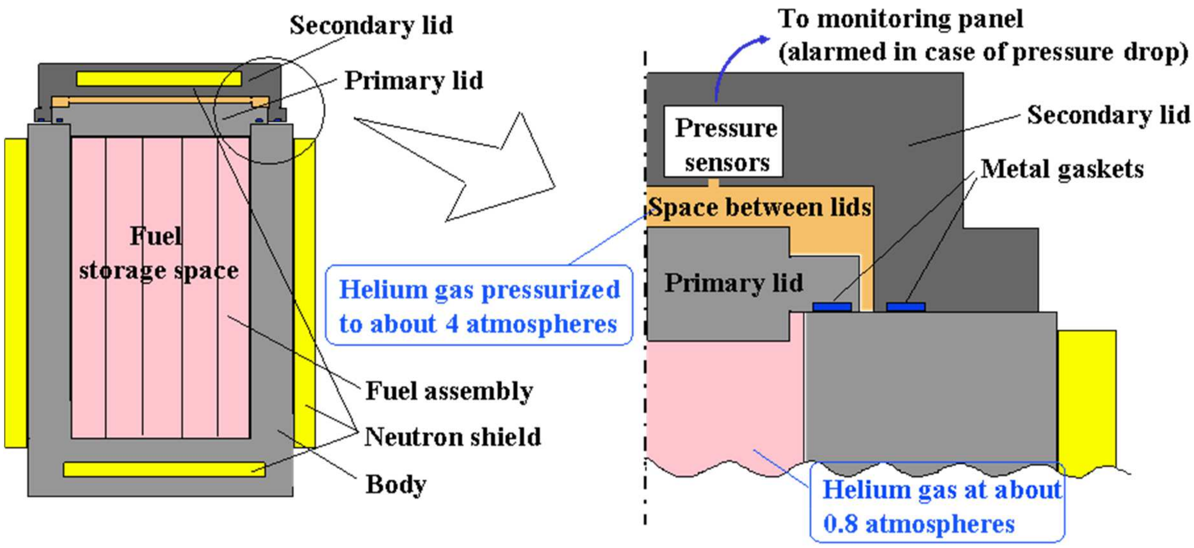


FIG. 3. Design features concerning containment.

5. INTEGRITY INSPECTION OF DRY STORAGE CASKS AND SPENT FUELS

As a basis of our quality assurance, structural soundness is investigated by various manufacturing tests. Since little stress can be generated during its fuel storage period, it seems very unlikely that any defects during manufacturing process can grow rapidly even if it existed. However since our dry storage facility is one of the few facilities in our country and the technology is still unproven, the integrity of storage casks was carefully inspected in 2000 and 2005. At the inspections, both sealing performance of metal gaskets and fuel cladding integrity were checked cautiously.

The integrity inspection of the sealing performance was composed of a visual inspection of metal gasket of the primary lid and the flange seal surface and a leak test. The examination of fuel claddings was composed of a visual inspection of a spent fuel and a Kr-85 detection within the cask by a gas sampling.

5.1. Inspection in 2000

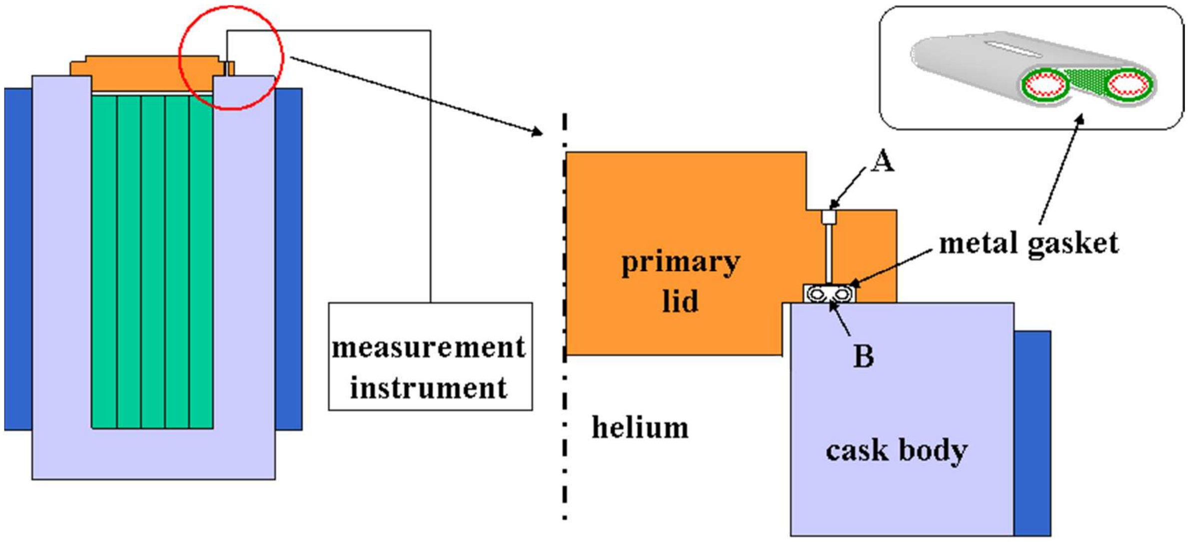
The first sampling inspection was conducted in 2000 after a storage period of five years. One of the four casks for unit 4 (37 assemblies in a cask) was selected for the test which contained the maximum amount of heat among the nine dry casks. The target cask was transported to an operational floor in a reactor building and the secondary lid was open in advance.

At the first stage of the test, sealing performance of a metal gasket for the inner lid was checked by a leak test. As shown in Fig. 4, the gap between the doubled spring of the metal gasket was vacuumed and the amount of helium gas which leaked through the gasket was measured. The result was 5.3×10^{-8} Pa/m^3/s, which was far below the required leakage rate of 1×10^{-6} Pa/m^3/s. It should be noted that although the measured leakage rate is very low, it is still considered that the value should indicate higher value than the real leakage rate. The reason for this is that since the gap between the doubled spring of the metal gasket for the inner lid were normally kept in a helium-rich condition during its storage period, some small amount of helium could still remain in the area at the time of the leak rate measurement.

Following the leak test, integrity of a fuel cladding was verified by measuring Kr-85's concentration in the gas phase. The measurement result showed no significant change in the value compared to the background level as is shown in Fig. 5 (a). It convinced us that all the fuel cladding has been kept in good condition after the five-year storage in the dry cask.

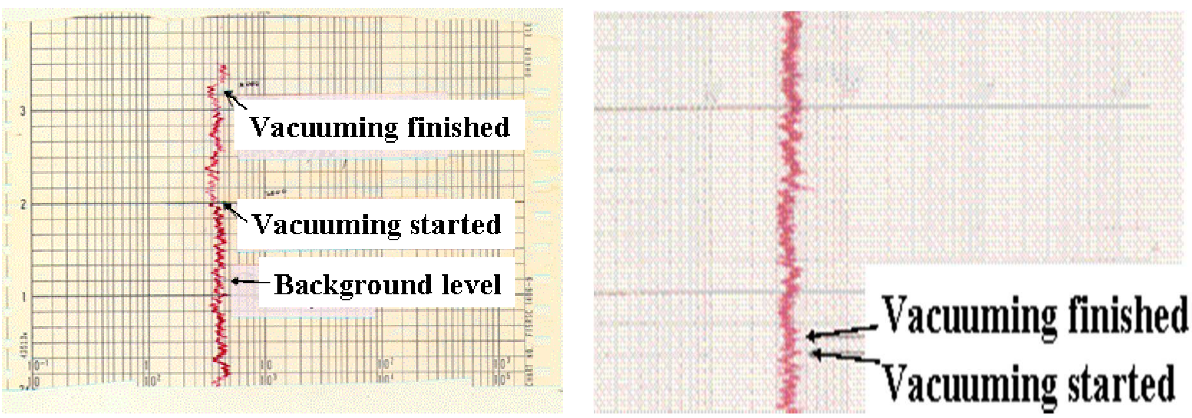
After the internal gas concentration check, the cask was moved into the spent fuel pool. Then the inner lid was open and the sealing parts were inspected visually. On this examination, it was found that the surface of the metal gasket was partially whitened as shown in Fig. 6 (a). The reason for the whitening is considered to be induced by a little amount of residual water on the metal surface when the secondary lid was closed. On the process of installing spent fuels in a cask, the space between two lids was vacuumed after the double lids were closed. During this process, residual moisture around bolt holes of the primary lid might not be removed perfectly, and it could cause of the whitening of the aluminum coating. Based on the lessons and reflections from this, our procedure manual was improved so that any bolting should be done after drying up all the bolt holes carefully. Also, metal gaskets of both primary and secondary lids of all of nine casks were replaced by new gaskets in 2001 just to make sure.

In addition to the visual inspection of the sealing system, the integrity of its fuel cladding was also verified by visual inspection tests of two fuel assemblies. These two target fuel assemblies were selected as the ones whose heat generation are among the highest in the cask, that is, about 32 GWd/t. The result showed no significant change on the appearance as shown in Fig. 7 (a).



Procedure
1. A flexible pipe is connected to a detection hole (A).
2. The other end of the pipe is connected to the measurement instrument.
3. Helium among the doubled spring of a metal gasket (B) is vacuumed by a vacuum pump which is installed in the measurement instrument.
4. Flow rate of helium gas passing through the metal gasket is measured.
5. The measured amount is converted into leak rate.

FIG. 4. Schematic diagram of the seal test equipment.

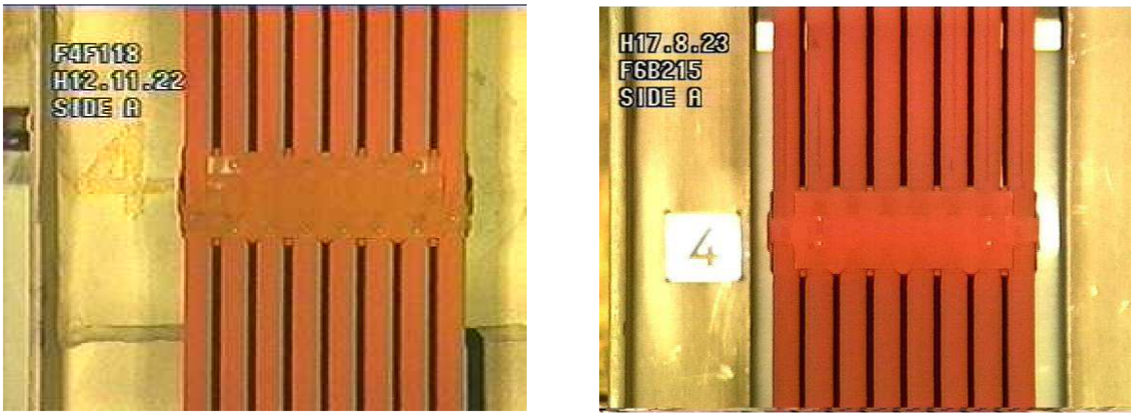


(a) Inspection in 2000 *(b) Inspection in 2005*

FIG. 5. Data sheet of Kr-85 detection.

(a) Inspection in 2000 (b) Inspection in 2005

FIG. 6. Surface condition of metal gaskets observed at visual inspections.



(a) Inspection in 2000 (b) Inspection in 2005

FIG. 7. Picture of fuel bundles after 5/10- year storages in dry casks.

5.2. Inspection in 2005

Our second sampling inspection was conducted in 2005 after a ten-year storage. Note that its integrity inspection of the seal performance was actually conducted after a four-year storage because all metal gaskets were replaced in 2001 as mentioned above. This time, the inspection was planned for five casks for unit6 fuels, which stored 52 assemblies in each cask. A cask which contained the largest heat quantity among the five casks was selected for detailed inspections.

As a result of the leak test, the leak rate was found to be 1.6×10^{-7}Pa/m^3/s which was far below the required rate. The following inner Kr-85 gas concentration measurement showed no significant change compared to the background level as shown in Fig. 5(b). Both results indicated us that even after the ten-year storage in dry condition, fuels could be stored safely.
In addition, two types of visual inspections were conducted, one for sealing materials and the other for fuel cladding. The visual inspection of the sealing material clarified that the

betterment of our procedure manual worked well and no change was observed due to residual water. However, because of other reasons, it was again found that the surface of the metal gasket was whitened as shown in Fig. 6 (b). The reason for this is estimated that the target cask spent a few days in a fuel pool before opening its primary lid. This estimation was confirmed to be true by a reproductive experiment. On that experiment, it was verified that oxide film was formed on the surface of the gasket by the contact with pool water. It was also found that the growth of the thickness became saturated in a few days in such condition.

Another visual inspection was performed in order to verify integrity of fuel cladding. The test was conducted for two fuel assemblies, which was selected as two of the highest heat-generating fuels within the cask (28 GWd/t). As a result, no any sign of degradation was found as shown in Fig. 7 (b).

Next sampling integrity inspection is planned within a few years.

6. CONCLUSIONS

Our 15-year operating experience in the dry storage facility encourages us to challenge the Mutsu project, which will be the first off-site interim spent fuel storage facility away from any reactor site in Japan. On designing the Mutsu facilities, our previous experience at Fukushima-Daiichi NPS has been reflected in many ways. In addition, our effort to gain more information about the long-term storage of spent fuels will be continued steadily. For further progress, we are planning to have periodical inspections of dry storage casks in the future too. The next inspection is going to be held within a few years and the result attained there will be reflected to the design and maintainance plan for the Mutsu project.

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INSPECTION OF FUEL CLADDING AND METAL GASKET IN METALLIC DRY CASK AT TOKAI NO. 2 POWER STATION

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Abstract

The metallic dry cask storage of spent fuel started in December 2001 at TOKAI No.2 power station. The cask that had served for 7 years was inspected in January 2009. The objective of this inspection is confirmation of fuel cladding and metal gasket integrity. This cask accommodates 8 × 8 zirconium liner type fuel. The gasket applied to this cask consists of aluminum outer lining and Inconel spring. This inspection confirmed that there had been no damage in fuel cladding and metal gasket during the storage for 7 years.

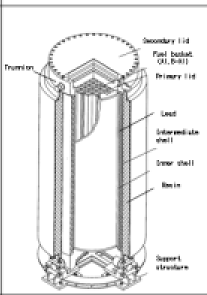
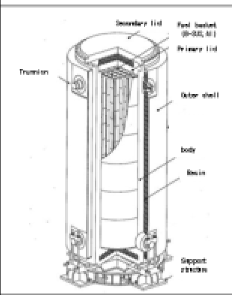
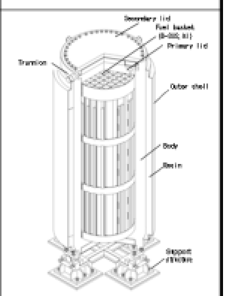
1. INTRODUCTION

The Japan Atomic Power Company has four nuclear power plants. TOKAI No.2 power station, one of four, has the spent fuel dry cask storage facility. The metallic dry cask storage of spent fuel started in December 2001 at TOKAI No.2 power station. The inspection of cask and spent fuel assemblies was conducted in January 2009.

2. SPECIFICATION OF DRY CASK STORAGE FACILITY

The storage facility at TOKAI No.2 power station can contain 24 casks. Seventeen casks are in charge in the facility. The cask cooling uses natural air circulation. The storage casks are vertically put, tied to the floor. The air view of storage facility and current state of facility inside are shown in Fig 1.

TABLE 1. THE SPECIFICATION OF CASK

Cask	No.1～15	No.16～17	No.18～21
Cask type (Air view)			
Manufacturer	Hitachi Zosen	Hitachi GE	Toshiba
Number of fuel	61		
Storage fuel type	8×8, new8×8, new8×8Zr liner, High burnup8×8		
Cooling time of fuel	7-9 years		
Status	Loaded	Empty and Reserve	Under manufacturing

The specification of cask is shown in Table 1. The manufacturer of No1.-15 casks is Hitachi-Zosen. The manufacturer of No.16-17 casks is Hitachi GE. Each cask can accommodate 61 fuel assemblies of BWR and can store 8 × 8, new 8 × 8, new 8 × 8 Zr liner and high burn up 8 × 8. The cooling time required spent fuel to load, which depends on fuel type, is about seven to nine years. The gasket applied to this cask consists of aluminum outer lining and Inconel spring.

3. INSPECTION OF METALLIC DRY CASK

3.1. Background of inspection

The metallic dry cask storage of spent fuel started in December 2001 at TOKAI No.2 power station. The Japan Atomic Power Company promised regulation authority to investigate casks and spent fuel in the licensing of metallic dry cask.

The Japan Atomic Power Company and Tokyo Electoric Company are planning intermidiate spent fuel facility away-from-reactor. In 2002, the Nuclear Safety commission in Japan demanded that electonic company should continuously investigate metallic dry cask and spent fuel in NPP for intermidiate spent fuel facility.

The investigation of metallic dry cask storage of spent fuel was done only in Idaho National Laboratory and FUKUSHIMA DAI-ICHI NPP of Tokyo electoric company.

The Inspection at TOKAI No.2 Power station is a valuable contribution to the metallic dry cask and spent fuel.

3.2. Specification of inspection cask and spent fuel

The objective of this inspection is to confirm fuel cladding and metal gasket integrity. The specification of inspection cask and spent fuel is shown in Table 2.

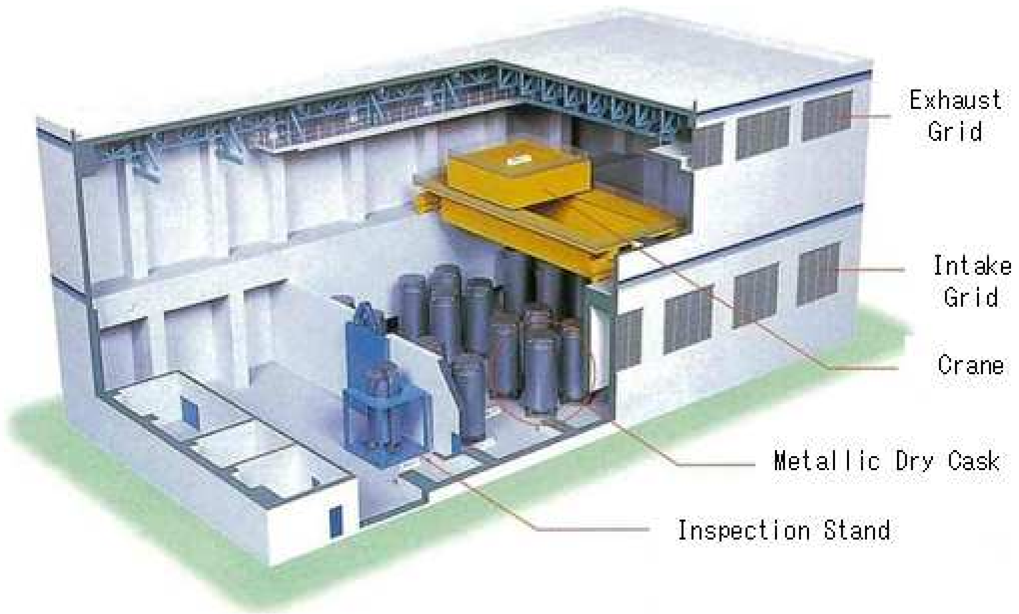
The cask that had served for 7 years was inspected. This cask is one of the first storage casks. The fuel decay heat was approximately 13 kW at the beginning of storage and approximately 10 kW at the inspection, on calculation basis.

This cask accommodates 8 × 8 Zirconium liner type fuel. The burn up of spent fuel is approximately 31,800–33,500 MWd/t. Two spent fuel assemblies were inspected in detail by underwater camera. The cooling period in wet pool before storage of these fuel assemblies were approximately 8 and 9 years.

3.3. The result of inspection

The inspection was composed by following five items:

(1) Cover gas cover gas sampling (to detect Kr-85);
(2) Visual inspection of spent fuel (two assemblies);
(3) Visual inspection of sealing surface on the primary lid;
(4) Visual inspection of metallic gasket for the primary lid;
(5) Sealing performance test of primary lid.



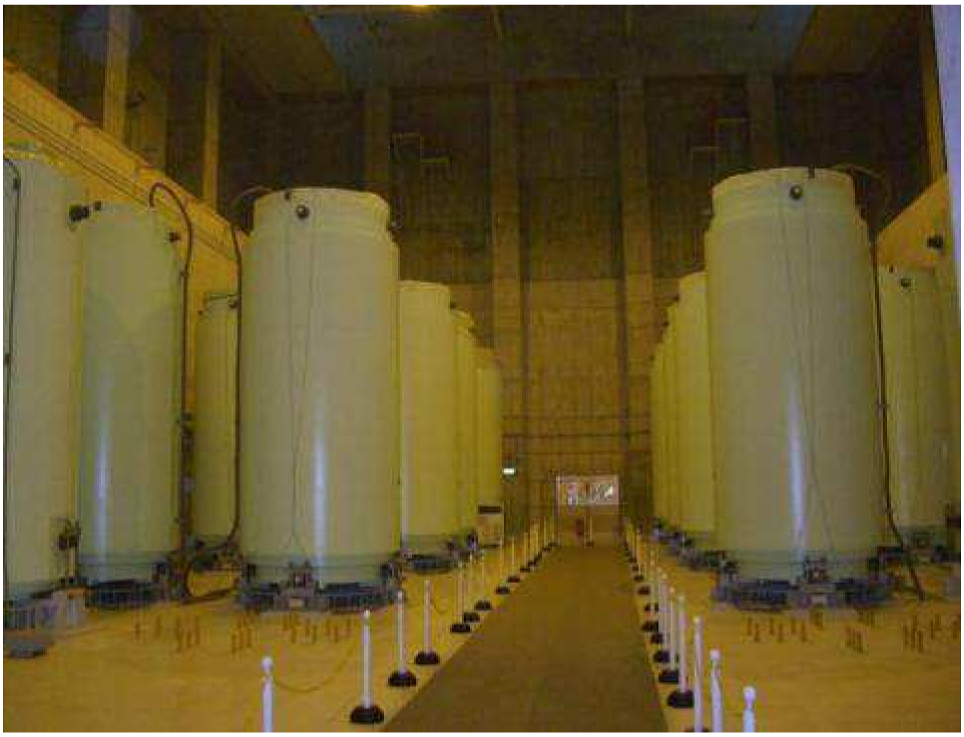


FIG. 1. The air view of storage facility and current state of facility.

The first and second inspection items were done for confirmation of spent fuel integrity. The rest three inspection items were done for confirmation of primary lid sealing performance.

3.3.1. Cover gas sampling （to detect Kr-85）

Before the cask was put into the cask pit of spent fuel storage pool, the cover gas was displaced by fresh helium gas at first and then the cavity of metallic dry cask was filled with water.

When the cover gas was displaced by fresh helim gas, all of the cover gas was led into Kr-85 detecter for gas sampling. Kr-85 is a fission product which indicates fuel cladding leakage. The concept of cover gas sampling is shown as Fig. 2. When the cavity of metallic dry cask was filled with water, all of the cover gas was led into Kr-85 detecter for gas sampling. Kr-85 was not detected for either two situations. The integrity of spent fuel cladding in the cask was confirmed.

3.3.2. Visual inspection of spent fuel

The appearance of two fuel assemblies was inspected by underwater camera. The burn up of these fuel assemblies was the highest in the cask which was inspected. The appearance is shown in Fig. 3. The appearance of spent fuel remains the same as observed at the storage starting. The integrity of spent fuel appearance was confirmed.

3.3.3. Visual inspection of sealing surface on the primary lid

The sealing surface on the primary lid and cask body was inspected by visual observation. Fig.4 shows the photograph of sealing surface. No scratch, crack or oxidation was observed on the sealing surface. The integrity of primary sealing surface was confirmed.

3.3.4. Visual inspection of metallic gasket for the primary lid

The metallic gasket in primary lid was inspected by visual observation. Fig.5 shows the photograph of metallic gasket. No scratch, crack or oxidation was observed in the metallic gasket. The integrity of metallic gasket was confirmed.

3.3.5. Sealing performance test of primary lid

The sealing performance of primary lid was inspected. The sealing performance was nearly the same as obtained at the storage starting. The leak tightness of sealing was confirmed.

TABLE 2. THE PROFILE OF INSPECTED CASK AND SPENT FUEL

Cask	Storage start		January 2002
	Period of storage		Approx. 7years
	Heating value (calculated)	Storage start	Approx. 13kW
		The time of investigation	Approx. 10kW
	Burn up of loading spent fuel		Approx. 31,800～Approx. 33,500 MWd/t
	Type of loading spent fuel		new8×8Zr liner

Spent fuel	ID	HTK016	TLJ011
	Type of spent fuel	new8×8Zr liner	
	Burn up	Approx. 33,500 MWd/t	Approx. 33,500 MWd/t

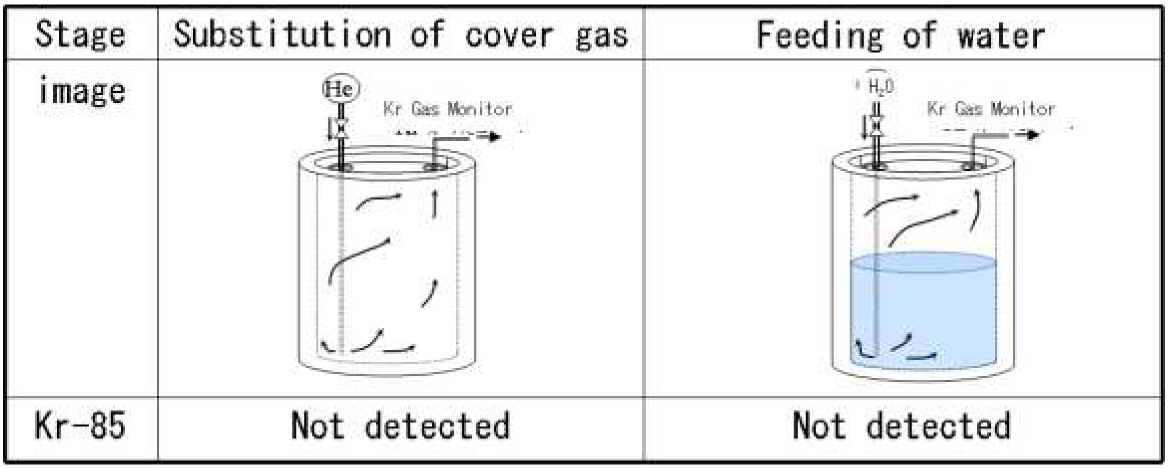


FIG. 2 The concept of cover gas sampling.

FIG. 3. Visual inspection of spent fuel.

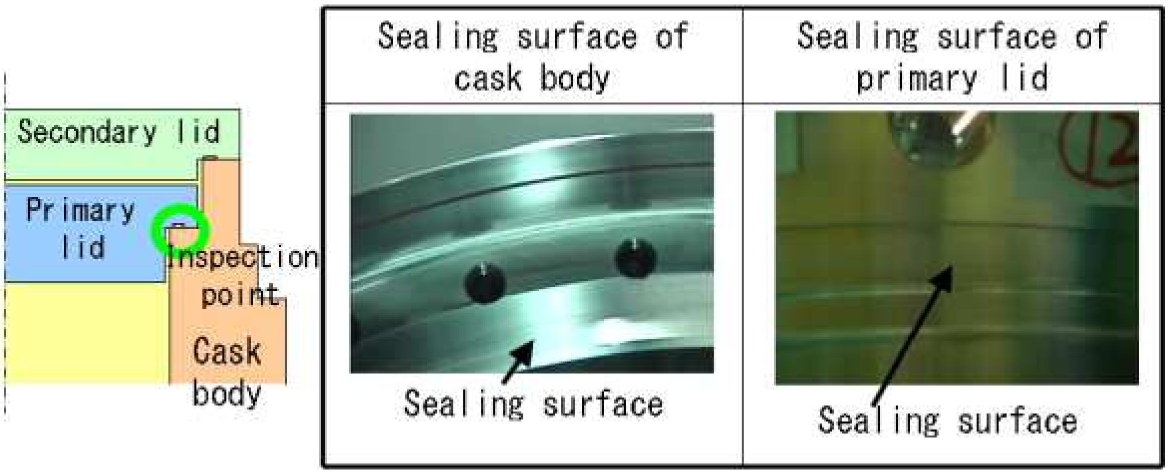


FIG. 4 Visual inspection of sealing surface on the primary lid.

FIG. 5 Visual inspection of metallic gasket for the primary.

4. CONCLUDING REMARKS

The inspection of metallic dry cask which had served for 7 years indicates following items:

(1) The integrity of spent fuel cladding in the cask was confirmed. No change in appearance of spent fuel was observed;
(2) The integrity of primary sealing surface and metallic gasket was confirmed. The leak tightness of sealing was confirmed;
(3) Since no oxidation or no sealing degradation was observed, the fuel loading procedure considering the desiccation was validated.

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ENHANCEMENT IN THE STORAGE CAPACITY OF KANUPP SPENT FUEL STORAGE BAY

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Abstract

KANUPP completed its nominal design life of 30 years in the year 2002. After refurbishment and safety upgrades KANUPP operational life has been extended up to 2019. Therefore, it is imperative to take steps either to augment the storage capacity of the bay or arrange alternative storage for spent fuel bundles that will be discharged during plant operation in future. A dry storage facility is planned for interim storage of KANUPP spent fuel bundles. This facility will serve as interim storage for next 50-60 years till shifting of the fuel bundles to repository which is the ultimate storage. This is a long term planning for the storage of spent fuel bundles. The existing storage capacity of spent fuel storage bay is approaching its limit. Therefore, there is an urgent demand to generate the storage capacity for the spent fuel bundles discharge in near future. On the basis of the studies, it is planned to place six more trays above the existing stacks of 18 tiers of trays. Therefore, two stacks consisting of 24 trays will be loaded into a High Density Tray Rack which will be placed in storage area of spent fuel storage bay. Sixty racks could be arranged in layout of 6 × 10. A total of 31680 spent fuel bundles instead of 23760 can be stored in proposed layout in existing spent fuel storage facility. By implementing HDTR system, the storage capacity of spent fuel storage bay would be enhanced for about 7900 more spent fuel bundles. This augmentation in bay storage capacity will provide the enough time to build an interim spent fuel dry storage facility for KANUPP.

1. INTRODUCTION

Karachi Nuclear Power Plant (KANUPP) has been operating since 1972. The irradiated fuel bundles discharged from the reactor are stored in spent fuel storage bay. With the completion of 30 years of nominal design life in the year 2002, it was decided to extend the KANUPP's operational life up to 2019.

The KANUPP spent fuel storage bay was built to accommodate 23,760 spent fuel bundles at the maximum. The existing storage capacity of spent fuel storage bay is approaching to its limit. Total of 23151 spent fuel bundles have been stored in the storage bay so far (up to 1st January, 2010). Therefore, it is indispensable to take steps either by enhancing the storage capacity of the existing bay or developing a substitute for the storage of cooler spent fuel bundles in order to make space for spent fuel bundles freshly discharged from KANUPP core. To handle the severe storage problem, a dry storage facility is planned that will act as an interim storage of KANUPP spent fuel bundles for next 50–60 years. However this is a long term project that will be take time for its completion. It is therefore necessary to ensure the availability of storage space in spent fuel storage bay until KANUPP spent fuel dry storage facility becomes operational.

Considering future plans and current limitation of KANUPP spent fuel storage bay, a study was conducted for enhancement in the storage capacity of KANUPP spent fuel storage bay keeping the radiation level within allowable limits set by the National Regulatory Authority. For this purpose, source term of KANUPP spent fuel bundles at different cooling times was determined using computer code. The dose rate at water surface was determined with the help of a shielding code.

The bay water cooling capacity was also re- evaluated to ascertain that the existing system is capable to remove the excess heat generated due to increased number of spent fuel bundles in storage bay beyond its designed storage capacity. Based on the study, an optimal option has been selected to augment the existing storage capacity of spent fuel storage bay by using the high density tray racking (HDTR) System.

In this way, an additional storage space for spent fuel discharge during next few years would be achieved i.e. the bay capacity would be increased. This augmentation in storage capacity of the spent fuel bay will provide the enough time to develop an interim spent fuel dry storage facility for KANUPP.

2. EXISTING STORAGE LAYOUT IN KANUPP SPENT FUEL STORAGE BAY

KANUPP spent fuel storage bay is divided into four areas, namely storage area, inspection area, shipping cask area and decontamination area [1]. All the fuel bundles discharged from the reactor core are stored underwater in storage area. This area has water depth of 5.94 m. which acts as a heat removal medium and as a shield from gamma radiation radiating from the spent fuel. 3.96 m water shield thickness helps to keep radiation level of spent fuel tray loaded with two months cooled spent fuel bundles to 8.7E-3 mSv/hr at 30.5 cm (1 foot) above the water surface [2].

The storage bay was accommodate spent fuel bundles that were expected to be discharged during 20 years of operation with 80% capacity factor, and short term storage for small quantities of booster rods [3]. At KANUPP, eleven spent fuel bundles are kept in a storage tray, which is transferred from discharge bay to storage bay. 18 trays are piled in one stack. Storage bay can accommodate at maximum of 120 stacks. This means that total 23760 spent fuel bundles can be accommodated in storage bay. The existing layout of spent fuel storage bay is shown in Fig. 1.

3. PROPOSED AMMENDMENT IN EXISTING STORAGE PATTERN

It is planned that the storage capacity of SFB will be enhanced by placing six more trays above the existing 18 tiers of trays, i.e., number of trays per stack will be increased from 18–24. In order to attain the seismic stability, it is planned to place these trays in a “High Density Tray Rack”. The rack is designed to hold 2 columns each of 24 spent fuel trays. Each rack therefore holds 528 spent fuel bundles.

In this enhancement scheme, total of sixty (60) high density tray racks are to be placed in spent fuel storage bay in an array of 6 × 10 as shown in Fig. 2. In this way, the total storage capacity would be extended up to 31680 spent fuel bundles. This implies that the storage capacity of the bay would be enhanced exactly 1/3 of design capacity (7920 spent fuel bundles). With the estimated fuel discharge rate in future, storage space for next few years would be attained. In other words, the existing bay is expected to be used to store spent fuel bundles that would be discharged during next few years.

The appendix demonstrates a comparison of enhanced storage scheme with the existing storage scheme.

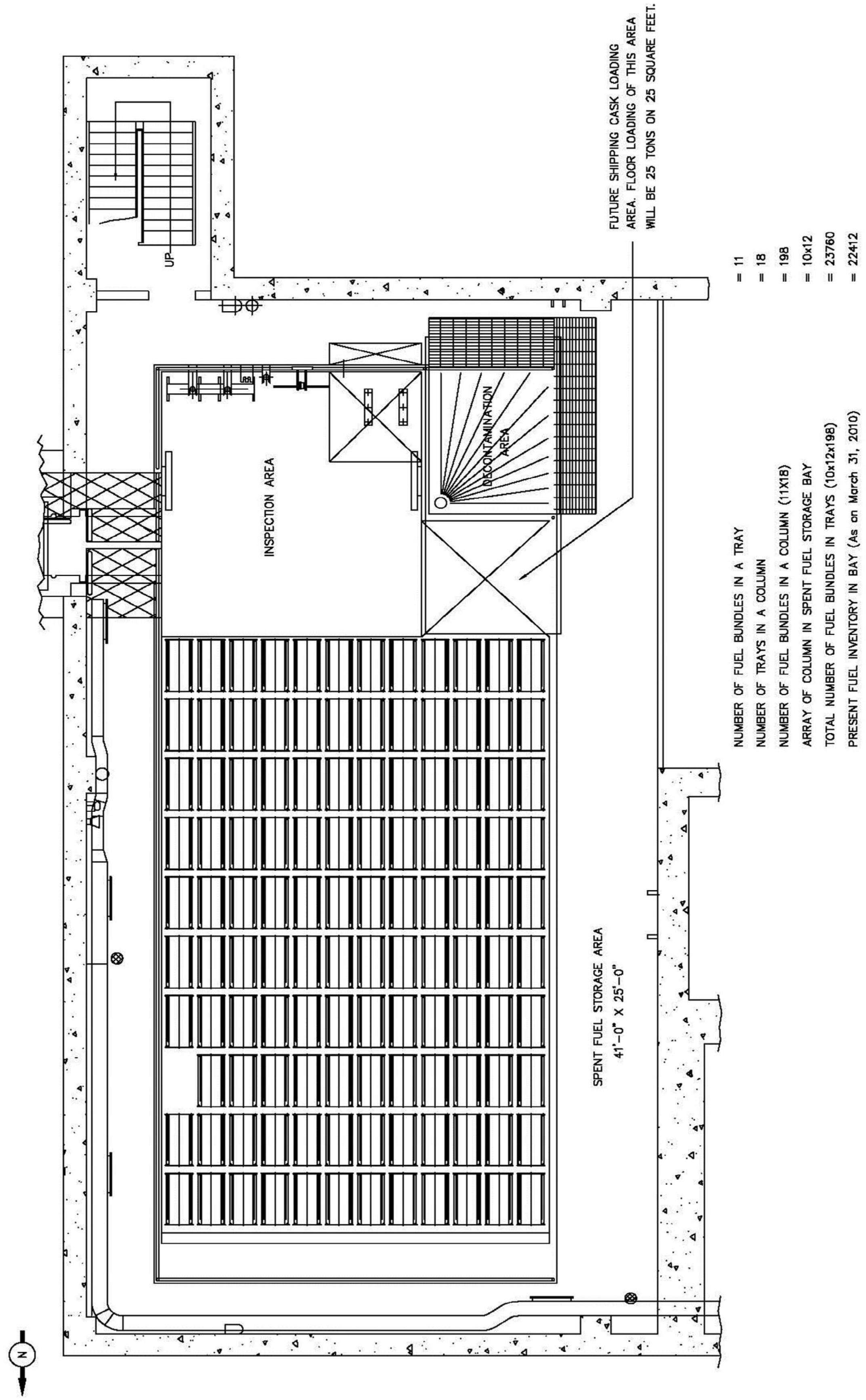

FIG. 1. Present fuel inventory in KANUPP spent fuel storage bay.

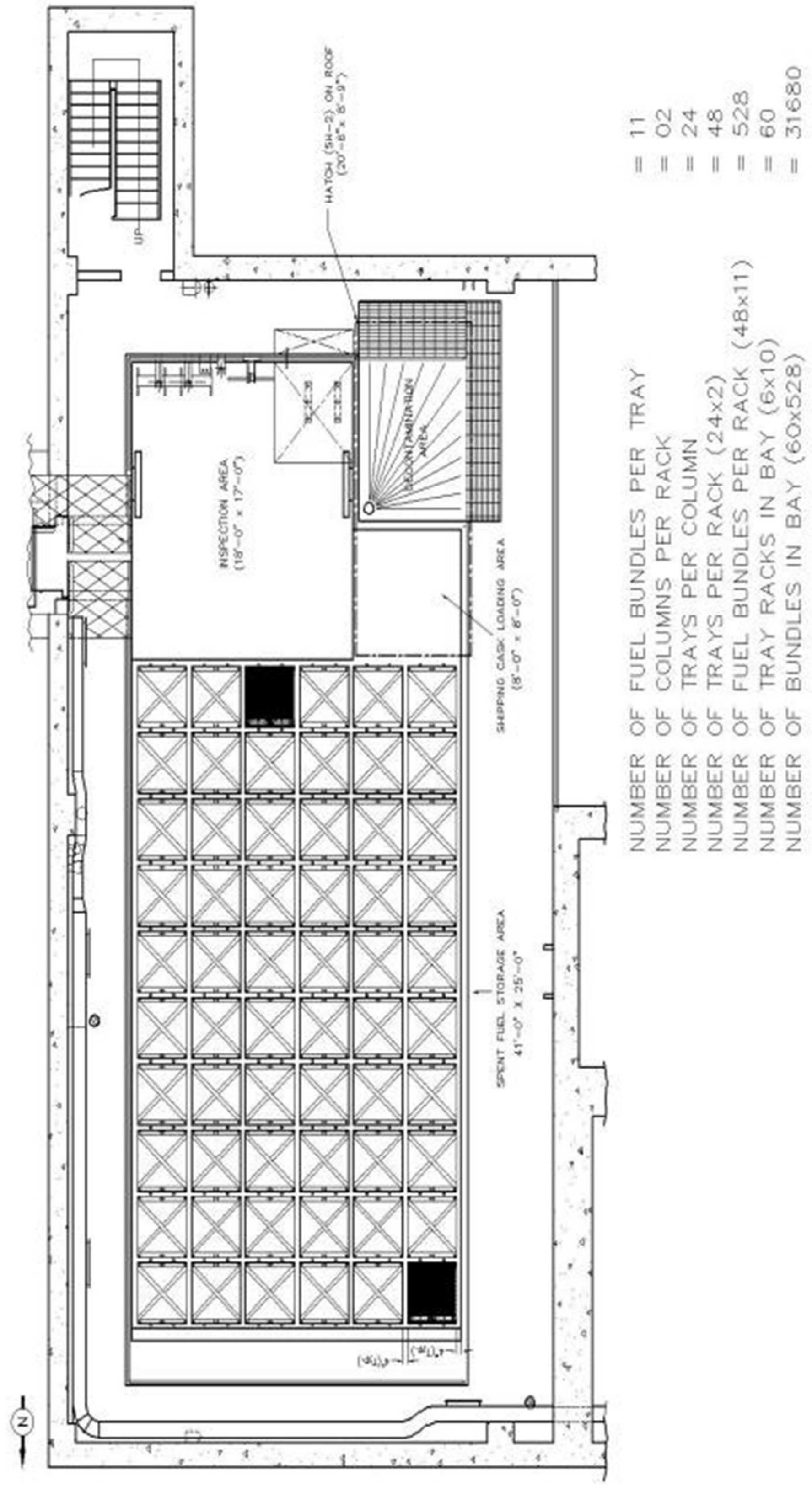

FIG. 2. Enhancement scheme (HDTR) in KANUPP spent fuel storage bay.

4. HIGH DENSITY TRAY RACK

The high density tray rack is a seismically and structurally qualified stainless steel frame to be placed in storage area of spent fuel storage bay. A typical loaded high density tray rack is shown in Fig. 3. These racks are handled using the spent fuel bay crane with the rack handling tool.

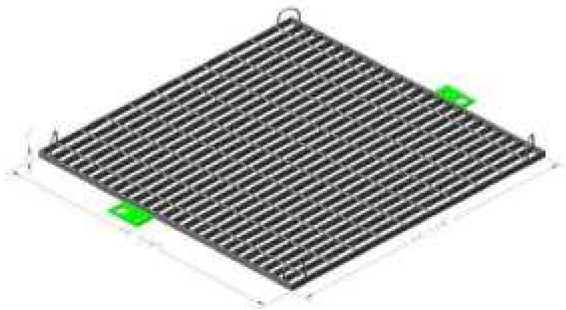

FIG. 3. Loaded High Density Tray Rack for 48 Spent Fuel Trays (528 Bundles).

5. SAFETY FEATURES OF HDTR SYSTEM

To design and operate HDTR system in both normal and abnormal conditions, following analyses have been performed to achieve the following safety objectives:

5.1. Computation of thickness of water column for shielding

Water acts as a shielding medium to provide protection for plant personnel in all handling and storage operation of HDTR system. A shielding analysis has been performed for HDTR system in spent fuel storage bay.

5.1.1. Evaluation of source term

Source term of spent fuel bundles is evaluated by employing ORIKAN computer code (modified version of ORIGEN 2 for KANUPP core). The yearly average discharge burnup profile of spent fuel bundles (Fig. 4) provides burnup variations during 1972–2009.

It can be seen in the figure that during 1977–1988, the average discharge burnup was greater than the average discharge burnup value (7400 MWD/TeU) and only once it crossed the value of 8000 MWD/TeU. Therefore, the average value of 9000 MWD/TeU is selected as a representative burnup [4]. It can provide an envelope for all average discharge burnup variations. This representative burnup value is used in source term calculations.

The power profile of bundles in a central channel and corresponding residence time is modeled in computer code. Gamma photons emanated as a result of decay of fission products and actinides is output as a function of post irradiation time.

5.1.2. *Shielding calculations*

Shielding calculations are carried out by using a shielding code. Contribution of all spent fuel bundles stored in storage bay is modeled. At the time of discharge from the reactor core, a spent fuel bundle contains large amount of several fission product radio-nuclides with half-life ranging between a few seconds and several years. These radio-nuclides continue to disintegrate, therefore with the passage of time, the activity of fuel diminishes so heat generation will be reduced. The rate of decrease of spent fuel activity and decay heat is very fast within 10 years of cooling time. Afterwards, activity and decay heat decrease rate slow down. Spectrum of a 10 years cooled spent fuel bundle in Fig. 5 reflects the photon flux due to long lived fission products only. Short-lived fission products have been decayed earlier.

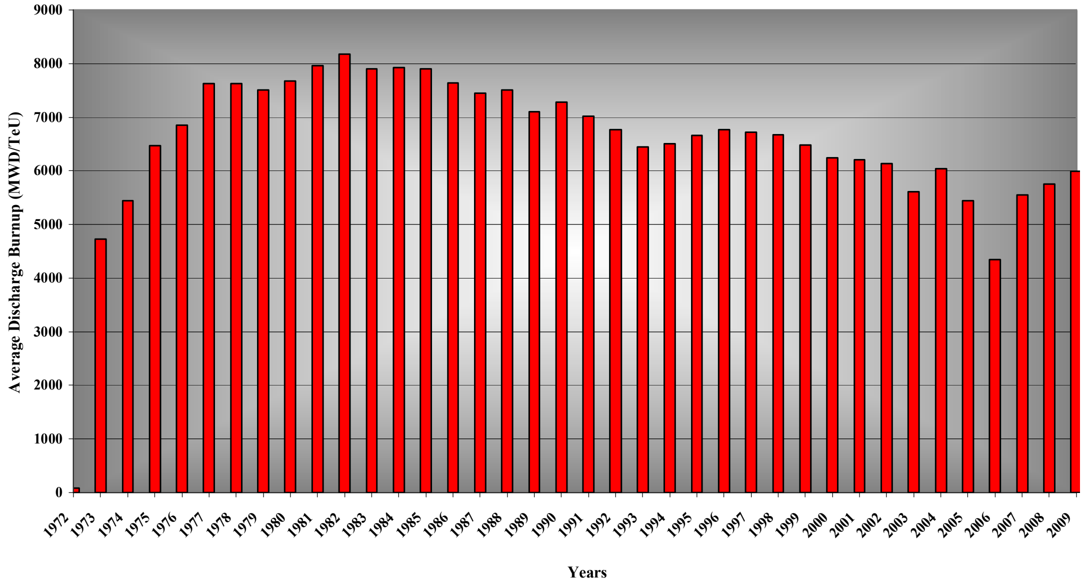

FIG. 4. Average BU distribution of spent fuel discharged during 1972–2009.

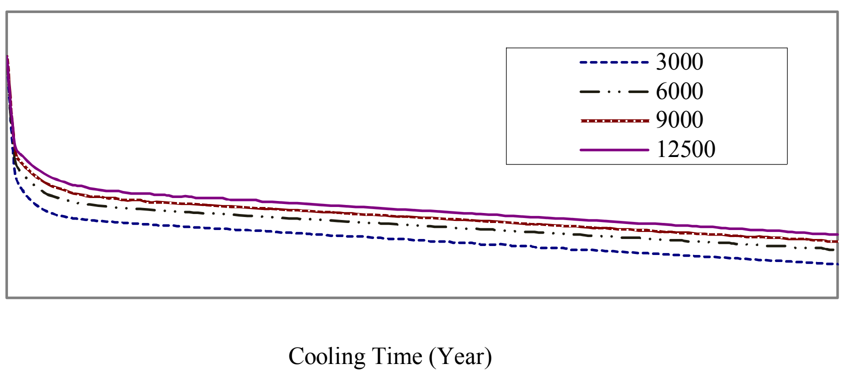

FIG.5. Source term evaluated over cooling period of 100 years at different BUs.

Total of 23151 spent fuel bundles have been stored in the storage bay since 1972. More than 72% of total spent fuel bundles have cooling time greater than 10 years as shown in Fig. 6.

For the sake of safety, it is assumed that all spent fuel bundles in every stack consisting of 24 tiers of trays are 10 years cooled and have burnup of 9000 MWd/TeU.

The values of gamma photon are used as input in shielding code to compute minimum water shield required to limit dose rate ~8.7E-3 mSv/hr at the center of SF storage bay 30.5 cm (1 ft) above the water surface. Result indicates that 2.13 m water shield thickness is sufficient to minimize dose rate below 8.7E-3 mSv/hr [5].

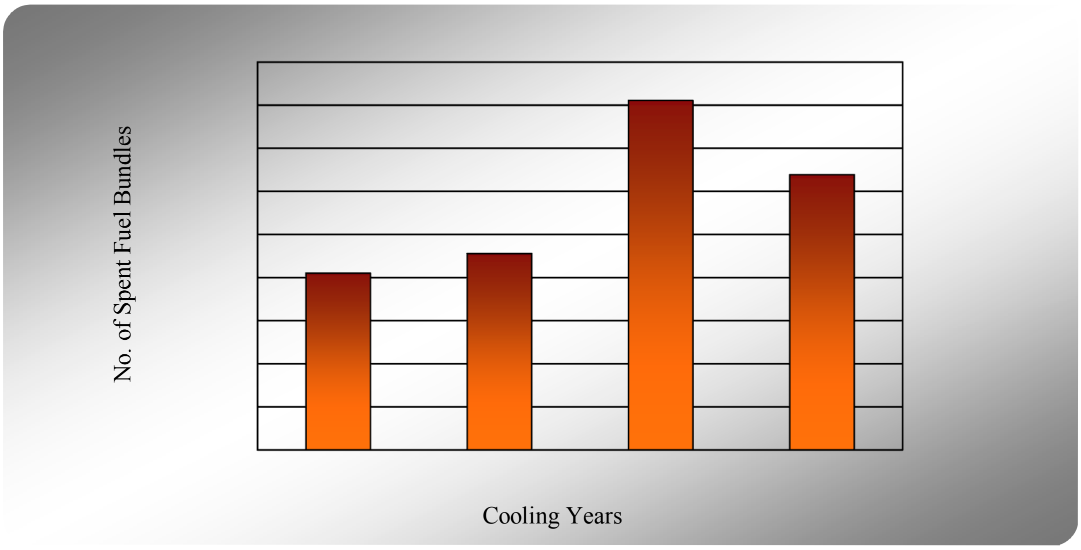

FIG. 6. Number of spent fuel bundles w.r.t. various cooling time.

The active height of stack with 24 fuel trays is about 2.44 m, so 3.51 m water column is still available to shield the spent fuel. The dose rate with available shield thickness comes out to be 2.8E-6 mSv/hr for 10 years cooled bundles having burnup of 9000 MWD/TeU. Calculations are repeated for the same dimensions of source assuming spent fuel of 5 years and 1 year cooling period and aforementioned burnup. On account of decreasing cooling time, the dose rate with 3.51 m water shield thickness increases slightly, however remain below the dose rate limit of 8.7E-3 mSv/hr as presented in Table 1.

However, during tray loading operation into the HDTR, the thickness of water shielding will be decreased momentarily to about 3.20 m. Though this momentary lessening of shielding thickness owe to a single tray, yet for the sake of safety, exposure due to one rack with 3.2 m water shield thickness is assumed. Dose rates due to 10 years, 5 years and 1 year cooled spent fuel bundles are tabulated in Table 1.

TABLE 1. DOSE RATES (mSv/hr) W.R.T. VARIOUS COOLING PERIODS AT AVAILABLE SHIELD THICKNESS

Cooling Period (Years)	Dose Rates (mSv/hr)	
	Available Water Shield Thickness (m)	
	3.51	3.20
10	2.78E-06	4.35E-06
5	6.09E-05	7.22E-05
1	1.22E-03	1.48E-03

The tray at the top of the storage rack contributes mainly in the surface dose rate. Radiations emanating from spent fuel in lower trays beneath the top tray are well shielded due to self-attenuation feature of UO_2 within the bundles as well as attenuation in steel of the tray and UO_2 of spent fuel placed over them. This feature will be used while filling of HDTR. The spent fuel trays containing spent fuel with quite short cooling periods will be stacked at bottom of rack, however top trays will consist of at least 10 years cooled spent fuel bundles.

5.2. Cooling capacity analysis

The gammas and betas are emitted continuously within irradiated fuel due to decay of fission products even after discharge from the core. These gammas and betas interact with the fuel materials and cause heat production in fuel. Spent fuel should be provided adequate cooling for the removal of decay heat. KANUPP spent fuel storage bay water provides necessary cooling and shielding medium to the spent fuel bundles.

The decay heat of a spent fuel bundle depends upon various parameters like discharge burnup, power and residence time of the fuel bundle in reactor core. The decay heat is also calculated by using ORIKAN computer Code.

Calculations show that 0.21 MW_{th} decay heat will be generated in the spent fuel storage bay due to over all 31680 spent fuel bundles. A situation may arise, which demands unloading of in-core fuel bundles. It is assessed that at least three months will be required to transfer all 2266 in-core fuel bundles to the spent fuel bay. Decay heat due to these bundles (three months cooled), is computed to be 0.27 MWth. The bay cooling system is capable to cope with heat removal requirement of 31680 spent fuels in the storage bay stored during normal course of operation as well as heat generated as a result of dumping of the whole reactor charge into the spent fuel storage bay. The total heat removal capacity of bay cooling system is 1.8 MWth [6].

The calculated total decay heat 0.48 MWth is well in limits of total heat removal capacity of Bay Water Cooling System.

5.3. Criticality assessment

One of the fundamental safety objectives of all designs for spent fuel storage facilities is to ensure that subcriticality of the entire system will always be maintained. KANUPP fuel is made up of natural uranium; these bundles are stored in light water medium after discharge from the reactor core. Criticality of spent CANDU fuel bundles in light water bay is very unlikely due to high absorption cross-section (σa) for thermal neutron of light water. However the criticality assessment of KANUPP spent fuel storage bay for HDTR system has been carried out as highly enriched booster fuel assemblies enclosed in steel cans are also stored under water in the bay. The spent fuel placed in HDTR in proposed layout in the spent fuel storage bay will remain subcritical in operational and accidental conditions. Furthermore, the use of steel in spent fuel trays, racks and liner in the surrounding walls of the bay make Keff even lesser.

5.4. Seismic analysis

A seismic analysis has been carried out to assess the stability against seismic event (ground acceleration 0.2 g):

The result of analysis reveals that overturning will not take place under the specified seismic loading. However sliding will take place which is much less than the clearance available between two adjacent racks or between a rack and bay wall. By considering conservative approach, the rack will move a maximum distance of 2.916 mm (0.115 inch), which is less than even the available clearance of 12.7 mm (0.5 inch) between trunnion of adjacent racks. A stress analysis is performed to ensure that the axial, bending and shear stresses are within the allowable limits [6].

6. HDTR SYSTEM IMPLEMENTATION

Followings activities are required to be accomplished prior to the commencement of HDTR operation. Once the activities have been completed, the placement of high density tray racks will be started.

(1) Design and development of high density tray rack and rack handling tool;
(2) Design and development of rack top cover with tool and seal;
(3) Fabrication of full scale model of rack and tool for their qualification;
(4) Fabrication of qualified high density tray racks and rack handling tool;
(5) Fabrication of rack top cover with tool;
(6) Placement of two (02) racks in bay.

A comprehensive work plan is prepared to complete the relevant tasks within time frame. All these activities have been accomplished successfully by the end of March 2010.

7. HDTR SYSTEM OPERATION

The HDTRs placement in the spent fuel bay and tray loading operation has been commenced in the month of April 2010. At first step, five adjacent stacks of trays were transferred from their storage position to the inspection area in the spent fuel bay in order to make space for the placement of rack in the storage area. The space thus created was utilized to accommodate a high density tray rack. By using service building hatch and crane, a rack was brought into the shipping cask loading area of spent fuel storage bay. From there, the bay crane picked the rack and placed in the empty space created in storage area of spent fuel storage bay. 12 spent fuel trays with least cooling period were loaded at the bottom of the rack; six in each column of the rack. These trays were covered by loading 22–23 years cooled 36 trays; 18 trays in each column.

This strategy has not only generated the space for spent fuel bundles freshly discharged from reactor core and room for placing another rack in the storage area but also helped to keep the radiation dose above the water surface within acceptable limits. Two high density tray racks have been successfully loaded so far in the presence of the IAEA Safeguards inspectors. Next two HDTRs are expected to be filled during forthcoming IAEA Safeguards inspection.

8. IAEA SAFEGUARDS

KANUPP fuel is under IAEA Safeguards. Therefore the high density tray rack and its top cover (Fig. 7) have been designed to facilitate the provision for IAEA safeguards seal (Fig. 8). Two seals have been incorporated on to the top cover of each rack by the IAEA safeguards inspectors. In addition to that the clearance of 4 inches (100 mm) between two adjacent racks and between rack and bay wall will be available to accommodate the Collimator used for annual spent fuel verification measurement carried out by IAEA inspectors.

9. FUTURE PLAN

In order to achieve ultimate solution for spent fuel storage space problem in existing bay, an interim spent fuel dry storage facility has been planned to construct within plant premises. It is also planned that the operation of HDTR will be stopped, once the dry fuel storage facility would become operational.

10. CONCLUSIONS

By implementing HDTR system, the storage capacity of spent fuel storage bay would be enhanced for about 7900 more spent fuel bundles. This augmentation in bay storage capacity will provide the enough time to build an interim spent fuel dry storage facility for KANUPP.

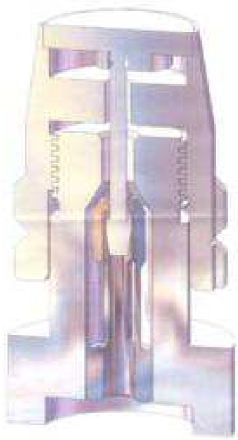

FIG. 7. A View of rack top cover.

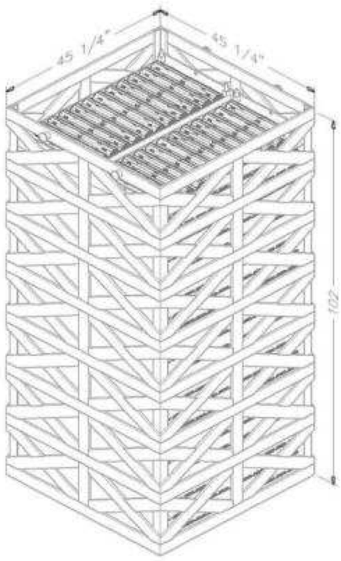

FIG. 8. A Cut view JCS seal provided by IAEA.

Appendix

Comparison between existing and enhanced storage scheme

Enhanced storage (HDTR system)	Existing storage	Parameters
48 fuel trays rack (2 × 24 trays)	18 fuel trays[1] stack	Single storage unit in bay
528	198	Number of bundles in single unit
6 × 10	12 × 10	Array in bay
60	120	Number of units in bay
31680	23760	Number of bundles in bay
33.3	-	Fuel storage advantage (%)
320	381	Available water shielding (cm)
Qualified	Not qualified	Seismic qualification (0.2 ground acceleration)

(1) A fuel tray contains 11 fuel bundles

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INVESTIGATION OF THERMAL PROCESSES AT DRY STORAGE OF SPENT NUCLEAR FUEL

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Abstract

For good safety dry storage of the spent nuclear fuel of the Zaporizhska nuclear power plant (Ukraine) investigations of thermal processes are carried out. Researches were carried out by the solution of the conjugate problems of heat exchange. The free convection of ventilating air and helium into basket of storage, and radiative heat exchange in the container was considered. As a result temperature fields of the container and air in windless day and temperature fields of the spent fuel assemblies with identical and various energy-release inside the container of a storage are obtained. Results of investigations can be used for construction of safety near-stations dry storages in other atomic power plants of Ukraine or at open dry storages of the spent nuclear fuel in other countries.

1. INTRODUCTION

The structure of Ukrainian nuclear power industry includes four nuclear power plants. Today in Ukraine works fifteen nuclear plant units — thirteen with reactor WWER-1000 and two with WWER-440. Therefore the problem of handling with the spent nuclear fuel (SNF) for Ukraine is actual. Only on Zaporizhska Nuclear Power Plant (ZNPP) are formed every year 134.4 tons of SNF. In consideration of ZNPP's power, it is almost half from all SNF in Ukraine.

For the decision of a problem of handling with SNF the technology of interim dry storage because its advantages are low cost and simple realization has been chosen. Today this technology is fixed in the “Energy strategy of Ukraine till 2030” [1].

Today in Ukraine on the Zaporizhska nuclear power plant exists unique near-station storage of spent nuclear fuel. From August 2001 the spent nuclear fuel of six reactors WWER-1000 is stored by a dry method on the open area. The dry SNF storage (DSNFS) is designed for arrangement of 380 ventilated containers, each of which have 24 spent fuel assemblies. It will provide the storage of spent fuel for all period of operation ZNPP.

General safety of DSNFS includes nuclear safety, radiation safety and safe thermal conditions for facilities operation. Full researches of thermal processes required for safe operation of DSNFS and its further modernization. This information is necessary for making thermal conditions of storage which will guarantee the nuclear safety of containers with SNF.

The work purpose is definition of safe thermal conditions of dry storage of SNF on open area at Zaporizhska NPP.

2. PROBLEM STATEMENT

Ventilated storage container (VSC) [2], which used in dry spent nuclear fuel storage (DSNFS) at ZNPP, has prototype is VSC-24 cask of American company of Sierra Nuclear Corporation. Containers are located on the open area storage at territory ZNPP.

VSC structure is shown at Figure 1. Twenty four spent fuel assemblies (SFA) is placed into tight cluster storage basket (TCSB). TCSB filled with inert gas (helium), which circulates in the internal space of the basket due to natural convection caused by temperature difference of hot SFA and cold casing of basket.

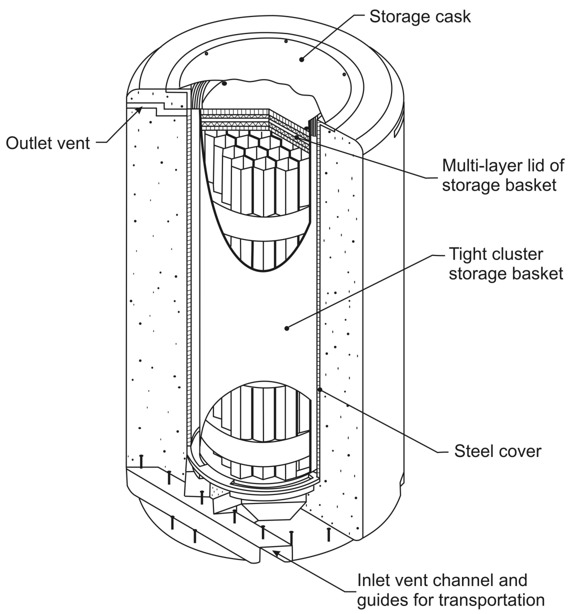


FIG. 1. Structure of ventilated container with spent nuclear fuel.

TCSB is placed into a concrete casing which is the radiation protection barrier and protects the basket from mechanical damages and other effects of the environment. Heat from basket surface is removed by natural air draft through cylindrical ventilating channel which is formed by basket casing and steel cover of concrete container casing. Heat transfer from SFA to the basket casing and from casing of TCSB to the container steel cover is carrying out by radiative heat exchange also.

The spent fuel assemblies are loaded into a basket after 5 years of storage in cooling pond. The calculated value is 0.909 kW [3] and energy-release of each SFA is not more than 1 kW.

From 2003 on Zaporizhska NPP are used the spent fuel assemblies (SFA-A) with higher energy-release. After six years of storage in cooling pond the SFA-A has energy-release 1.22 kW [3]. Thus there is a problem of storage of assemblies with energy-release more than 1kW.

Containers with spent nuclear fuel are located by groups on the open area of storage. For the control of thermal state of VSC the registration of temperature measurements on an exit from ventilating channels of the container is conducted. The analysis of these measurements has allowed disclosing factors which influence thermal condition of VSC [4]. The wind influence on thermal conditions of containers is the greatest.

Criteria of safety establish a limit 350°C for the temperature of spent nuclear fuel cladding inside TCSB. As the system of VSC cooling is passive then cooling only by air take place. The ventilating air on an exit from channels should not be heated more than on 61°C [5].

For research of thermal processes on DSNFS it is necessary to generate mathematical model which will take into account influence of a wind on temperature inside TCSB and will determine the temperature fields of gas and solid environments.

3. METHODOLOGY

Research of thermal processes on DSNFS was carried out by the solution of the conjugate problems of heat exchange.

The mathematical model viewed stationary process of thermal physics includes the following equations:

- Continuity;
- Motion of viscous fluid;
- Energy;
- Heat conductivity.

The model has two differential equations of transport:

- For turbulent kinetic energy k;
- For velocity of its dissipation ε.

For radiation heat transfer calculations used the equation:

$$Q_{12} = \sigma_0 \varepsilon \left(T_1^4 - T_2^4\right) H_{12},$$

where $\sigma_0 = 5.672 \cdot 10^{-8}$ Wt/(m^2·K^4) — Stefan-Boltzmann constant; ε — reduced of emissivity factor; T_1 — temperature of one surface (more heated) body; T_2 — temperature surfaces of the second (less heated) body; H_{12} — area of a cross surface of an irradiation which can be found from expression

$$H_{12} = \int_{F_1} dF_1 \int_{F_2} \frac{\cos\theta_1 \cos\theta_2}{\pi R_{12}^2} dF_2 ,$$

where F_1, F_2 — the areas of surfaces participating in heat exchange; θ_1 and θ_2 — corners between direct, connecting the centres of the partial surfaces dF_1, and dF_2, and the relevant normal lines to these platforms; R_{12} — and distance between these partial surfaces.

Reduced of emissivity factor may be identified as

$$\varepsilon = [1 + (1/\varepsilon_1 - 1)\varphi_{12} + (1/\varepsilon_2 - 1)\varphi_{21}]^{-1},$$

where ε_1 and ε_2 — emissivity factor of surface 1 and surface 2 accordingly; $\varphi_{12} = H_{12}/F_1$ and $\varphi_{21} = H_{12}/F_2$ — coefficients of an irradiance or the angular coefficients of radiation showing, what part of the radiation which are let out by one body, impinges on another.

Computer programs which allow solving the equations of mathematical model (ANSYS Fluent, STAR-CD, PHOENICS, etc.) can be used for definition of thermal fields and fields of air and helium velocities.

In researches two types of calculation areas were used:

(1) The concrete container with TCSB. In VSC the simplified structure of the inlet channels was considered (hydraulic resistance has been maintained). The TCSB as solid body with equivalent heat conductivity [6] was considered;
(2) TCSB with detailed structure and a concrete cover with the simplified geometrical structure of channels.

The atmospheric pressure and the temperature of atmospheric air were set as boundary conditions in calculation areas. In calculations with influence of wind the velocity of wind and the temperature of atmospheric air were set as boundary conditions.

For calculations with first calculated area the equivalent heat conductivity of TCSB is necessary. Determination method of equivalent heat conductivity is based on decision inverse conjugate problem of heat conductivity [6, 7]. Proposed method consists in the following. Two tasks are considered, one of them uses the detailed geometrical model and the other — the simplified one when the object of investigation is replaced on a homogeneous isotropic body with equivalent heat conductivity (λ_e). The determination of the last one shall be performed by the formula:

$$\sigma(\lambda_e) = \sqrt{\frac{\sum_{i=1}^{N}\left(T_i^n - T_i^y\right)^2}{N}} \to \min .$$

Where, N is the number of so called reference points which consider temperature deviation; T_i^n is temperature in the i-th reference point, received in the result of the decision of a problem with the detailed geometrical model of the investigated object; T_i^y is temperature in the i-th reference point, received in the result of the decision of a task with the simplified geometrical model of the investigated object.

This approach allows receiving the value of equivalent heat conductivity for temperature distribution in the investigated object with simplified geometry which is maximally similar to the one with the detailed geometry.

4. RESULTS

Results of researches [6] have allowed to receive value of equivalent heat conductivity — 3,9 W/mK. This value was used at calculations of temperature fields of the container and air in windless weather and with wind influence. These data are used for modelling a thermal condition of containers group on a platform of storage also.

The maximum of temperature on the territory of Zaporizhska NPP is 40°C and this temperature is most dangerous. Temperature fields of the container and air in windless day at temperature of ambient air 40°C shown in Fig. 2.

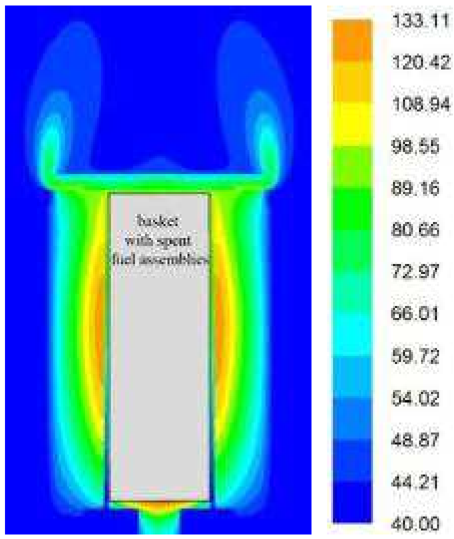


FIG. 2. Temperature fields of the container and ventilating air (°C).

The ambient air passes through inlet channels, passes between a basket and a steel cover of the container, is heated up and leaves from the outlet channels. Heated air rises vertically upwards above the container. At calculations the energy-release of each SFA was 1 kW. The maximum of temperature at TCSB is 311.5°C. Heating of air does not exceed 61°C, criteria of safety are observed.

The similar results are observed in windless conditions at other temperatures of ambient air.

The solution methodology has been checked up by comparison of measured and calculated temperatures on an exit from container channels. For example, the energy-release of container No.35 is 20.024 kW and average observed temperature on exit from outlet channel is 29.625 C (windless, ambient temperature is -10.4 C). For this container the calculated average temperature is 29.055 C. For other measurements in windless day the calculated temperatures are not differed more than on 2%. This verification of calculations has shown adequacy of model.

The researches of thermal state of single container with SNF for three directions of wind (Fig. 3) have been carried out.

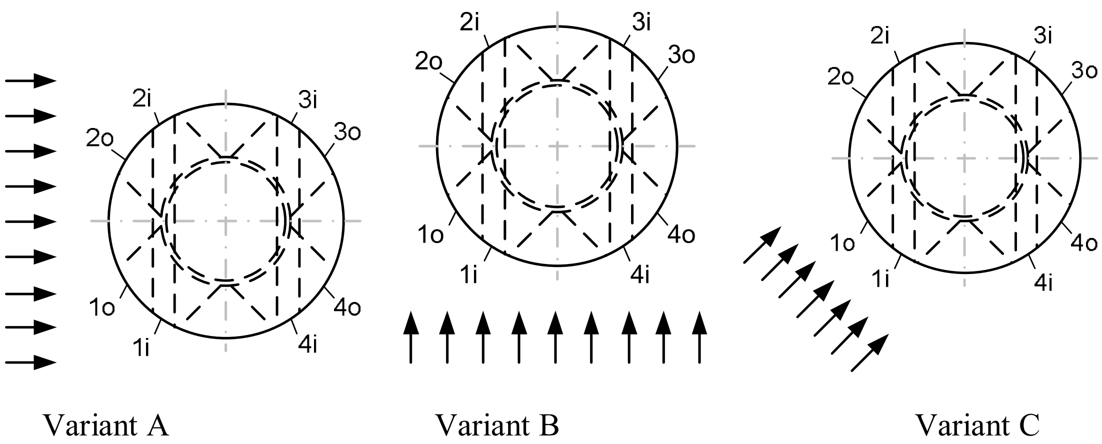


FIG. 3. Variants of wind direction (o — outlet, i — inlet ventilation channels).

Results of calculations at various velocities of a wind are given in Table 1. Small velocities of a wind (about 1 m/s) render feeble influence on the maximum temperature in VSC in comparison with calm conditions. The maximum of temperature in VSC near 3 m/s wind velocity and this speed of wind are most dangerous at storage of spent nuclear fuel. At strong wind (10 m/s and higher) the maximum temperature in the VSC is decreasing.

TABLE 1. MAXIMUM TEMPERATURE IN VSC AT VARIOUS VELOCITIES OF A WIND (T_A = 24°C, ENERGY-RELEASE OF TCSB IS 24 KW)

Velocity of wind, m/s	Maximum temperature in VSC, °C		
	A	B	C
0		294.3	
1	302.5	297.2	301.5
3	325.6	312.7	297.7
5	305.1	318.3	288.9
10	256.8	302.9	249.1

Temperature fields of the spent fuel assemblies with identical energy-release of each SFA inside the container of storage are presented on Fig. 4. At calculations the energy-release of each SFA was 0.909 kW.

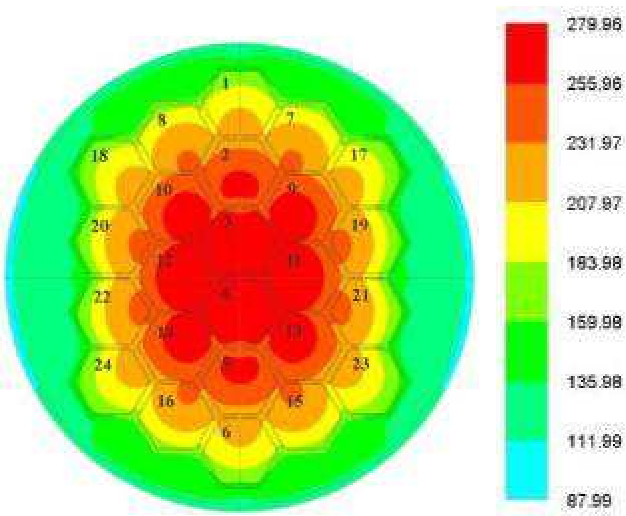


FIG. 4. Temperature field of basket with the spent fuel assemblies (°C).

In TCSB the small velocities of helium are observed. The maximum temperatures of SFA at the centre of a basket are observed. On height of TCSB the maximum of temperatures is approximately on the middle of a zone of thermal emission of SFA.

The other forms of temperature fields are observed at placement SFA with various energy-releases. It is shown in Fig. 5 and Fig. 6. Variants of filling TCSB have been considered:

(1) The SFA-A are placed in the centre of TCSB, near with axes (Fig. 5);
(2) The SFA are placed in periphery, near the cover of TCSB (Fig. 6).

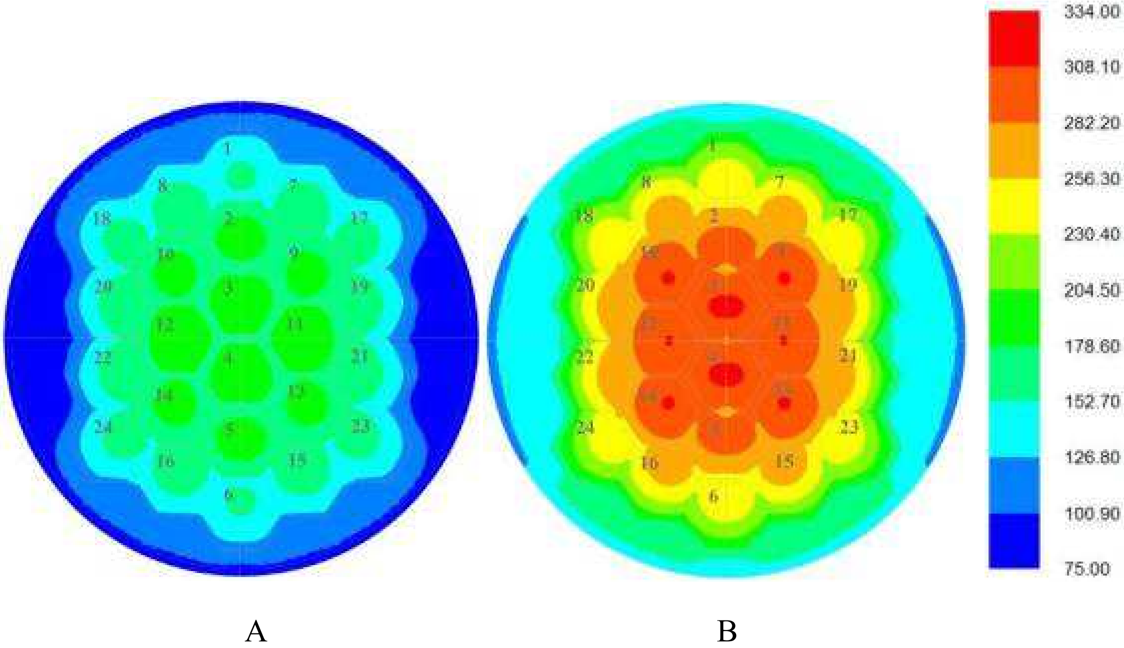
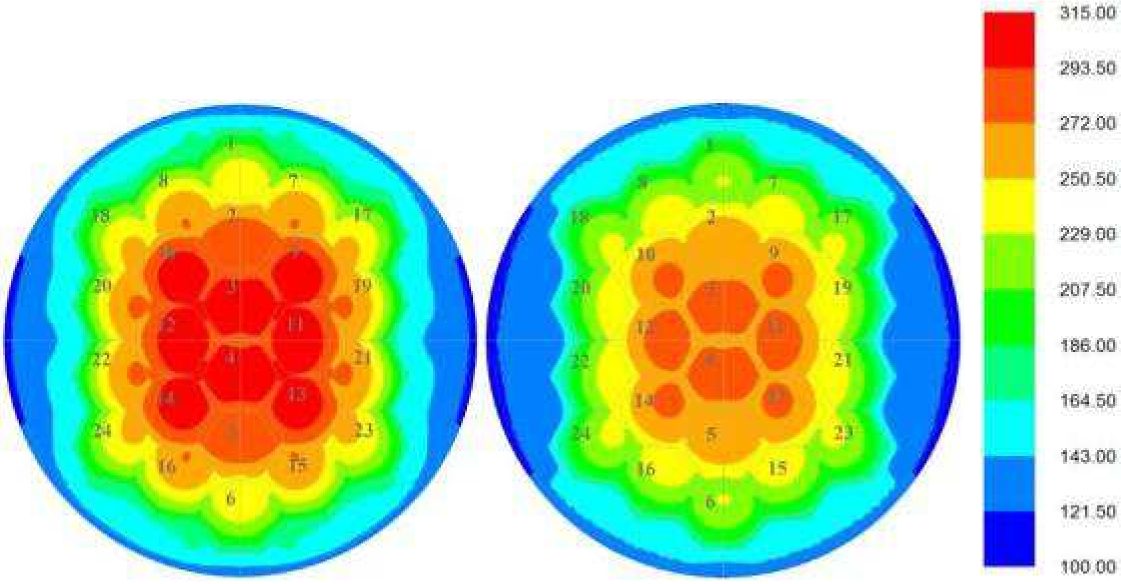


FIG. 5. Temperature field of basket with the spent fuel assemblies with various energy-releases.

In Fig. 5 are shown two variants:

(1) Variant A — six SFA-A with energy-release 1.2 kW (No. 2-5, 11, 12) and eighteen SFA with energy-release 0.909 kW (No. 1, 6-10, 13-24) (Fig. 5, A). The SFA serve as protection for SFA-A with higher gamma intensity. The maximum of temperature in TCSB does not exceed 210°C;
(2) Variant B — eighteen SFA-A with energy-release 1.2 kW (No. 2-5, 7-16, 19-22) and six SFA with energy-release 0.909 kW (No. 1, 6, 17, 18, 23, 24) (Fig. 5, B). At quantity-increasing of SFA-A in TCSB the maximum of temperatures in TCSB increase too. The temperature gradient was incremented; the maximum of temperature in TCSB is 334°C.



A B

FIG. 6. Temperature field of basket with the spent fuel assemblies with various energy-releases.

In Fig. 6 are shown two variants also:

(1) Variant A — six SFA with energy-release 0.909 kW (No. 2-5, 11, 12) and eighteen SFA-A with energy-release 1.2 kW (No. 1, 6-10, 13-24) (Fig. 6, A). The SFA were placed in centre of the basket in zone of high temperatures. The maximum of temperature in TCSB is 315°C;
(2) Variant B — eighteen SFA with energy-release 0.909 kW (No. 2-5, 7-16, 19-22) and six SFA-A with energy-release 1.2 kW (No. 1, 6, 17, 18, 23, 24) (Fig. 6, B). For best cooling spent fuel assemblies with higher energy-release were placed near the cover of basket. The maximum of temperature in TCSB does not exceed 289°C.

The SFA placing by variant which shown on Fig.6 can be only at observance of radiation safety norms.

5. CONCLUSIONS

(1) The mathematical model for definition of thermal condition of containers with spent nuclear fuel on a platform of storage is created. The conjugate problems of heat exchange were solved. The thermal influence of solid bodies and gas environments were taken into account;
(2) Verification of calculations by taken temperatures on exits from ventilating channels of the container has model adequacy;
(3) Results of modelling have shown operation safety of containers at high temperatures of a free air (40°C), because heating of air in the ventilating channels does not exceed 61°C;
(4) The placing of containers on storage platform is necessary in view of an annual wind rose for lower influence of a wind on their thermal state. The most dangerous of wind

directions is direction "A" (perpendicularly of inlet channels) and with velocity about 3 m/s when the ventilating air does not leave freely from outlet channels. Results of investigations can be used for construction of wind protection system for containers;

(5) The placing in TCSB the spent fuel assemblies with energy-release more than 1 kW was considered. The temperature in TCSB is lower when the SFA-A are placing near the cover of basket. This variant of placing can be only at observance of radiation safety norms;

(6) Results of investigations can be used for construction of safe near-stations dry storages in other nuclear power plants of Ukraine or at open dry storages of the spent nuclear fuel in other countries.

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