POSTER SESSION

MANAGING AGEING IN SPENT NUCLEAR FUEL STORAGE FACILITIES

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Abstract

Spent fuel pools (SFP) that are outside containment system without redundancy whose failure could release radioactive material that exceed allowable limit. If SFP have to continue to operate for long term after power plant shutdown it is essential to develop an ageing management program within the general life management program of the nuclear power plant. This work refers to the Atucha I nuclear power plant (NPP) SFPs. The fuel assembly (FA) of Atucha NPPs is 6 meter long and encompasses 36 Zircaloy-4 cladded fuel rods. For these spent fuel assembies (SFA) there are two storage buildings located adyacent to the reactor building. One of the alternatives considered at the end of Atucha I operation ist to transfer all SFAs to dry storage, another one is to continue the operation of the SFPs and to transfer to dry storage just a selected amount of SFAs. For the selection of the dry technology it should be kept in mind the characteristics of the Aturcha SFA, in particular, its length and burnup which differs according to the discharge date because of the use of natural uranium (NU) or slightly enriched uranium (SEU). Therefore, the fundamental point here is to keep in mind that it is the effect of ageing due to time and use that cause net changes in the characteristics of a System, Structure and Component (SSC). We employ formal processes to systematically identify and evaluate the Critical Systems, Structures and Components (CSSCs) in the facilities. A Technology Watch Programme is being established to ensure that degradation mechanisms, which could impact on facilities life, are promptly investigated so that mitigating programmes can be designed. With this methodology we analyse the following components of the pools, concrete wall stability, integrity of concrete structure, pool lining, and integrity of metal structure, pipe failures, degradation in storage racks and SFA degradation.

1. INTRODUCTION

Nowadays, Argentina has two nuclear power plants (NPP) in operation, Atucha I (CNA I) since 1974 and Embalse (CNE) since 1983, and one NPP under construction, Atucha II (CNA II). All three NPPs are based on natural uranium as fuel with heavy water as coolant and moderator.

The Atucha I (370 MWe) and Atucha II (750 MWe) NPPs are almost unique in their type, both employ a pressure vessel. The Embalse NPP (600 MWe) is a typical CANDU which employs pressure tubes. Since September 1994, the organisation of the Argentine nuclear activities, confirmed in April 1997, is as follows:

— “Nucleoeléctrica Argentina Sociedad Anónima" (NA-SA), which operates the Atucha 1 and Embalse NPPs;

— "Autoridad Regulatoria Nuclear" (ARN), which is the national regulatory board of nuclear activities;

— "Comisión Nacional de Energía Atómica" (CNEA), which is the national atomic energy commission.

Within this scheme, one of the main activities undertaken by CNEA is to provide technological assistance to NA-SA concerning NPP operation. Works on life management of NPPs are included within these activities.

Atucha I has two SFP buildings (houses) where the SFAs are stored. From August 2001, all SFAs removed from the Atucha I core are of slightly enriched (0.85%) Uranium (SEU) dioxide. Because of the use of SEU instead of NU the average burnup increased from 5900 MWd/tU to 11300 MWd/tU and, consequently, the yearly amount of SFAs discharged decreased from about 430–230.

Atucha fuel assembly (FA) characteristics:

- Fuel material: NU dioxide;
- Number of FA in core: 252;
- FA array: Bundle;
- Number of fuel rods: 36;
- Number of empty rods: 1;
- NU dioxide pellet diameter: 10.6 mm;
- NU dioxide pellet length: 12 mm;
- Total FA weight: 210 kg;
- Total FA length: 6180 mm;
- FA cross-section diameter: 108 mm;
- Fuel rod length: 5650 mm;
- Active length: 5295 mm;
- Cladding material: Zircaloy-4;
- Fuel material weight: 152.5 kg NU/FA;
- Structural components (head and foot) material: Austenitic steel;
- Spacers material: Inconel;
- Reloading of FA: During full power service.

The handling, transfer and storage of SFAs is done under light water. In Pool House 1 the facility has three interconnected pools. One is used to assemble and disassemble FAs and the other two to store SFAs.

In Pool House 2 there are five interconnected pools. One of them is used for operation and the other four for storage. The watertight sheathing of the pools is built of steel plates and has concrete embedded supporting structure. It is designed for a temperature difference of about 35°C. The SFPs are equipped with a cooling system and emergency spillways.

The sheathing plates are welded to the supporting structure and to the anchoring plates, and the water tightness is tested. All parts in contact with SFP water are made of stainless steel. SFAs are immersed and hang vertically from stainless steel frames. The SFAs are cooled through natural circulation of the SFP water.

The cooling system of the SFPs has the purpose to remove the decay heat from the SFAs stored. The decay heat is finally discharged through a heat exchanger to the water of the river. The extraction of SFP water is carried out by means of two overflow systems per pool which drive the water to coolers that re-inject it as cold water in the SFPs. The feeding pipes end near the bottom of the SFPs, so that the water flow in the SFPs is upwards. In the coolers the pressure of the secondary side is higher to avoid that radioactivity be discharged to the water of the river. The extracted SFP water is subjected to a purification process in the sedimentation filters and once cooled it is re-injected in the SFPs.

The SFP water as well as the water added to replace losses flow through an ion exchange system, automatically regulated by floating valves. To avoid SFP drainage by siphon effect in case of an eventual break of a pipe, each pipe is filled with a pipe of usually open ventilation. The water purification system also supplies the pipes that surround the SFPs which clean the SFP walls and reduce the radioactivity associated to aerosols.

The purification and cooling system of the SFPs is designed to maintain the SFP water temperature below 32°C under normal conditions of operation when the pool contains SFAs discharged during ten years of operation at a NPP load factor of 80% and for a cooling water temperature of 22°C.

The main purpose of the purification and cooling system is:

(1) To maintain the temperature of the SFP water — in which failed as well as intact SFAs are stored — below 32°C for a cooling water temperature of 22°C;

(2) To maintain the purity and cleanness of the SFP water to allow the observation of the SFA handling operations under the water;

(3) To maintain an appropriate capacity of extraction of radioactive material in suspension and dissolved in the SFP water to allow the access of personnel to the working areas;

(4) To maintain the minimum level of water in the SFPs to assure appropriate shielding during all the phases of the SFA handling procedures and SF storage;

(5) To reduce to the minimum acceptable the mixture of the water of the SFP with that of discharge and reception of the fuel. Also to reduce to the minimum the mixing of water from the SFP storing failed SFAs to the water of the SFP storing intact SFAs and the water of the fuel reception pool. In this way the dispersion of radioactive material from failed SFAs to the water of the SFPs is avoided.

There are two separated purification and cooling systems: One for the SFP and another for the fuel reception pool. The system of the SFP is primarily designed to remove the SFA decay heat.

The objective of this design is to provide operational flexibility to satisfy different purification needs without making an unnecessary duplication of equipment.

To control the transfer of the SFAs, underwater lighting is available in all SF and reception pools. The lighting can be vertically and horizontally adjusted. The handling system is made of a travelling crane with p devices to assemble and disassemble FAs. The travelling crane can be displaced along all SF and reception pools.

2. METHODOLOGY

It is fundamental importance to keep in mind that the effects of ageing due to time and use cause net changes in the characteristics of a System, Structure or Component (SSC). We employ formal processes to systematically identify and evaluate the Critical Systems, Structures and Components (CSSCs) in the facilities. Afterwards, we apply a plan to ensure the facilities surveillance, operation, and maintenance programs, monitoring and control of the component degradation within the original design specifications, essential for the facilities life attainment.

A Technology Watch Programme is being established to ensure that degradation mechanisms, which could impact on facilities life, are promptly investigated so that mitigating programmes can be designed.

The methodology we will use for these studies is shown on Figure 1.

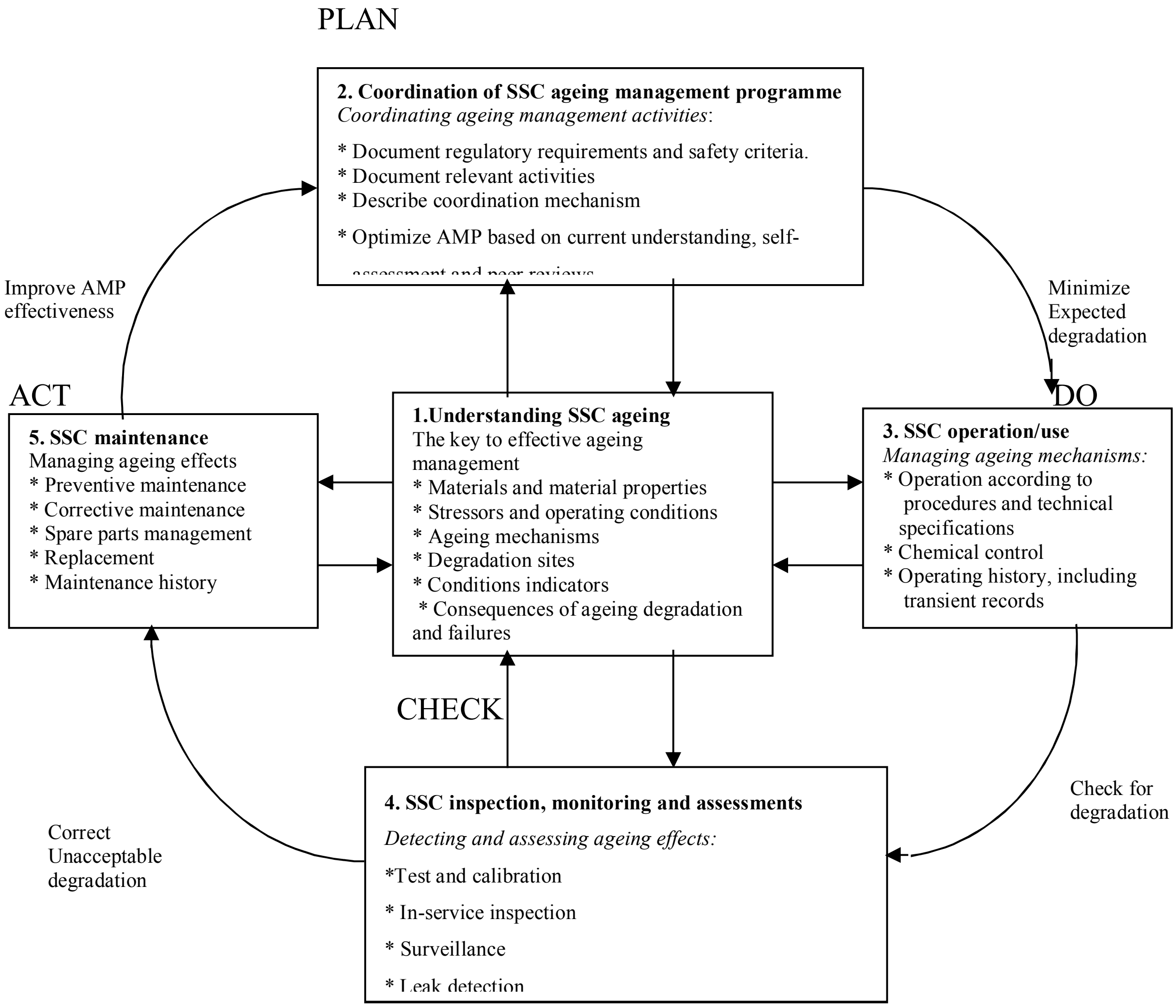

FIG 1. Methodology used for ageing management.

3. RESULTS

After analysing both Pool Houses at Atucha I NPP, we have obtained the following results:

— Concrete wall stability: For all pools, structures were carried out, only visual inspection showed that they do not present fissures, ruptures or shelling material;

— Integrity of concrete structure: The structure is in a perfect state;

— Pool lining: It does not show signs of corrosion;

— Integrity of metal structure: It does not show signs of corrosion;

— Degradation in storage racks: They do not show signs of degradation;

— Pipe failures: Flaws of the system have not been reported.

4. CONCLUSIONS

The Pool House II at Atucha I NPP can be operated independently of the reactor system, except in the case of the water of the water cooling systems. Therefore it would be necessary to carry out a modification in the facilities of the cooling water to be able to continue with their operation after the end of operation of the NPP. The Pool House I can be operated from the Pool House II with few modifications in the systems. Both Pool Houses are in very good conditions.

If the country decides to tranfer SFAs to dry systems we think that the better solution is a canister-based system. The canister is loaded with SFAs within the SFP. The canister is drained, vacuum dried, decontaminated and pressurised with helium. The canister is transfered to a vertical concrete cask system and the casks may be enclosed in buildings. The system must be modified for the characteristics of the Atucha fuel, low burnup, large fuel (length: 6 meters) and long cooling time.

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SPENT FUEL MANAGEMENT IN BULGARIA

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Abstract

The report presents the legislative framework in the Republic of Bulgaria for spent fuel (SF) management; storage facilities for spent fuel (at reactor spent fuel storage/reactor pond, away from reactor spent fuel storage facility (SFSF) and the dry storage facility), as well as the SF transportaion back to Russia. The policy of the Republic of Bulgaria regarding the management of SF and radioactive wastes (RAW) has been based on the moral principle of avoiding to impose undue burdens on future generations.

1. GENERAL INFORMATION

The Act on the Safe Use of Nuclear Energy (ASUNE) and the secondary legislation regarding its application has governed the public relations as regards the safety of spent nuclear fuel. The ASUNE, the regulations and the guide drafts based on the best international practices have been taken into consideration as well as the recommendations ensuring the observance of internationally approved principles such as: Maintaining the exposure as low as reasonably achievable, clear assigning of responsibilities regarding the use of nuclear energy and the management of SF, considering the interdependence between the separate stages in the management of SF, implementing the defence in depth principle for nuclear facilities (NF), improvement of the radiation protection and the emergency planning.

The policy of the Republic of Bulgaria regarding the management of SF is defined in the ASUNE and the Environmental Protection Act.

On 1 January 2007, Bulgaria became a member of the European Union and since then the national nuclear programme has been developed in cooperation with the European institutions. Harmonization of the national legislation with the EU one has been undertaken and representatives of the Republic of Bulgaria have already taken part in the work of a number of organizations, commissions and work groups dealing with the problems of safety within the frames of the Union. The obligations to inform the European Commission about the implementation of the requirements of the ‘acquis communautaire’ have been observed. A document entitled National Report on the Radiation Protection in the Republic of Bulgaria has been issued to present the current status of radiation protection in our country, the existing problems in this area and the actions required for their resolution. By the end of 2009 it has been envisaged to develop, revise and update a total of 18 regulatory documents on radiation protection, including amendments to the ASUNE.

2. EXISTING NUCLEAR FACILITIES IN BULGARIA

The existing nuclear facilities in Bulgaria are as follows:

- Unit 1 (standard WWER-440/V – 230):

- Commissioned in Oct. 74;
- 23rd fuel cycle completed;
- Final shut-down in Dec. 2002;

* Unit 2 (standard WWER-440/V – 230):
  - Commissioned in Nov. 75;
  - 24th fuel cycle completed;
  - Final shut-down in Dec. 2002;
* Unit 3 (enhanced WWER-440/V – 230):

— Commissioned in Dec. 80;

— 22nd fuel cycle completed;

— Final shut-down Dec. 2006;

* Unit 4 (enhanced WWER-440/V – 230):

— Commissioned in Jun. 82;

— 21st fuel cycle completed;

— Final Shut-down in Dec. 2006;

* Unit 5 (standard WWER-1000/V – 320):

— Commissioned in Sep. 88;

— 17th fuel cycle (at present the 17th fuel loading cycle is undergoing);

* Unit 6 (standard WWER-1000/V – 320):

— Commissioned in Dec. 93;

— 15th fuel cycle in operation;

* Wet spent fuel storage facility with 4 pools:
  - Design: 1978–1982;
  - Construction: 1982–1988;
  - Commissioning: 1990;
  - Max. capacity is 4920 Fuel Assemblies in 168 Baskets;

- Dry spent fuel storage facility project:

— Under construction.

3. MANAGEMENT PRACTICES IN BULGARIA

- According to the design requirements of WWER-440 reactors at KNPP SF is stored for a period of three years in the reactor ponds before transportion back to Russia for processing. In 1985 a decision was taken that the reactor pond storage period for SF from WWER-440 reactors should be increased from 3–5 years. This necessitated the construction of a wet storage facility on the site of Kozloduy NPP and it was commissioned in 1989.

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The SF from units 1–4 is stored in the spent fuel ponds by the time it is transferred to the SFSF or to Russia. The SF pond for WWER-440 assemblies is designed to store the fuel assemblies in two rows of racks: the upper racks can be removed, while the lower ones are fixed. Currently, units 1–4 hold operational licences for operation in wet mode storage of SF in the reactor ponds. There is no any fuel in the reactor ponds 1 and 2, they are fully empty. The fuel in the reactor ponds 3 and 4 occupies one row of racks.

- SF from units 5 and 6 is being stored in reactor ponds 5 and 6 pending its transport to Russia or the SFSF away from reactor. The ponds are located in the containment of the respective unit. They consist of 4 bays which are physically separated by partition walls. Three bays are allocated for immediate storage of the spent assemblies, while the fourth bay is used for transport and handling operations with fresh and spent fuel. The racks and hermetic panels for placing and storage of defect assemblies are inside the fuel storage areas.

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The spent nuclear fuel storage (SFSF) away from reactor on the site of KNPP is a wet type, storage. The fuel is stored underwater in four ponds. The spent fuel assemblies are located in transport baskets. In 2006 the storage facility was furnished with a refuelling machine for handling of spent fuel from WWER-440 and WWER-1000 reactors.

3.1. Construction of Dry Storage facility (DSF)

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In 2008 the design of Dry Spent Fuel Storage Facility (DSFSF) on the site of Kozloduy NPP was approved and a construction permit for the facility was issued.

- Pursuant to the national strategy for SF and RAW management, the Updated Decommissioning Strategy of KNPP Units 1–4 and the Framework Agreement with EBRD for funding, Stage 1 of the dry spent fuel storage facility is currently being built. This stage envisages storage of 2800 WWER-440 assemblies for a period of 50 years. Commissioning is scheduled for 2010. The storage technology will use a system of CONSTOR 440/84 casks cooled through natural air convection and having a load capacity of 84 assemblies. The casks will be loaded with spent fuel in the existing SFSF and after that will be located in the DSF. The capacity for handling and preparation for storage is 420 assemblies per year, which means 5 CONSTOR 440/84 casks per year.

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3.2.Facilities for SF management and accounting

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The Republic of Bulgaria has the following SF management facilities with the respective SF quantities and characteristics (by 31.12.2009):

Facilities of Kozloduy NPP Plc.:

- SF reactor storage at unit 3 (Reactor pond - 3):
  - Location: the central reactor hall of units 3 and 4, adjacent to unit 3;
  - Purpose: storage of SF from unit 3;
  - Storage method: under water in two racks;
  - Storage capacity (number of assemblies): 728;
  - SF assemblies stored: 365 pcs.
- SF reactor storage at unit 4 (Reactor pond - 4):
  - Location: the central reactor hall of units 3 and 4, adjacent to unit 4;
  - Purpose: storage of SF from unit 4;
  - Storage method: under water in two racks;
  - Storage capacity (number of assemblies): 726;
  - SF assemblies stored: 373 pcs.
- SF reactor storage at unit 5 (Reactor pond - 5):
  - Location: the central reactor hall of units 5, adjacent to the unit;
  - Purpose: storage of SF from unit 5;
  - Storage method: under water in one rack;
  - Storage capacity (number of assemblies): 612;
  - SF assemblies stored: 265 pcs.

- SF reactor storage at unit 6 (Reactor pond - 6):
  - Location: the central reactor hall of units 6, adjacent to the unit;
  - Purpose: storage of SF from unit 6;
  - Storage method: under water in one rack;
  - Storage capacity (number of assemblies): 612;
  - SF assemblies stored: 317 pcs.
- Away from reactor SFSF for storing spent nuclear fuel under water (SFSF):
  - Location: On the site of KNPP, close to units 3 and 4;
  - Purpose: to store SF from all reactors on the site;
  - Storage method: under water, in a pool divided in 4 sections;
  - Capacity (number of baskets): 168 or under specific conditions — 200;
  - SF assemblies stored: 4518 pcs. from WWER-440 and 96 from WWER-1000.

4. SHIPMENT TO RUSSIA

The shipment of spent nuclear fuel from Kozloduy NPP started in 1979 from Unit1 and Unit2 under a contract signed on October 25, 1979 between Atomenergoexport, Moscow and Energoimpex, Sofia. According to the contract there have been carried out 21 shipments of spent fuel from Units 1-4 (total number of WWER-440 spent fuel assemblies 3018 using 102 casks type TK-6). Every year additional annexes to the main contract were signed and shipments of spent fuel were carried out under the respective annual annexes.

In 1998 the old contract for shipment of WWER-440 SF from Bulgaria to Russia was replaced with new one signed on March 03, 1998 between Kozloduy NPP and the Russian enterprise "Mayak" reflecting new economical conditions for every shipment under additional annexes. Spent nuclear fuel from WWER-1000 reactors (Units 5-6) is shipped to Russia under another contract signed on June 06, 2000 between Kozloduy NPP and the Russian enterprise "Mayak". Every shipment is carried out under additional annex to the contract. The table 1 below shows the shipments of WWER-440 and WWER-1000 SFA from Kozloduy NPP in the period of 1979-2009.

TABLE 1. SHIPMENTS OF WWER-440 AND WWER-1000 SFA FROM KOZLODUY NPP IN THE PERIOD OF 1979–2009

Year	WWER-440	WWER-1000, Unit - 5	WWER-1000, Unit - 6
1979-1988	3018		
1998	240	-	-
1999	-	-	-

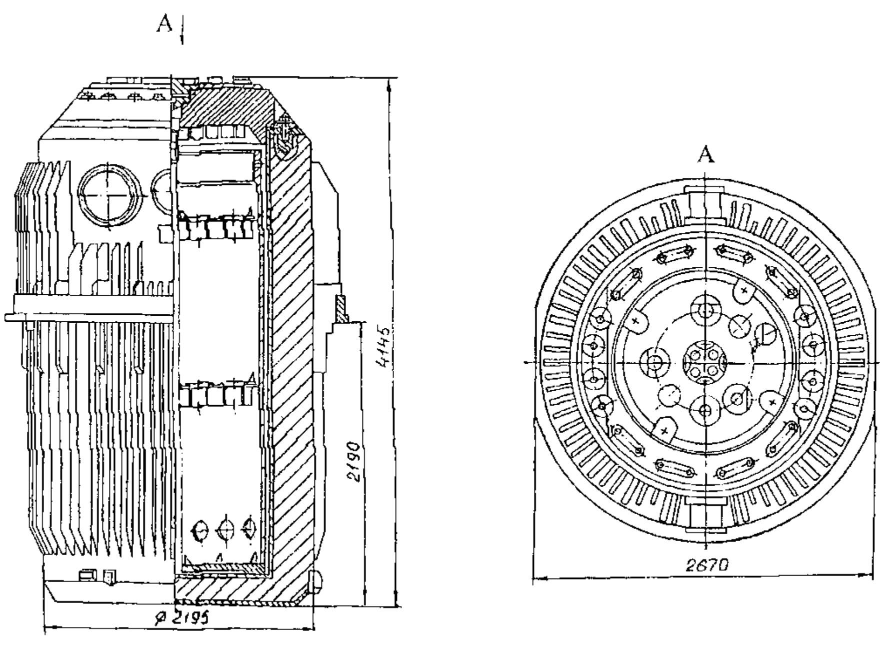
2000	-	-	-
2001	-	96	-
2002	-	-	96
2003	208	-	96
2004	240	96	96
2005		144	48
2006	240	24	71
2007	240	96	
2008	480	96	
2009	240		
Total SFA	4906	552	407

The transportation of SF from Kozloduy NPP to Russia is carried out by a specially designed river type barge called "Nautilus". The barge transports SF for processing in Russia. The barge is suitably equipped to transport 8 casks loaded with SF from WWER-440 (240 assemblies) or WWER-1000 (96 assemblies).

4.1. Transport casks

The following figures show spent fuel transport casks:

- TUK-6 for WWER-440 spent fuel (Fig. 1):
  - — Packaging capacity, FA, pcs. 30;
  - — Mass of a packaging, t, not more than 81.0;
  - — Mass of a loaded package, t, not more than 92.0.



Cask TUK-6 with WWER 440 SNF. Handling operations of TUK-6 within the storage facility.

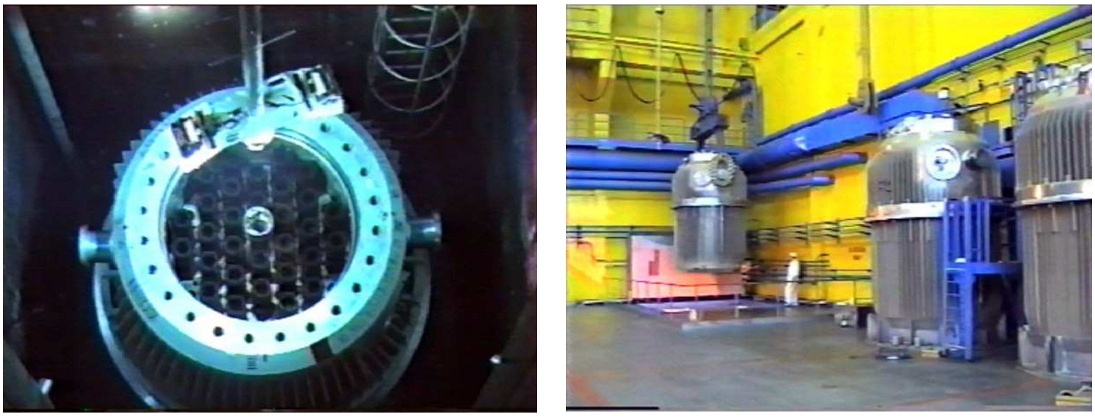
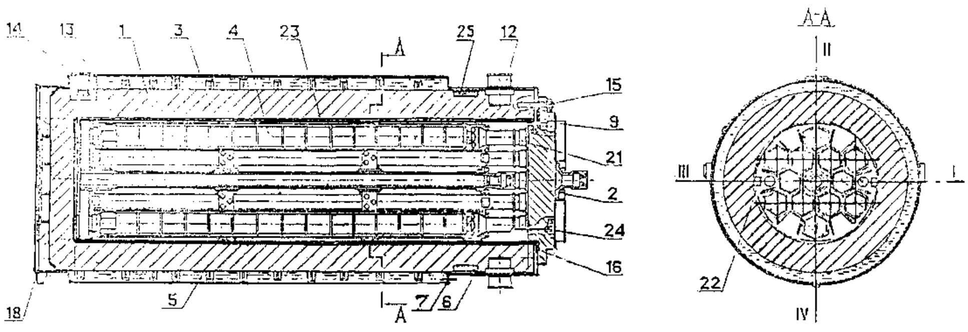
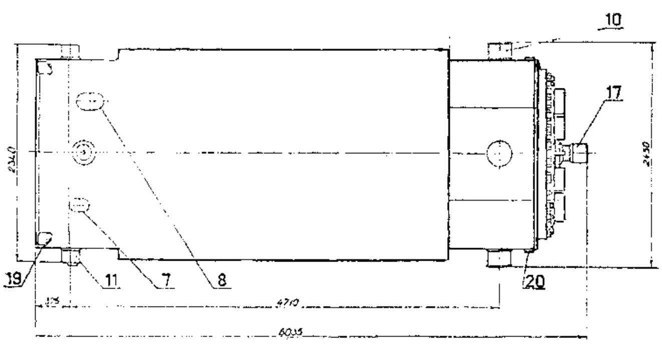


FIG. 1. TUK-6 cross section and general view.

- TUK-13 for WWER-1000 (Fig. 2):
  - — Packaging capacity, FA, pcs. 12
  - — Mass of a packaging, t, not more than 104.0
  - — Mass of a loaded package, t, not more than 113.0





1. Body; 2. Lid; 3. Canister; 4. Fuel assembly of VVER-1000 (without canister); 5. Neutron shielding; 6. Expansion capacity; 7. Device for filling (drain) of the neutron shielding; 8. Device for drain of the inner cavity of container; 9. Device for overflow of the inner cavity of the container; 10. Loading trunnion; 11. Trunnion of fastening and turning over of the container; 12. Safeguard trunnin; 13. Lug weld-in; 14. Plug; 15. Key; 16. Seal; 17. Grip device; 18. Support ring; 19. Fixing lug; 20. Wedge stop; 21. Pressing device; 22. Guiding key; 23. Hexahedral tube of boron steel; 24. Thermal converter of resistance; 25. Ring cutting.

FIG. 2. TUK-13 cross-section view.

4.2. Transport of packages

4.2.1. Road transport

During transportation a special convoy is arranged, including fire engine, ambulance, security and police vehicles. Special trailers are used for transport of spent fuel casks (Fig. 3).

TUK-6 trailer

TUK-13 trailer

FIG. 3. Transportation trailers.

4.2.2. Water transport

For spent nuclear fuel shipment along the Danube River a special trail barge is used with a capacity of 8 casks TUK-6 or TUK-13 (Figs. 4 and 5). The barge is equipped with cooling system; decontamination system; radiation monitoring system and communication equipment.

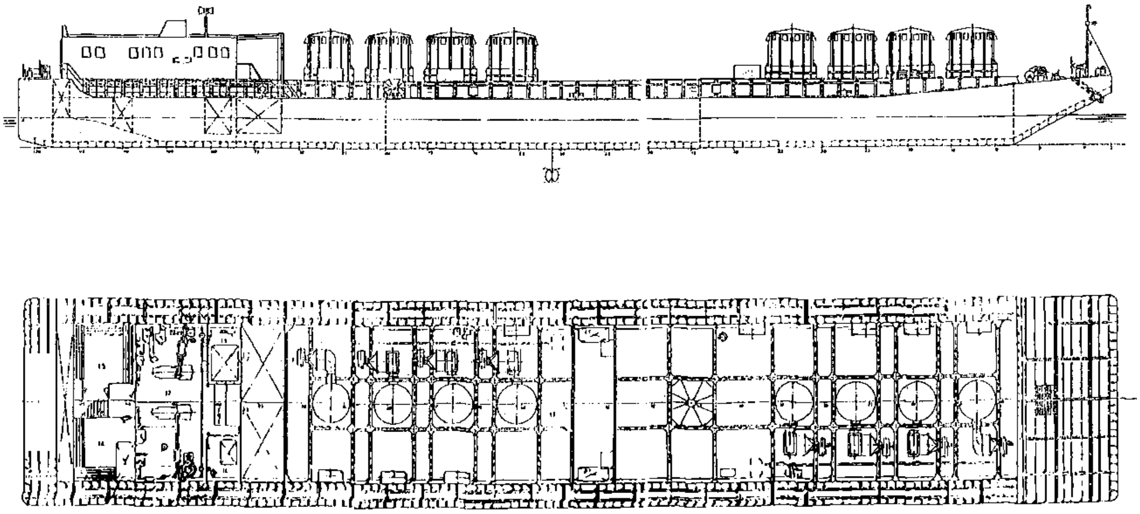


FIG. 4. General view of the barge.

FIG. 5. Barge with a pusher.

4.2.3. Railway transport

The casks with SNF are transported by a special railroad train consisting of up to eight freight wagon-containers with ventilation system, two escort wagons for accompanying persons, means for control of casks' parameters and two protection wagons (Fig. 6).

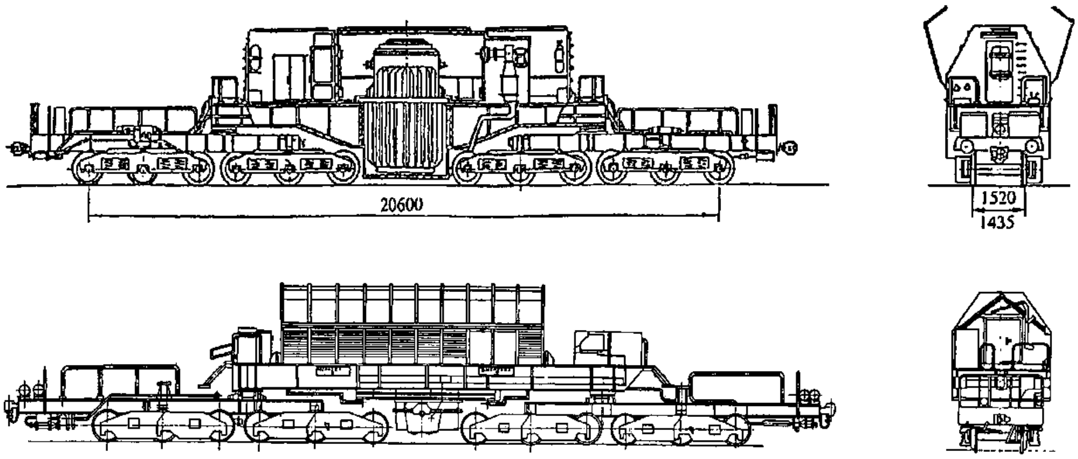


FIG. 6. Railway transport general view.

4.2.4. Emergency preparedness

The emergency plan covers the route from Kozloduy NPP to the Ukraine-Russian border. It covers the road, water and railway transport schemes. The plan defines the activities of the accompanying and emergency centre teams, provides instructions aiming the limitation and elimination of the consequences in case of accident during transportation. This plan is harmonized with Bulgarian, Romanian and international legislation.

5. CONCLUSIONS

The Republic of Bulgaria signed the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management in Vienna, on 22 September 1998.

Since 1979 the transportation scheme of SF has been successfully applied 39 times.

Substantial experience in transportation of spent nuclear fuel has been gained. The fact that no accident during transportation occurred illustrates the capabilities of the transportation system, consisting of transportation scheme, equipment, personnel training and emergency planning.

The policy of the Republic of Bulgaria regarding the management of SF and radioactive wastes (RAW) has been based on the moral principle of avoiding to impose undue burdens on future generations.

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DIFFERENCES OF TECHNICAL REQUIREMENTS BETWEEN TRANSPORTATION AND STORAGE METAL CASKS

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Abstract

The worldwide demand of storage facilities for spent fuels discharged from nuclear power stations is increasing to maintain sustainable operation of the nuclear power stations. The spent fuels are stored at first in the fuel pools (wet storage). When the spent fuels exceed the pool storage capacity, the fuels are transferred to the other storage facility located at reactor or away from reactor, which often adopts a dry storage technology. To use metal casks is one of the options for the dry storage facilities, and some storage facilities have already utilized large metal casks, whose original design concept were developed to transport the spent fuels from nuclear power stations to a reprocessing plant by trains, trucks or by sea-going vessels. It is widely understood that the technology of transportation casks developed up to now is able to apply to the storage casks without any significant design changes. Technical requirements on the design are discussed between the storage cask and the transportation cask to confirm of the understanding based on the assumption that the large metal cask is used for transportation and storage respectively.

1. DEVELOPMENT OF METAL CASKS AND ITS STRUCTURE

- A development of large spent fuel transportation casks such as transporting LWR spent fuels from a nuclear power station to a reprocessing plant by rail or by sea-going vessel were started at the beginning of 1970s. The large transportation casks have a robust cask body which combines the radiation shielding, and a single bolted lid with elastomer seals that form the containment boundary of the casks. The spent fuels are contained in the cavity of the cask body in a fuel basket to configure the fuels at their individual positions to keep the fuel sub-critical. Trunnions and shock absorbers are equipped on the outside of cask for handling and reducing impact force at the accident drop condition. Sometimes fins are attached on the surface of the cask body for dissipating decay heat of spent fuels. A transport frame is used for both transportation and handling of the cask. These are designed to conform to the regulatory requirements of each country, most of which is prepared based on the IAEA regulations for the safe transport of radioactive material [1]. An example of the transportation cask is shown in Figure 1, and its outline structure is shown in Fig. 2.

2. COMPARISON OF DESIGN REQUIREMENTS BETWEEN TRANSPORTATION CASKS AND STORAGE CASKS

2.1. Design requirements for transportation casks

The design requirements for transportation casks had been prepared, and applied to the cask design at the end of 1960s. The design requirements are specified in Section VI and VII of the IAEA regulations. For large transportation casks such as B(M) type package, the four fundamental performances, subcritical, radiation protection, heat removal and containment, shall be maintained under the design conditions listed below:

(1) General conditions of transport;

(2) Normal and abnormal conditions of transport;

(3) Design basis accident condition of transport;

(4) Critical safety in isolation and arrays.

The transport regulations quantitatively specify the design requirements under these conditions as general and normal and accident test conditions.

FIG. 1. Transportation cask.

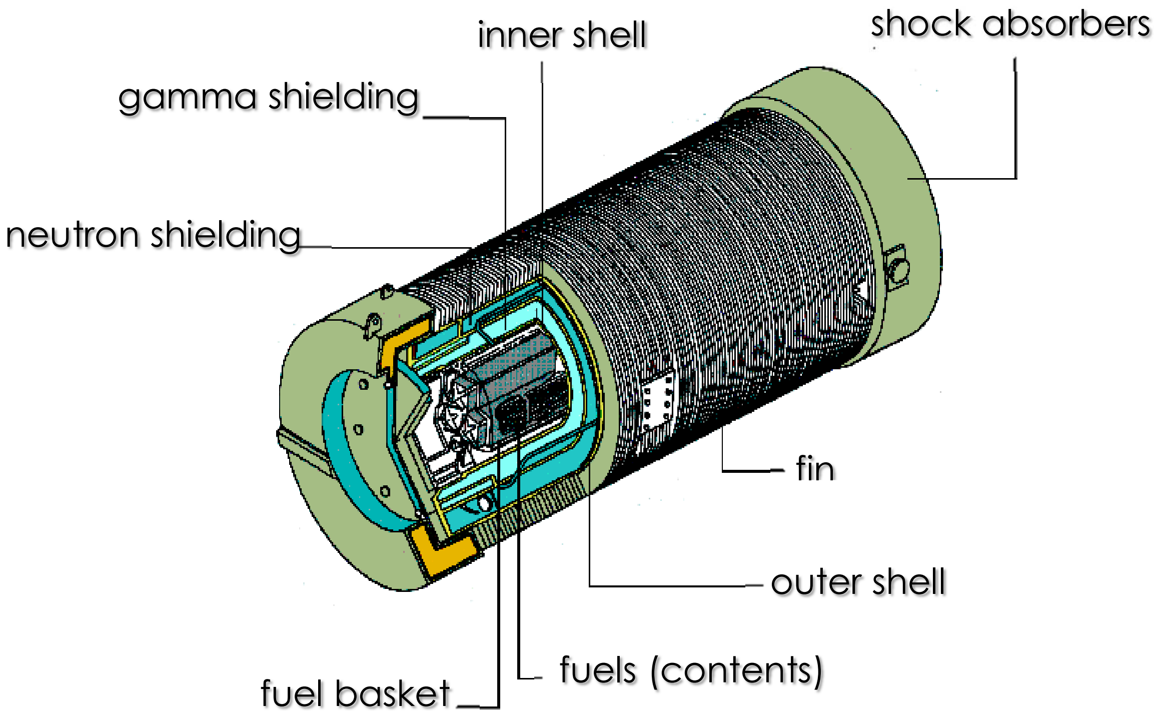

FIG. 2. Structure of transportation cask.

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2.2. Requirements for storage casks

The regulations for spent fuel storage facility were provided first for the wet storage facility (storage pool) as an annex facility to the nuclear reactor. This situation might be same for many countries, where neither requirements for dry storage facility nor those for dry storage cask have been specifically prepared for. In addition, there are not so quantitative design conditions as transportation casks are provided, because the most design conditions for the storage facilities are site-specific. When a dry cask is used for the dry storage facility, quantitative design conditions are necessary to design the storage cask that conforms to the storage regulations. Many storage cask designers tried to apply the design conditions and requirements for transportation casks to the storage cask design, since many transport design conditions are quantitatively specified as mentioned above, and seem easier to apply them for the storage cask design.

The design conditions for the storage casks, which are not always clearly defined, can be specified as below by referring to IAEA Safety Series, Design of Spent Fuel Facility [2] and AESJ F002:2008 [3], the latter is provided for the design of storage casks. Four requested performances, subcritical, radiation protection, heat removal and containment shall be maintained under these design conditions:

(1) General conditions of storage;

(2) Normal and abnormal conditions of storage;

(3) Design basis accident condition of storage;

(4) Requirements for critical safety;

(5) Material degradation over the design lifetime.

Except No. 5, the design conditions are similar to those for the transportation cask.

2.3. Comparison of design conditions between transport and storage casks

Fig. 3 shows the diagram showing the relation between regulatory requirements (regulations), design conditions and necessary performances for transportation casks or storage casks.

As discussed in 2.1 and 2.2 above, the necessary design features and the role of structural design to maintain these performances are same between transportation casks and storage casks except item (5) for storage casks. This result implies that the design conditions for transportation cask are applicable to the storage casks.

These four performances shall be maintained not only at normal or abnormal conditions but also design basis accident conditions such as 9m drop test for transportation and seismic condition for storage. The robust body is necessary to maintain the performances at any design conditions, which is achieved by the structural design. The design conditions were compared between transportation and storage, which are shown in Table 1.

TABLE 1. COMPARISON OF DESIGN CONDITIONS BETWEEN TRANSPORT AND STORAGE CASKS

<table>
<tr><th colspan="2">• conditions</th><th>• for transportation cask</th><th>• for storage cask</th></tr>
<tr><td>• operational occurrences</td><td>• normal</td><td>✓ environment conditions including solar insolation
✓ transportation condition such as vibration at transport and lifting load</td><td>• (environment conditions)
✓ temperature
✓ atmospheric pressure
✓ wind
✓ snow
• (handling)
✓ lifting load</td></tr>
<tr><td></td><td>• abnormal</td><td>• (test conditions)
✓ water spray
✓ free drop
✓ stacking
✓ penetration</td><td>• (handling)
✓ drop
✓ collision
✓ tip over
• (natural disasters)
✓ tornado/typhoon
✓ earthquake
✓ tsunami
• (disasters)
✓ fire/explosion
✓ aircraft crush
• (storage period)
✓ material degradation</td></tr>
</table>

• design basis accident conditions	• (test conditions) ✓ 9 meter drop ✓ 1 m drop ✓ fire ✓ water immersion ✓ enhanced water immersion	✓ no specified conditions. ✓ the design bases accident condition is selected from the abnormal conditions taking into consideration with the site specific conditions. •

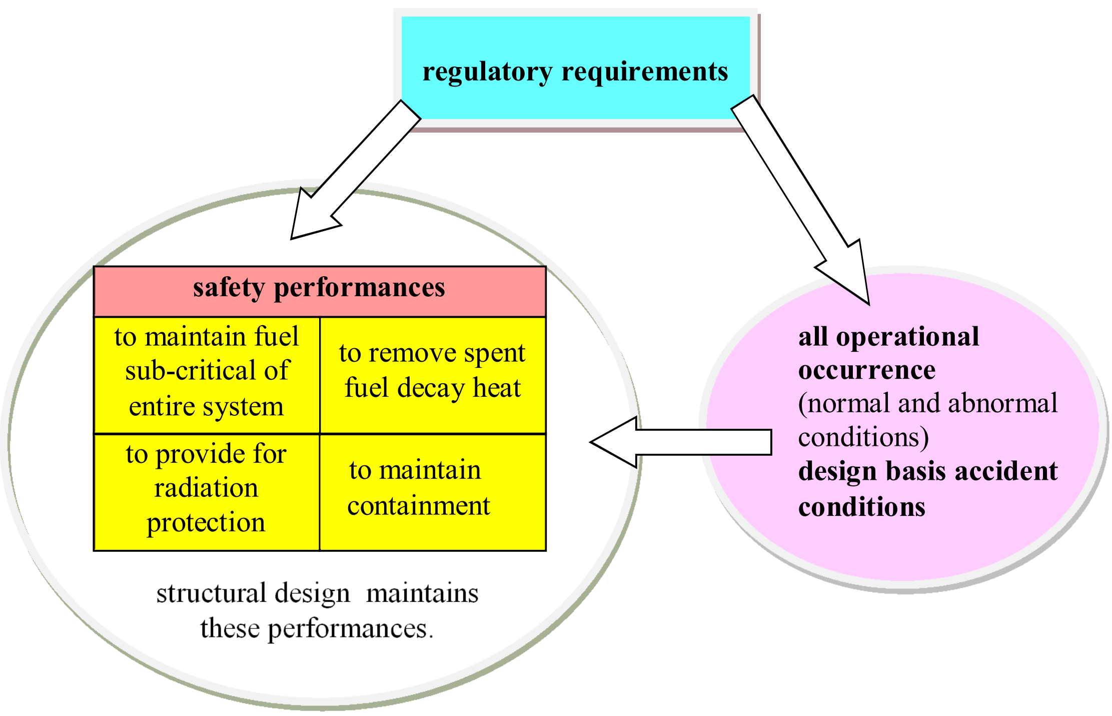

FIG. 3. Design requirements, conditions and performances.

As the phenomena occur during transport can be generalized, the design conditions for transportation cask are very concrete and quantitative. On the other hand, as those for storage casks are site-specific, they are more abstract and qualitative. For example, the accident conditions are clearly specified for transport as test conditions, but those for storage are requested to determine from the abnormal storage conditions taking into consideration with the possibility of radioactive release. It is easier for many designers to design the storage cask in accordance with the transportation regulations and requirements than to consider the site-specific design conditions precisely at the beginning stage of the design. The site-specific design can be add to the design later.

Other reason is that a transportable storage cask is sometimes considered for the spent fuel storage facilities. This is a good reasoning for the designers to design the cask first in accordance with the transport regulations, and to add the site specific designs latter.

3. ADDITIONAL CONDITIONS FOR THE STORAGE CASK DESIGNED ACCORDING TO THE TRANSPORT REGULATIONS

Fig. 4 shows the two factors, which are important to compare the design conditions between transportation and storage casks. Those are "duration time" and "load intensity". For the transportation cask, duration time is rather shorter and its load intensity operating to the casks is rather larger comparing to those of storage casks. On the other hand, the duration time is extremely longer for the storage casks comparing to the transportation casks. These are key factors that should be considered to use transportation cask design for the storage casks. The load relevant to the storage conditions can be said to occur slowly.

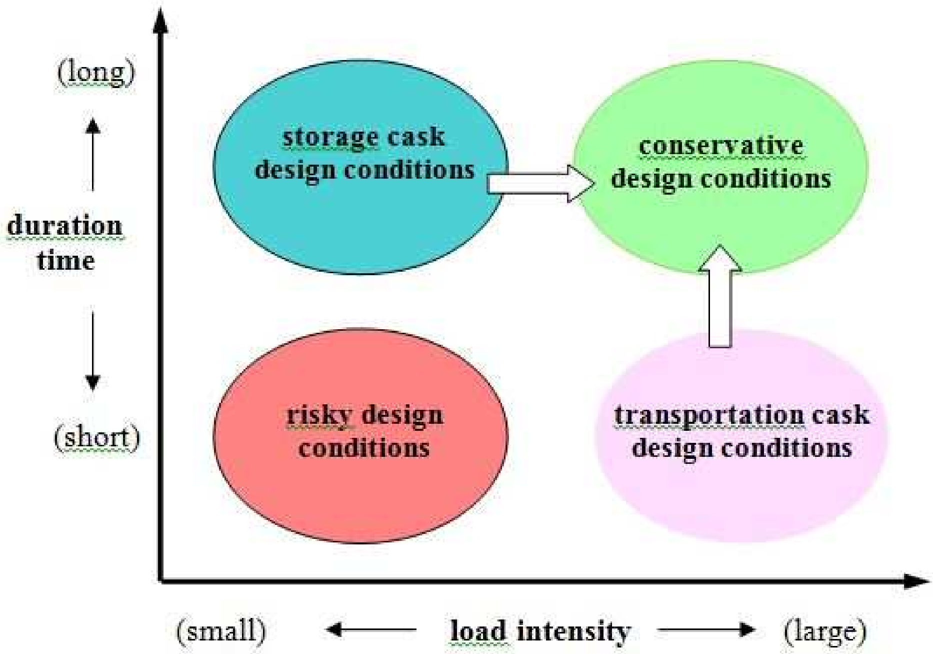

FIG. 4. Load intensity and duration time.

For example, 9m drop test and earthquake are similar phenomena, both of which induce impact load and acceleration on the cask. The duration time of earthquake (tens of seconds) is shorter among the abnormal conditions for storage in Table 1, but it is still extremely longer comparing with the duration time of 9m drop (tens of mille-seconds). On the other hand, the acceleration at 9m drop is a hundred larger than that of earthquake.

In structural design of transportation casks, the duration time of load is usually ignored (static design), so that the design is valuable for the storage casks too. The design is performed in the area of "conservative design conditions", which locates the upper right area of Fig. 4.

The material degradation throughout the anticipated storage lifetime is the item which should be considered as one of the affect to the cask performance, containment. To prevent the radioactive materials release during the storage lifetime, the designer shall take care of corrosion and degradation of materials.

The transportation casks usually use elastomer o-rings as the seal material, which is very good material for sealing, but it will degrade during rather shorter time comparing to the storage lifetime It is difficult to expect its sealing performance throughout the anticipated storage period.

Considering the degradation, the designer of storage casks should use metal o-rings, which are developed to have similar performance and characteristic as elastomer o-rings. The other alternative would be to adopt welded lid system instead of bolted lid.

In the case that the metal o-ring is selected for the sealing materials, the double lid system should be adopted to the cask design in order to monitor the sealing performance throughout the anticipated lifetime. No monitoring is necessary for the welded lid system.

The designer shall also take care of the degradation of fuel cladding. The cask inner cavity gas should be substitute to inert gas like helium gas to prevent corrosion. The maximum cladding temperature should be limited to prevent creep and hydrogen reorientation, which was currently pointed out to be considered for the storage cask design [4].

The storage casks are shown in Fig. 5, and the outline structure of the storage cask is shown in Fig. 6, which incorporates the considerations discussed above to the design.

FIG. 5. Storage casks in storage facility.

4. CONCLUSIONS

Transport regulations and design conditions are widely used not only for the design of transportation casks but for those of storage casks. To justify this validity, the design requirements and conditions of transportation casks are compared with those of storage casks.

Almost all design conditions of both transportation and storage are shown to be compatible, and clarified the two items that should be considered for the storage cask in addition to transport cask. One of the items is to prevent the release of radioactive materials due to material degradation over the anticipated lifetime. The lid seal material should be changed from elastomer to metal, or to adopt the welded lid system. The double lid system is necessary to monitor the sealing performance if the former consideration is incorporated in the storage cask design. The other one is that the cask cavity gas should be substituted to inert gas to prevent the degradation of fuel cladding over the anticipated lifetime.

- All storage design requirements can be satisfied when storage casks are designed in accordance with the requirements specified in the transport regulations, provided that the two considerations mentioned above are added in to the design.

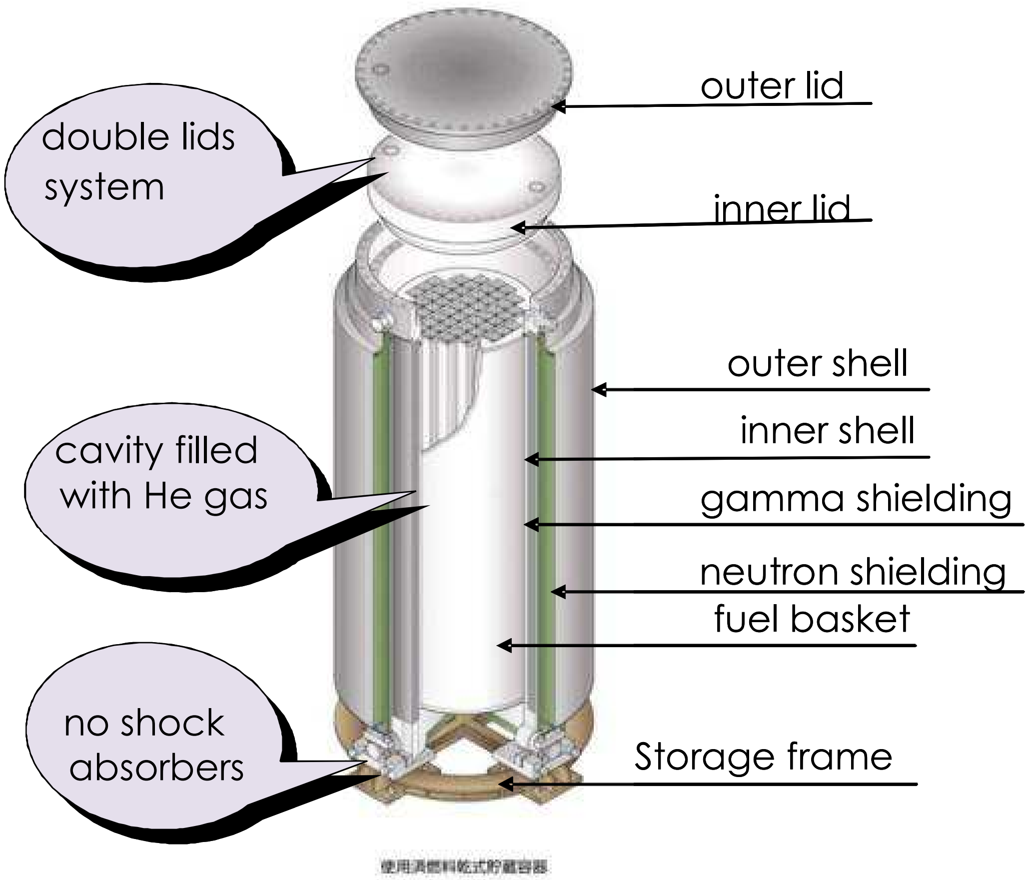

FIG. 6. Structure of storage cask.

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- [1] Regulations for the Safe Transport of Radioactive Material, 2005 Edition Safety Requirements, IAEA Safety Standards Series No. TS-R-1.
- [2] Design of Spent Fuel Storage Facilities, IAEA Safety Series No. 116, 1994.
- [3] Standard for Safety Design and Inspection of Metal Casks for Spent Fuel Interim Storage Facility. AESJ-SC-F002:2008 (Japanese).
- [4] AOMI M., KAMIMURA K., BABA T., MIYASHITA T., High Burnup Fuel Cladding Tube Property Test for Evaluation of Spent Fuel Integrity in Interim Dry Storage, IAEA-CN-144/34.

NUCLEAR ENERGY AND PUBLIC OPINION: CHILE'S CASE

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1. HISTORY AND THEORETICAL FRAMEWORK

Public opinion is as old as history itself. Its origins date back to the ancient Greece where the Agora was consulted about matters of interest and at the same time it was practiced the art of persuasion through dialogue. Philosophers such as Socrates, Plato and Aristotle gave proof of their knowledge and skills of persuasion. These aspects were inherited by the Roman Empire, which sought through the senate the idea of transmitting what we know as "Vox Populi" (The Greek doxa), a term that together with the "Consensus" used by Medieval thinkers, constitute its pre-modern origin. From a conceptual point of view, public opinion comes alongside the creation of the idea of the state and as a result of the historical process called the Enlightenment. Thinkers such as Rousseau, Locke, Montesquieu, Kant and Hegel contextualised it within the legal system of the state. For Rousseau, it should be understood as a expression of the general will; for Kant it was the highest realization of the Enlightenment era and a result of the use of reason and law and for Locke human behaviour was defined by three fundamental laws: the divine, civil and…Public Opinion!. Hegel was much more specific and expressed that public opinion was called to be the instrument by which society expressed its support or rejection to the rulers' decisions. It was in coffee shops and gatherings where opinion was born; for this reason, it was only the elite who was able to expressed about general interest's topics, therefore the importance of sociability during modern times. This is how we arrive to the contemporary definition of the term. Since the French Revolution the concept has been associated with the sovereignty of the states and people, i.e. the rule of the majority[1]. Today due to the massive participation of people in issues of importance, the opinion delivered by the media, and the globalization of communications, it is difficult to arrive at a definition of the term we have been dealing with. Nevertheless, it is possible to deliver an idea about this concept, though preliminarily, but not less useful.

For Habermas, this becomes the core of social cohesion, construction and legal political justification; therefore a central element of any government action. [2] For this reason the studies on public opinion have become an essential element in the designing and implementation process of political decisions. For some authors, on the other hand, such as Horacio Botero Montoya, this has been nothing but the result of a series of spokespersons of a minority who, on behalf of the majority, say that they represent collective and public interests. [3]However, there is a consensus by which public opinion is what a group of people say regarding public affairs' issues, i.e. those issues that are relevant to the citizens according to

[1] Hans, Speier: ¨ Desarrollo Histórico de la Opinión Pública ¨ Roble Publishers. 1969. Page 56.

[2] Habermas, Jurgen: ¨ Historia y Critica de la Opinión Publica ¨ G. Gili Publishers, Barcelona 1981. Page 67.

[3]Luis Horacio Botero Montoya: ¨ La Opinión Pública en la Democracia ¨. Uni magazine Vol 1, Number 3. July 2006.

their perceptions, information and knowledge. For this reason is that public opinion is formed by citizens who converged on their similes and who formed an interested audience –precisely- in public affairs. They generate a series of perceptions, not necessarily correct or based on solid arguments, which are expressed in a public space. As it can be seen, we should keep in mind all the concepts that exist about it in order to achieve an objective approach about the subject under study.

Since it is not possible to arrive at a unitary definition of this socio-political phenomenon called public opinion, we can say at least that it has two meanings: firstly, it can be seen as the set of existing flows in a specific time and space, and, secondly, as the opinion of a majority, i.e. the prevailing opinion among several other already existing. From this understanding we can identify the elements that characterize the public opinion:

(1) A plurality of individuals who express an opinion, which is not and cannot be considered as a unique opinion, but an opinion from the most part of society;

(2) The affinity of this plurality in terms of judgments and attitudes, which allow some convergences from situational factors in relation to issues of public interest;

(3) The awareness of the plurality which forms a group, even though it is informal;

(4) A point of contrast and / or resistance to other groups of opinion.

2. TOPIC IN PUBLIC OPINION

There is a consensus that public opinion is one of the pillars of social cohesion and at the same time is a central element of any government action. The process of forming public opinion is a highly complex one that involves the presence of at least three actors: politicians, media and opinion leaders. In addition to this, it exists a fourth element which obtains a vital importance and it is an essential component of all this; it is called the "topic" on which an opinion is created[4]. From the topic it is derived the concept of theming, which is the labelling of the events that happen in the world. Labelling is a process that makes easy to understand, and study Public Opinion. Upon a specific topic it can also exist a debate, which can lead to the creation of an opinion.

The aim of this study is to know about public opinion in Chile concerning nuclear energy. For this reason we should try to identify the most important aspects related to the debate that has been carried out (including main actors and opinions) and knowing whether this will create or not a serious and well informed national opinion. [5]

3. NUCLEAR ENERGY AND PUBLIC OPINION IN CHILE

Articles and interviews in relation to nuclear energy that have appeared in the national press during the last five years as well as public opinion's surveys have been used as the source of this research. Press has given abundant information on this matter basically through statements and interviews but, on the other hand, surveys have not given enough information. This is due to the fact that during the last five years only one survey concerning nuclear

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[4]Luhmann, Niklas: ¨ La realidad de los Medios de Masas ¨. Barcelona. Anthropos Publishers. 2000.

[5] Author's Note: In order to make this document general works concerning Public Opinion have been consulted as well as media and internet sources during the last five years.

energy has been carried out[6]. That demonstrates the lack of relevance and interest concerning this "topic" given by those who organize and plans opinion's surveys. Keeping this in mind our first conclusion is that so far there is no a real necessity to know more and until now this topic has not become a subject of general interest so as to justify investment and efforts in surveys in this matter[7]. However, it is important to highlight, that the nuclear issue has been included in surveys in which public opinion has been consulted about other subjects of interest such as energy crisis and environmental projects[8]. Just as an example, due to this kind of opinion's survey, it has been possible to know the level of acceptance of people about the construction of nuclear energy plants (it has oscillated between 7 and 12%).

Regarding the "debate" that has been carried out concerning the specific aspect of nuclear energy we found a completely different scenario. In recent years a series of opinions have appeared focused mainly –but not exclusively- on the uses of nuclear energy and, by extension, the construction of nuclear power plants. In this debate many people has participate. Mainly representatives from the academic, business, and governmental world, as well as NGO's, parliament and even candidates for the Presidency of the Republic[9], all of them have been able to openly express their opinions and arguments several times. Opinions are summarized in the following points:

Presidency of the Republic: Two positions –though quite opposite- are combined complementarily: first, the political commitment not to acquire nuclear energy, and second, the interest of studying this subject through the setting up of the Presidential Commission for the feasibility of using this energy source (Zanelli Commission).

Parliament: Some opinions have been said in favour of nuclear energy by two senators (Orpiz and Nunez). In the House of the Representatives the opinion is rather negative.

Ministry of Energy: Minister Tokman thinks the study on nuclear energy cannot be excluded as a source of electric power production.

Chilean Nuclear Energy Commission: Due to the nature of its work it has openly supported the use of nuclear energy.

Academics: Three important universities (Universidad de Santiago, Universidad Catolica and Universidad de Chile) have declared that it is necessary to discuss the matter.

Business: The business world view, except for Endesa (the company that generates energy to the central interconnected system), generally supports the use of nuclear energy. Companies that support nuclear energy include: Gas Atacama, Transelec, Chamber of Industry and Trade and National Mining Society.

NGO's: "Terram" and "Chile Sustentable" foundations estimate that the country should not use the nuclear energy as a source of electricity arguing, among other reasons, environmental and economic security.

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[6] Appendix I.

[7] Results obtained from the only available survey on the internet related to nuclear energy and public opinion in Chile in 2006 are included in apendix I.

[8] Adimark, June 17, 2008. Survey relating to the construction of a hydroelectric plant in Aysen, southern Chile.

[9] It has also been observed the presence of international actors such as the Nuclear Electronics Corporation.

Presidential Candidates: Mr. Eduardo Frei considers necessary to study the subject and not a priori rule out its use. Mr. Sebastian Pinera has appeared open to the possibility of diversifying the Central Interconnected System, but with some caution about the nuclear energy[10].

The debate has been intensified from time to time coincidently with energy crisis, and as it has been mentioned before, this debate has involved academic people, businessmen, and public figures (most of them with education and preparation on this subject) in a discussion of high level and knowledge. These two factors have given certain properties to the debate, such as those set out as follows: first, it has been circumstantial, which means that it has been considered only when a particular event happens. This means that at the same time nuclear energy has been considered important only after a particular event related occurs. Therefore it has not been permanent. Second, we can say that debate is closely linked to the electric power crisis that hit the country in recent years (Argentine gas crisis, high oil prices). Finally, this debate has not reached popular areas of the country, because only people with proven skills and knowledge have participated. In other words, it has been an elite debate preventing to make generalizations. Debate is also strongly influenced by the images of accidents and nuclear explosions, which can't allow an analysis devoid of passion. A clear presence of lobby pursued by companies that promote nuclear energy can be seen. Some of those companies are directly involved in the preparation of the feasibility study mentioned above; this fact compromises the study's objectivity[11].

"Pro and against" arguments have been told in the debate on the use of nuclear energy. The "pro" arguments are linked to the country's needs in the energy field. In this context, the diversification of the Central Interconnected System - SIC - has occupied a central place. Also for those who hold this position, nuclear energy will serve as a permanent source of supply of the SIC. In this line are also those who justify its use given the increasing electricity demand by the large mining, the main economic activity. Other arguments in its favour, are related to the independence granted by the nuclear cycle (compared to the price and gas and oil reserves), the security of the facilities and its contribution to the environment by promoting the reduction of CO 2. Another group of arguments invoke the so called "nuclear industry reborn", the promotion of investments by the use of nuclear technology and the installation of energy production centres. Finally, they strongly supported its price competitiveness in the international market. On the other hand, those who argue against its use are focused mainly on aspects of lack of security involved in this kind of energy in general and in particular the specific case of Chile focused on highlighting the high seismic activity presented in the country. Another set of arguments also associated with insecurity, are related to the fear that a possible Chernobyl-type accident may happen in the country; the disastrous effects that would produce radioactive waste on human health and its consequences on the environment. Moreover this posture gives importance to the high dependence that would be generated by the use of uranium 325, which would leave nuclear energy in the same conditions as other energy sources such as gas, charcoal and oil. High production costs would make it a very expensive technology and therefore subject to government subsidies. It has also been remembered the slowdown process in the development of nuclear energy in the world given

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[10] This list is not exhaustive and does not compromise the view point of the people mentioned.

[11] The Chapter on Risk and Safety was awarded by the Nuclear Corporation, a subsidiary of the Russian company Intermash; the Regulatory Framework was awarded by the Company STUK, which is the regulatory agency of the nuclear industry in Finland; for the analysis of the fuel life cycle was hired the British firm AMEC; the public and private role was awarded by Universidad Adolfo Ibáñez with SENES, a Canadian consulting firm specializing in nuclear energy; finally, the chapter on Public Opinion was awarded by a national company called Tironi y Asociados.

its popular rejection in Europe as a result, among other things, the still unresolved problem of radioactive waste.

Nevertheless, both positions are at the antipodes, we have been able to appreciate as well some common aspects that, on the very debate, exist among the actors who have participated in it. These kind of, let say, “consensus” are related mainly to the following ideas: the necessity to reached a well-informed debate, for this same reason to elaborate serious and objective studies on the subject; achieve an on-going debate in order to avoid that this debate takes place only when there is crisis; there is a consensus that sustainable development can go hand in hand with economic growth; to promote the debate to be conducted among various social actors and not be restricted to a particular group; make the decision reached on the subject of the debate as a “Country’s Decision”. These coincidences allow us to see with optimism the future of this national debate and, therefore, the formation of a serious public opinion.

To sum up, we can say that there is not public opinion related to nuclear energy. Only there is a debate comes from an elite group, which is nowadays a characteristic of the national debate. The references concerning opinion surveys have been too elementary and have failed at the moment of capturing if there is a rejection or support popular opinion. The opinion about nuclear energy has been collected in surveys related to other aspects (environment and power generation). For this we can ensure that there is a not public opinion on the subject. In Chile it has been developed a high-level academic debate but which presents little participation on behalf of regular citizens. We believe that, although what we have just mentioned, there is fertile ground for the formation of a national public opinion on nuclear energy. We believe that it is possible to achieve a solid public opinion about nuclear energy, if the current debate carries out certain conditions and requirements occurs, those are presented below:

- More information available about all the aspects of the nuclear cycle and the development of an educational process, which includes not only elementary school and high school, but also universities, colleges, and institutes;
- Create an impartial instance that can guide and coordinate the national debate; therefore it is necessary the establishment of a neutral testing centre, which could be used to help the objectivity of the debate that is carried out;
- Give a sense of belonging to the members of our society on an issue that should be seen as something of national interest;
- Reach a “National Agreement” in which are combined the technical and value aspects in line with international standards;
- Include in the debate current and potential international commitments related to the topic under discussion (IAEA-OECD, etc...);
- Perform a parallel task to the discussion of collective socialization and education for the population.

The “National Agreement” is achievable regardless of whether people are pro or against the use of this energy source. Efforts are need in order to publicize the arguments of a debate that should produce a serious and informed national public opinion and at the same time give the necessary legitimacy to the decision taken.

THE STATE OF THE WWER NUCLEAR SPENT FUEL MANAGEMENT IN UKRAINE AND TRENDS ON THE OPTIMAL CHOICE OF SPENT FUEL MANAGEMENT STRATEGY

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• **Abstract**

• The paper gives an overview of the operational state of the away from reactor interim spent fuel storage facility at the Zaporizhzhya NPP site, commissioned according to the dry storage cask technology that is developed by the US company Sierra Nuclear Corporation. The trends of the SFM at the WWER nuclear units were noted, which are caused by the advanced nuclear fuel implementation in order to improve the nuclear fuel utilization.

1. INTRODUCTION

Due to the shortage of explored and commissioned traditional energy resources the Ukraine's nuclear power energy development plays the key role in electrical power supplying for Ukraine.

In view of the steadily continuous rising of the price on natural gas and imported oil products, since 1991 Ukraine put into operation three WWER-1000 units (Zaporizhzhya NPP-6 in 1995, Khmelnytsky NPP-2 and Rivne NPP-4 in 2005) in addition to 11 units with WWER-1000 reactor and 2 units WWER-440 reactor to redress the current energy demands.

Today, 15 energy units are in operation at Nuclear Power Plants with total capacity make up 13835 MW, and it can run up to 15835 MW after 2016 by putting into operation two new nuclear energy units at the Khmelnytsky NPP site, as well as 10 years lifetime extension of old WWER nuclear units, which run up 30 years age limit as provided by initial design documentation.

However, the growing of the nuclear energy utilization requires a reasonable SFM that is in strong relation with the choice of type of nuclear fuel cycle in Ukraine.

At present, the final stage policy of WWER spent nuclear fuel management still hasn't been defined in Ukraine, and so all planning activities in the SFM are based on application of the well-known approach, so-called "deferred decision" [1]. The ecological and economic studying for reprocessing or direct final disposal of WWER spent fuel (into a geological repository) have place in Ukraine.

2. ACTIVITY ON THE WWER NUCLEAR SPENT FUEL STORAGE

In view of the reduction of the capacities of at-reactor (AR) storage pools and the evident necessity of a balanced development of the national nuclear energy industry, the Ukraine built and operate the away from reactor (AFR) interim spent fuel storage facility (ISFSF) at the

Zaporizhzhya NPP for storage period up to 50 years and plans to build the Centralized AFR ISFSF for Rivne, Khmelnytsky and South-Ukraine NPPs with a storage period up to 100 years. In 2005 NNEGC "Energoatom" signed an agreement with the engineering company "Holtec International" to perform the design of Centralized Interim Spent Fuel Storage Facility (CISFSF) for long-term dry storing of the spent fuel assemblies (FAs) discharged from WWER-1000 and WWER-440 reactors at the South Ukraine, Rivne and Khmelnytskyy NPPs. CISFSF will be located into the so-called Chornobyl Exclusion Zone and will be operated not less than 100 years. Its designed capacity has to give a storage capacity of 12010 spent FAs from WWER 1000 and 4519 spent FAs from WWER 440.-

To date of the Centralized AFR ISFSF commissioning (after 2012), spent FAs from Rivne, Khmelnytsky and South-Ukraine NPPs will ship to reprocessing plants in Russia on terms to get back the extracted recycled materials and the high level radioactive wastes, which have radioactive equivalent of wastes from reprocessing, for final disposal.

2.1. Spent fuel storage facility at the Zaporizhzhya NPP

In 1993, after annual blocking of spent fuel removal due to erroneous interpretation of Russian Radwastes Low, the Zaporizhzhya NPP management undertook active measures with respect to the construction of on-site interim spent fuel storage facility (ISFSF).

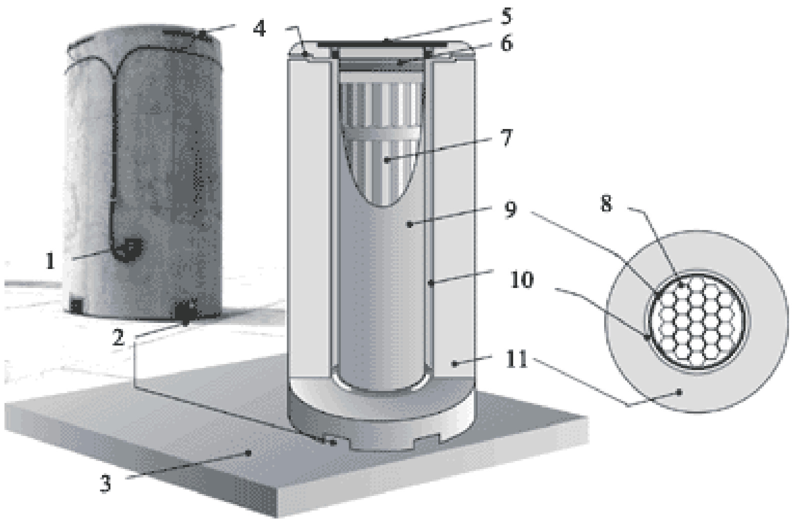

FIG. 1. Design of Ventilated Storage Cask.

1. Temperature-sensitive element
2. Air inlet holes and guide channels for transportation
3. Concrete pad of spent fuel storage facility
4. Air outlet hole
5. Lid of ventilated concrete cask
6. Structural and shield lids of storage basket
7. Unit of hexagonal storage sleeves for 24 spent fas
8. Drain tube
9. Shell of multi-seater basket (msb)
11. Ventilated concrete cask (vcc)

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In 1994, after the assessment of proposals from different companies, the decision was made to build ISFSF on the basis of the VSC-24 storage system, which had been developed by the US company Sierra Nuclear Corporation (SNC) and licensed by US NRC. This year the Zaporizhzhya NPP signed a contract agreement with the engineering company Duke Engineering & Services to perform the design of the storage containers for the WWER-1000 spent fuel assemblies (FAs), the technology transfer and the service support of the ISFSF construction. The designing of the remaining components was carried out by Kharkov Design Institute “Energoproject”, which performed functions of the general designer for the Zaporizhzhya NPP.

Ventilated Storage Cask (VSC) Designs and Major VSC Loading Operations with main components for the Zaporizhzhya ISFSF are shown in Figs 1 and 2.

The pilot commercial operation of the dry storage facility started on August 24, 2001 after the first loaded storage cask was installed on the Zaporizhzhya ISFSF site.The Zaporizhzhya dry storage facility commercial operation license was obtained by NNEGC “ENERGOATOM” on August 10, 2004.

Dynamics of annual spent fuel loading at the Zaporizhzhya ISFS is given in Fig. 3.

As of April, 2010, AFR ISFSF at the Zaporizhzhya NPP contained 84 loaded storage casks VSC with 2010 spent FAs, 1188 control rods, 801 burnable adsorber rods assemblies (BARAs) and 21 nutron adsorber insertions. The first phase commissioning of this ISFSF will be accomplished after the hundredth loaded storage cask will be installed at the storage site. The secont phase commissioning of this storage, designed up to 280 storage casks, has come to the final stage.

Outline of facility location is shown in Fig.4.

2.2. Facility requirements for loading spent fuel assemblies

The spent fuel assemblies scheduled to loading into a multi-seater storage basket (MSB) of the Zaporizhzhya ISFSF shall meet the following requirements (restrictions) [2]:

- Spent fuel assembly cooling time in reactor pool before loading into MSB - no less than 5 years;
- Maximum heat-generating of any spent fuel assembly before loading into MSB - at the most 0.99 kw;
- Maximum initial enrichment for every fuel assembly — at the most 4.4 wt% of ^{235}U;
- Fuel assembly must be adequately intact without geometric discrepancies over limits appointed by project;
- Violation of acceptability criterion on leakproofness of fuel rods was not detected (cladding is considered as enclosure);
- Should not be damaged fuel cladding with the direct contact between fuel pellet and cooling water.

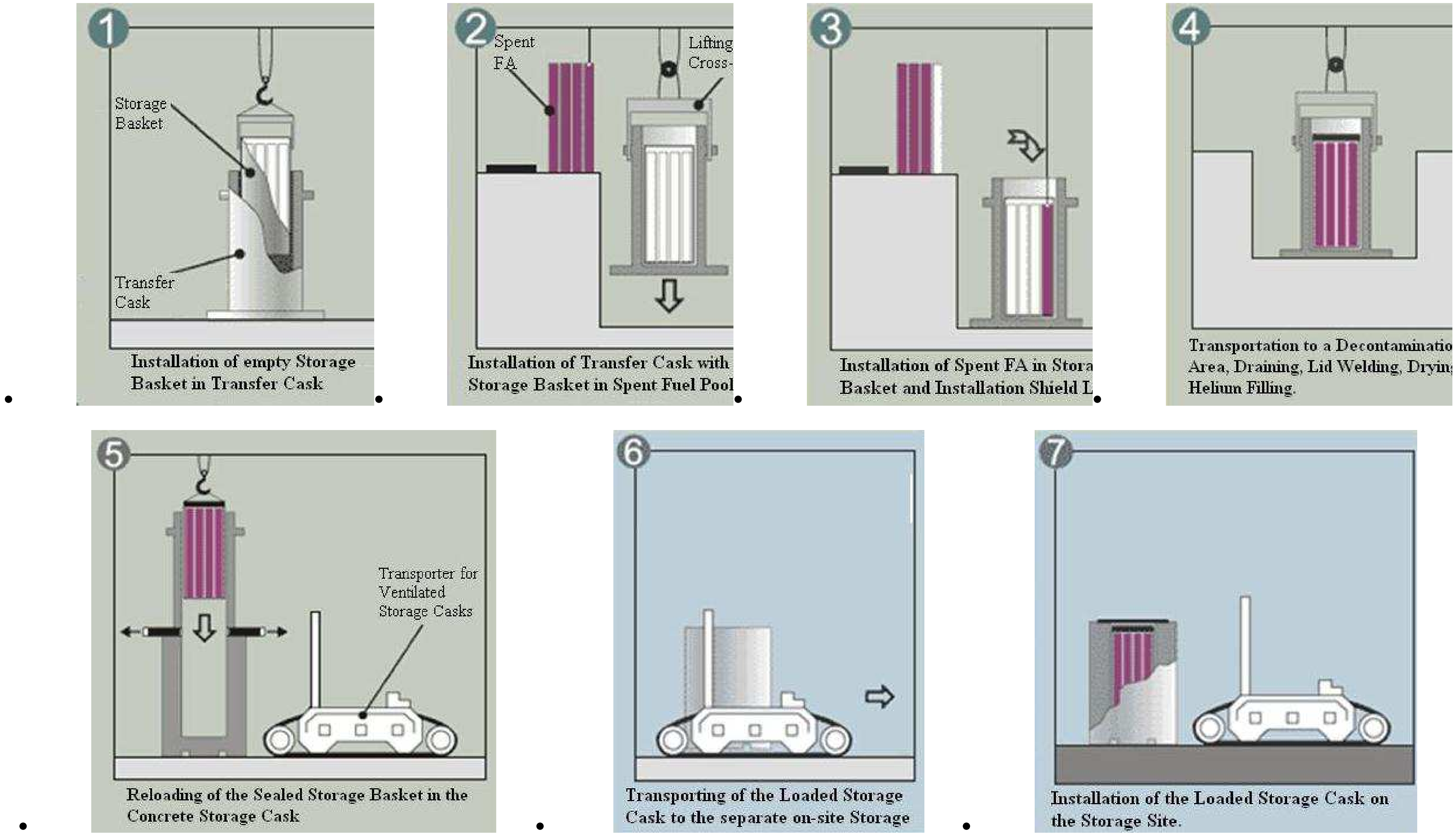
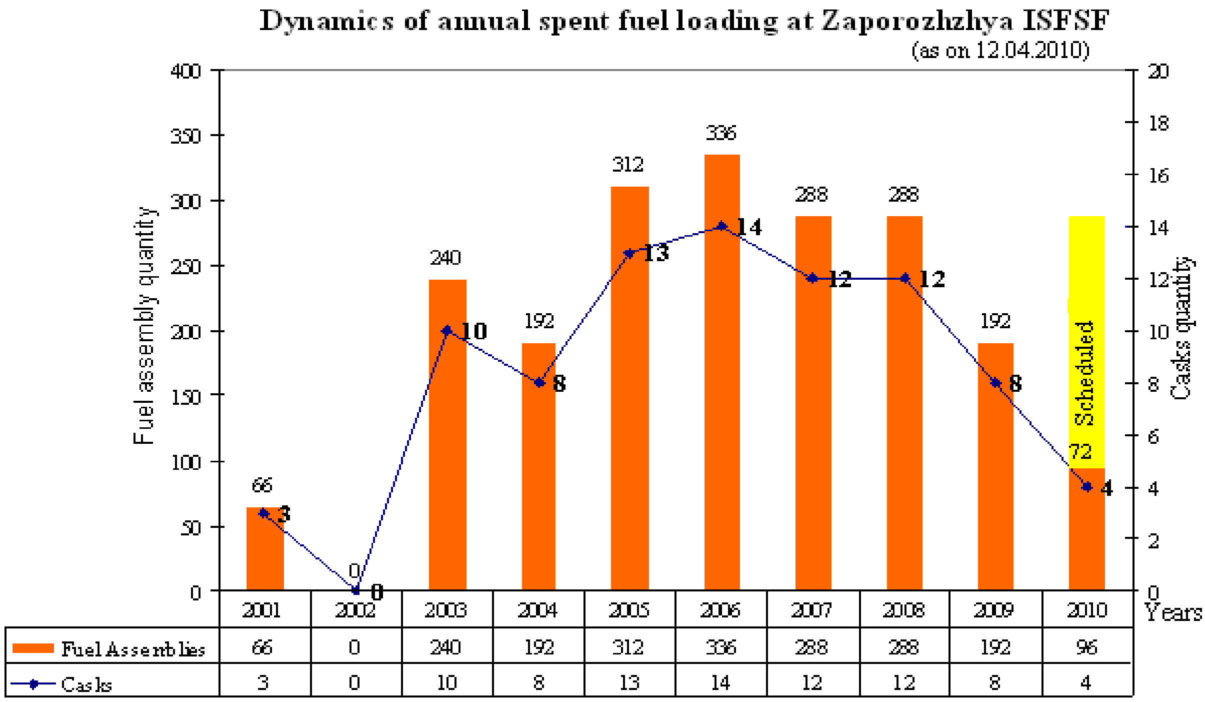


*FIG. 2. Major VSC Loading Operations***.**



	2001	2002	2003	2004	2005	2006	2007	2008	2009	2010
Fuel Assemblies	66	0	240	192	312	336	288	288	192	96
Casks	3	0	10	8	13	14	12	12	8	4

FIG. 3. Dynamics of annual spent fuel loading at the Zaporizhzhya ISFSF.

FIG.4. The Zaporizhzhya NPP dry storage system can be divided into three zones: Casks Loading (nuclear units), Casks Transportation (roads) and Storage Site for Installation of loaded Storage Casks.

1. Reactor facility
2. Turbine installation
3. Diesel generator
4. Unite pumping plant
5. Special buildings 1 & 2
6. Solid radioactive waste storage
7. Conjunct-auxiliary building
8. Laboratory building 1 & 2
9. Administrative building, check-point 1
10. Check-point 2
11. Isfsf site
12. Sprinkling pools
13. Staff restaurant
14. Full-scale simulator
15. Training centre

2.3. Nuclear safety

In compliance with the national nuclear safety regulations [3], the effective multiplication factor should be less than 0.95 for all normal conditions and design accidents. A reasonable choice and substantiation of the cask loading pattern has to be accomplished for every loading MSB. This subcriticality requirement is ensured by technical measures are shown in Table 1.

TABLE 1. USE OF TECHNICAL METHODS TO MEET SUBCRITICALITY REQUIREMENT FOR LOADING VSCS

	Technical measures to meet subcriticality requirement		
MSB numbers	Reduced capacity	Using burnup credit	Heterogeneous neutron absorber loading
VSC1 – VSC3	+	–	+
VSC4 – VSC6, VSC14	–	–	+
VSC7 – VSC13, VSC15 – VSC84	–	+	+

2.4. Modification of the initial design of storage project

To meet the nuclear safety regulations regarding the effective multiplication factor limitation, which should be less than 0.95, there were worked up and applied in operation:

— Checkout methodology of residual spent control rods’ efficiency;

— Methodology of defining and verification of spent fas burnup value;

— Alternative nutron adsorber insertions (alternative nutron adsorber insertions were loaded into VSC78, VSC83 and VSC84).

Yet the operating organization (NNEGC "Energoatom") hasn't the right to load VSCs without the regulating authority permission, which be obtained on the basis of examination results of the loading pattern.

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With respect to the VSCs' fabrication method:

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— The domestic research organizations elaborated and implemented the eigensolutions concerning the welding technology for the MSB sealing;

— The Carbo Zinc MSB's coating was replaced by the Silicone Varnish "KO-828M" that is not reacted chemically with the acidic borated water from the spent fuel storage pool to produce hydrogen;

— The American steel SA516 replaced by domestic steel "10ХСНД";

— MSB assembling technology and vacuum-drying system was upgraded;

— To mitigate the radiation impact on the working area the protective wall in the area of storage perimeter was constructed.

2.5. Radiological survey at the ISFSF site

TABLE 2. SYSTEMATIZED RESULT OF RADIOLOGICAL CONTAMINATIONS INSPECTIONS FOR LOADED VSCS [4]

Data names	Unit of measurement	Measurement result	Operational reference level
Air activity concentration in air outlet holes of loaded VSCs	Ci/l (Bq/l)	$< 1\cdot10^{-9}$ (MDA[12])	$1\cdot10^{-9}$ (37)
Aerosols concentration in air outlet holes of loaded VSCs (indication)	Ci/l (Bq/l)	$< 5\cdot10^{-14}$ (MDA)	$5\cdot10^{-14}$ ($18.7\cdot10^{-5}$)
Non-fixed contamination on lattice of air outlet holes of loaded VSCs[13]	dpm/cm²	There are no α and β contamination	At the most 1 for α. At the most 100 for β.

[12] minimum detectable activity (MDA)

[13] unit of measurement is *dpm/cm²*, here *dpm* is *disintegrations per minute*

In the course of the casks loading and handling, as well as ISFSF maintenance the skilled workers' external exposures are considerably less than the operational reference level 15 mSv/a.

The maximal annual dose recorded for workers, who participate in handling of reloading spent FAs from spent fuel storage pool to VSC and ISFSF, corresponded 0,847 mSv/a. According to the permanent monitoring data of the informative-telemetry survey system "Koltso" (Ring) Dose Rates of gamma-rays correspond to the natural rate inside of the Zaporizhzhya NPP perimeter and outside of ISFSF site.

The permanent survey for radio nuclides contents demonstrates their natural level in free air, atmospheric precipitation, groundwater and manufacturing water, i.e. null influence on environment.

In accordance with spectrometric measuring ("Screener 3M") the internal exposures of the workers are under the operational reference level (3700 Bq/a).

TABLE 3. DOSE RATES (γ+N) FROM CONTROL POINTS OF LOADED VSCS AT THE ZAP. ISFSF IN MSV/H (AS ON 30 DECEMBER 2009)

Data names	Side face on the height of point 1.5 m	Centres of inlet and outlet air holes of loaded VSCs	Centres of upper lids of VSCs
Peak value	55.6	155.6	82.5
Average value	34.5	87.2	43.6
Minimum value	13.4	18.8	4.7
Designed criterion	100	1000	100

3. NUCLEAR FUEL UTILIZATION TREND AND ITS IMPACT ON IMMEDIATE TASKS OF SPENT FUEL MANAGEMENT

- The past two decades of the nuclear power plants' operation are characterized by the sustainable improvement of economic indicators of the nuclear fuel utilization. It is provides with implementing of safe and cost-beneficial fuel cycles based on improved nuclear FAs modifications.

- Due to the fuel reliability enhancement of WWER-1000 FAs the spent fuel burnup increases permanently (See Fig.5) with reducing of fuel consumption, i.e. spent FAs' generation. So the required cooling time of such spent FAs extends exceedingly to meet the decay heat limit of spent FA according to restrictions for safety of transportation and dry storage.

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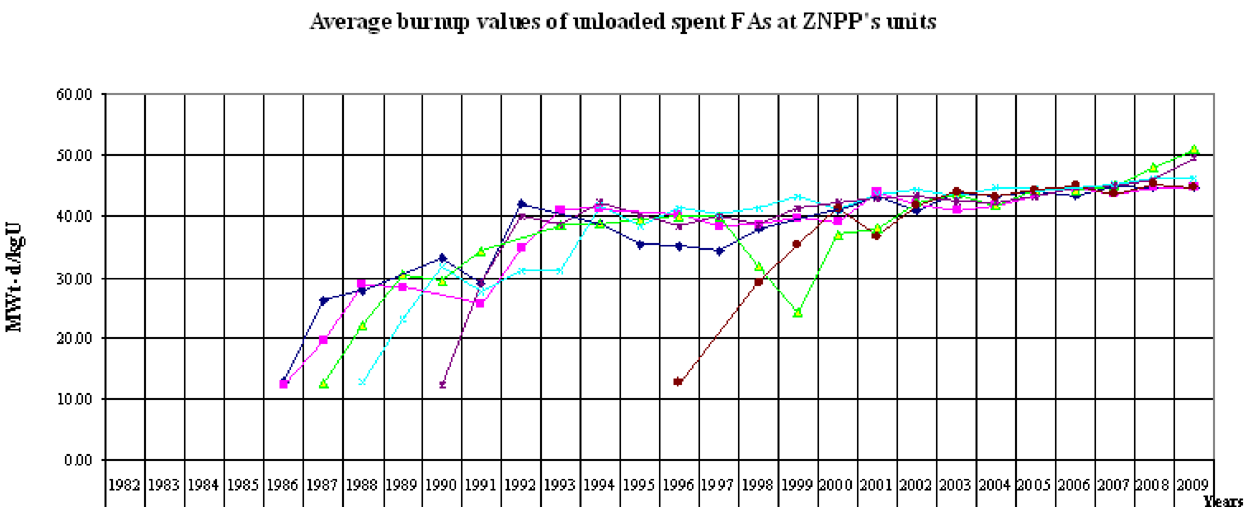

	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995
ZNPP-1					12.78	26.33	27.94		33.11	28.88	41.87		38.82	35.35
ZNPP-2					12.25	19.62	28.76	28.38		25.46	34.67	41.08	41.27	
ZNPP-3						12.47	22.09	30.33	29.53	34.16		38.45	38.67	39.37
ZNPP-4							12.70	23.10	31.80	27.67	31.02	30.93	41.64	38.36
ZNPP-5									12.28	29.21	39.90	38.50	42.18	40.22
ZNPP-6														

	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006	2007	2008	2009
ZNPP-1	35.23	34.23	37.78		40.81	43.11	40.94	43.73	43.05	43.99	43.44	44.89	44.74	44.62
ZNPP-2	40.06	38.31	38.54	39.63	39.00	43.73	41.94	40.95	41.75	43.37	44.49	43.52	44.75	44.73
ZNPP-3	39.85	39.85	31.68	24.12	37.03	37.86	42.21	43.53	41.80	44.20	44.31	44.91	47.96	50.77
ZNPP-4	41.26	40.04	41.32	43.09	41.05	43.64	44.20	43.46	44.67	44.51		45.23	46.00	45.98
ZNPP-5	38.41	39.99	38.46	41.35	42.27	42.78	43.37	42.45	42.13	43.06	44.67	44.86	45.82	49.43
ZNPP-6	12.72		29.28	35.47	41.36	36.80	41.77	44.01	43.10	44.31	44.86	43.51	45.06	44.65

FIG. 5. Average burnup values of unloaded spent FAs at ZNPP's units.

Due to rising the initial fuel enrichment, planning fuel burnups, as well as uranium content in the fuel assembly with a view to meet the long term storing requirements the handling with spent fuel is complicated. The implementation of such new nuclear fuel can be considerably hampered by restrictions with regard to at reactor cooling pool capacities and FAs storage racks abilities to support the sub criticality of the pools under the limits.

- For example, to meet the Zaporizhzhya ISFSF acceptance requirement (criterion) on spent FA decay heat limit, the cooling time of this spent FA, if it had got burnup 62 MW·d/kgU, has to be no less than 13 years.

- It is not realizable even for at reactor cooling pool with 612 places of capacity, taking into account the capability reservation for emergency core unloading to be maintained at all times.

- So, some safety reassessment and spent FAs storage racks modernization at reactor cooling pools can be required for a transition from 4-years to 5-years fuel cycle, which is preliminary characterized (EFPD=325) by:

— Fas quantity in reload batch — 32–36;

— Reload batch average enrichment — 4.83 wt% of ^{235}U;

— Average FA design burnup — 57,8 MW·d/kgu;

— Maximum FA burnup — up to 68 MW·d/kgU.

Taking into consideration the outlook for the further FA burnup increase, which is supported by the basic designs of the Russian FAs (TVSA and TVS-2), in order to proceed to 5-years fuel cycle and next 6-years fuel cycle for Ukrainian WWER-1000 units, ought to mitigate the excessive conservatism of the Ukrainian safety regulations in respect of the safety substantiation of spent nuclear fuel management systems by means of the calculations of the fuel burnup effect on the criticality. With all this going on the existing conservatism of regulations ought to keep in respect of location of low burnup fuel. Under such conditions, it is necessary to divide the at-reactor cooling pool into individual zones for high burnup spent FAs and for low burnup (or fresh) fuel, included the capability reservation for emergency core unloading. In this connection it is necessary to note that due to replacement of the TVS-M FAs on TVSA FAs in the period 2003-2008 the income of the leaking WWER-1000 FAs unshipped before the appointed time was decreased noticeably. The dependence of annual income of leaking WWER-1000 FAs, unloaded ahead of schedule, from the TVSA loading is shown in Fig.6.

In order to reduce the cost of SFM the NNEGC “Energoatom” performs a set of activities among of which the following ones are the most promising:

— Implementation of 5-year fuel cycle with reduced consumption of the natural uranium and fuel assemblies in the reload batch;

— Operation of dismountable fuel assembly that allows to load back in the core the repaired fuel assembly after removal of the leaking fuel pin;

— Improving of prediction of the on-power primary coolant activity based on the results of the fuel assembly sipping control (using of the rtop-ca code as a tool for solving the acceptable prediction of coolant activity when parameters of failure are known[14]);

— Studying of the fuel cycles and reactor technologies for utilization of fissile materials recovered from the spent fuel.

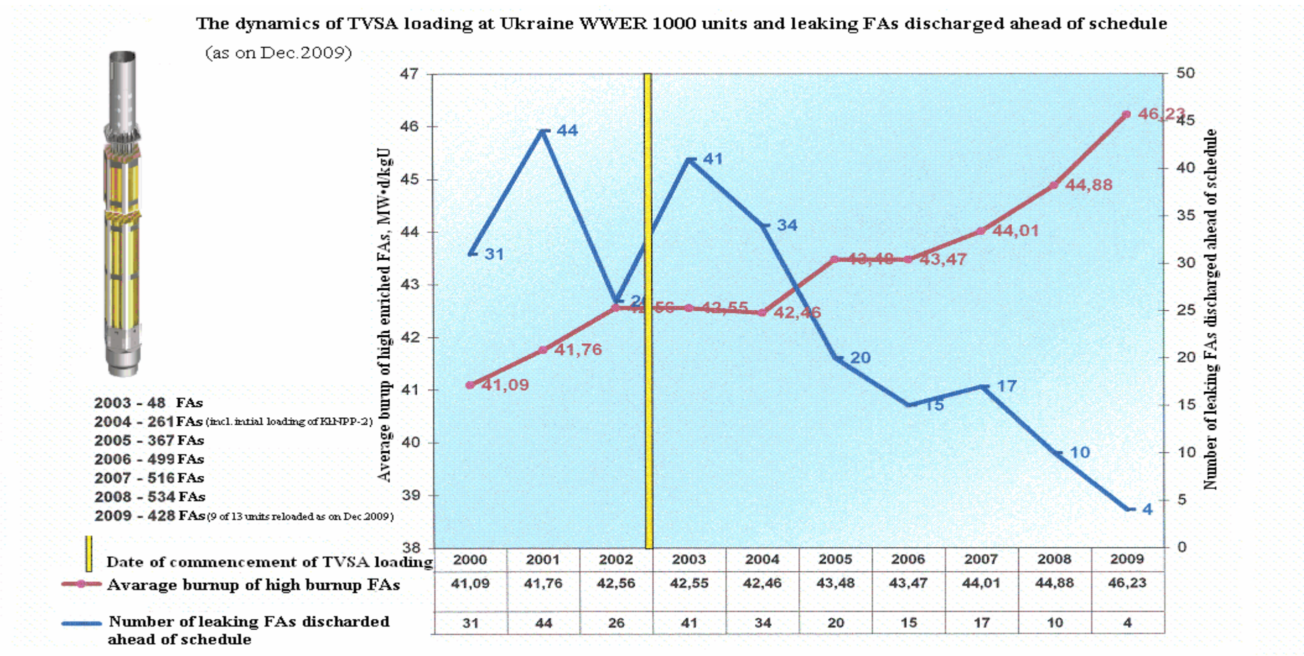

FIG. 6. Dependence of annual income of leaking WWER-1000 FAs, unloaded ahead of schedule, from the TVSA loading.

4. CONCLUSIONS

• The positive almost ten-year experience of the Zaporizhzhya ISFSF operation confirms safety, reliability and cost efficiency of the dry storage cask technology.

• Expansion of the dry interim spent fuel storage should be considered as the primary option for the next several decades. At near-term outlook there is no any other alternatives in Ukraine a long time interim storage. The choice of the option between spent fuel reprocessing and extended storage could be done just after the long time interim storing. However, it is necessary to keep in mind that Ukraine already has Chernobyl's spent fuel that does not have any perspective for reprocessing and should be disposed in the geological formation, so the

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[14] Developer - SRC RF TRINITI, Troitsk, Moscow Region, Russian Federation (Proceedings of the 7-th International Conference "WWER fuel Performance, Modelling and Experimental Support", 17-21 Sep. 2007)

option of WWER spent fuel disposal in the geological formation might be preferable as the domestic ability of Ukraine without the severe dependence from foreign services.

- Improvement of the fuel utilization at the existing NPPs and development of the fuel cycles with technologies for the recovered fissile materials utilization are the current priorities in reduction of the spent fuel generation.

Realization of the named activities facilitates the nuclear spent fuel generation decrease per energy generating unit, but does not call off the major task, which is the choice of the optimal strategy for the nuclear SFM in Ukraine. One should take into consideration that spent fuel reprocessing for recycling of fissionable products has no reason without new special nuclear reactors in the long-run plans of the Ukraine's energy strategy.

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[1] "Strategic Directions on Spent Nuclear Fuel Management of Ukraine's Nuclear Power Plants under the Energy Strategy of Ukraine for Period up to 2030", approved by Ministry of Fuel and Energy of Ukraine on 13 May 2008.

[2] "Safety Analysis Report of Dry Spent Fuel Storage Facility at the Zaporizhzhya NPP", Revision 3.01.1, approved by NNEGC "Energoatom" on 22 March 2006.

[3] "Regulations for the Safe of Storage and Transport of Nuclear Fuel at Nuclear Energy Installations", ПНАЭ Г-14-029-91.

[4] "Annual Report on Safety State of the Zaporizhzhya ISFSF for Year 2009, approved by NNEGC "Energoatom" on 26 March 2009.

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ATUCHA I: NPP SPENT FUEL DRY STORAGE CONCEPTUAL DESIGN

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• **Abstract**

• The present report shows a Spent Fuel Dry Storage Conceptual Design to emptying the former oldest fuel elements pool and storage them in a Dry Storage System, in order to reach the 32 (Full Power Years) FPY of Atucha I Nuclear Power Plant (CNA I) End Of Life [1]. The project consist mainly in the enlargement of one of the Pool Buildings of the Station (there are two of them) where the (Spent Fuel Elements) SFE will be stored in vertical underground silos. Each silo is composed of two storage units that contains 9 fuel elements each (18 fuel elements in each bin). This design allows a vertical storage of 2016 spent fuel elements (7 rows by 16 columns). The SFE must be transferred from the Pool Building to the Dry Storage Building through a dedicated shield for lifting and transporting the SFE. To move the shield, the actual 60 Ton capacity crane will be used. The operation time to emptying a complete pool will be approximately one year (1998 SFE). Therefore the storage system should be finished by 2013, in order not to penalize the continue operation of the Station. This conceptual design meets the basic principles of Nuclear Safety, protecting workers, public and the overall environment of ionizing radiation and radioactive contamination. This is achieved by transport and storage shielding, operation procedures and comply key conditions like subcriticality of the system, SFE monitoring and SFE heat removal.

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1. INTRODUCTION

According to the scenario projected by Nucleoeléctrica Argentina Sociedad Anónima (NA-SA) for CNA I operation, considering a power factor of 85%, the pool storage capacity will be exhausted in March 2015. Plant End of Life by design will be reach in December 2017.

So, in order not to penalize plant operation, it is required to have at least 614 free positions in the pool before March 2015. If not, the station will be out of service 33 months before the Plant End of Life (PEL).

To avoid interferences with the normal operation of the plant it was evaluated to make the installation of a temporary Dry Storage Fuel Elements as an extension of controlled area. The project provides the transfer of longer decay SFE in Station Storage Pools.

Under the constrains of the daily operation, the project was focused on the Building Pool Nº1 enlargement, in order to built an underground SFE storage. It is necessary to design specifics devices to transfer the SFE from the underwater pool to the storage place in a safety way.

This system ensures the operation of CNA I to reach the design end of life and possibility a link with the Atucha II Nuclear Power Plant (CNA II) Dry Storage System. The Comisión Nacional de Energía Atómica (CNEA) of Argentina is in charge of implementing the conceptual design of the Spent Fuel Elements Dry Storage (SFEDS).

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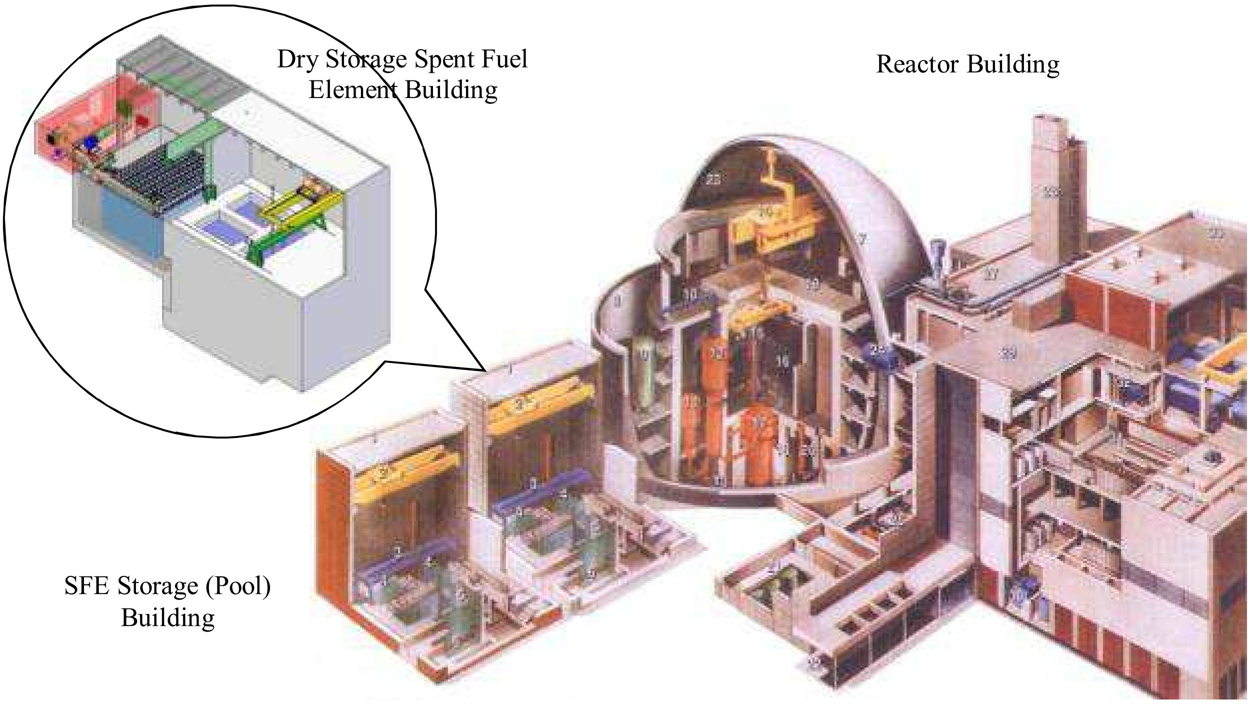


FIG. 1. View of the location of the Dry Storage Building in the Atucha I Nuclear Power Plant.

2. CNA I CURRENT SITUATION

The fuel assembly of CNA I have an active length of 5.3 m and a circular cross section of 0.10 m diameter, with 36 fuel rods plus one structural rod. Each Fuel Assenbly (FA) is loaded with approximately 176 kg of UO_2.

CNA I was fuelled with natural uranium during the first 27 years of operation, the average burn up of the spent fuel was approximately 6.000 MWd/tU. In January 1995, the utility started a program to gradually convert the fuel to Slightly Enriched Uranium (SEU), using an enrichment of 0.85% U-235.

The program was completed in august of 2001, since then the whole core is fuelled with SEU and the average burn up of the spent fuel element is approximately 11.300 MWd/tU. This change produced an important saving in fuel consumption: from approximately 395 FA/FPY to approximately 210 FA/FPY.

The Fuel Elements Management Division of CNA I determined that with the proposed system i.e. 2016 Fuel Elements (FE) storage capacity the PEL would be exceeded in 5.27 FPY or 6.19 calendar years **(this is assuming a 0.72 FE daily consumption, power factor of 85% and 250 reserves positions to empty the reactor core).**

With this FE storage capacity the station could run until February 2024, if a Life Extension is got, time enough to build a Dry Storage System compatible with CNA I and CNA II.

3. FUNDAMENTAL REQUIREMENTS TO DRY STORAGE SYSTEM

(a) Safe SFE confinement for a minimum of 50 years;

(b) Subcriticality: The storage configuration must be subcritical;

(c) Biological shield: The radiological limits should be respected;

(d) Heat balance: The FE temperature should never exceed, inside the silo, the maximum allowable by the FE sheath;

(e) System must be reversible, wet-dry-wet;

(f) Passive cooling methods ensuring maximum temperature not to be exceeded;

(g) Loading and unloading of the containers underwater;

(h) The loaded transport shield must not exceed the capacity of the existent crane;

(i) Possibility to isolate both buildings;

(j) To Include in the design the possible life extension of the CNA I;

(k) Must to be constructed and licensed by 2013 to allow sufficient time for transfer SFE from pool I to SFEDS;

(l) The conceptual design includes containment barriers which increase the safety of the storage system.

4. CRITICALITY OF THE CONFIGURATION ADOPTED

The calculation of the system criticality should be performed for different scenarios proposed (Montecarlo code used), both for normal operation and accidental situations during storage maneuvers (calculation based on the regulations NUREG-1536, *"Standard Review Plan for Dry Cask Storage Systems"* and NUREG-1617, *"Standard Review Plan for Transportation packages for spent nuclear fuel"*). These regulations establish criteria for acceptance a multiplication factor $K_{eff} \leq 0.95$.

Calculations will be performed for natural uranium SFE (0.711% U-235) and for enrichment uranium as used in CNA I (SEU 0.85% U-235).

In the calculation model the study area is in red.

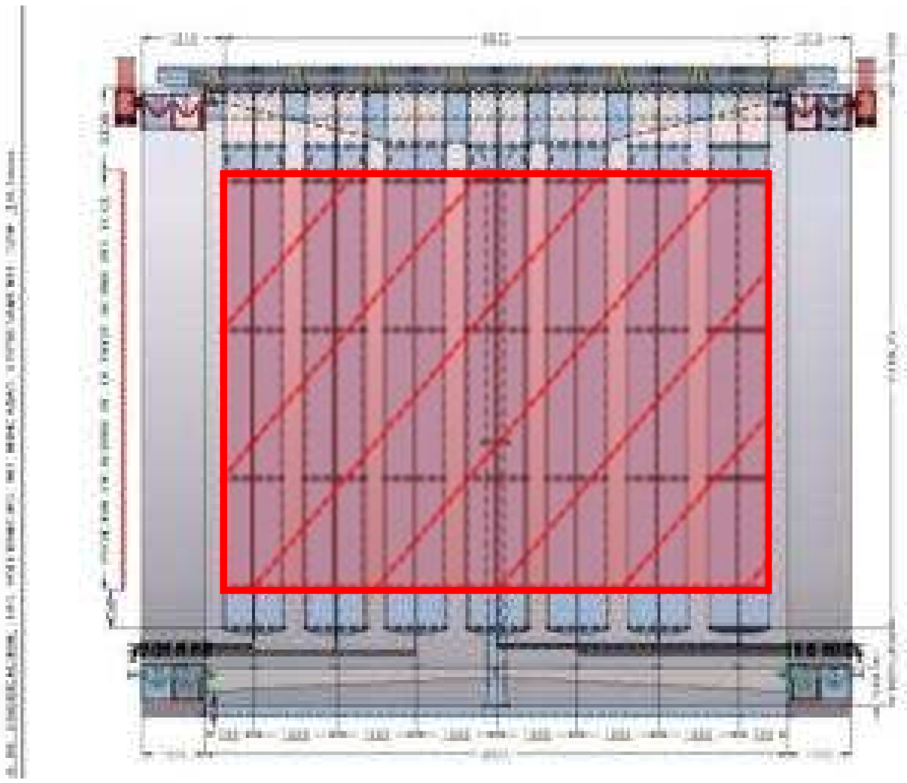


FIG. 2. Sectional view storage silos. Area for criticality calculations in red.

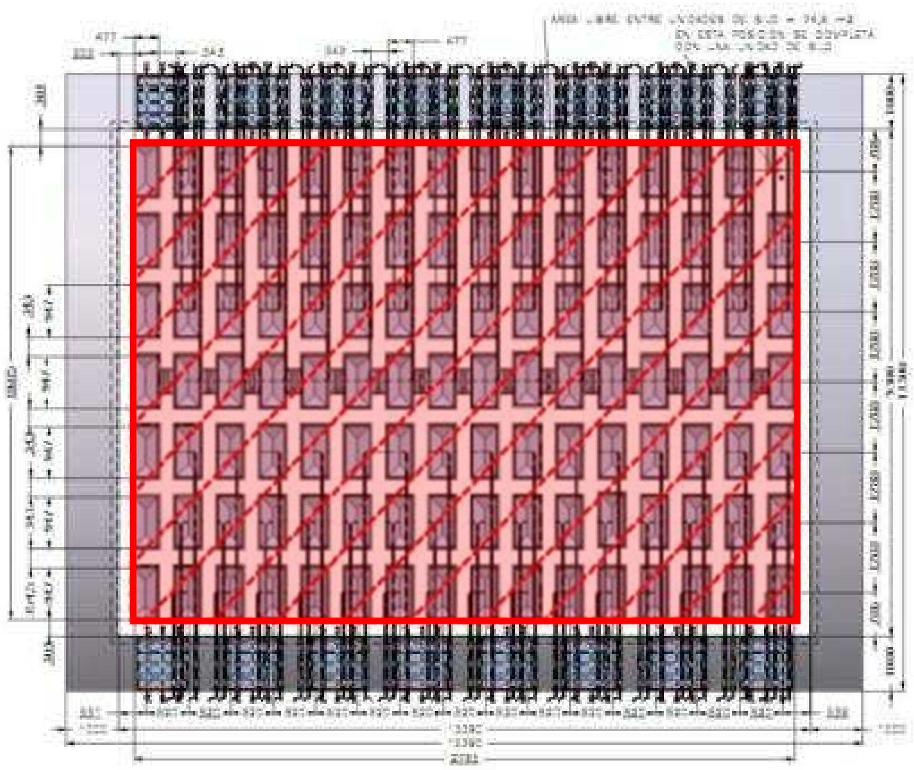


FIG. 3. Plan view of storage silos. Area for criticality calculations in red.

Installation overview:

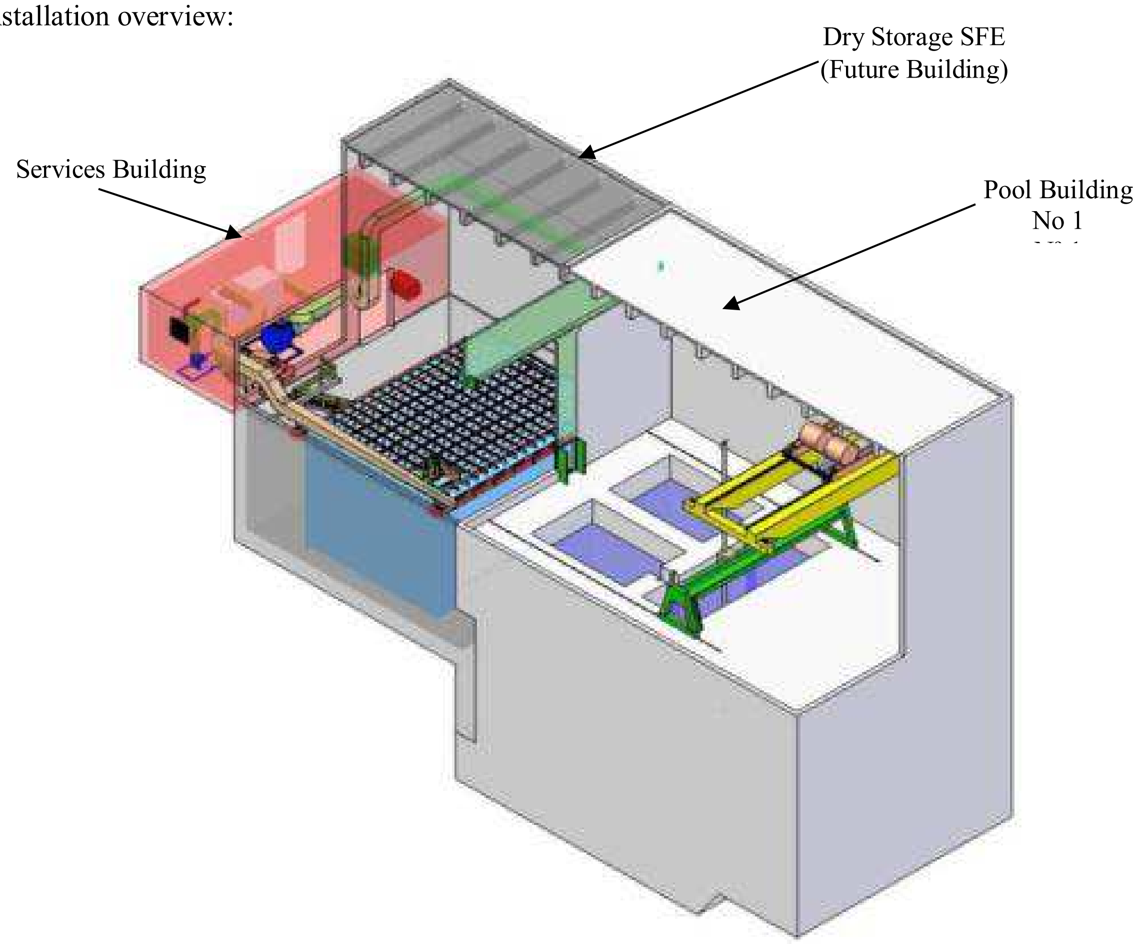


FIG. 4. SFE Storage building layout.

To avoid interferences with the plant normal operation, it is preferable to perform the SFEDS in vertical silos in the Pool Building number 1, (see Fig. 4). The SFE of that pool are stored there since the plant start operation. The underground silos should be an extension of the existing building and with a load capacity of 2016 SFE (7 rows and 16 columns).

As explained before, the necessary time to transfer 1998 SFE (a complete pool capacity) to the dry storage place will take approx one year.

5.1. Silos building

The installation provides for the enlargement of SFE Building I, towards the west side, a building which will contain two underground silos (as is shown in Figs. 4 and 5), with 2016 SFE capacity, including internal components of the reactor, now deposited in SFE Building I and II (cooling channels, control rod guide tubes, control rods, etc.).

The new building should have the services of the present SFE Building (i.e. water, compressed air, ventilation, electric power, radiation monitoring systems, and enlargement of the crane bridge rails) to be able to carry out the correct transfer, maintenance and complete control of the system.

5.2. Silo unit

A Silo Unit (SU) will have capacity to store two Storage Baskets (SB) for 9 SFE each and will have stainless steel wall, drying piping and SU monitoring. Figures 6 and 7 show the empty and loaded SU.

Remnant decay heat on SFE will be removed using natural thermal convection inside the closed SU.

This heat is transferred trough the SU walls to the air between silos and cooled with a new ventilation system. Figure 8 shows the SU support grid and Fig. 9 shows a sectional view of the SU in the silo building.

Some SU will have instrumentation provided to allow getting information of the equilibrium temperature which the SFE bundle will reach, as well as its radiological status.

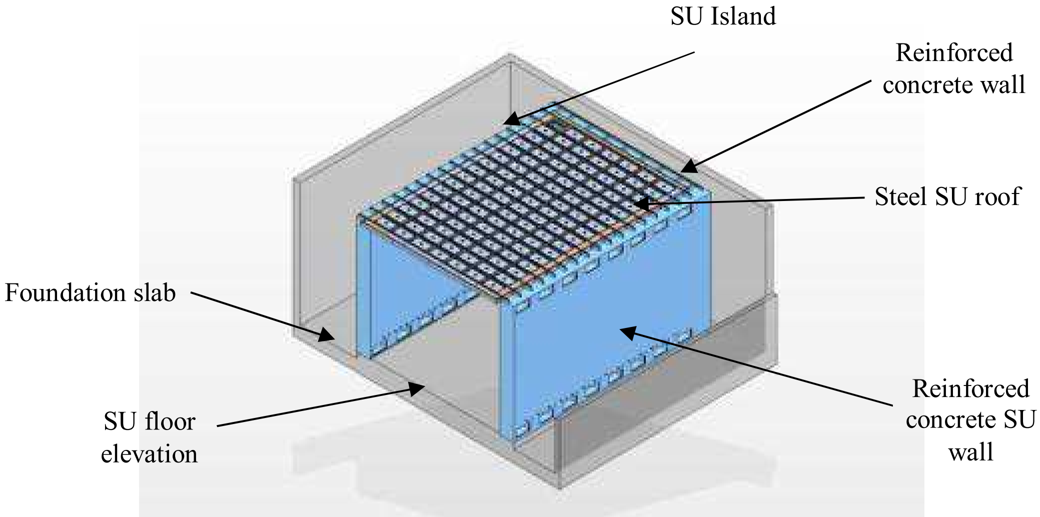


FIG. 5. Isometric view of SU location.

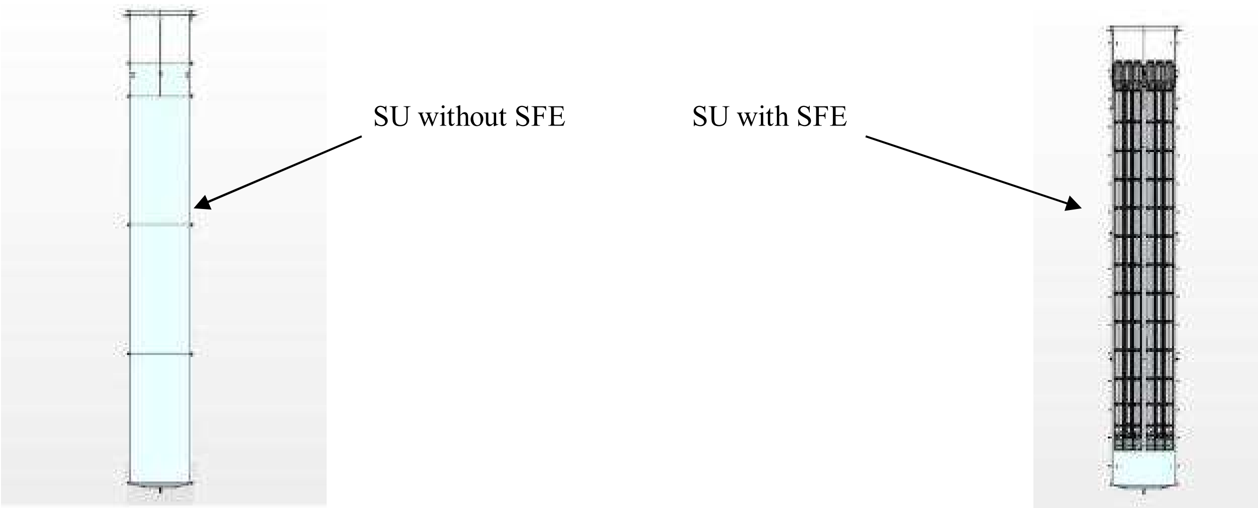


FIG. 6 and FIG. 7. Empty and Loaded SU.

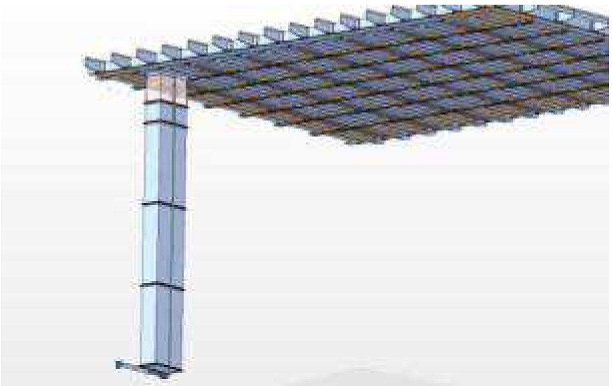


FIG. 8 Support grid.

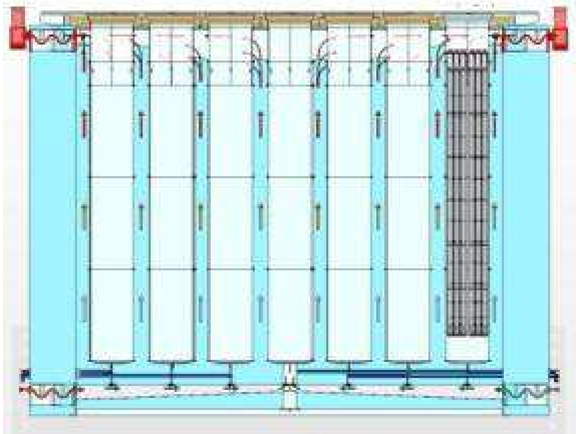


FIG. 9. Sectional view.

5.3. Storage Basket

The SFE will be stored in steel rectangular SB, with 9 units' capacity each. As shown in Figs. 10 and 11 the SB has a support system on top which allows locking with the lifting tool.

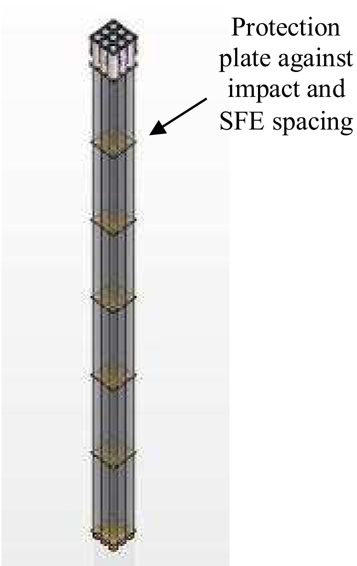


FIG. 10. SB protection plate.

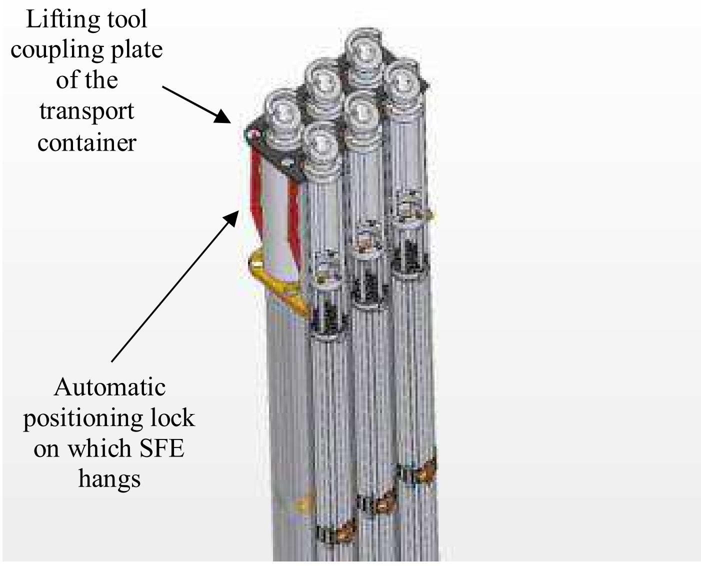
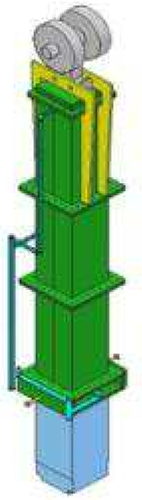


FIG. 11. SB lifting tool and positioning lock.

5.3.1. Shielding for lifting and transportation

To remove the SB with the SFE inside them, the facility has a structure, shown in Figs. 12 and 13, with two functions, store the SB inside them and provide adequate shielding level for the workers who are performing the SFE transport operation to the dry storage. Besides, it allows drying the SFE and the SB in their resting position and SFE monitoring integrity.



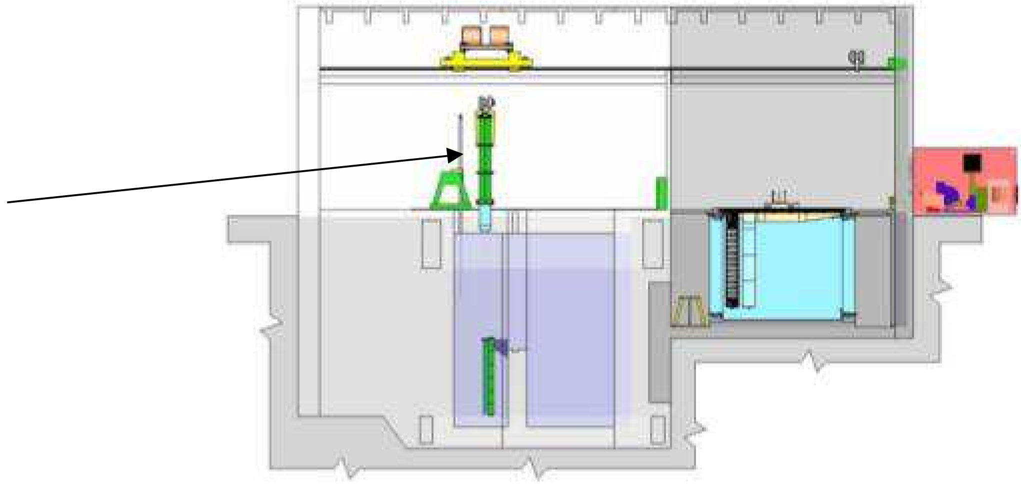


FIG. 12. Detailed view of the shielding for lifting and transportation device

FIG. 13. General view of the shielding for lifting and transportation

- 6. COOLING

6.1. Natural convection internal cooling

The gas inside the silo (not defined yet) will flow from bottom to the top by natural thermal convection as shown in Figs. 14 and 15, and will transmit the accumulated heat on top by conduction through the walls to the silo building.

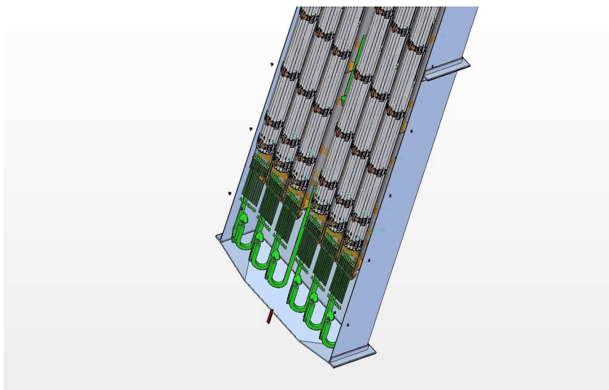


FIG. 14. Bottom of the SU

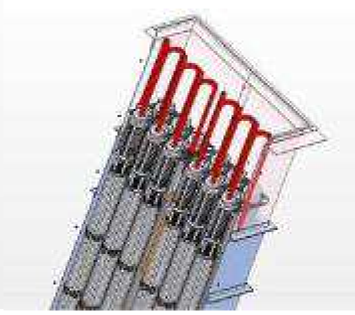


FIG. 15. Top of the SU

6.2. External cooling by means of forced convection and conduction through the silo wall

Silos cooling are made by forced convection air circulation through the ventilation system, which extracts the heat produced in the silo by conduction as shown in Fig. 16. The heat is transmitted by conduction through the concrete silo wall. The atmosphere between silos is under depression related to building environment

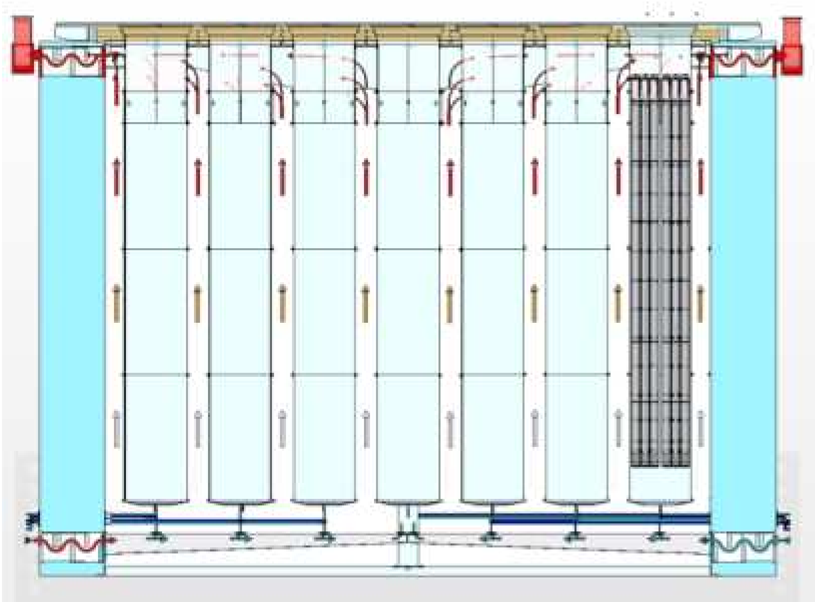


FIG. 16. Sectional view of silo unit (SU). Air distribution among silos trough the air injection duct at the bottom. Air extraction is through the upper area.

7. CONCLUSIONS

Temporary Dry Storage in an enlargement of Pool Building I is shown as the most economical and simplest alternative from the licensing point of view as SFE are not removed from the controlled zone, allowing the operation of CNA I until the Dry Storage System for CNA II is put into service.

Construction details will be analyzed at the detail engineering stage, which should be done parallel to the licensing stage.

With this system the following is achieved:

- SFE will not leave the Controlled Zone (which means, lower demands for licensing) and allows additional containment barriers (Silo, Silo Structure and Reinforced Concrete Wall of the Silo) which decrease potential radiological impact on the Environment;
- SFE can shift from wet storage to Dry Storage, and eventually go back to wet storage, according to CNEA’s decision after discontinuing its operation in 2017;
- It allows to store irradiated internal components of the reactor (channels, control rod guide tubes), which now are taking place in the decay pools;
- Decrease investment regarding other proposed systems;

- It allows normal operation of CNA I until reaching the end of its life according to design and considers a possible life extension, enough to connect to the future CNA II Dry Storage System.
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MANAGEMENT OF CASKS IN JAPAN'S INTERIM STORAGE FACILITY

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Abstract

Recyclable-Fuel Storage Company (RFS) is planning to construct Japan's first interim storage facility for spent fuels from light water reactors. This facility is designed not to have hot cells and to store the transported casks as-is. During storage, it is checked if spent fuels are appropriately stored by monitoring of pressure between lids of metal casks, periodical radiation measurement of casks, and others. The planned storage period is 50 years. After storage, casks are carried out to reprocessing facilities without opening the lids and refilling spent fuels. Management works during carrying-in, storage and carrying-out of casks at the storage facility are described in this document.

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1. OUTLINE OF RFS AND STORAGE FACILITY

RFS was jointly invested and established by electric power companies (TEPCO and JAPC) to serve as a company specializing solely in spent fuel storage. RFS plans to eventually store about 5,000 tU of spent fuels in Aomori Prefecture in Japan (Fig. 1). First, RFS plans to construct a building with storage capacity of up to about 3,000 tU that is large enough to store up to 288 dry storage metal casks (Fig. 2). RFS takes charge of cask storage and electric power companies take charge of cask transportation between facilities.

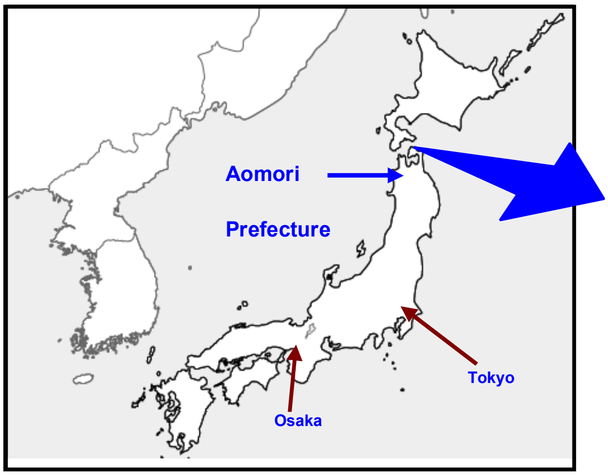


FIG. 1. Location of Aomori Prefecture in Japan.

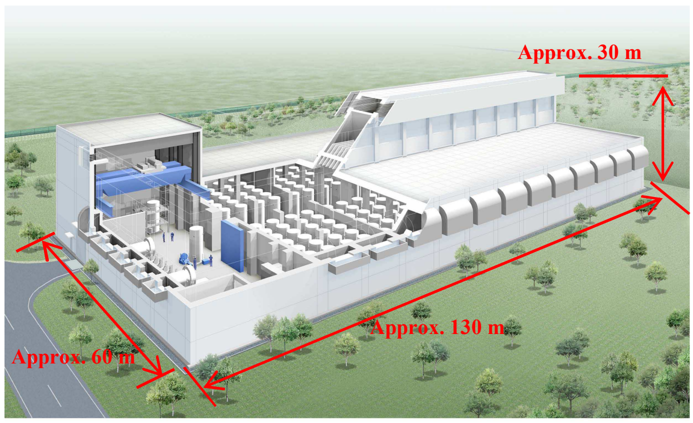


FIG. 2. Schematic diagram of storage facility.

2. STRUCTURE OF METAL CASK

The metal cask has a multiple containment structure that consists of primary and secondary lids. Radioactive substances are contained in the metal cask by making the pressure in the space between lids positive in advance to create a pressure barrier and by making the pressure in the metal cask negative. Metal gaskets are used to seal the lids from the viewpoint of maintaining the containment function over a long period of time. The pressure in the space between lids is continuously monitored during storage, and any leakage from the primary lid or the secondary lid can be detected as the pressure in the space between lids decreases due to such leakage. For information, the metal gasket used in this storage facility is the same type as the one used in the dry metal casks stored in nuclear power stations in Japan. The dry storage casks stored in nuclear power stations have not caused any problem in containment function during their use of more than ten years.

At the time of transportation and an anomaly of containment function, the structure is designed to allow a tertiary lid that uses an elastomer gasket to be mounted. The elastomer gasket has been actually used in the present transportation casks over many years.

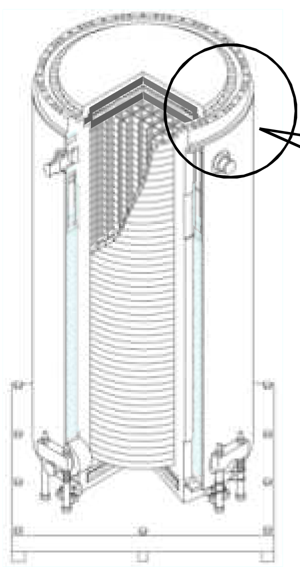


FIG. 3. Schematic diagram of metal cask.

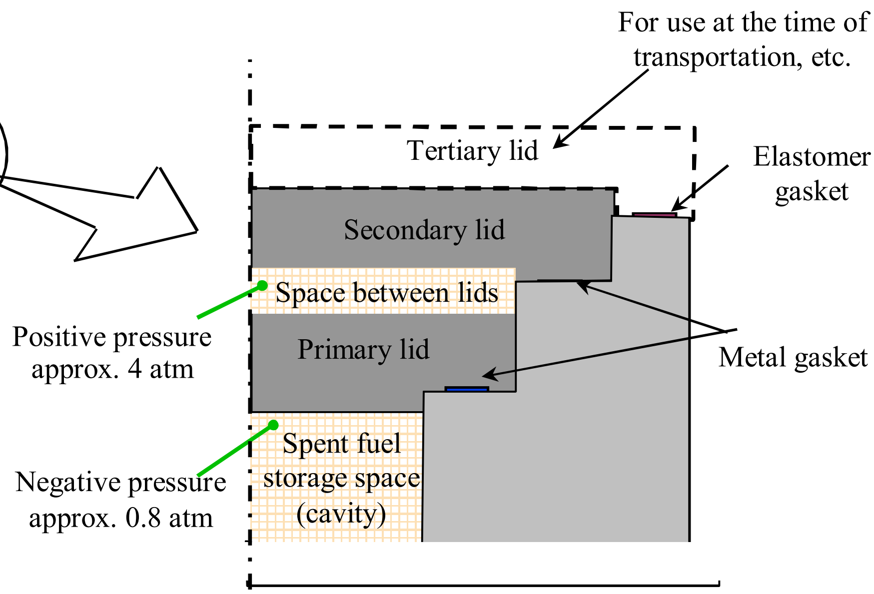


FIG. 4. Detail of lids of metal cask.

3. MANAGEMENT OF METAL CASK

Fig. 5 shows the plan of storage building. The storage building consists mainly of three areas, i.e., an acceptance area where transported metal casks are accepted, a storage area where metal casks are stored, and an auxiliary area where supervisory panels to indicate and record signals from the detectors in the storage area are installed. The management works mainly done by RFS are to inspect and monitor metal casks and to make and save records thereof from the acceptance to carrying-out of metal casks in the storage facility.

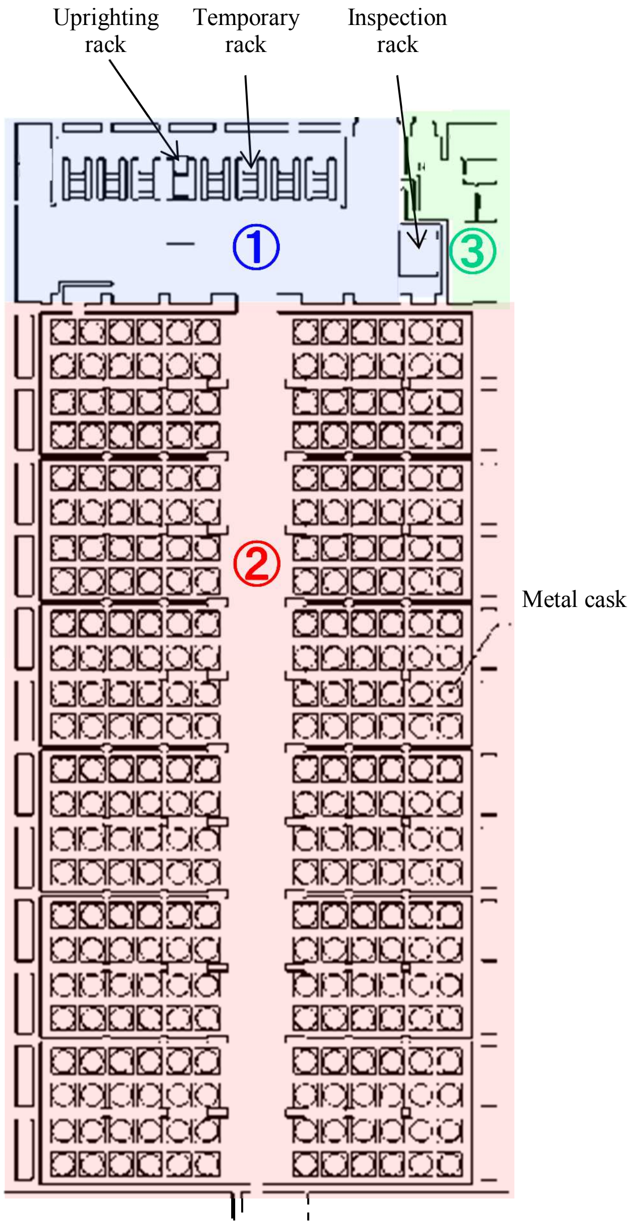


FIG. 5. Plan of storage building.

General description of each area:

① Acceptance area (casks are moved by using overhead traveling cranes and cask transport vehicle):

- Temporary racks: there are seven racks where casks accepted are temporarily placed;
- Uprighting rack: there is one rack that is dedicated to upright horizontally placed casks;
- Inspection rack: there is one rack where casks are inspected, etc.

② Storage area (casks are moved by using cask transport vehicle):

The area where metal casks are stored vertically. The area has a space where 288 casks can be stored.

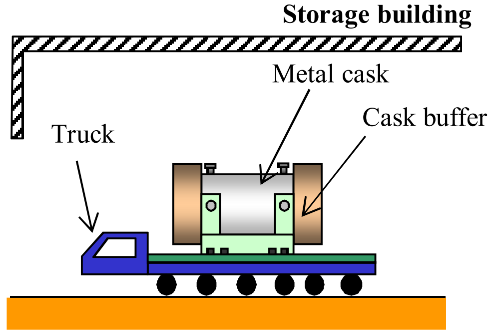
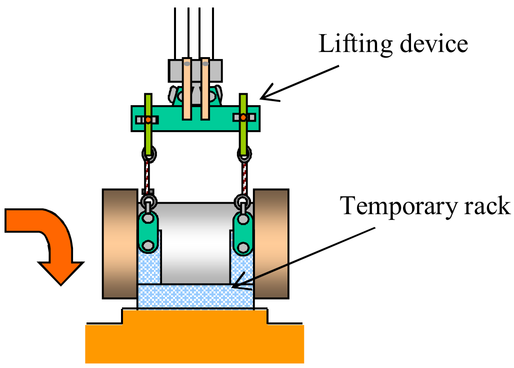
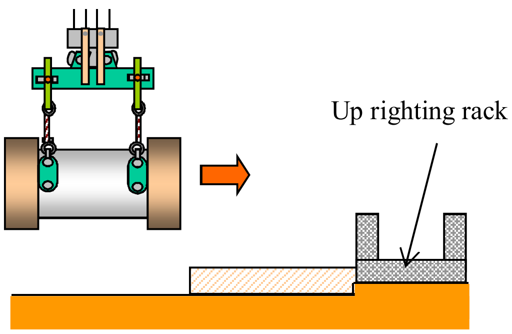
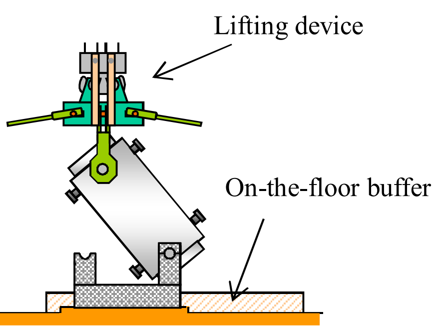
③ Auxiliary area:

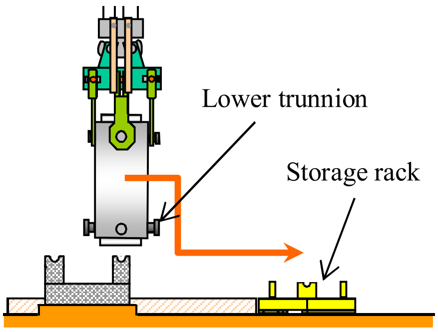
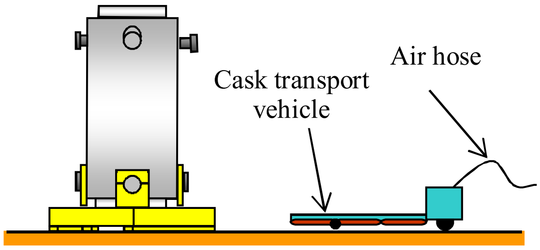
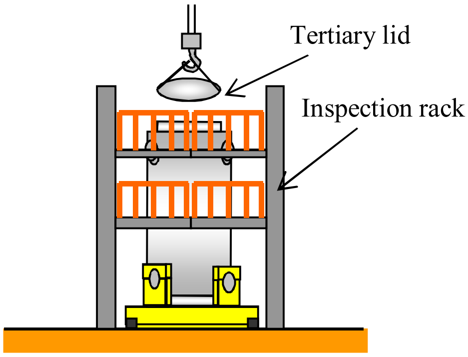
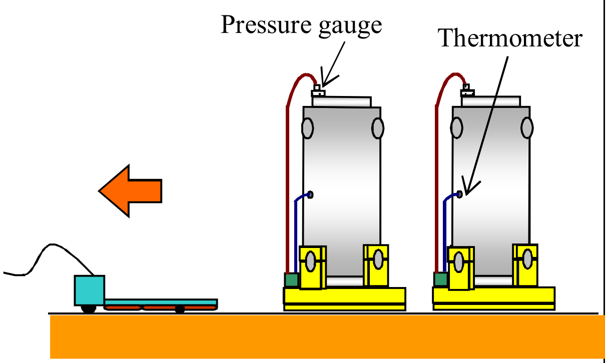
The area has supervisory panels and others to indicate and record the signals from the detectors to check the pressure in the space between lids of casks and the temperature of air supplied into and exhausted from the storage building.

The works to manage metal casks at each stage from the acceptance to carrying-out of metal casks in the interim storage facility are described as follows.

3.1. Acceptance of metal casks into the storage facility

Electric power companies load their spent fuels into casks at their power stations and transport them to the storage facility. Records of inspections (pre-shipment inspection records) that electric power companies are responsible to implement at their power stations are delivered to RFS together with casks at the time of carrying-in. The handling process of metal casks in the storage facility is described as follows.

	Handling process of metal cask	
1	[Acceptance area] 	A truck that carries a metal cask enters into the acceptance area, and appearance inspection, radiation measurement, surface contamination measurement and others are conducted on the truck.
2	[Acceptance area: temporary rack] 	The metal cask is moved by using an overhead traveling crane from the truck onto a temporary rack and horizontally mounted on it.
3	[Acceptance area: up righting rack] 	The metal cask is moved from the temporary rack onto an up righting rack to be vertically mounted.
4	[Acceptance area: up righting rack] 	Cask buffers are removed at the up righting rack. Then, the cask is vertically mounted by using the overhead traveling crane. On-the-floor buffers are always placed around the up righting rack in preparation for the event that the cask is toppled over.
5	[Acceptance area: uprighting rack]	

		The vertically mounted cask is moved onto a storage rack. The cask is moved above the on-the-floor buffers, while limiting the lifting height. The cask is placed on the storage rack and then fixed on the storage rack through four lower trunnions.
6	[Acceptance area] 	A cask transport vehicle is inserted under the storage rack, and the vehicle is floated. (The cask transport vehicle is designed to move a metal cask horizontally by floating the vehicle with pneumatic pressure and thereby reducing the friction between the floor surface and the vehicle.)
7	[Acceptance area: inspection rack] 	The cask is moved onto an inspection rack by using the cask transport vehicle. The cask is placed on the inspection rack, and then the tertiary lid for transportation is removed by using the overhead traveling crane, and dose equivalent rate inspection and others are conducted. Then, the devices to measure the pressure in the space between lids of the cask and the temperature on the side of the cask are installed.
8	[Storage area] 	The cask and the storage rack are moved again by using the cask transport vehicle into the storage area to be unloaded onto the floor. Then, the storage rack is fixed on the floor by bolts.

3.2. Management of metal casks during storage

The metal casks that are carried into the storage area are emplaced one by one from the outside (air inlet side) of the storage building toward the centre (air outlet side) thereof. At the time, the total quantity of heat from metal casks per line is managed to maintain the air temperature around metal casks and the concrete temperature of the storage building at standard values or lower. Moreover, the air inlet and air outlet are equipped respectively with temperature sensors and the temperatures thereof are measured to monitor if the heat removal function of the storage building is properly maintained.

Fig. 6 shows the conceptual scheme of metal cask management in the storage facility. During the storage period, in addition to the temperature of air supplied into and exhausted from the storage building as mentioned above, the data of area radiation, pressure in the space between lids of metal casks, surface temperatures of metal casks and others are transmitted to the supervisory panel. RFS responsibly saves these records throughout the storage period together with the records delivered at the time of acceptance of metal casks. If any anomaly is found in metal casks such as leakage from primary lids during the storage period, necessary additional measures are taken and such casks are returned to electric power companies.

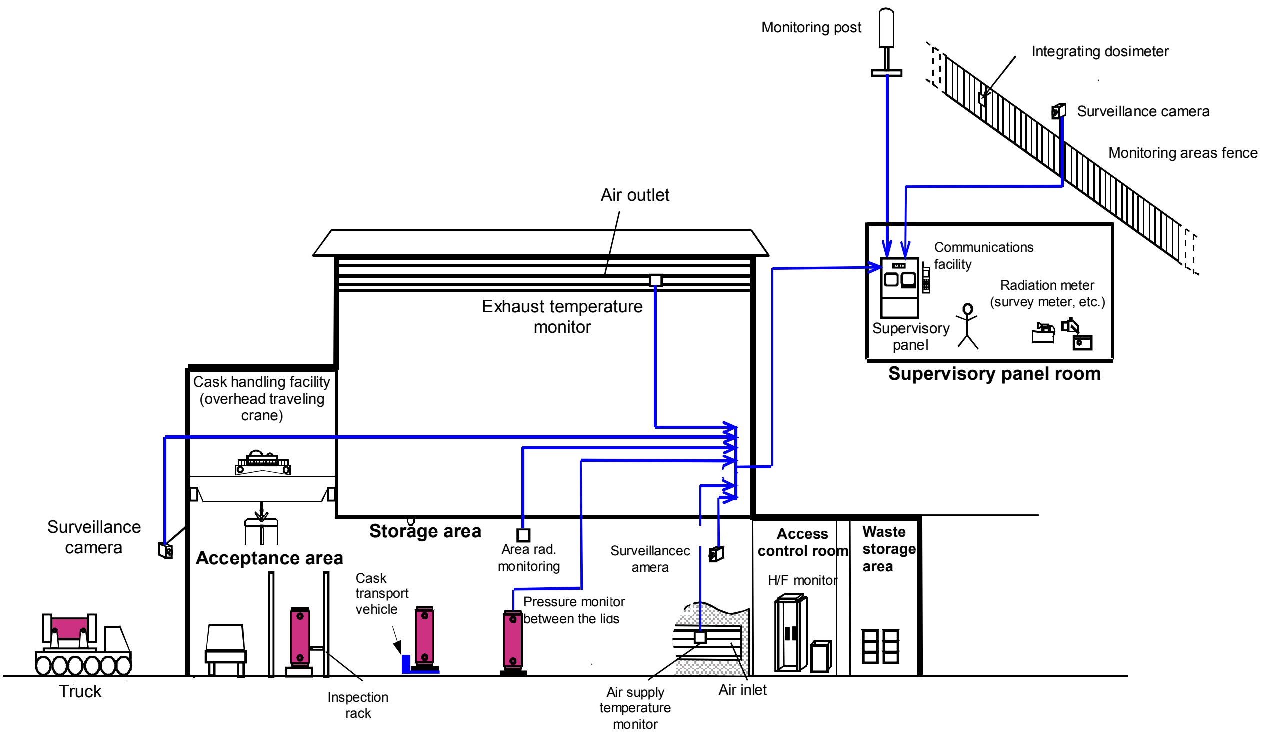


FIG. 6. Configuration of facility.

3.2. Carrying-out of metal casks after storage

After storage period, RFS will return casks to electric power companies at the storage facility, and electric power companies will transport them to reprocessing facilities. In the event of returning casks, records at the time of acceptance mentioned in 3.1 and records during storage mentioned in 3.2 are also returned to electric power companies.

4. FUTURE PLANS

RFS submitted an application for safety review of the basic design of its interim storage facility to the regulatory authority, and received permission in May 2010. RFS will submit an application for review of the detailed design of the facility to the regulatory authority, and then start construction of the storage building and production of metal casks after approval of the detailed design. RFS plans to start the spent fuel storage business in 2012 and accept about 30 metal casks a year.

IMPROVEMENT OF OPERATIONAL SAFETY OF DUAL-PURPOSE CASK FOR SNF IN STORAGE

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Abstract

By now more than 100 dual-purpose packaging sets TUK-108/1 are in operation in the mode of interim storage and transportation of SNF from decommissioned nuclear powered submarines (NPSs). In accordance with certificate, spent fuel is stored in TUK-108/1 on the premises of plants involved in NPS dismantlement for 2 years, whereupon it is transported for processing to PO Mayak. At one Far Eastern plant Zvezda involved in NPS dismantlement there arose a complicated situation due to necessity to extend period of storage of SNF in TUK-108/1. To ensure safety over a longer period of storage of SNF in TUK-108/1 it is essential to modify conditions of storage by removing of residual water and filling the inner cavity of the cask with an inert gas. This report describes key issues of cask drying technology, justification of terms of dry storage of naval SNF in TUK-108/1 and RBMK-1000 SNF in UKhK-109, design features of the mobile drying facility, results of operation of the pilot facility at the Far Eastern plant Zvezda and cold testing on the shop factory for Leningrad NPP.

1. INTRODUCTION

One of the strategic goals in the area of nuclear energy development in the Russian Federation is SNF and radioactive waste management [1]. In accordance with the waste management concept adopted in the Russian Federation it is supposed to process the waste at the final stage of nuclear fuel cycle. Around one fifth part of the SNF unloaded is to be processed. The remaining quantity of SNF is subject to be stored for its further processing.

Around 10 years ago the works related to developing a new technology with regard to the SNF management were initiated. The key point of such technology is a double-purpose package intended for storing and transporting of the SNF on base of metal concrete cask (MCC) [2,3].

During that period of time the most crucial situation was created for unloading of SNF from the Russian decommissioned nuclear submarines, as the existing transport infrastructure and process enterprises were not able to satisfy with the requirements of removal and conditioning the SNF arriving from both already utilized nuclear power submarines and their reactors which are subject to be utilized. The USA and Norway mutually with the Russian Federation took part in a trilateral Project 1.1 which was aimed at creation of a pilot cask intended for temporary storage and transporting of the SNF from the utilized nuclear submarines. The project was an integral part of the Program of Military and Technical Cooperation in the area of the Artic Environment Protection (AMEC Program). In December of 2000 the Project was successfully completed by issuing a certificate permitting design of the SNF carriage cask and transporting it from the nuclear power submarines. It was the first certified double-purpose transport package of a metal concrete cask type, which was given the index TUK-108/1.

Design and certification of double-purpose metal concrete casks have been finalized recently for the following types [4]:

- TUK108/1 — for transporting and storing of the SNF from submarines;

- TUK-120 - for storing and transporting of the SNF from nuclear ice-breaking fleet;
- TUK-104 and TUK-109 — for transporting and storing of the SNF coming from the RBMK-1000;
- TUK-123 — for transporting and storing of the SNF coming from the BN-350 (town of Aktau, Kazakhstan);
- TUK-121 — for transporting and storing of highly radioactive waste.

The main stage of the TUK-108/1 putting into operation with regard to the SNF transporting was completed in 2003. More than 100 casks of such type were manufactured and supplied to the “Mayak” Facility, Federal State Unitary Enterprise “Zvezdochka” and Federal State Unitary Enterprise “DalRAW”.

Accumulating sites aimed at temporary storage of the TUK-108/1 have been created and are being operated at the points of the SNF unloading from nuclear power submarine reactors and its loading into the casks, such as Federal State Unitary Enterprise MP “Zvezdochka”, Federal State Unitary Enterprise DVZ “Zvezda”, and Federal State Unitary Enterprise “DalRAW”. In compliance with the effective certificates the term of SNF technological storage inside the TUK-108/1 at the accumulating sites is limited by two years. Initially the technological storage term was 6 months. Due to positive experience gained with regard to the TUK-108/1 operation under short term conditions a decision about extension of the SNF period of storage up to 2 years without taking additional measures for enhancing of the TUK-108/1 protection was made.

Capacity of the accumulating site for TUK-108/1 storing at the Federal State Unitary Enterprise DVZ “Zvezda” was increased two times. Increase of capacity was mainly caused by absence of possibilities to transport the TUK-108/1 containing the SNF via railroad to the point of TK-VG-18. Section of railroad line between the plant site and the main railroad needed to be renovated (this length takes around 27 km). In the end of 2009 the reconstruction was completed. Within the period of reconstruction it was necessary to extend the terms of technological storage of the SNF inside the TUK-108/1. Analysis of the initial conditions for TUK-108/1 and availability of residial water in volume of approximately 3.2 L inside the interior space of the cask on having the cases with SNF stuffed in showed the impossibility of further extension of the storage term for the SNF inside the TUK-108/1 without taking the additional steps on enhancing protection and reliability of the SNF storage inside the TUK-108/1. A measure of such kind is to remove residial water from internal space of the cask and stuff it by inert gas.

Taking into consideration traditions of the AMEC Program, the decision was made by the Program Director with regard to implementation of the AMEC Project 1.1- 2 “Development of drying technology for casks intended for temporary storage of the SNF belonging to the Navy Fleet”, as a logical continuation of the AMEC Project 1.1 on designing of storing the TUK-108/1 prototype [5].

The goal of the AMEC Project 1.1-2 was to create at one of the Navy SNF sites a pilot model of the Installation for drying the container stuffed with SNF. That measure provided its secured temporary storage, enhancing its protection while transporting the container and it also allowed increasing its operational lifetime by means of improving the corrosion resistance of the constructive materials for bags, remote grid and container itself including a sealed unit. The Federal State Unitary Enterprise DVZ “Zvezda” was chosen as a priority site which is in need to solve a problem regarding the term extension for the SNF storage stuffed in the TUK-108/1. Later on the pilot model of the Installation was modified to solve a similar problem concerning removal of residual water from the packaging set UKhK-109 for RBMK-1000 SNF before its placement for temporary dry storing in cask-type storage facilities, created in Leningrad and Kursk NPPs.

2. SOURCES OF RESIDIAL WATER APPEARANCE INSIDE THE TUK-108/1

In accordance with the existing technology for SNF management at the sites aimed at submarine utilization (coastal unloading facilities — BKV) or floating technical bases — (PTB), the SNF loading is made from the submarines reactor or PTB water pools with help of base re-loading container. The SNF loading and unloading operations by the re-loading container into the TUK-108/1 are usually made outside in the fresh air. Analysis of residual water generation in the cask was made for the entire chain of TUK-108/1 handling.

The sources of water appearance inside the TUK-108/1 to be stuffed with the SNF at a floating technical bases or coastal unloading facilities are as follows:

- Residual water after cask decontamination at the “Mayak” Enterprise;
- Humid air created while transporting in absence of proper sealed cask covers;
- Ice creation while transporting in winter time;
- Ice or water above ice creation on metal surfaces while unloading empty cases and casks preparation for the SNF stuffing in winter time.

While getting ready the casks to be stuffed by the SNF the obligatory control of water appearance inside the cask should be provided. When the temperature of the air outside is lower than — 4°C and air humidity is 100% when the cover is open, the residual water cannot be checked up by using this method of control.

Analysis of sources of residual water appearance inside the empty cask, and existing technology of its control, the design of cask internal space and remote grid made us to conclude that in the casks arriving for loading by cases stuffed with the SNF some quantity of water estimated as minimum possible volume and equal to 2.5 L (taking into consideration space size of the cask bottom and method of control) is always available.

Appearance of water inside the cask while stuffing by cases with spent nuclear fuel assemblies from floating technical base pools using the base container is theoretically possible due to availability of stretching water films, corrosion products in shape of slime on the cases surface.

3. RESIDUAL WATER IMPACT TO THE TUK-108/1 PROTECTION

Availability of residual water leads to generation of a number of physical processes:

- Corrosion of MCC constructive elements (sealing, internal room of MCC, remote grid), cases and ampoules containing the SNF, fuel rods of spent nuclear fuel assemblies;
- Hydrogen and hydrogen peroxide generation in the residual water radiolysis process under the SNF gamma radiation impact inside internal space of MCC, in cases and ampoules;
- Creation of nitric acid inside internal room of MCC, in cases and ampoules in humid air conditions under the SNF gamma radiation impact.

There are the following types of corrosion of the cask constructive elements, cases, ampoules containing the SNF, fuel rods, due to availability of residual water:

- Homogeneous;
- Nodulose;
- Slotted;
- Pitting;
- Inter-crystal and corrosion cracking.

The same chemical and physical processes take place in all the types of casks intended for storage and transportation of SNF.

With the time being corrosion processes will take place in the cask. In connection with this water volume will gradually slow down and be spent for creation of slime and corrosion products. Taking into consideration conservative points of view the phenomenon can lead to corrosion overgrowing on the cask cover which can lead to some difficulties while its opening and this problem on its turn can also lead to difficulties while casks with spent nuclear fuel assemblies are being removed on having the container open.

Slime quantity takes approximately 0.6 in comparison with full volume of cask in relation to total corrosion and cannot be changed with the time being. Taking into account speed of oxide films growth and availability of gap in the unit of main joint which is of a few millimeters the overgrowing with corrosion is also possible. The film density can lead to difficulties with the protective cover opening and possible stucking of the cover. This situation can require use of the other tools and appliances aimed at the cask opening with a view of unloading the cases with SNF at a process plant. At the expense of interaction reaction between metal and hydrogen as radiolysis product creation of pitting corrosion of stainless steal is possible. Cases for spent nuclear fuel assemblies are made of stainless steal.

Analysis of maximum acceptable quantity of residual water existing in MCC with SNF while preparing for a protective long term storage taking into account level of corrosion allowed for the storage period shows that acceptable volume of residual water inside MCC should take for the TUK-108/1 containing SNF arriving from the Navy and ice-breaking ships approximately 13 g.

Argon is used as an inert gas for stuffing internal space of the TUK-108/1 containing SNF after its drying [5].

4. JUSTIFICATION OF SELECTION OF DRYING TECHNOLOGY

In actual practice two methods of drying of cask inner cavity are used:

- Ventilation;
- Evacuation.

Ventilation method involves arrangement of hot air circulation in the MCC (metal-and-concrete cask) inner cavity.

Among advantages of this technique as regards casks are the following:

- It does not matter in what temperature conditions is the loaded cask since this method does not require any long warming up of the cask's structure;
- Good heat insulation properties of walls of the metal-and-concrete cask will play a positive role allowing for reducing the amount of delivered heat in heating its inner cavity due to delivery of hot air.

Among disadvantages one has to point out:

- Necessity in a special technological lid for the cask;
- Potential release of activity from the tuk with continuous long air circulation;
- Complication of technology of activities related to rearrangement of lids including unfeasibility of combination with operations to fill the inner cavity with an inert gas.

Evacuation technique involves provision of a necessary vacuum in the inner cavity to ensure intensive boiling of residual water at sufficiently low temperatures (10–15°C). Among advantages of this technique as regards casks are the following:

- Simplicity of design;
- Minimum amount of additional technological tooling;
- Reliable control of residual water content.

While developing the technology of MCC management the selection of drying technology has been made with consideration for international experience such as the vacuum drying facility at Ignalina NPP (Lithuania) for drying of casks CASTOR and CONSTOR with RBMK-1500 SFA, at NPP Dukovany (Czech Republic) for casks CASTOR with VVER-440 SFA [7–8]. Taking into account evaluation of different technologies of drying of casks with SNF and international experience, JSC KBSM has adopted and developed the technology of MCC vacuum drying both with RBMK-1000 SFA for Leningrad and Kursk NPP and with naval and icebreaker SNF [11]. Schematic of the cask drying facility by evacuation is shown in Fig. 1.

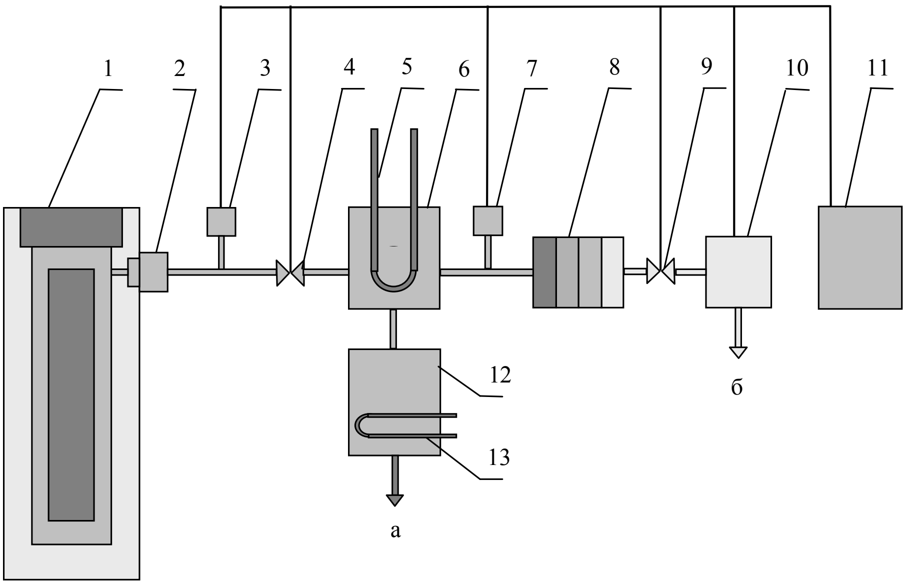

FIG.1. Drying arrangement. 1) cask; 2) connecting device; 3, 7) steam/gas environment pressure sensors; 4, 9) shut-off valves; 5, 13) cooling elements; 6) condenser; 8) filters for cleaning steam/gas environment from radioactive aerosols; 10) vacuum pump; 11) control panel; 12) condensate collecting tank; a) condensate discharge to special sewer system; б) gas discharge to special ventilation system.

A vacuum system of the drying facility is connected to the systems of special ventilation, special sewer, gas supply (inert gas) and power supply of the site. A need in having within the drying facility such elements as a condenser with the condensate collector and aerosol filters is defined on the basis of a potential amount of drained water and activity of discharged steam-and-water mixture in terms of radioactive aerosols.

The control of residual water volume is based on the value of steam and gas mixture in the cask inner cavity in the process of drying and with the switched-off pump.

2.1. Computational justification of drying process and selection of parameters of drying equipment

A drying process involves evacuation of the cask inner cavity and usage of heat accumulated by the cask mass as well as of residual heat release of SNF SFA for evaporation of water with discharge of formed vapour through the valve channel, the connecting device and pipelines to the vacuum pump and further to the special ventilation system. If necessary, prior to the vacuum pump a condenser is installed with a tank for condensate collection and aerosol filters.

At a low ambient temperature, it is essential to warm the cask before drying. It is achievable by putting the cask in a warm room or by heating the cask with the special heating equipment both at the stage of cask preparation for drying and at the stage of drying itself which allows for making the entire process independent of ambient temperature and ensuring cask drying in any external conditions.

It is also essential at a low temperature of the environment to exclude a potentiality of condensate accumulation and freezing in the pipes which can be attained either by heating steam and water mixture at the cask outlet (short pipes), or by warming up pipelines along the whole length using a special electric cable.

A general problem of cask drying includes consideration of the following stages:

- Determination of initial state of the cask, which has been formed under conditions of preliminary storage;
- Preparation of the cask for drying operation either in the heating facility or through sufficiently long weathering in a warm room;
- Performance of cask drying until a definite criterion of process completion is achieved.

As the possibility of measurements in the cask inner cavity is practically non-existent, a computational analysis of the above-mentioned stages is the only way to correlate the ambient conditions with the characteristics of the cask state before and during drying, to determine modal parameters of the process as well as to reveal influence of individual factors and get the possibility for justification of structure and characteristics of equipment.

In order to perform computational analysis JSC KBSM has developed a comprehensive mathematical model and routine for calculation of the cask drying procedure. Its significant feature is an intimate connection between non-steady heat and mass transfer while drying and non-steady temperature field of the cask. A thermal state of the cask experiences a continuous change in the drying process which has a direct impact upon its nature.

- A full mathematical model of drying includes the following main parts:
-
- A mathematical model of thermal state of the cask which describes in detail the heat exchange between all significant components and the environment including the path of external heating;
- A mathematical model of the inner cavity drying process, which contains description of dynamics of pressure, temperature and masses of water, water vapor and air in their interaction with elements of the cask and the drying facility;

- A mathematical model of the cask heating unit;
- A mathematical model of the condenser and the condensate collector;
- A mathematical model of pipe system with built-in different type heaters;
- A flow rate of the vacuum pump.

- The mathematical model and the program of comprehensive thermal calculation of the cask ("TRAK") has been developed specially for thermal analysis of transport casks and packaging sets with SNF both in normal operation and in emergencies, and has been widely used for evaluation of domestic and foreign casks [6]. The program envisages a possibility of detailed description of cask design geometry and processes of heat transfer by any combination of recognized mechanisms of heat transfer both within the computational area and on its boundaries.

The "TRAK" program has passed the stages of careful experimental and computational verification as regards the thermal state of different casks for transportation and storage of spent fuel of power nuclear reactors.

The "TRAK" program acts together with the developed unit of program modules "OSUSHKA". The last is actually implemented as a part of the "TRAK" program.

- In development of the drying model the following assumptions have been made:

- The parameters of the systems are concentrated in a definite number of limited ("control") volumes, i.e. Within each control volume the heat exchange parameters are thought to be identical;
- Overheating of saturated water in the cask cavity results in the practically instantaneous generation of a corresponding volume of steam;
- A drop fraction of liquid phase in the cask atmosphere is not taken into account. Although volumetric and surface forms of condensation are considered, it is supposed that condensate after formation becomes a part of liquid phase without any delay.

- As control volumes, the following volumes are considered:

- A layer of water in the cask bottom;

- A steam and air volume in the cask inner cavity;
- A channel of valve of cask drying;
- A built-in heater of steam and water mixture at the outlet of the cask;
- A condenser;
- A condensate collector;
- A distributed model of the pipe system.
-

- Each control volume is described by a system of equations of mass and energy conservation and by equations of state of water vapor and air. An intensity of steam generation in the bottom is determined on the basis of equations of conservation assuming that all heat supplied to saturated liquid is spent on steam generation. Heat-transfer coefficient with boiling has been calculated on a basis of recommendations [7], verified at least to the pressure of 10 kPa.
-

- Heat exchange of steam and air volume of the cavity with the cask walls and devices inside the body is described in distributed parameters with consideration of possible vapor condensation.
-

- It is worth noting that the process of water boiling at low pressures (below 20 kPa) has its peculiarities which were many times described in literature [8, 9] — considerable water overheating occurs, action of steam generation centers is extremely irregular, heat-transfer coefficient with pressure drop is reduced noticeably. One has to take into account a potential breach of the continuous evaporation surface and associated reduction of the area of heat supply from the cask bottom to the boiling water layer. At the same time, there are evidences that in thin liquid layers (below 5 mm) heat-transfer coefficient reveals a tendency for growing [10] (approximately by 20-30%). Intensity and character of boiling depend also on the material of the cask bottom and nature of heating. Elimination of uncertainties related to the listed factors is possible by studying in the processes of drying of real objects.
-

- A distributed pipeline model is constructed on the bases of solution of equation of heat- and mass transfer of steam/air mixture along the channel with consideration of heat exchange of pipe walls with the flow of mixture and the environment.
-

- A very important element of the drying facility is a set of equipment for cask heating aimed at provision of required temperature in the cask inner cavity and elimination of vapor condensation on the inner surface of the cask cavity and in the channel of drying valve. In the course of development two options of heating unit have been investigated which ensure a sufficiently intensive and uniform warming-up that excludes both formation of hot spots on the cask surface and overheating of its inner area.
-
- In the first option the heating unit involves a special metal casing on which inner surface the screens of electric heaters are located. The casing is installed on MCC during warming-up.
-
- In the second option the heating unit involves a closed air loop for long-term blowing-off of the entire outer surface of the cask by a hot air flow, the circulation and heating of which are provided by an electric heating element (radiator). A set of warming-up equipment consists of the mass-produced radiator, a cloth envelope which is installed on the cask with a gap defined by the cask design, the pipeline, a closing loop of circulation of air heat-transfer agent.
-
- The thermal calculation of heating loop reduces to the solution within the framework of the “TRAK” program of the equation of transfer of warming air medium on the calculation grid of the cask and along its blown-off surface. Heat exchange with air flow specifies boundary conditions in each calculation point on the cask outer surface.
-
- The software complex which implements this described mathematical model may be applied for calculation of drying process and selection of parameters of equipment of different types of cask drying facility. The software complex has been, particularly, used as a computational tool in development of the cask drying facility for spent fuel of icebreaker fleet, which has passed cold and hot tests at FSUE RTP Atomflot.
-
- As a consequence of calculation of dynamics of drying of TUK-108/1 at FSUE DVZ Zvezda the selection of equipment has been substantiated. As a warming-up unit, the second option has been chosen which involves air loop for long-term blowing off. As a heating unit, a radiator EKOTs-16 has been selected which is characterized by:
-

- Capacity of fan electric motor is 0.55 kw;
- Capacity of heaters in three sections is 15.5 kw;
- Air flow rate is 1900 m^3/hr;
- Maximum air temperature at the outlet is 50°C;
- Maximum air warming-up is 35°C.
-
- A permissible resistance of air path should not exceed 400 Pa requiring a careful optimization of the air circulation channel.
-
- The plant includes a vacuum oil-free spiral pump XDS35i with a rated discharge 35 m^3/hr.
-
- As a maximum amount of water in the cask prior to commencement of drying does not exceed 3–4 liters, while SFAs are put in sealed canisters, the equipment does not include a condenser with a condensate collector and aerosol filters resulting in significant simplification of control of drying process.
-
- A total length of pipelines which connect the cask with the vacuum pump is no more than 2.5 m, thus to prevent freezing of condensate in pipes at a low temperature in the drying workroom (however, not lower than 5°C), at the outlet of the cask it is envisaged to pass the pumped off steam and air mixture through the screw channel with a built-in heater with regulated setting of temperature and capacity within the range of 50–250 W.
-
- A total amount of residual heat release of 49 SFA in the cask was taken as 59 W.
-

As a basis for further computations related to warming-up and drying of the cask we consider a steady thermal state of the cask which is established in an unheated storage facility where a temperature may be within the range of minus 30°C to plus 40°C.

- Calculation of initial states covers a part of this range from minus 30°C to plus 15°C. A temperature 15°C is determined on a basis of computational assessment as a minimum mean temperature during at least 10 last days of storage, when a preliminary cask warming-up before drying is not a must. In this case, a temperature in the cask bottom (not less than 15°C) and stock of heat accumulated by the cask mass are enough for formation and support of the process of water boiling in the bottom during drying. At a mean storage temperature below 15°C to do drying one has to perform the preliminary cask warming-up.

-

- The calculations performed have allowed for building a generalized assessment of duration of cask warming-up in the heating unit depending on the temporary storage temperature (ambient air temperatures) before drying (Fig. 2).

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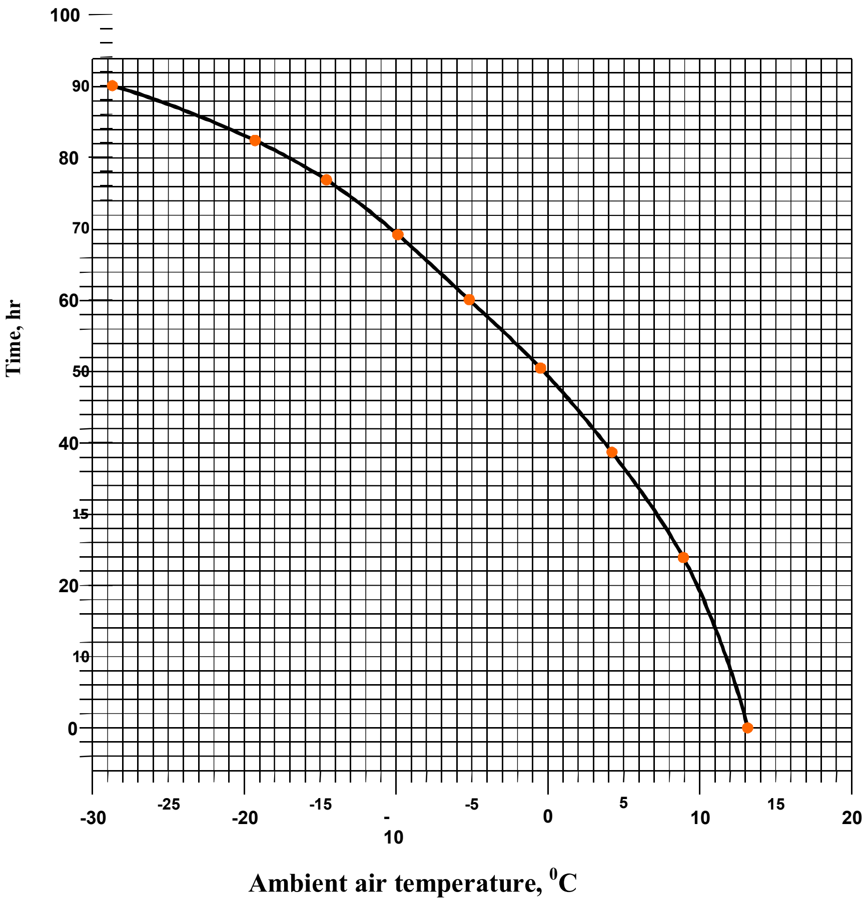

FIG. 2. Duration of cask preliminary warming up in the heating unit before drying as function of air temperature during preliminary storage.

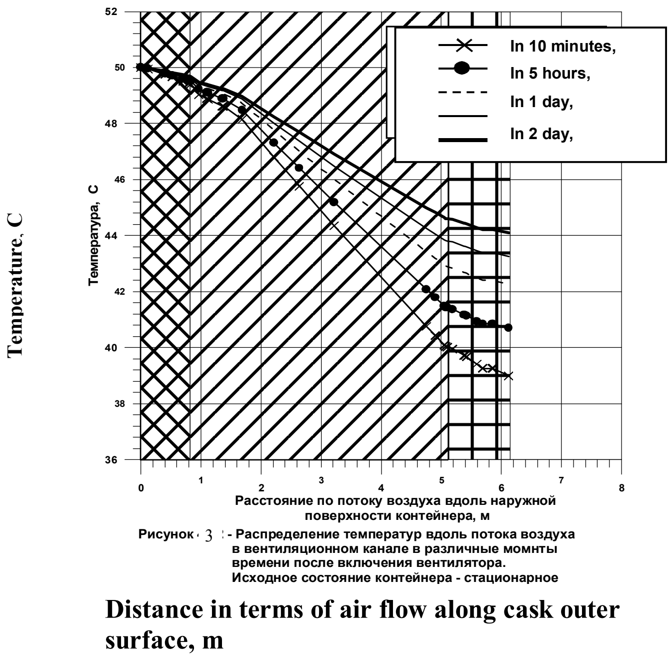

FIG. 3. Temperature distribution along air flow in ventilation channel in different points in time after fan actuation. Initial cask state — steady at -20°C.

A plot of this type may be used for assessment of required time of warming-up of any cask prior to drying in a cold season of the year. In case when a warming-up init is unavailable, the duration of the cask weathering prior to drying in a room with a mean temperature 15°C exceeds that shown in Fig. 2 approximately by a factor of 2.5.

Fig. 3 illustrates the distribution of the temperature in the flow of heating air of the warming-up unit along all heated surfaces of the cask within 10 minutes, 5 hours, 1 day, 2 days and 3.5 days after commencement of warming-up at a temperature of the cask storage of minus 20°C. As it seen from the figure, even in the very beginning of warming-up of the cold cask the required heating of air in the radiator is about 12.5°C, that corresponds to no more than 36% of rated capacity of heaters (that is, all together no more than 6.1 kW). By the end of warming-up this share is reduced to 17% (in other words, all in all no more than 3.2 kW is needed). Insignificant power expenditures make practicable to perform drying process without shut-down of the warming-up unit, as doing this the duration of drying is noticeably decreased while the stability of the process is increased.

Figs 4–8 show the dynamics of the drying mode of the cask which has been stored at temperature minus 20°C and subjected to the stage of forced warming-up for 82.5 hours.

Initial state prior to drying: temperature of the bottom of the cask cavity is slightly higher than 15°C, gas environment is air under pressure of pressure ~ 0.1 MPa.

With turning the vacuum pump on the pressure in the cask in ~ 27 minutes is exponentially decreased to the pressure ~ 1600 Pa, corresponding to the saturation temperature ~ 14°C, when water in the bottom begins to boil (Fig. 4, initial stage of the process — Fig. 5).

Pressure of steam and air mixture is stabilized, air pressure keeps on dropping. Pressure in the pipe system is stabilized at a level ~ 400 Pa. A stage of stable process sets in when steam pressure raises very slow, basically under influence of a total warming-up of the cask.

From Fig. 6 it follows that a temperature of central part of bottom where boiling is more intensive due to greater overheating within the first 10 hours decreases by 3.5°C, despite the proceeding heating of the cask, and afterwards slowly increases. Due to the reduced intensity of boiling the rate of growth is increased by the end of the process.

Fig. 7 shows the dynamics of the temperature of boiling water in the bottom, the temperature of steam and air volume and the saturation temperature. As it may be seen, steam coming to the steam and gas mixture is somehow overheated which hampers the growth of its temperature. When a thickness of water layer becomes less than 1 mm that corresponds to about 0.4 kg of water, a total surface of evaporation is split into individual areas, a rate of evaporation decreases resulting in reduction of pressure.

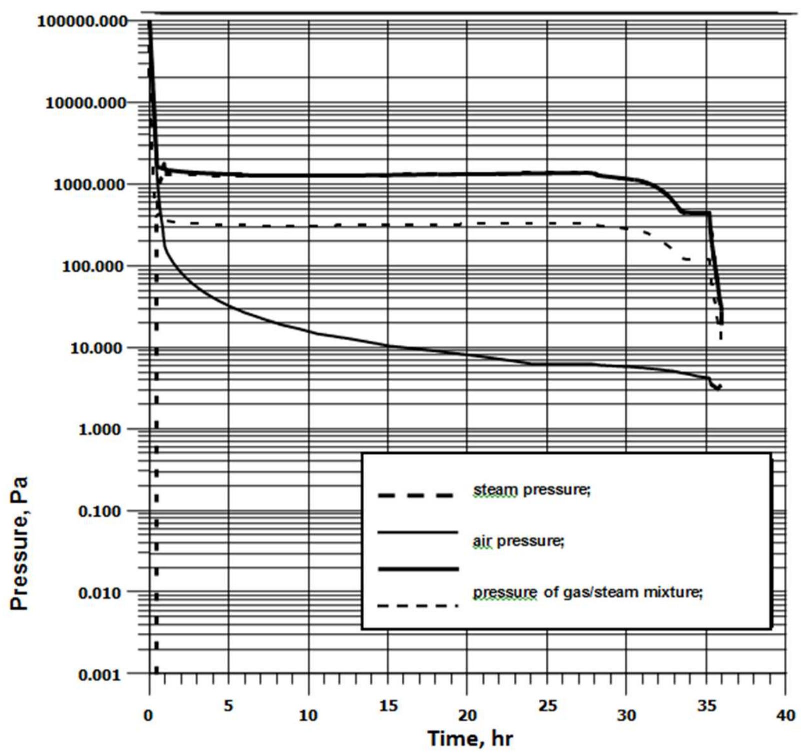

FIG. 4. Change in partial pressure of steam and air and pressure of gas/steam mixture in cask TUK-108 and pipeline in the process of drying with heating.

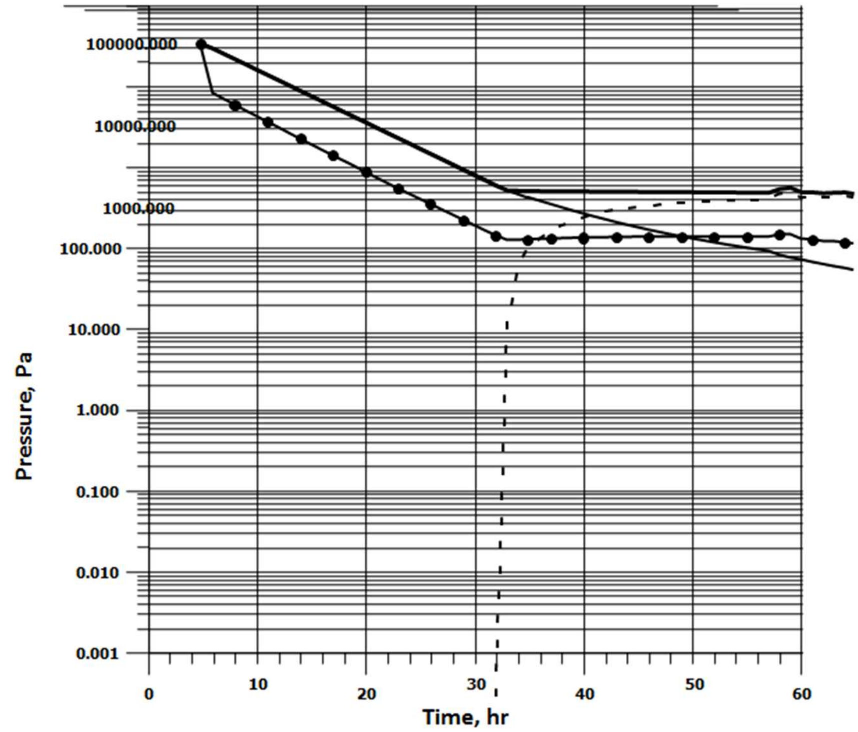

FIG. 5. Change in partial pressure of team and air and pressure of gas/steam mixture in the initial period of drying with heating.

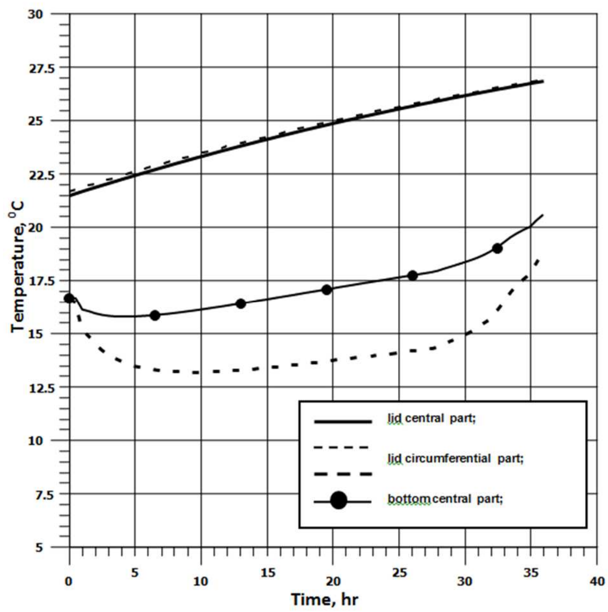

FIG. 6. Change of temperatures of inner surfaces of bottomand lid of cask TUK-108 in the process of drying with heating.

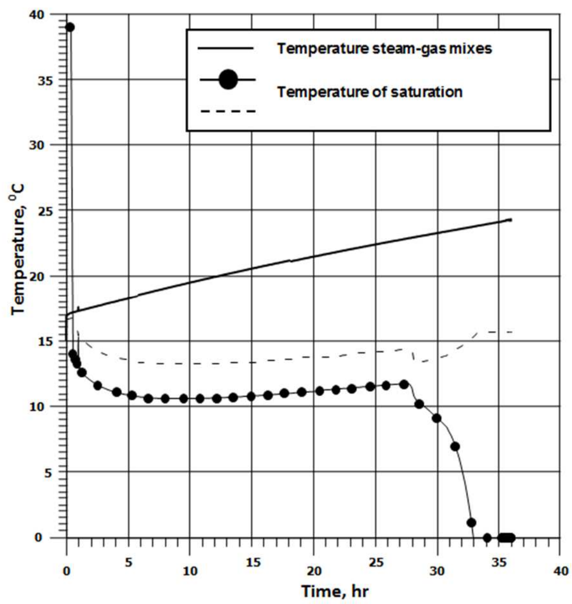

FIG. 7. Change of temperature of gas/steam mixture, saturation temperature and temperature of water in cask bottom in the process of drying with heating.

A change of mass of water in the bottom is presented in Fig. 8. Once all water is evaporated, the pressure of mixture (at this stage practically pure vapor) having achieved a level of 440 Pa, drastically drops again, within several minutes lowering to the level of less than 100 Pa. This instant can be thought to be the moment of completion of the basic drying process. For the option under consideration the duration of drying is 35.5–36 hours.

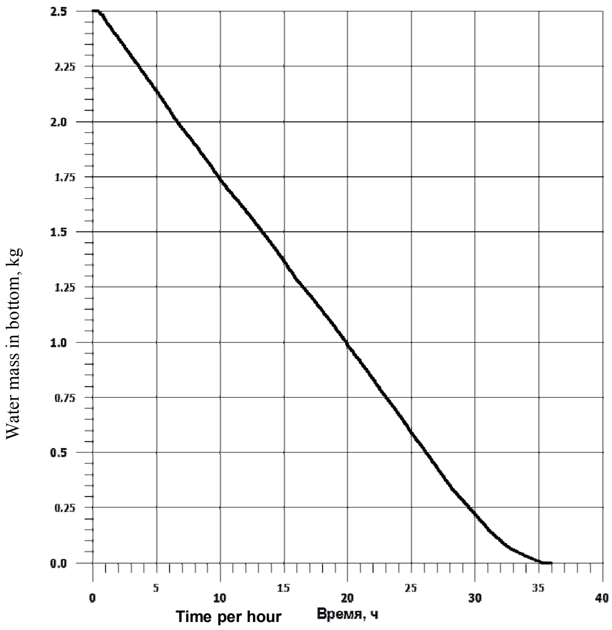

FIG. 8. Change of water mass in cask bottom in the process of driying with heating.

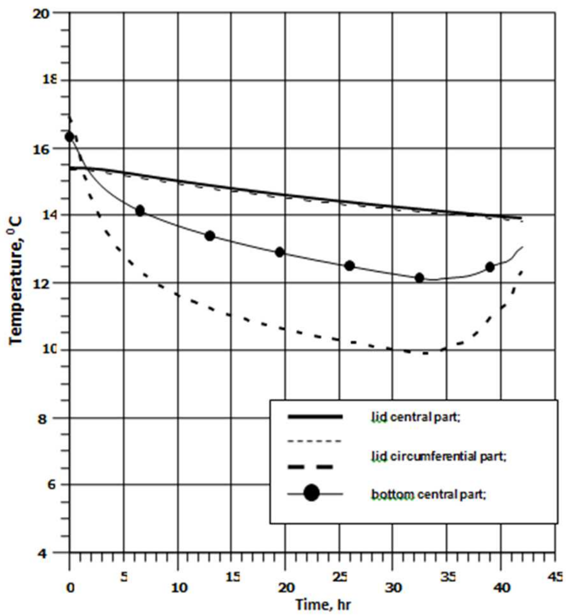

FIG. 9. *Change of temperatures of inner surfaces of bottom and lid of cask TUK-108 in the process of drying without heating. Drying starting in steady state at 15°C without preliminary warming up.*

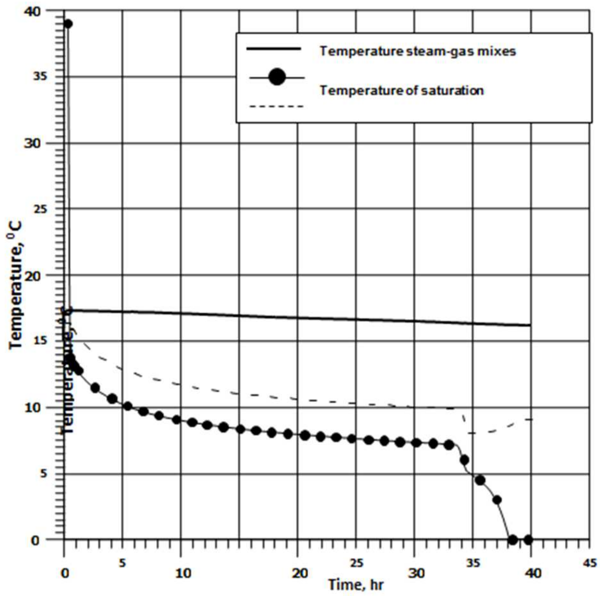

FIG. 10. *Change of temperature of gas/steam mixture, saturation temperature and temperature of water in cask bottom in the process of drying without heating.*

As drying has been initiated in the cask the condition of which is far from equilibrium, this process proceeds with a rather strong dependency on the rising temperature of the cask.

Figures 9–10 show the dynamics of parameters in the process of drying of the cask the thermal condition of which does not require any preliminary warming up. It has been established that the minimum ambient temperature at the stage of preliminary storage in this case is no less than 15°C. The drying process of such cask proceeds at lower temperature and pressure than in previous cases, while the duration of drying increases. Temperature of the bottom drops in the drying process by more than 7°C below 10°C. As this takes place, temperature of steam and gas mixture and surfaces of the whole inner cavity including lids is reduced, which in the practice can result in appearance of condensation sources. Drying process becomes less stable while the duration of drying increases.

As it follows from the presented plots of change of parameters in drying, the main part of drying process occurs under pressure of the mixture which is stabilized within the range specified mostly by the temperature of the cask (in the presented case within 600–1800 Pa). Depending on expendable nature of the valve of the cask and the drying system this level of pressure is constantly supported over the course of sometime (in this case about 30 hours).

By the moment of full evaporation of water in the bottom of the cask, vapor pressure begins to reduce more and more quickly, and after full boiling out it relatively fast (in this case in 20–30 minutes) drops to the level of 100 Pa and probably even lower.

Therefore, a reliable criterion of completion of the main drying process can be a sufficiently great, up to 100–200 Pa, reduction of pressure behind the cask valve.

Once the vacuum pump is shut down the pressure in the cask inner cavity, as a rule, again starts growing due to impact of latent sources of water vapor (apertures, microcracks of different origin, etc.) and accompanying gases as well as a result of non-absolute tightness of the system. A rate of pressure increase due to incomplete tightness (C_l, Pa/hr) to a first approximation can be considered as constant one while the intensity of vapor and gas sources should drop. For this reason repeatable switching vacuum pump on/off and measurements of the rate of pressure build-up in the cask inner cavity should give decreasing results. Thus, assuming that partial pressure of water vapor (and hence its concentration) within a certain control time $\Delta\tau$ should not exceed a maximum allowable value $p_{\lim}$, we have as a criterion of completion of dryin

$$(p_{изм})_n + \frac{\delta p}{\delta\tau} \cdot \Delta\tau \le p_{\lim} + C_l \cdot \Delta\tau - \Delta p_s ,$$

where

$$\delta p = (p_{изм})_n - (p_{изм})_{n-1} ,$$

$(p_{изм})_n$ — pressure measured in point of time τ_n,

$$\delta\tau = \tau_n - \tau_{n-1} ,$$

Δp_s - chosen value of pressure safety margin

According to the above data of residual maximum allowable amount of water vapor in the cask, a value of vapor pressure $p_{\lim}$ is determined to be equal to ~ 1700 Pa for casks with naval and icebreaker SNF, and ~ 2500 Pa for casks with RBMK-1000 SNF.

The developed methodology, a mathematical model and a program of calculation of the process of drying by evacuation of the cask inner cavity by the equipment which provides also preliminary warming-up of the cask and formation of parameters of the outgoing steam and air flow has made possible to consider the problem in combination with calculation of temperature fields in the cask and to define properties of drying process with a high probability as well as choose the equipment parameters.

3. TUK-108/1 DRYING FACILITY

Based on the aforesaid justification, the design documentation has been developed for the mobile TUK-108/1 drying facility. In December 2007 the test facility was manufactured, subjected to factory tests and later delivered to FSUE DVZ Zvezda. At CUC (coastal unloading complex) of the plant a set of activities was performed aimed at connection of the facility to a power supply, ventilation and waste collection systems, a station for storing of argon cylinders was constructed as well as a line for delivery to the cask for filling its inner cavity. At one of the areas of SNF loading in TUK-108/1 at CUC (all together two areas) the facility was assembled with a warming-up unit. Preliminary tests of working capacity of the TUK-108/1 drying facility (Fig. 11) were carried out.

In March 2007, the State tests of the UK-108/1 drying facility were performed. Testing was carried out with a TUK-108/1 loaded with SNF and stored at the pad for 2 years. Test results of the TUK-108/1 drying facility confirmed all above mentioned principles and technical solutions, computational procedures and programs utilized and newly developed in the course of implementation of AMEC 1.1-2. A mobile TUK-108/1 drying unit which has been commissioned at CUC of FSUE DVZ Zvezda ensures a chance of using it in both areas of SNF loading into TUK-108/1. The decision on extension of the period of SNF storing in TUK-108/1 up to 5 years has been drawn up.

FIG. 11. TUK-108 drying facility with heating area.

4. CONCLUSION

As a result of putting into operation of the mobile TUK-108/1 drying facility the following problems have been resolved:

- The SNF storage period in TUK-108/1 has been extended to 5 years;
- The safety of storage and subsequent transportation of SNF to PO Mayak has been improved through disposal of residual water and filling the TUK-108/1 inner cavity with an inert gas;
- The flexibility of managing the transport and technology scheme of management of SNF from decommissioned NPS has been ensured, for example, in the case of temporary lack of possibility of shipment of the casks from storage pads in such situation as with CUC of FSUE DVZ Zvezda. Thus, CUC of FSUE Zvezdochka may cease to operate due to the similar problem. The storage pad is located in the Yagri Island which is separated from the railroad by the bridge which needs major repairs or new construction;

- Smooth process of safe SNF unloading from the dismantled NPS in the Far East has been provided in the situation when for an indefinite time it cannot be transported for reprocessing.

The construction and principles of work of the drying facility have been used when creating the mobile facility concerning removal of residual water from UKhK-109 with SNF. The drying facilities have been manufactured, subjected to factory tests and later delivered to newly manufactured dry cask storage facilities in Leningrad and Kursk NPPs. During 2010 starting-up and adjustment and acceptance tests concerning putting into operation cask storage facilities should be completed. UKhK-109 Drying Facility will be used for its preparation for long-term storing.

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SPENT FUEL MANAGEMENT PROGRAM IN THE 3MW TRIGA MARK-II RESEARCH REACTOR OF BANGLADESH

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Abstract

Bangladesh Atomic Energy Commission (BAEC) has been operating a 3 MW TRIGA MARK II research reactor since 1986. The reactor was installed in the campus of the Atomic Energy Research Establishment (AERE) at Savar, Dhaka. It is one of the main nuclear research facilities in the country. The reactor uses TRIGA LEU fuel with uranium content of 20% by weight. The enrichment level of the fuel is 19.7%. The reactor has so far been operated for 7834 hours with a total cumulative burn up of 15898 MWh (662.5 MWd). The total burn up life of the present core is 1200 MWd. The main areas of use are: training of man-power for nuclear power plant applications, radioisotope (RI) production, neutron activation analysis (NAA), neutron radiography (NR) and neutron scattering. The government of Bangladesh has taken decision to establish nuclear power programme in the country. There is an ADP (Annual Development Project) to accomplish necessary activities for construction of medium size nuclear power plant (NPP) in the western zone of the country. Now, with regard to the safe management, storage of spent fuel and disposal of radioactive waste arising from operation of the research reactor and also from the proposed NPP expected to be constructed in future, BAEC is drawing up short and long-term plans and programs. At present, there does not exist any spent fuel element in the reactor facility. It is to be mentioned that Bangladesh is aware of the US DOE's 'Take Back Program' in connection with the research reactor spent fuel of US origin, and is very much interested to take part in this program. The paper presents the current status of handling and storage facilities available for spent fuel and strategy for the safe management of spent fuel to be generated from the research reactor in near future.

1. INTRODUCTION

- The TRIGA Mark-II research reactor of BAEC is a light water cooled, graphite reflected reactor designed for steady-state and square wave operation up to a power level of 3 MW (thermal) and for pulsing operation with a maximum pulse power of 852 MW [1]. The reactor was made critical at 50W for the first time on 14 September 1986 and was commissioned to steady state power of 3 MW in October 1986. Since then, it has been used for manpower training, radioisotope production and various R&D activities in the field of NAA, NR and neutron scattering. During the period, operation of the reactor was interrupted several times due to different incidents encountered in the cooling system, Dry Central Thimble (DCT) and at the beam ports of the reactor. The most severe of these incidents was the "N-16 Decay Tank Leakage Incident" that took place in 1997 due to pitting corrosion on the bottom of the tank. It is to be mentioned that the corroded bottom portion of the tank was in direct contact with the concrete saddle. As a result of this incident, full power operation of the reactor remained suspended for several years. During that time, the reactor was, however, operated at 250 kW under natural convection cooling mode, so as to cater the needs of the reactor users (e.g., NAA and NR groups) who require lower neutron flux. Operation of the reactor at lower power level was made possible by establishing a temporary by-pass connection across the decay tank using local technology. To take the reactor back to normal operation, BAEC implemented a government funded ADP with a total project cost of about 0.9 million US dollar. Under the project, renovation and upgrading of the entire cooling system of the reactor were carried out. The renovated cooling system was successfully commissioned in June 2002 and through this, it was possible to restore the full power operation of the reactor after a long period of about five years. Since July 2004, the reactor was being used for production of I-131 on routine basis. Such reactor operations were continued until the end of October 2008, when there occurred an incident at the DCT (Dry Central Thimble) with the irradiated Al-can containing a quartz vial filled with TeO2 powder. What actually happened is that an aluminum (Al) sample container (*can*) containing a quartz vial filled with about 40g of TeO_2 powder was found to get stuck to the bottom the DCT while it was irradiated in the DCT during the last week of October 2008. When ROMU personnel were unloading the Al-can after a cooling period of two days, the top cap of the can got detached and came out with the specimen lifting tool leaving the body of the can containing the irradiated TeO_2 inside the DCT. The sample was highly radioactive and such needed quite long cooling time before it could be taken out of the DCT. ROMU had to design and fabricate a special lifting tool which was used successfully to take the *can* out on 13 January 2009. The DCT was then decontaminated with the assistance of health physics group of Institute of Nuclear Science and Technology (INST).

After that, it became necessary to remove the plugs (polyethylene, lead, graphite, etc.) from inside the Radial Beam Port #1 (RBP-1) so as to make it ready for installation of the High Resolution Powder Diffractometer (HRPD), which has been procured under a different ADP project of BAEC. While doing the removal work, the 101.6cm (40in) long inner graphite plug of the beam port got broken. The broken part of the graphite plug (33cm long), which was found to get swelled, was tightly sticking to the inner wall of the RBP-1. ROMU along with the relevant divisions of INST took measures to take the plug out. As the radiation dose rate in front of the RBP-1 was quite high (greater than the permissible dose rate by a factor of more than 100), reactor operations were suspended so as to allow the dose rate to come down to a lower level such that maintenance personnel could work safely to clear the RBP-1. After removal of the inner 13 inches long graphite plug, the RBP-1 was found leaking at a rate of about 500 ml/day. Detail investigations showed that the aluminum part of the RBP-1 located within the reactor tank had developed corrosion leakage on its bottom part. With the approval of regulatory authority, a split type encirclement clamp (STEC) was designed, fabricated locally and then installed around the RBP-1 on 26 February 2010 to stop the leakage of the water. Due to the DCT incident and the beam port leakage problem there were no reactor operations during the period November 2008 to February 2010. Reactor operations were started again from 01 March 2010. It is to be noted that the HRPD was installed in RBP-2 instead of RBP-1. BAEC is now working to replace the old analog control console of the reactor by a new digital one under an ADP project entirely funded by the government of Bangladesh.

2. FEATURES OF THE BAEC TRIGA REACTOR

- The TRIGA Mark-II research reactor of BAEC is a light water cooled, graphite reflected reactor, designed for steady-state and square wave operation up to a power level of 3 MW (thermal) and for pulsing operation with a maximum pulse power of 852 MW [2]. The reactor core is located near the bottom of the reactor tank. The reactor tank is made of 6061-T6 aluminum alloy and has a length of about 8.23 m (27 ft) and a diameter of about 1.98m (6.5ft). It is filled up with about 24 865 liters (6 578 gallons) of demineralized water. The reactor core consists of a total of 100 fuel elements (including 5 fuel follower control rods and 2 instrumented fuel elements), 6 control rods, 18 graphite dummy elements, 1 Dry Central Thimble (DCT), 1 pneumatic transfer system irradiation terminus and 1 Am-Be neutron source (strength: 3Ci). The general characteristics of the reactor are summarized in Table 1.

- TABLE 1. MAIN DESIGN CHARACTERISTICS OF THE BAEC TRIGA REACTOR

Characteristics	Parameters
General:	
Reactor Type	Pool Type
Thermal Power	3000 kW
First Criticality	14 September, 1986
Neutron Flux	7.46×10^{13} $n\cdot cm^{-2}\cdot s^{-1}$ (Max.)
Fuel:	
Contents of uranium	20%
Enrichment	19.7%
Cladding	SS 304
Chemical Composition	Er-U-$ZrH_{1.6}$
Moderator	$ZrH_{1.6}$ and Demineralized Water
Coolant	Demineralized Water
Reflector	Graphite
Control Rod Material	B_4C

3. DESCRIPTION OF FUEL ELEMENT

- The fuel element of the BAEC TRIGA reactor is a homogeneous mixture of Er-U-$ZrH_{1.6}$, containing about 20% by weight of uranium enriched to about 19.7% U-235 and about 0.47% by weight of Erbium (burnable poison). The hydrogen-to-zirconium atom ratio of the fuel-moderator material is about 1.6–1. The active section of the fuel-moderator element is 38.1 cm (15 in.) long and 3.63 cm (1.43 in.) in diameter. The active fuel section together with the top and the bottom graphite reflector pieces are contained in a 0.5 mm (0.02 in.) thick stainless steel cladding. The cladding is welded to the top and bottom end fittings. The top end fitting is grooved and specially shaped to fit and lock into the fuel-handling tool. The overall weight of the fuel element is about 3.64 kg (8 lbs) [2]. The U-235 content is about 100 gm (0.05 lb). Details of the TRIGA fuel are shown in Figure 1.

4. PRESENT STATUS OF THE FUEL BURNUP

- The annual burnup for the BAEC research reactor fuel is shown in Fig. 2. The reactor has been operated for 7834 hours with a total cumulative burnup (BU) of 15898 MWh (662.5 MWd) since commissioning. The present fuel loading of the BAEC research reactor is good for an accumulated burnup of about 1375 MWDs. The Reactor Physics and Engineering Division (RPED) of the Institute of Nuclear Science and Technology (INST) is performing the burnup calculations. At present no spent fuel has been generated in the BAEC reactor facility. However, with the RI production and newly installed powder diffractometer utilization programs undertaken, it is expected that the reactor will start to generate spent fuels from the year 2015.

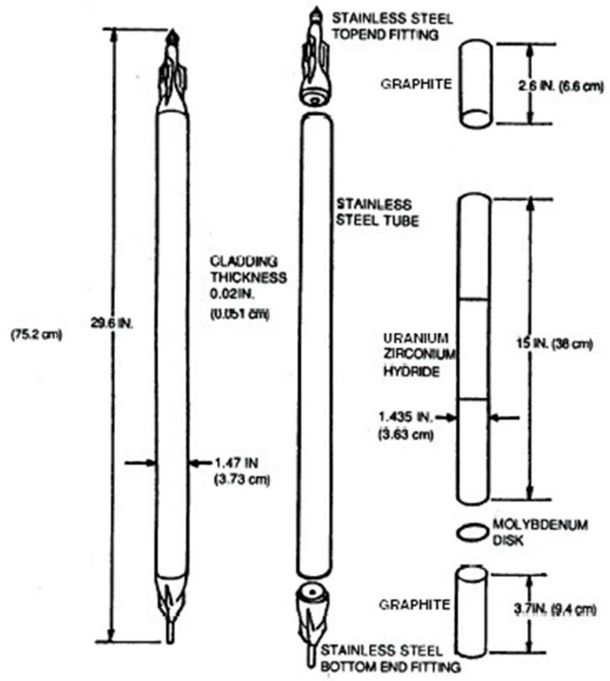

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FIG. 1. BAEC research reactor TRIGA fuel element.

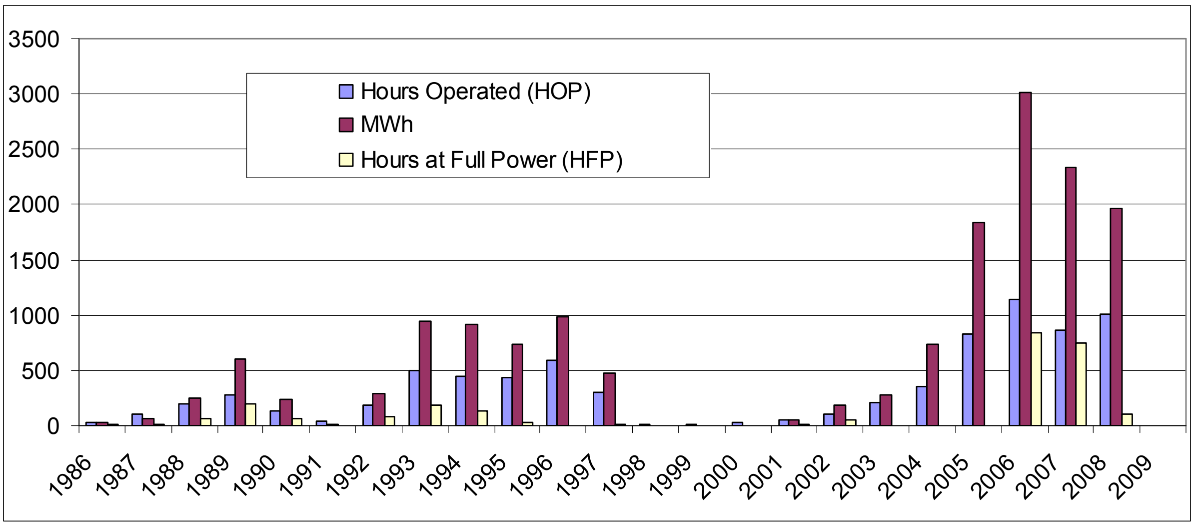

FIG. 2. Reactor operation data (Sept. 1986 – December 2009).

5. DESCRIPTION OF STORAGE FACILITY

5.1. Spent fuel storage facility

• The reactor facility is equipped with three spent fuel storage pits located at the ground floor of the reactor hall. The pits are made of stainless steel pipes of diameter 25.4 cm (10 inches) and of depth 457.2 cm (15 feet). Each one of the pits is provided with a lock on its stainless steel cover plate to limit access to the pit and also an M.S. cover plate (with lifting hook) that fits flush with the floor. Each storage pit is capable of storing 19 spent fuel elements (total capacity: 57). Fig. 3 shows the detailed drawing of the fuel storage pit. For storing the fuel elements into these pits, suitable storage racks are needed. At present the BAEC reactor facility does not have any rack of such kind. However, efforts have been undertaken to design and develop the storage racks compatible with the storage pits and also with the handling and lifting facilities available in the reactor hall. Besides these pits, there are three submerged fuel storage racks located along the inner wall of the reactor tank at a depth of about 610 cm (20 feet). The purpose of these racks is to provide temporary storage for the spent fuel and for the graphite dummy elements.

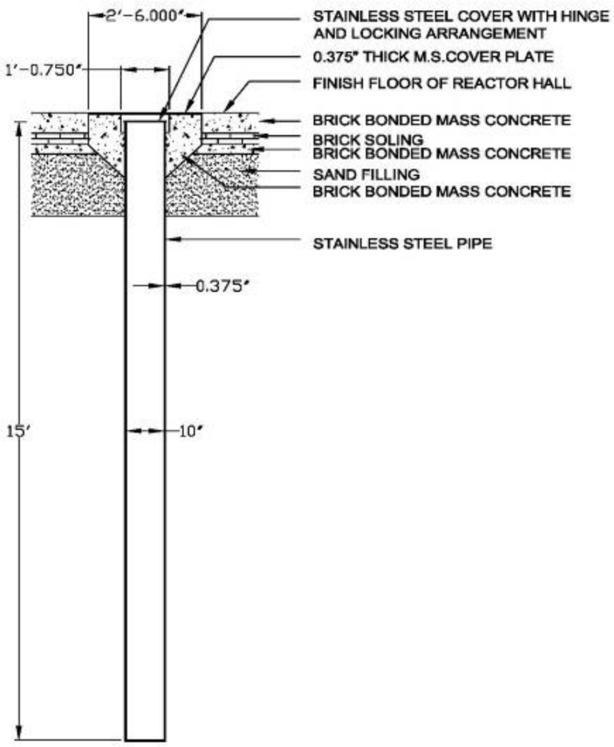

FIG. 3. Spent fuel storage pits of BAEC TRIGA LEU fuel.

- Each one of the racks is capable of holding 10 fuel elements (total capacity 3 × 10 = 30). Fig. 4 represents the photographic view of the storage rack.

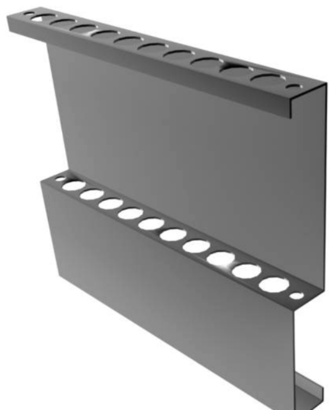

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FIG. 4. Fuel storage rack.

5.2. Central waste processing and storage facility (CWPSF)

• A Central Radioactive Waste Processing and Storage Facility (CWPSE) has been constructed in AERE campus located near the research reactor facility. The activities of this facility include: collection, handling, segregation, characterization, classification, treatment, conditioning, storage and disposal of all kinds of radioactive wastes generated in the country from nuclear installations, and also from application of radioactive materials in medicine, industry, research, agriculture, education, etc.

• The reactor facility is not susceptible to produce liquid waste in bulk quantity. However, any liquid waste produced in the facility will be processed and, if required, stored in the CWPSF. The layout plan of the CWPSF, which has a floor area of about 1 160 Sq. m (12 480 Sq. ft), is shown in Fig. 5. The CWPSF has a liquid waste treatment facility (Aqua-Express) designed for treatment and purification of Low and Intermediate Level Waste (LILW) at a rate of about 300 liters (79.4 gallon) of liquid radioactive wastes per hour using ion-exchange-cum-ultra-filtration technique.

• As there does not exist any comprehensive national program for radioactive waste management, the CWPSF will be used as the storage for all sorts of radioactive wastes produced in the reactor and other facilities. CWPSF is equipped with facilities such that solid wastes could be categorized, compressed for volume reduction, and immobilized in the steel drums each having a capacity of 200 liter (52.9 gallon) [IAEA Std. 200 liter steel drum having diameter and height of ~58 cm (22.8 in.) and ~88 cm (34.6 in.) respectively]. The filled up radioactive waste drums will be stored in appropriate storage room of the CWPSF for further decision to be taken in future (disposal). It is to be mentioned that the CWPSF has a provision for storing 112 IAEA standard 200-liter capacity radioactive waste storage steel drums. For conditioning and handling of the radioactive wastes, the facility (CWPSF) is equipped with the followings:

•

— 40 tonnes (40 000 kg) capacity low force compactor 1 no.

— In-drum cement mixture 1 no.

— Commercial cement mixture 1 no.

— 3 tonnes (3 000 kg) capacity forklift truck 1 no.

— 3 000-liter (794 gallon) capacity LAD (low activity drainage) tank 1 no.

— Sorting machine/box 1 no.

— Decontamination machine/box 1 no.

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• Other equipment/machine/apparatus available in the CWPSF include gamma, beta, alpha spectrometers, UV-visible spectrophotometers, compressive strength testing machine, pH and conductivity meters, weighing scales/balance, water bath, centrifuge, drying oven, sieve shaker, miscellaneous glass and plastic apparatus, etc.

• It is to be mentioned that all spent Radium-226 sources (35.6 GBq) collected from different hospitals and industries of the country have been conditioned and safely stored at the CWPSF using 200 liter (52.9 gallon) capacity mild steel drums. The drums are designed and developed in Seibersdorf, Austria and supplied by the IAEA [3], [4]. Used Co-60 sources of the gamma irradiator of the Institute of Food and Radiation Biology (IFRB) have also been stored in the CWPSF.

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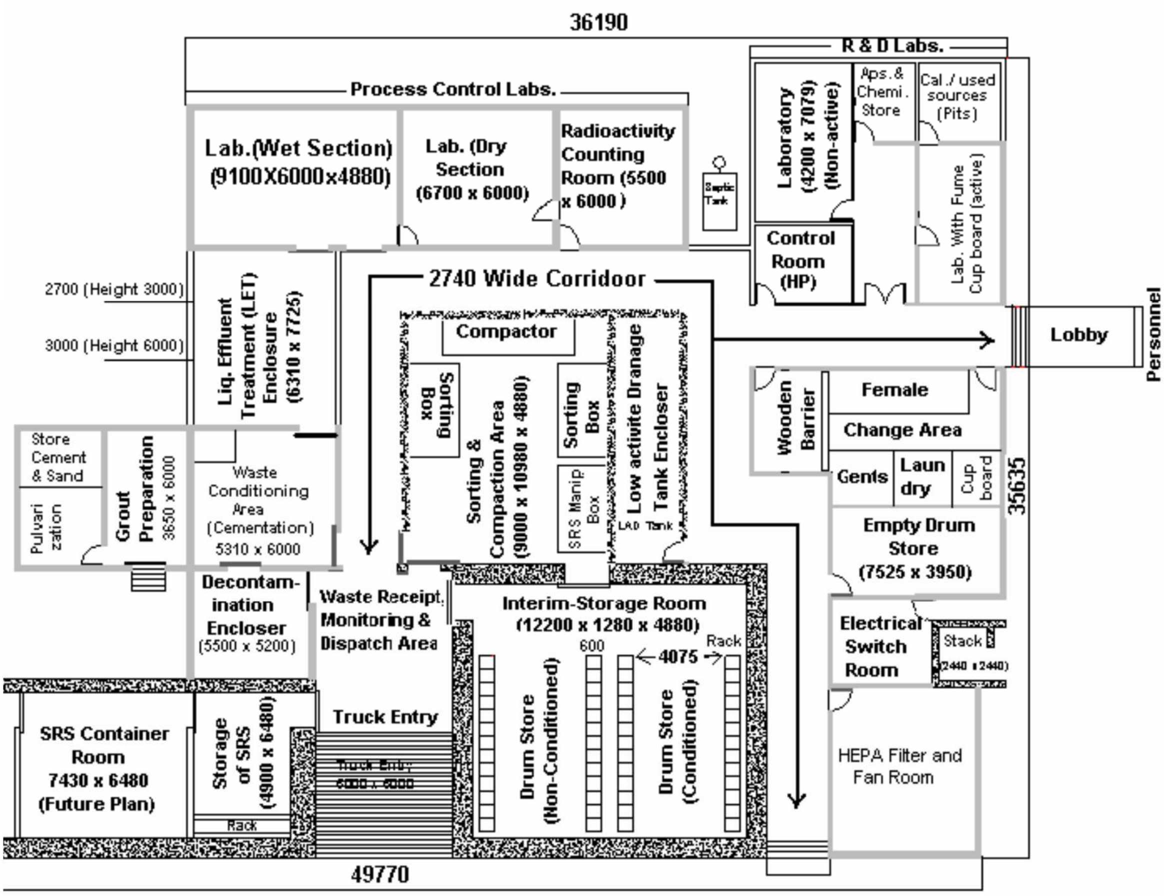

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FIG. 5. Central radioactive waste processing and storage facility (CWPSF).

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6. FUEL HANDLING INFRASTRUCTURES

6.1. Fuel handling tools

• Fuel handling tool is used to handle the fuel elements, graphite dummy elements and the fuel inspection tool calibration elements. Two types of fuel handling tools are available in the reactor facility. One is flexible type and the other is rigid type. Both of the fuel handling tools were supplied by General Atomics of USA, the reactor supplier.

6.2. Fuel transfer cask

• At present the BAEC reactor facility has no spent fuel transfer cask for transferring irradiated fuel elements from the reactor pool to the spent fuel storage pits. BAEC has taken measures to design and develop such cask for the reactor facility. Fig. 6 shows the proposed fuel transfer cask.

6.3. Crane

• An overhead crane of 5 tons (5 000 kg) capacity is used for transferring irradiated samples (TeO2 targets for I-131 production). An IAEA supplied lead transfer cask is used to transfer irradiated samples from the reactor top to the U-cask of the RI production laboratory. The weight of the lead transfer cask is about 1.5 (1 500 kg) tons. The crane can also be used for handling the proposed spent fuel transfer cask during transfer of spent fuel from reactor pool to the spent fuel storage pits.

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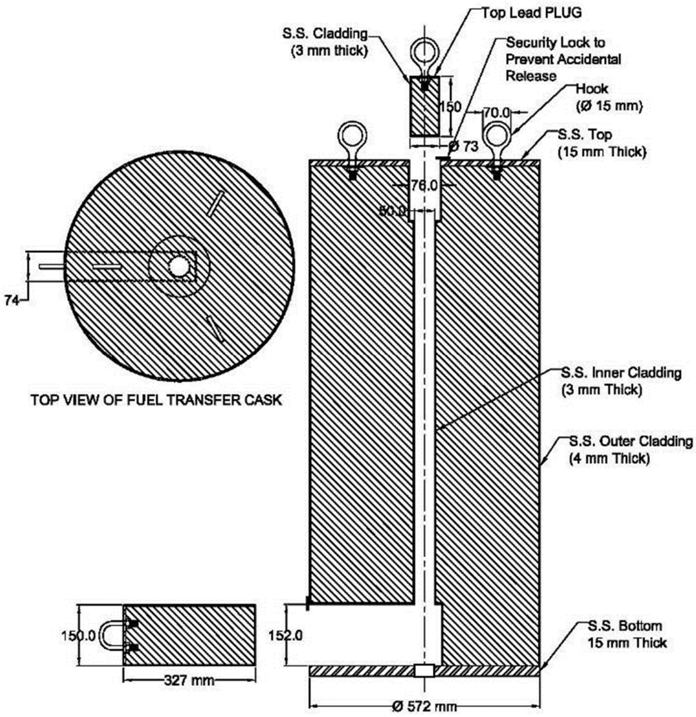

FIG. 6. Cross-sectional view of the proposed fuel transfer cask.

7. SPENT FUEL MANAGEMENT POLICY

• Bangladesh has strong commitment towards nuclear nonproliferation and as such, it has signed almost all multilateral and bilateral agreements, protocols, treaties, etc. prevailing in the International Nuclear Non-proliferation regime. Bangladesh signed a Nuclear Cooperation Agreement with the USA on 17 September 1981 (which was later extended up to 2012). This facilitated import of nuclear technology from the USA. Bangladesh would like for extend the tenure of the Agreement further when it will expire in 2012. It is to be mentioned that Bangladesh is aware of the US DOE's 'Take Back Program' in connection with the research reactor spent fuel of US origin. Bangladesh is very much interested to take part in this program. But as per the present arrangements of the DOE program, the spent fuel take back 'Window' would be closed in 2016. So, it would be very much helpful for Bangladesh if the said ‘Window’ could be extended beyond 2020 such that all the spent fuels generated in the research reactor of Bangladesh could be sent to the USA under the DOE’s program.

- With regard to the management of the spent fuel from the future nuclear power plant(s), the plan is that, the spent fuels will be stored in water pool storages at the reactor site for a period of about 20 years. The reason for water pool storage option is based on the fact that water is a convenient storage medium, inexpensive and available in desired quantities. Moreover, it can cool down by natural circulation; provide shielding from radiation and suitable for handling. After sufficient cooling in the water pool storage, the spent fuels will be shipped for reprocessing to regional/international reprocessing facility, if such facility is available at that time or to some country with reprocessing facility. Medium and high level wastes generated in the NPP will also be stored at reactor site for a considerable period of time and after that the wastes will be conditioned and then finally disposed off in the crystalline hard rock mine located in the northern part of the country. In this connection it is to be mentioned that the Bangladesh government has taken measures to rectify the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. It is expected that soon Bangladesh Government will be a party to the convention.

8. NATIONAL LEGISLATION AND REGULATORY INFRASTRUCTURE

- The authority for the control of all radiological and nuclear practices in Bangladesh is vested on the BAEC as the competent authority. The legal basis for this control are the ‘Nuclear Safety and Radiation Control (NSRC) act 1993 [5] and the Nuclear Safety and the Radiation Control Rules, 1997 [6] which essentially incorporate the requirements of the international Basic Safety Standards [7]. For addressing safety of radiation sources, protection of man and the environment and in compliance with the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management [8], the national NSRC Act (1993) and the NSRC Regulation (1997) are in operation; while the Radioactive Waste and Spent Fuel Management Rule-2005 is awaiting for the approval of the Ministry. BAEC is speeding-up the process of becoming a party to the Joint Convention.

9. CONCLUSION

The future nuclear power program and the associated SFM program in the Peoples Republic Bangladesh will be reviewed. An interim spent fuel storage facility will be set up to manage spent fuel safely and economically for many decades, thereby providing adequate time to develop an integrated SFM system. The current major issues are to secure a site for the facility and to optimize the storage concept. Considering the delicate public acceptance issues and the pursuit of universally accepted methods for safe SFM, close international co-operation is desirable to resolve the upcoming challenges. Efforts have been undertaken to design and develop spent fuel storage racks compatible with the existing 3 spent fuel storage pits located at the ground floor of the reactor hall. BAEC has also taken up measures to design and develop spent fuel transfer cask. BAEC expects that the US DOE’s ‘Take Back Program’ would of great use in determining the ultimate fate of the spent fuels generated from the operation of the BAEC research reactor.

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SAFETY ASPECTS AND ECONOMIC IMPACTS OF SPENT FUEL MANAGEMENT POLICIES IN PWR NUCLEAR FUEL CYCLE

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Abstract

With the decision to introduce nuclear power for electricity generation in Egypt, the assessment of different nuclear fuel cycle strategies is of great importance. In this context, safety and economic aspects of nuclear fuel cycle options are topics of global importance relevant to the development of nuclear technology. As a part of nuclear fuel cycle evaluation studies in the department of nuclear fuel cycle safety, NCNSRC-AEA, this paper evaluates the safety and economic aspects of PWR nuclear fuel cycle options. The once through or direct spent fuel disposal and the "self-generated recycle" fuel cycle concepts have been considered in this assessment. Effect of increasing reactor fuel irradiation level on nuclear fuel cycle requirements as well as its impact on the radioactive waste volumes arising have been estimated. The results showed a remarkable decrease in uranium requirements, while radioactive waste volumes increased. Fuel-reprocessing costs have been estimated as functions of the spent fuel disposal costs and the natural uranium prices to determine the justifiable fuel-reprocessing costs. Environmental safety aspects of the nuclear fuel cycle with the two options have been evaluated and discussed.

1. INTRODUCTION

1.1. General

Safety and economic aspects of nuclear fuel cycle (NFC) are topics of global importance relevant to public acceptance of nuclear energy and to the development of nuclear technology. In this context, nuclear waste, especially SFM, has become one of the crucial problems of nuclear energy utilization in electricity generation. Recently, the issue of SFM is of special interest; so there must be a safe and effective way to handle this problem. On the other hand, the choice of the nuclear fuel cycle option and SFM policy have a strong influence on sustainability of nuclear energy as well as on radioactive waste arising and associated environmental and safety issues. The once-through fuel cycle economics has major emphasis only on reactor fuel performance, whereas the nuclear fuel cycle with recycling option is based on an integrated fuel cycle sustainable system [1, 2]. For Egypt, as a developing country having limited natural energy resources, and with the governmental decision to use nuclear power for electricity generation, the choice of a reliable, safe and economic nuclear fuel cycle option and SFM policy is of great importance. For these reasons, and in the light of future expected shortage in natural energy resources and environmental impacts of radioactive wastes arising, the assessment of different nuclear fuel cycle strategies is an essential request. Currently, the PWR is fully proven, dominant, safe and reliable reactor and fuel cycle system and has become widely used in many countries. This study evaluates the safety aspects and economic impacts of PWR back-end nuclear fuel cycle options: the direct spent fuel disposal and the "self-generated recycle" fuel cycle concepts. Effect of increasing reactor fuel irradiation level in new reactor design systems on nuclear fuel cycle requirements and its impact on the radioactive waste arising have been estimated. Figure 1 shows a schematic diagram for PWR nuclear fuel cycle options.

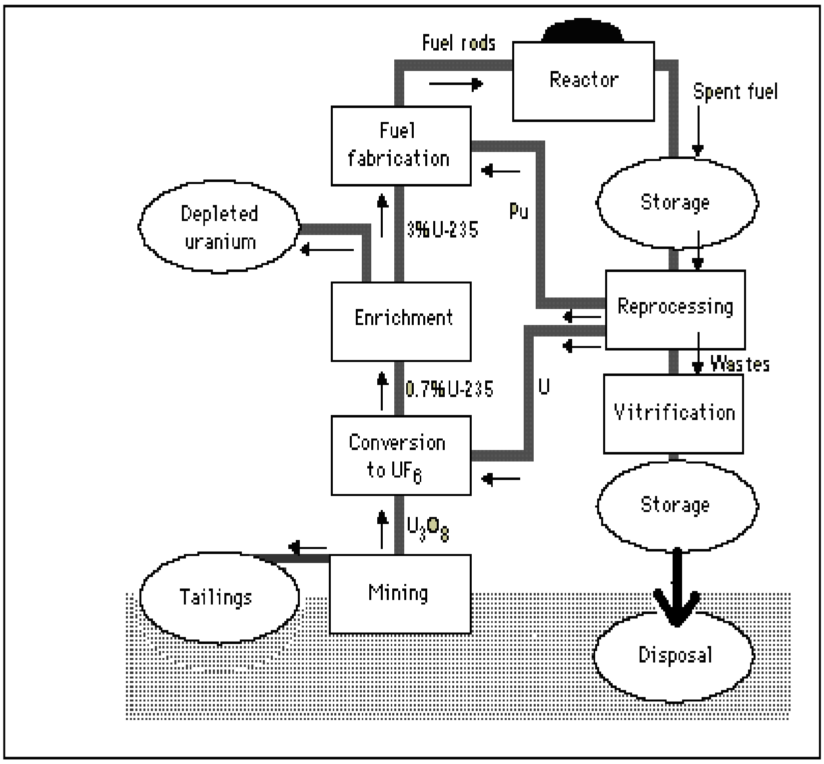

FIG. 1. PWR nuclear fuel cycle options schematic.

1.2. PWR-NFC spent fuel management policies

1.2.1. Once-through option (OT)

After removal from the reactor, the spent fuel is very hot and highly radioactive, so, it is normally stored in pools at the reactor site and then be transferred to an interim storage facility. The interim storage period is the time interval after the minimum cooling period following discharge from the reactor until reprocessing or spent fuel encapsulation prior to disposal. Interim storage of spent fuel could take place at the reactor site in cooling pools or in cask storage. In this case, storage costs are often an integral part of the power plant operating costs. Alternatively, it could take place at a separate interim storage site or in storage pools at the reprocessing site. In the latter case, interim storage costs are usually included in the reprocessing price. A number of different approaches have been developed for interim storage in which the fuel assemblies, either intact or dismantled to reduce the volume they take up, are stored in cooling pools either on the reactor site or at separate sites. Additionally, dry stores have been developed in which the fuel assemblies, with or without pre-treatment and special packaging, can be safely held in either an air or inert gas atmosphere. In most countries, interim storage is also required for the wastes during the period between conditioning and final disposal. Specially constructed facilities already exist for this purpose. Fuel assemblies may, after a period of cooling, which may be 30–50 years, be encapsulated in metallic canister directly or be disassembled using remote handling techniques so that the fuel pins can be packed together more closely prior to encapsulation.

1.2.2. Reprocessing or recycling option (RO)

The spent nuclear fuel contains about 95% U-238 and about 1% U-235 that has not fissioned, about 1% plutonium and 3% fission products, which are highly radioactive, with other transuranic elements formed in the reactor. In the reprocessing facility the used fuel is separated into its three components: uranium, plutonium and waste, containing fission products. Reprocessing enables recycling of the uranium and plutonium into fresh fuel, and produces a significantly reduced amount of waste compared with treating all the spent fuel as waste. During reprocessing, the main waste product is the high level waste solution containing most of fission products from the spent fuel. These wastes are vitrificated for final disposal. ILW/LLW is also generated at the reprocessing plant and these wastes are treated as solidified slurries in cement or asphalt, compacted waste, incinerated ash or packaged solid waste [3–4].

1.3. Calculation procedure and reference input data

Table 1 depicts the reference reactor and NFC input data used for the calculations. A computer program has been developed to estimate the material flow of the NFC with uranium and plutonium values or credits**.** The flow chart of the general structure of calculation procedure using ENFCE and ORIGEN2 computer programs [5] is shown in Fig. 2.The material balance flow requirements for PWR-NFC options are calculated by ENFCE computer code. The natural uranium requirement was calculated using the data for a reference nuclear plant material balance, with 4.5.0% ^{235}U enrichment for the equilibrium cycle charge. In this calculation the tail ^{235}U concentration of the enrichment process is 0.25%. The ^{235}U concentration of recovered uranium is 0.9% as shown in Table 1 of the reference input data for PWR –NFC.

These requirements are calculated based on reactor fuel burnup and other plant parameter. Regarding the recycling option of the spent fuel nuclear fuel, the value of the recovered uranium and plutonium produced is the value of the fuels that can be made from them minus the costs of fuel fabrication. Because fuels made with recovered plutonium and uranium would substitute for LEU fuels, their value is determined by the price of LEU fuel with the same design burnup, which in turn depends on the price of uranium [11].

TABLE .1. REFERENCE REACTOR AND NFC DATA FOR CALCULATIONS

Parameters	Reference Values
- Reactor type	PWR
- Electric power (MWe)	1000
- Thermal efficiency (%)	34
- Thermal power (MWt)	2940
- Specific power (MWt/ton U)	40.2
- Capacity factor (%)	80
- Fuel Loading per core(MTU)	69.5
- Burn up (MWd/MTU)	45000
- Fuel feed enrichment	4.5
- Feed enrichment (%)	0.7/0.81
- Enrichment/Tails (%)	4. 5/0.25
- Uranium requirement/(kg) enriched uranium	
-Natural uranium price ($ /(kg)	7.267,(5.982)
-Conversion price ($ /(kg)	70.0
-Enrichment price ($ /SWU)	7.0
-Fuel fabrication price ($ /(kg)	130.0

- MOx fuel fab. price ($ /(kg)	350.0 1100.0

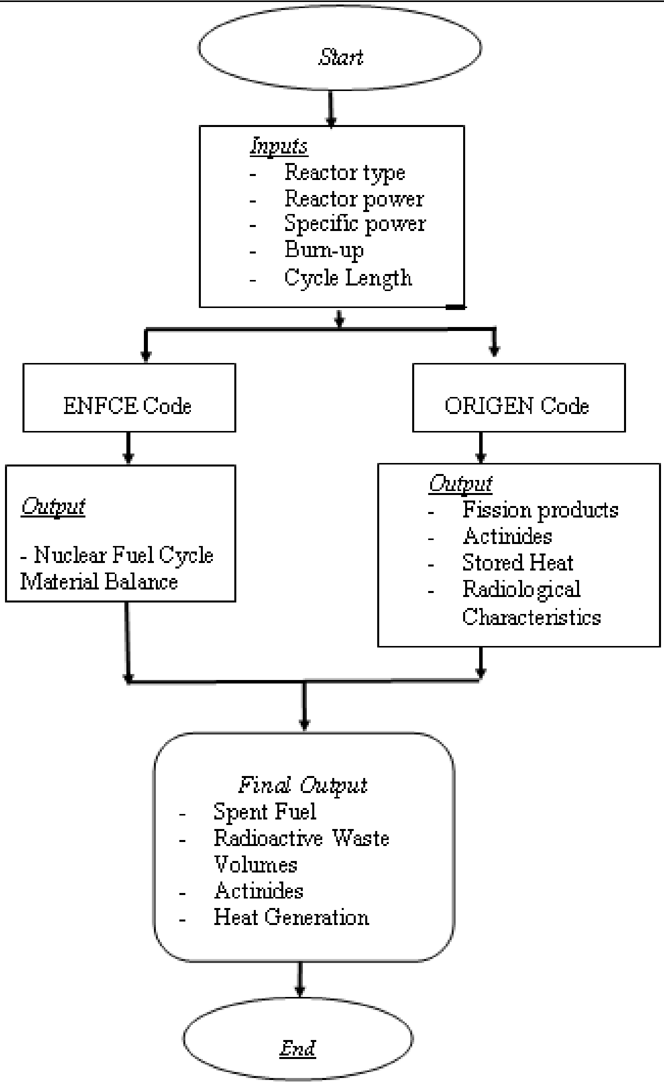

FIG. 2. Calculation procedure flow diagram.

2. RESULTS AND DISCUSSIONS

2.1. Nuclear safety considerations

The spent fuel, after removal from the reactor, will normally be stored in pools at the reactor site and then be transferred to an interim storage. When spent fuel is in pool storage or packaged in casks or canisters for dry storage radioactive wastes are generated [7]. During reprocessing, the main waste product is the high level waste solution containing most of fission products from the spent fuel. These wastes are vitrified for final disposal. ILW/LLW is also generated at the reprocessing plant and these wastes are treated as solidified slurries in cement or asphalt, compacted waste, incinerated ash or packaged solid waste. The waste volumes generated from reprocessing, storage, conditioning (including vitrification in the case of HLW) and disposal have been assessed using different references as shown in Table 2.

Waste volumes for all steps in each fuel cycle are estimated from the unit volumes as described in Table 2. As shown from this Table, the PWR-MOX fuel cycle has the smallest mill tailing which is ~30% lower than that for the PWR-OT fuel cycle. Also, PWR-MOX option has the highest ILW volume among the options, mainly due to the wastes generated in the reprocessing plant. The ILW of PWR-OT option is the lowest; however HLW is generated in the reprocessing plant for the PWR-MOX option. The volume of the waste of HLW and spent fuel is not a critical measure for waste disposal. The decay heat and the resulting disposal area are dominant factors in controlling the efficiency for overall waste management in the nuclear fuel cycle [8, 9].

TABLE 2. RADIOACTIVE WASTE VOLUMES FROM PWR -NFC OPTIONS (M^3/GWE-YR).

Fuel cycle options	Waste type				
	Tailings	LLW	ILW	HLW	SF
PWR-OT	60	600	50	-	19. 0
PWR-RO	40	650	90	3	-

2.2. Decay heat generation

As shown from Table 3**,** the safety and key cost driver for low level wastes is the waste volume; while for the spent fuels and high level wastes (HLW) the decay heat is as depicted in Table 3. The decay heat generated affects dry storage or disposal waste spacing. This spacing is important in disposal, because they affect the number spent fuel canisters that can be placed in the repository of a given size and thus the disposal cost [10]. Table 3 shows a comparison of the isotope decay heat output at 50 years and 300 years of cooling time for direct disposal and reprocessing concepts. The first column depicts the heat generated by seven isotopes that are producing almost all of heat in the fuel cycles. The decay heat of MOX spent fuel is the highest at both 60 years and 300 years. As can be seen from Table 3 , at 60 years of cooling time, the decay heat of the PWR spent fuels is mainly due to ^{90}Sr and ^{137}Cs. On the other hand, the decay heat of the MOX spent fuel is governed by ^{241}Am. The decay heat of PWR-OT concept is nearly 36% ~ 20% of the decay heat of the PWR-MOX spent fuel. The main difference between the two concepts is the decrease of fission products: ^{90}ST and ^{137}Cs. Also, it could be seen from Table 3 that, for 60 year to 300 year, the decay heat of the PWR-RO option clearly has the higher heat output which is about 2.75- 4.60 times higher heat output compared to PWR-OT option.

2.3. Economic assessment

Table 4 depicts the material flow for the two PWR -NFC options. The uranium credit has been calculated on the basis that the recovered uranium is produced in the form of UO_2 and only a relatively small premium is involved in converting it to UF_6 when it is recycled. If the uranium was recovered in another chemical form, higher conversion costs would be incurred with a potentially higher premium. This would have the effect of reducing the credit worth of the recovered uranium.

TABLE 3. DECAY HEAT OUTPUT OF SPENT FUELS (W/MHM)

Isotopes	PWR –OT option		PWR -RO option	
	60years	300 years	60 years	300 years
^{238}Pu	83	12	500	73
^{239} Pu	13	13	20	20
^{240} Pu	21	20	61	60
^{241}Am	160	118	841	620
^{242}Cm	14	0	215	0
^{90}Sr*	206	0	96	0
^{137}Cs*	216	2	218	2
others	5	2	30	16
Total	721	170	1984	788

TABLE 4. MATERIAL BALANCE FOR PWR -NFC OPTIONS (MTU/1GW-YR)

NFC Stage	PWR-OT	PWR-RO
1. Yellow Cake	161	111
2. Conversion	135	94
3. Enrichment	92	64
4. Fuel Fabrication	19	16
5. .Interim Storage	19	16
6. Reprocessing	_	15
7. Waste conditioning	19	3

Based on these reference unit prices, the cost of an enriched uranium fuel element for PWR (4.5% enrichment) is \$1394 per kg. On the other hand, the cost of an equivalent mixed oxide made from 933 g of natural uranium and 44 g of fissile plutonium will be \$ 1 172 per kg made up of \$65 for uranium purchase, \$7 for conversion and \$ 1100 for fabrication. The value of \$1100 per kg HM for MOX fuel fabrication is consistent with Ref. [6].

The reprocessing price (C_r) at which the costs of the two fuel cycles are equal, is the breakeven price, given from the following Equation [4]:

$$C_{is} + C_{ds} = C_r + C_{dw} + (C_f - C_e)$$

where:

C_{is}: cost of interim storage
C_{ds}: disposal of spent fuel
C_r: cost of reprocessing
C_{dw}: cost of wastes disposal
C_f: cost of recovered Pu and U
C_e: cost of equivalent LEU fuel

The breakeven fuel reprocessing costs calculations are carried out in terms of spent-fuel disposal costs (in mills/kWh) and varying natural uranium (yellow-cake) prices (in US$/lb.) are shown in Fig. 3. The calculations assumed a utility financing real after-tax return on equity of 10.0%. Unit costs for other fuel-cycle expenditures are assumed fixed. Fig. 3 demonstrates that at lower SF disposal costs and lower yellow cake prices, the break-even fuel reprocessing cost tends to be decreased. For example; for a spot yellow-cake price of US$20 per pound and a spent-fuel disposal fee of 1 mill/kWh, the break-even fuel reprocessing cost will be much less than US$500.0 per kgHM. This value is much lower than what is actually recently paid by the utilities for reprocessing their spent fuel abroad [6].While, for a yellow-cake price of US$ 40.0 per pound, and the cost of spent-fuel disposal in a geologic repository would have to be as high as 6.0 mills/kWh, fuel reprocessing cost tends to be increased up to US $2000 per kgHM. This means that the break-even fuel reprocessing costs are strongly dependent on both uranium and SF disposal costs.

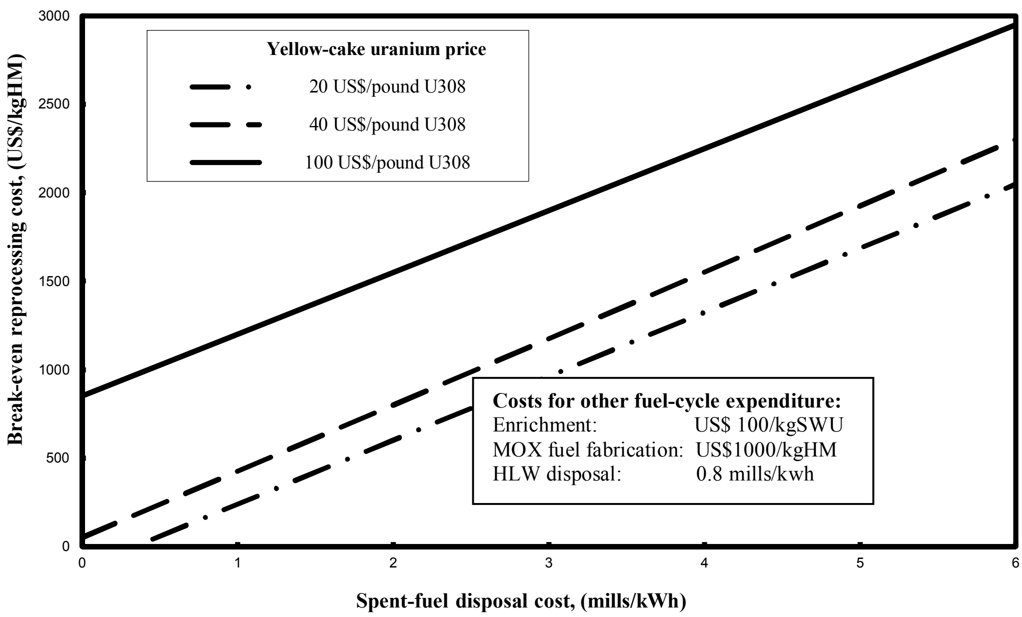

FIG.3. Break even reprocessing unit price as functions of yellow-cake and spent fuel disposal prices.

3. CONCLUSIONS

This paper evaluates the safety and economic aspects of PWR nuclear fuel cycle policies. The study is focusing on the spent fuel radiological characteristics e.g.: decay heat, which could be a measure for the effectiveness of waste management and the environmental considerations for various nuclear cycles. The decay heat and activity properties could be used for the design of transportation cask, interim storage, and final disposal facilities. The paper also estimated and compared the fuel cycle material flow of the direct spent fuel disposal and the "self-generated recycle" fuel cycle concepts to determine the justifiable fuel-reprocessing costs. Spent fuel break-even reprocessing costs have been estimated in terms of the spent fuel disposal costs and the natural uranium prices. The break-even fuel reprocessing costs have been estimated as a function of uranium supply and spent fuel disposal costs.

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EXPLORING ALTERNATIVES FOR NUCLEAR FUEL DISPOSAL IN MEXICO

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Mexico

Abstract

Spent fuel is one of the most important issues in the nuclear industry, currently SFM is been cause of great amount of research, investments, constructing repositories or constructing the necessary facilities to reprocess the fuel, and later to recycle the plutonium recovered in thermal reactors. What is the best solution? Or what is the best technology for an specific solution? Many countries have deferred the decision on selecting an option, while others works actively constructing repositories and others implementing the reprocessing facilities to recycle the plutonium obtained from nuclear spent fuel. In México the nuclear power is limited to two reactors BWR type and medium size. So the nuclear spent fuel discharged has been accommodated at reactor's spent fuel pools. Originally these pools have enough capacity to accommodate spent fuel for the 40 years of designed plant operation. However currently, the plants are under a process for extended power uprate to 20% of original power and also there are plans to extend operational life for 20 more years. Under these conditions there will not be enough room for spent fuel in the pools.

1. INTRODUCTION

The management of spent fuel from nuclear power plants has become a major policy issue for virtually every nuclear power program in the world. For the nuclear industry, to find sufficient capacity for storage and processing or disposal of spent fuel is essential if nuclear power plants are to be allowed to continue to operate. At the same time the options chosen for SFM can have a substantial impact on the political controversies, as: proliferation, radiological risks, environmental hazards, and economic costs of the nuclear fuel cycle.

Today, in some countries, the spent nuclear fuel is seen as a valuable energy resource, since most of its mass is uranium and plutonium that could be recovered and re-used for additional energy production, these countries have decided to reprocess the spent fuel and recycle the plutonium recovered while vitrifying the residual waste to be buried in deep repositories. Other countries tend to view the spent fuel as a waste, arguing that the cost of recovering its energy content is more than that energy is worth, and that reprocessing and recycling weapons-usable plutonium creates unnecessary proliferation hazards. Today the spent fuel is a potential energy resource whose exploitation could not be economically justified at present, but may become important at some unknown point in the future. Countries that think on that way have plans to direct disposal of the nuclear spent fuel in deep repositories.

In such context, many countries have not defined yet a policy regards the option to manage the nuclear spent fuel, instead have in some way to wait and see what happens in the near future as technology continue been developed.

2. THE CASE OF MEXICO

México has two BWRs reactor operating at Veracruz in the Gulf of Mexico. These reactors have an electrical output of 654 MWe each, these two units had been operating since 1990 and 1995 respectively, so the unit 1 has operated 14 fuel cycles and the unit 2 has been operating for 9 cycles.

Currently the cycle length selected for LVNPP is 18 months, to reach this cycle lengths, an average enrichment of 3.7% is required and the average burnup is between 9 500–11000 MWD/tU. So the nuclear fuel when is discharged has an average burnup of around 40 000 MWd/tU. The BWRs for LVNPP have a core containing 444 fuel assemblies, so a fuel reload is required each cycle, typically a quarter of core is replaced during reload which means 120 fuel assemblies on the average.

After the operation during 14 cycles the unit 1 has discharged on the order of 1560 fuel assemblies and unit 2 about 1000 FA. That makes a total of 2560 FAs. Each reactor has his own pool for spent fuel; both pools are identical with a capacity of 3177 FAs each. The total capacity of the whole plant is then 6354 fuel assemblies, which is enough to operate the reactors for 32 more cycles before the spent fuel pools will be totally fulfilled. However if the facility wants to extend the life of the plant, some kind of solution will be necessary. By now the capacity for spent fuel storage in our plants is enough for at least 15 more years. However each interface to implement a solution when time arrives has many problems to solve, and is very convenient to start planning now for the future.

The next figure shows the accumulated amount of spent fuel in reactor pools, and the limit at which the pools will be full of fuel. The figure shows too that this will take place in cycles 23 and 24 respectively for each unit.

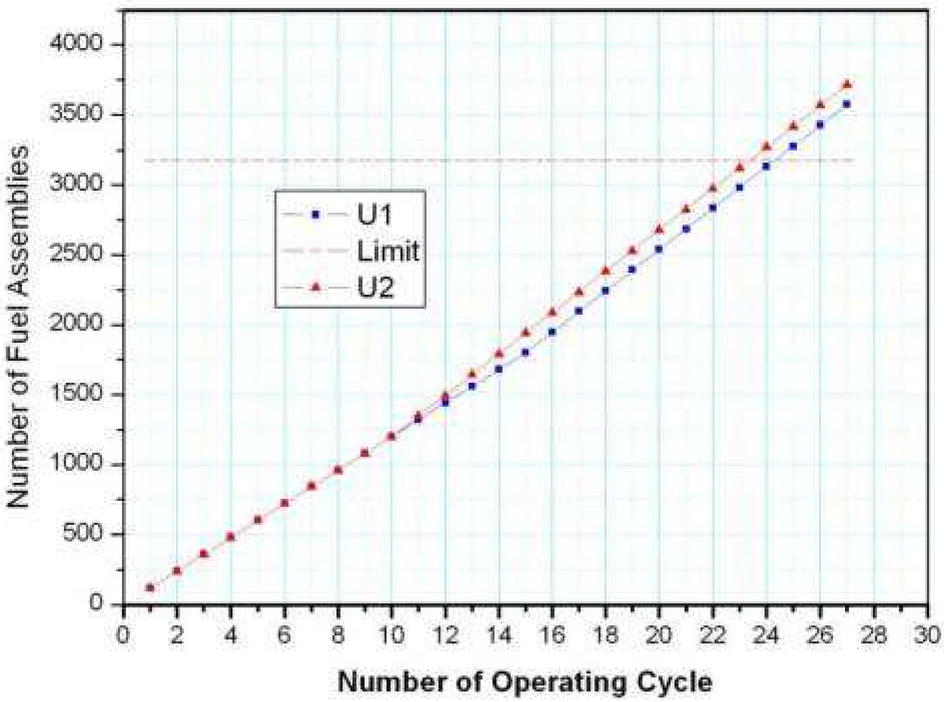

FIG. 1. Accumulated amount of Spent Fuel and limit.

Table 1 shows a projection at 40 years of reactors operation for the spent fuel accumulated at the plant site.

TABLE 1. PROJECTION FOR SPENT FUEL ACCUMULATED IN 40 OPERATING YEARS

	Assemblies/Cycle	Unit-2	Unit-1	Assemblies
Current	120	10 cycles	15 cycles	3000
EPU	148	30 cycles	25 cycles	8140
TOTAL		40 cycles	40 cycles	11140

Some studies had been performed to determine which will be the most promising solution in the long run and which the best technology to be implemented. The studies performed include the option for fuel reprocessing and later recycling of recovered plutonium, investigations about direct disposal and interim storage.

• 3. REPROCESSING AND PLUTONIUM RECYCLING

3.1. Feasibility of plutonium use in BWRs

Exploring the reprocessing option, several studies has been performed at ININ which is the institute for nuclear research in Mexico, these studies include:

- Core performance using plutonium;
- Possible amounts of plutonium existing in the spent fuel;
- Design of MOX fuel;
- Optimum plutonium concentrations;
- Maximum MOX fuel load in core;
- Amount of Spent fuel needed to implement MOX.

3.2. Rights over the fuel

Up to now, the spent fuel storage in reactor pools at the site belongs to the utility; however, as the utility belongs to the government the actual proprietary of the spent fuel is the government, who can decide what to do trough the Energy Secretariat, but the policy has not been defined yet. Another point to take into account, concerned to the international treaties in nuclear and proliferation matters.

3.3. Reprocessing

Mexico do not have reprocessing facilities, if the policy about spent fuel is defined, and the recycling option is selected, the spent fuel would have to be sent to some foreign facility to be reprocessed to obtain the plutonium and later the fabrication of MOX fuel must be contracted. At this point still is not clear, what in that kind of contracts is included and what is not for instance: what happens with radioactive waste and recovered uranium, in addition the transport of fuel and waste should be solved.

3.4. Fuel fabrication

The institute for nuclear research in Mexico, constructed and operated a small nuclear fuel fabrication plant to develop the fabrication technology for UOX fuel, that plant was close in 1996, however the plant still exist and could be adapted to work with mixed oxides to start tests.

3.5. Licensing of MOX fuel

Typically each reload of the nuclear reactor must be licensed by the regulatory body before the new fuel be introduced to the core, for MOX fuel the process should be the same as for UOX fuel.

3.6. Economics

Several scenarios for fuel recycling had been calculated, and finally we arrive that recycling is more expensive that direct disposal of spent fuel, however, as the calculation are based on international assumptions and prices it is possible that real costs differed from calculated.

A summary of the results is found in the Table 2. There it may be noted that the "Once Through" scenario is less expensive than the "Recycling" scenario, with a non-trivial difference of 263 million dollars. Note that a constant money calculation has been used to obtain these results.

TABLE 2. ECONOMIC EVALUATION OF FUEL CYCLE POLICY

	Unit Cost		Once Through		Recycling	
	USD/Kg HM	Kg/FA	# F.A	Cost USD	# F.A.	Cost USD
UOX fuel assembly	2123	180	6488	2.479E+09	5868	2.242E+09
MOX fuel assembly	1500	180			620	1.674E+08
Reprocessing	1140	180			4340	8.906E+08
Final disposition.	840	180	6488	9.810E+08	2180	3.296E+08
Uranium recovered.	140	169.2			4340	1.028E+08
High level waste	-300	7.258			4340	-9.431E+06
TOTAL				**3.460E+09**		**3.723E+09**

The costs for final conditioning and disposition have uncertainties attached. These costs were obtained from the literature and are considered generic. Thus the costs for the recycling option can be even higher.

4. REPOSITORY FOR FINAL DISPOSAL

The concept of a deep repository has been investigated as a technological option for spent fuel disposal, however, the number of reactors operating in Mexico up to now, probably do not justify the amount of investments needed to consider a deep repository viable since the economical point of view. Even that several proposals for a possible site has been taken into account for a possible site studies and some additional studies on engineering barriers and packing of containers are being planned for the future.

5. DRY STORAGE OPTION

Recent Practices in several countries, is the interim storage of spent fuel into special containers designed for that purpose, in such a way that the fuel can remain from 40-50 years before to take a decision what option will be best for final disposition. This technology can be implemented in reactor site or away from site. In Mexico this kind of technology has a high probability to be implemented in a short term waiting for a policy definition; however that is not a cheap solution. Since in one or other way the spent fuel will not be storage in casks forever and something has to be done to solve the problem.

For LVNPP we have calculated taken into account the current operational conditions, will be necessary two BWR casks for each reload of the reactor, so, for 40 years we have 27 fuel cycles 18 months each and 54 spent fuel casks to storage the total spent fuel discharged from one reactor, plus the fuel remaining in the core which amounts for another 3 reloads, that makes a total of: 60 spent fuel casks.

This option will cost for each reactor, taken into account that each cask has a cost of 800 000.00 USD and the services of filling it which costs from 250 000–300 000 USD, that makes a total of 1.1 million dollars each, and for one reactor spent fuel during his useful life the cost will be 66 million dollars plus maintenance and surveillance. In addition an interim storage facility must have to be constructed near the reactor site or away from it with the corresponding costs.

6. RECOMMENDATIONS

The main recommendations for SFM in México are: first to prevent spent fuel pools at the LVNPP to be fulfilled and select one of the available technologies for final disposal, otherwise the best option will be to defer the decision and to use dry storage, then to take a decision 40-50 years from now. Thinking that at the time will be a better or more developed technology.

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THAI STRATEGIC PLAN FOR SPENT FUEL MANAGEMENT

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In the past few years, nuclear power has become a key part of the global energy solution. Many countries have been planning on expansion and embarking of the nuclear power in their countries. It is not surprising that Thailand is one of those countries recognizing that nuclear energy is one of the promising options responses to Thai energy policy on energy security, fuel diversification, and greenhouse-gas emission reduction.

According to a Thailand Power Development Plan 2010–2030 (PDP2010), nuclear power plants producing 5,000 MW are expected to be included into the power system during 2020–2030. With advices and guidance from IAEA experts, a nuclear power infrastructure establishment plan (NPIEP) was established considering all aspects related to an introduction and implementation on nuclear power program.

In 2008, the Nuclear Power Program Development Office (NPPDO) was established under the Ministry of Energy to coordinate all activities related to the national nuclear power infrastructure establishment which are planned in Phase 1, pre-project reactivity phase. During this stage, NPPDO and other involved national organizations have analyzed and developed basic infrastructures to the extent needed in the implementation stage, which includes a development program on industrial and commercial infrastructure, technology transfer, human resources, a safety and environmental protection program, a preparation of setting up the regulatory body, and a regulatory framework on nuclear power.

In the aspect of radioactive waste management, NPPDO will propose the strategic plans to the government as part of a "readiness" report, which will be submitted in 2011. Therefore, it should be noted here that, this paper presents a planned proposal, not a final policy from the Thai government. The strategic plans and burdens with respect to determining an appropriate national involvement of the interim storage and the approach to ultimate disposal for low and high-level wastes and/or spent nuclear fuel will be discussed.

One of the biggest obstacles in developing nuclear power in Thailand is the concern of high radioactive waste with or without reprocessing. The public feels that nuclear waste is fatal, unacceptable hazards and environmental risks because it has very long half-life radio-nuclides. These negative perceptions cause difficulties in addressing the policy for final or even temporary disposal of low and high radioactive waste generated from prospective nuclear power plants. However, we need to build public confidence how the nuclear waste and spent fuel will be managed and stored safely for many decades.

Currently, there are several options of the waste management for Thai considerations: (1) Onsite storage in wet pool and dry cask; (2) Centralized interim storage; and (3) Final Repository. The on-site storage seems to be the best option for Thailand. Firstly, the option 2 and 3 will raise the public's apprehension on the site location of the disposal, which will probably cause *a priori* refusal of the nuclear project. Secondly, Thailand is only in the beginning stage of introducing the nuclear power in the country, the generated spent fuel will not be in the considerable amount to develop the permanent storage of spent fuel and/or high radioactive waste for some periods. In addition, there will be no transportation on the spent fuel which will lessen the public concern about the safety and security of spent fuel shipments.

Until the government establish the definite policy on the final storage of high-level waste, all spent fuels which will be produced in a nuclear power plant can be safely stored on site for 100 years based on existing technologies, wet or dry. In the meantime, we expect that significant progress on geological disposal facilities will be accomplished. Thailand will wait and see how international waste program will be developed in the approaches and methodologies.

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