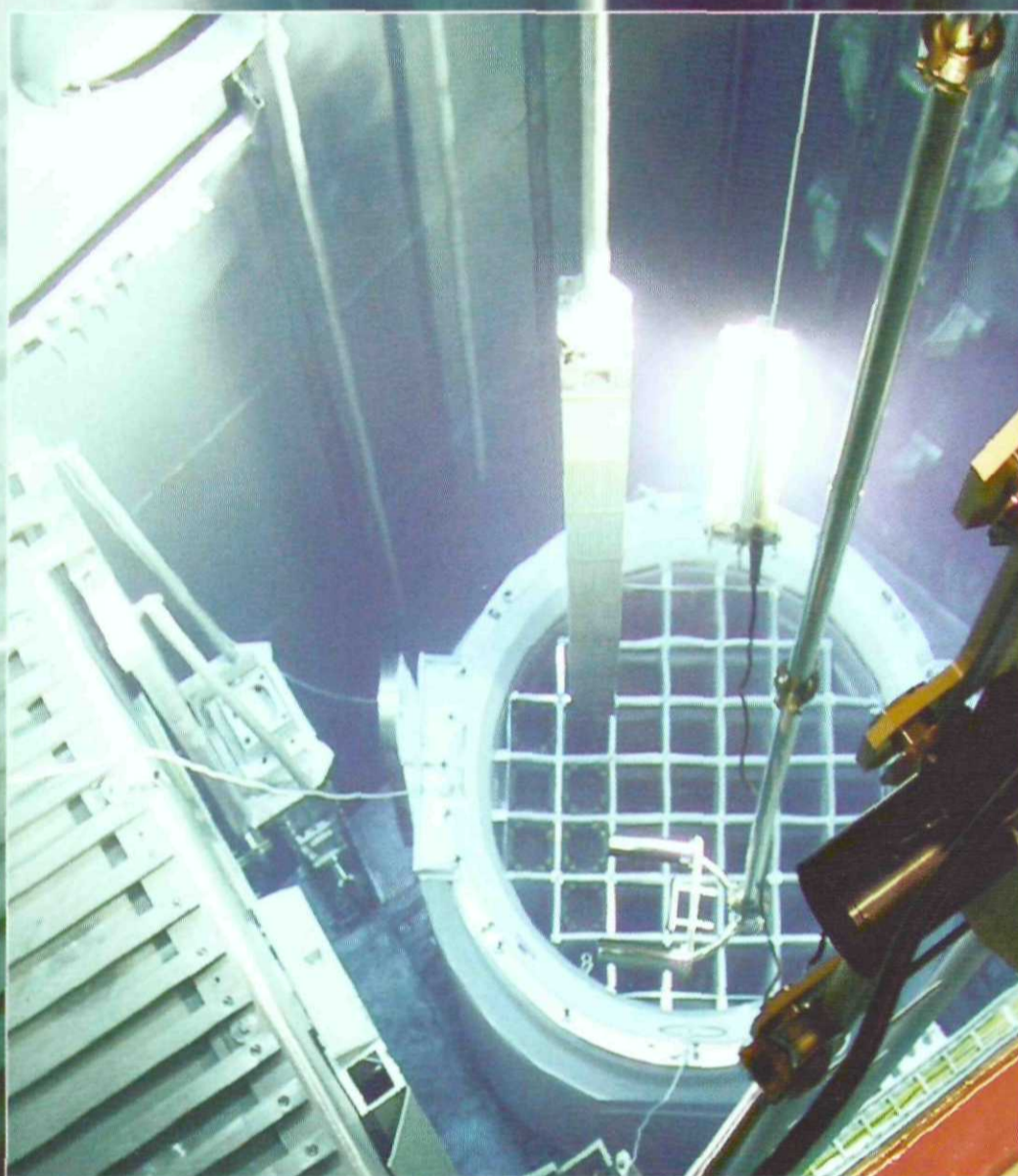




# International Conference on Storage of Spent Fuel from Power Reactors

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## BOOK OF EXTENDED SYNOPSES



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## LIST OF EXTENDED SYNOPSES

Number	Title of Paper	Name of Presenter	Session
IAEA-CN-102/1	Management of spent fuel from power reactors in Argentina	M. Audero	1
IAEA-CN-102/2	Conceptual design for an intermediate dry storage facility for Argentinean Atucha spent fuel	D.o. Brasnarof	2
IAEA-CN-102/3P	The status of storage of spent fuel from reactors: Bangladesh perspective	M. Monzurul Haque	3
IAEA-CN-102/5	Verification of the scale modular code system for criticality safety and depletion analyses of WWER spent fuel facilities	T. Apostolov	4
IAEA-CN-102/6	Regulatory approval of used fuel dry storage facilities in Canada – an Ontario power generation case study	A. Khan	3
IAEA-CN-102/7P	Spent fuel management in China	W. Zhang	3
IAEA-CN-102/8P	Status and prospects for spent fuel management in China	Y. Wang	3
IAEA-CN-102/9	Spent fuel storage facilities in the Czech republic	J. Coufal	1
IAEA-CN-102/10P	Computer simulation of 3D steady and 2D transient thermal loading of castor 440/84 using FEM	J. Hejna	3
IAEA-CN-102/11	Current status of the spent fuel management from power reactors in the Czech republic (licensing and operational experience)	P. Lietava	3
IAEA-CN-102/13	Key issues from the French R&D project on the long-term evolution of the spent nuclear fuel in conditions of interim dry storage	C. Ferry	4
IAEA-CN-102/14	Dry storage technologies: keys to choosing among metal casks, concrete shielded steel canister modules and vaults	V. Roland	2
IAEA-CN-102/15	Transport and interim storage casks in Switzerland	A. Verdier	3
IAEA-CN-102/16	Secondary phase formation during interaction of spent nuclear fuel and cladding material during long term storage and disposal leaching tests	M. Amme	4
IAEA-CN-102/17	Advanced spent fuel storage pools	B. Arndt	2
IAEA-CN-102/18	U <sub>o2</sub> and MOX fuel behaviour in long term dry storage	W. Goll	4
IAEA-CN-102/19	Management of spent fuel from power and research reactors using castor and constor casks and licensing experience in Germany	J. C. Neuber	4
IAEA-CN-102/20	The use of castor and constor casks for management of spent fuel from power (and research) reactors	A. Vossnacke	2
IAEA-CN-102/21	Transuranus simulation of WWER cladding creep under dry storage conditions	C. Gyori	4
IAEA-CN-102/22P	Safety analysis of the C30 spent fuel transport cask for the extended range of loading parameters	G. Hordósy	3

<b>Number</b>	<b>Title of Paper</b>	<b>Name of Presenter</b>	<b>Session</b>
IAEA-CN-102/23	Spent fuel dry storage in Hungary	F. Takáts	1
IAEA-CN-102/24	Spent fuel dry storage in India	H. B. Kulkarn	1
IAEA-CN-102/25P	Design of spent fuel storage of prototype fast breeder reactor	V. N. Sakthivel Rajan	3
IAEA-CN-102/26P	Spent fuel management strategy for future nuclear power plants operation in Indonesia	Z. Salimin	3
IAEA-CN-102/27	Thermal creep tests of BWR and PWR spent fuel cladding	K. Kamimura	4
IAEA-CN-102/28	Verification of dual-purpose metal cask integrity	S. Matsuoka	2
IAEA-CN-102/29	Post irradiation examinations of twenty years stored spent fuel	A. Sasahara	4
IAEA-CN-102/30	Current status of R&D program of spent fuel storage technology in CRIEPI	K. Shirai	2
IAEA-CN-102/31	Assessment of the storage concept for conditioned spent fuel	K. S. Seo	4
IAEA-CN-102/32	New interim spent fuel storage facility at Ignalina NPP	I. Krivov	1
IAEA-CN-102/33	Comparison of the main characteristics for castor and constor casks loaded with spent RBMK-1500 nuclear fuel	P. Poskas	3
IAEA-CN-102/34P	Dry storage of spent KANUPP-fuel and booster assemblies	W. Ahmed	3
IAEA-CN-102/35	The licensing process of Cernavoda interim spent fuel dry storage	V. Andrei	3
IAEA-CN-102/36	Implementation of Romanian NPP spent fuel management strategy - a regulatory approach	L. Biro	1
IAEA-CN-102/37P	Experience in performing of the Cernavoda spent fuel interim storage facility	M. Radu	3
IAEA-CN-102/39	Validation of dry storage modes for RBMK-1000 spent fuel assemblies (SFA)	I. M. Kadarmetov	4
IAEA-CN-102/40	Ensuring safety in handling the casks with irradiated nuclear fuel	T. Makarchuk	3
IAEA-CN-102/41	Research in corrosion resistance of structural materials of metal and concrete casks for spent nuclear fuel	N. Yanovskaya	3
IAEA-CN-102/42P	Comparison of subcriticality of the interim spent fuel storage before and after modification	V. Chrapčiak	3
IAEA-CN-102/43P	Capacity extension of the Bohinice storage facility	D. Belko	4
IAEA-CN-102/44	Storage of spent fuel in Slovakia	J. Vaclav	1
IAEA-CN-102/45P	The NPP KRŠKO rerecking project	B. Kurinčič	4
IAEA-CN-102/46	Update on spent fuel and HLW management in Spain	J. E. Martinez	1
IAEA-CN-102/47P	Evaluation of nuclear fuel cycle scenarios with respect to some parameters important for spent fuel storage	T. Akbas	4

<b>Number</b>	<b>Title of Paper</b>	<b>Name of Presenter</b>	<b>Session</b>
IAEA-CN-102/48	Multi-purpose canister storage of spent nuclear fuel in modular vault system	C. C.carter	2
IAEA-CN-102/49	National policy in the area of spent fuel management in Ukraine: Current status and trends (prospective)	A. Afanasyev	1
IAEA-CN-102/50P	Spent fuel storage facility of Zaporizhzhya NPP: creation, licensing, operation	O. Dvoyeglazov	4
IAEA-CN-102/53	The construction and operation experience of the interim spent fuel storage facility at the Zaporizhzhya nuclear power plant	Y. Trehub	3
IAEA-CN-102/55	U.S. Nuclear regulatory commission acceptance criteria and cladding considerations for the dry storage of spent fuel	K. A. Gruss	3
IAEA-CN-102/56	International experience of storing spent fuel in NUHOMS systems	A. Hanson	2
IAEA-CN-102/57P	The storage of spent fuel in Vietnam: present status and prospects	N. T. Sinh	4
IAEA-CN-102/58	SFCOMPO: a database for isotopic composition of nuclear spent fuel; current status and future development	K. Suyama	4
IAEA-CN-102/61	Optimization of cask capacity for long term spent fuel storage	W. Danker	2
IAEA-CN-102/62	Selection of AFR facilities for spent fuel storage	J. S. Lee	2
IAEA-CN-102/64	The German policy and strategy on the storage of spent fuel	P. V. Dobschuetz	1
IAEA-CN-102/65	Spent fuel management strategy in Japan	Y. Ikoma	1
IAEA-CN-102/66	Experience from extension and licensing of the Swedish central interim storage facility for spent fuel, CLAB, from 5000 to 8000 metric tonnes	I. Zellbi	3
IAEA-CN-102/67P	Gamma ray control of metal and concrete cask radiation protection	Shchigolev	4
IAEA-CN-102/68	Examination of the creep rupture phenomenon and the development of an acceptance criteria for spent fuel dry storage	J. Y. R. Rashid	4
IAEA-CN-102/70	The experience of the operation of the interim storage facility for spent fuel in Olkiluoto	K. Sarparanta	3
IAEA-CN-102/71	Canada's national policy on the long-term management of nuclear fuel waste	P. Brown	1
IAEA-CN-102/72P	Technique of monitoring cladding integrity of RBMK-1000 spent fuel assemblies after long storage	T. F. Makarchuk	4
IAEA-CN-102/73	On-site intermediate storage facilities in Germany	H. Flügge	3
IAEA-CN-102/74P	Concept of BN-350 reactor spent fuel handling during its storage after shutdown	S. Talanov	4

<b>Number</b>	<b>Title of Paper</b>	<b>Name of Presenter</b>	<b>Session</b>
IAEA-CN-102/75	Storage of spent fuel from the nuclear power plants Greifswald and Rheinsberg in the interim storage north	W. A. Birkholz	3
IAEA-CN-102/76	Licensing experience and technical issues related to wet storage of spent fuel in the United States	S. R. Jones	3
IAEA-CN-102/77P	Methods for WWER-1000 fuel testing under dry storage conditions	S. V. Pavlov	4
IAEA-CN-102/79	The Idaho spent fuel project update	R. Roberts	2
IAEA-CN-102/80	Some aspects of the Russian nuclear fuel cycle development	E. G. Kudryavtsev	1

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**MANAGEMENT OF SPENT FUEL FROM POWER REACTORS IN ARGENTINA**

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Argentina has two nuclear power plants (NPP) in operation, supplying 12% of the national electricity production. The Atucha-1 NPP started commercial operation in 1974, it is a 340 MWe Heavy Water Reactor with pressure vessel, of Siemens design. The Embalse NPP started commercial operation in 1984, it is a 600 MWe PHWR with pressure tubes of AECL design (CANDU). The nuclear power plants are owned and operated by the state company Nucleoeléctrica Argentina S.A.

The fuel assembly (FA) of Atucha-1 has an active length of 5.3 m, it has circular cross-section of 0,10 m diameter with 36 fuel rods plus one structural rod. Each FA is loaded with approximately 176 Kg of  $UO_2$ . Atucha-1 was fuelled with natural uranium during the first 27 years of operation, the average extraction burnup of the fuel was approximately 6,000 MWd/tU. In January 1995 the utility started a program to gradually convert the fuelling to slightly enriched uranium (SEU), using an enrichment of 0.85 % U-235. The program was completed in August 2001, since then the whole core is fuelled with SEU and the average extraction burnup of the fuel is approximately 11,300 MWd/tU. This change produced an important saving in fuel consumption: from approximately 395 FA/full power years (fpy) to approximately 210 FA/fpy.

In Atucha-1 the SF is stored in water pools located at the reactor site. The storage pools are made of concrete with stainless steel lining. The monitoring program of the whole installation has not detected any failure or degradation of the components neither of the SF. The original management strategy considered the transfer of the SF to dry interim storage after the final shutdown of the NPP. Nevertheless, it is foreseen the necessity of operate the wet storage installation during at least 10 years after the final shut down, to allow for thermal cooling and radioactive decay of the SF belonging to the last core.

At present Atucha-1 is planning a re-racking of the SF in order to enlarge the capacity of the wet storage pools. However, it is foreseen the necessity of additional storage capacity to comply with the 12.5 full power years of remnant life of the reactor. Therefore, it is necessary to anticipate the original planning in order to put in operation a dry storage installation before the final shutdown of the NPP. So, there is under development a conceptual design of a modular facility for dry storage of the SF to be located at the reactor site, using the concept of concrete silos.

The CANDU fuel assembly of the Embalse NPP has a length of 0.5 m and a circular cross section of 0,10 m diameter, with 37 fuel pins. This NPP is fuelled with natural uranium, achieving an average burnup of approximately 7,500 MWd/tU. The weight of UO<sub>2</sub> per fuel assembly is approximately 22 Kg, the fuel consumption is approx. 4,800 FA/fpy .

In Embalse NPP the SF is stored in water pools during 6 to 8 years for thermal cooling and radioactive decay and then are transferred to dry storage in silos, both interim storage facilities are located at the reactor site. The pools for wet storage are made of concrete with epoxy lining. The silos for dry storage are made of concrete, each silo contains 9 sealed stainless steel baskets with 60 fuel elements each one. This installation is of modular type, 120 silos were already built with a total capacity of 64,800 spent fuel assemblies, when needed, new silos will be added to the existing ones. It is planned that 6 to 8 years after final shutdown of the NPP all the SF will be in dry interim storage.

The inventory of SF stored at both nuclear power plants is shown in the following Table:

**Inventory of Spent Fuels from the Nuclear Power Plants**

NPP	STORED (*)		EXPECTED AT EOL
	WET	DRY	
<b>Atucha-1</b>	8,546	-----	11,600
<b>Embalse</b>	40,900	46,877	144,300

\* Status at December 31, 2002

CNEA is conducting a project to study the aging of materials in spent fuel storage facilities and is participating in an IAEA's coordinated research program on this topic.

In the management of spent fuel from the nuclear power plants, Argentina has not defined yet its fuel cycle back end strategy, although in principle the SF is considered as an asset. According to the current planning, a decision about this issue should be taken before the year 2030. In any case, a deep geological repository is considered necessary for the disposal of the SF or the high-level waste that would be generated in the case of their reprocessing. It is planned that the deep geological repository should be in operation in the year 2050, at present are being carried out geological studies for repository siting.



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## CONCEPTUAL DESIGN FOR AN INTERMEDIATE DRY STORAGE FACILITY FOR ARGENTINEAN ATUCHA SPENT FUEL

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The CNEA (Argentina National Atomic Energy Commission) is planning a new facility for the spent fuel of Atucha I according with the national policy to fulfill the requirement of the National Plan of Radioactive waste management with the lowest cost, having the flexibility to evaluate the fuel back end strategy in a wait and see approach.

Spent fuel elements can be stored in concrete for many decades economically and safety as intermediate step, thereby providing adequate time to develop an integrated fuel disposal system, this provides flexibility from the fuel to decay, thus facilitating final disposal with decrease of the decay heat.

A centralized storage for the NPP fuel elements (Embalse and Atucha I) with two very different fuel element and different enrichment was not considered, in order to minimize the radioactive waste movement.

Nowadays the total life Atucha I spent fuels are in two wet pools, having fuel elements with 28 years old. For Embalse fuel elements type dry vertical concrete silos were successfully implemented for intermediate strategy.

An intermediate storage for Atucha I was designed taking into account the following criteria:

- Assurance the fuel elements integrity for 30 year.
- Modular build-up to avoid over dimension systems.
- Low cost radiation shield (concrete and ground).
- Leak Monitoring system for the containment integrity.
- Possibility to take out the failed containment.
- Enable the re-encapsulation and the reentry for the fuel containment.
- Minimize the auxiliary systems with high maintenance cost (passive).
- Compatible with the national regulatory commission (ARN) regulation with monitoring systems, similar with the implemented in our dry silos at Embalse.
- Transfer systems and hot cell facility near the pool storage to use its water treatment systems.
- Minimize secondary waste during wet pool to the intermediate storage.

The Atucha I fuel element has 37 fuel rod in circular cluster geometry with an active length of 5,5 meters.

The storage systems must be compatible with the fuel elements with natural uranium and SEU (slight enrichment uranium) fuel elements, 28 years for the oldest and at least 5 year decaying time in wet pool for the new one.

Different storage systems for intermediate time were evaluated (silos, cavern, cask) [1] and the silo has the best cost effective solution, compatible with the strategy for CANDU type fuel dry storage facility.

The surface disposal had be chosen to minimize on-site works. Using concrete as radiation shielding, horizontal modular silos can be arrange can be to minimize surface and concrete volume and the instrumentation cost [2].

Re-racking is no necessary because cluster geometry is very compact, only an additional physical barrier to each fuel element is placed using a complete fuel canister corrosion resistant. The canister structural requirements are keep the integrity for 30 years period and need to be transported form the conditioning cell to the silo. The isolation of fuel element encapsulated can be tested by helium leak. The canister have special hook for the input-output of the silo for the end of life and failures.

A 5 x 5 square array not integrated fuel containment is used in the silo. Individual fuel allocation is used with elliptic rolling pin. This design enable an easy re-encapsulation movements via an input-output system if cladding fails.

The decay heat is cooled in the silo by a natural convection cooling system (passive). The air comes into the fuel room by one air inlet and an internal flow distributor by perforated plates. In the upper part of the silo are placed more than two air outlet. The natural convection system was designed to have the heat dissipation with sheath temperature up to 200 °C.

The fuel element conditioning is done in hot cell. To minimize fuel movement this is placed near the wet storage pool. The short distance between the hot cell and the wet pool enable the use of auxiliary systems (air purification, water treatment). A special cask is required for fuel element transport from the wet pool to the hot cell. Inside the hot cell is placed a rolling cut machine (without small metal scarps), a dryer, fill and pressurize machine, and the plug welding machine. The fuel element movement inside the hot cell are done by a remote handling system and a two dimension fuel allocation system, this one can transfer the fuel element encapsulated inside a special transport cask. This system can be used to the re-encapsulation situation if the external cladding fails.

The transport systems is a special cask with radiation shield and a mechanical ram system to push and pull the fuel into the silo and hot cell. This system enable to return the encapsulated fuel element to the hot cell eventually, or final waste treatment. This transport can be done by truck of 110 ton. with 14 axle trailer.

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**THE STATUS OF STORAGE OF SPENT FUEL FROM REACTORS: BANGLADESH PERSPECTIVE\***A. JALIL<sup>A</sup>, ♥M. MONZURUL HAQUE<sup>B</sup>, M. A. ZULQUARNAIN<sup>B</sup>, G. RABBANI<sup>A</sup>

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Bangladesh is a modest user of nuclear technology and its nuclear program is fully dedicated to the peaceful applications in the fields of medicine, agriculture, industry, research and development. Presently a 3 MW TRIGA MARK II research reactor, located in the campus of the Atomic Energy Research Establishment (AERE) at Savar about 40 km north-west of the capital city Dhaka, is the main nuclear research facility in the country. The reactor has so far been operated for about 3428 hours with a total burn-up of about 5750 MWh. The reactor uses TRIGA LEU fuel with normal uranium content of 20% by weight. The enrichment level of the fuel is 19.7%. At the same time, Bangladesh Atomic Energy Commission (BAEC) is seriously pursuing a project to establish a 600 MW PWR type nuclear power plant in the west-central part of the country in near future. With regard to the safe management, storage and disposal of wastes accruing from the present operation of the research reactor and the future operation of the power reactor, BAEC is going ahead with specific plans and program.

The management of spent fuel has always been one of the most important stages of the nuclear fuel cycle. Three approaches currently exist for safely managing the spent fuel. One is associated with the open, or once-through, fuel cycle and calls for the direct disposal of spent fuel in a deep geological repository that does not allow for its retrieval. The second approach is associated with the closed fuel cycle and involves the preprocessing of spent fuel and the recycling of recovered plutonium and uranium in new mixed oxide (MOX) fuels. The third approach is commonly called the 'wait and see' approach, whereby spent fuel is placed in long-term interim storage pending a decision as to its ultimate reprocessing or disposal. Approach selection procedures depends on the political, technical, economic and safeguard issues as well as environmental protection aspects. This selection process has been influenced by a number of developments which, taken together, are elevating the importance of spent fuel management and storage in many countries.

\* Work performed on Bangladesh Government's programme on management and disposal of spent fuel from nuclear reactor.

♥ Presenter author



These developments include the comparative costs of uranium and reprocessing. Today's low uranium prices, and the relatively high costs for reprocessing, mean that the price of fuel with recycled uranium and plutonium is higher than the price of fuel from newly mined uranium. At the same time, some countries are delaying decisions on reprocessing, while others are abandoning the option and there are evident delays in the availability of facilities for the direct disposal of spent fuel nearly in all countries. Consequently, operators in many countries are finding themselves in the 'wait and see' position. This provides them with more time to evaluate the available technologies before deciding of their spent fuel management approach. However, it also places greater pressure on storage requirements and we are seeing that storage problems are growing in many countries.

The current spent fuel management policy in different countries can be divided into three broad groups:

Countries that have been following the once-through fuel cycle and are focusing their attention on interim storage followed by disposal of the fuel (e.g. Canada, Sweden and the United States of America);

Countries that have selected the reprocessing option and are actively operating or constructing reprocessing plants or have contracts for reprocessing abroad and/or are returning some or all of their fuel to the country of origin (e.g. France, Japan, the Russian Federation and the United Kingdom);

Countries that are still evaluating their spent fuel management programmes (e.g. the Republic of Korea, Lithuania and Mexico).

Bangladesh is a peaceful country and has signed several multilateral and bilateral agreements, protocols, treaties, etc. in connection with the Nuclear Non-proliferation regime. In these regards, we will follow the policy A for fuel management. In addition to this, Bangladesh has also signed a Nuclear Cooperation Agreement with the USA on September 17, 1981, which facilitated export of nuclear technology from USA to Bangladesh. The research reactor of BAEC was procured under its provisions. The tenure of the Agreement has been extended up to 2012.

Presently, there is not a single spent fuel element in the reactor facility and with the present trend of burn-up, it seems that BAEC TRIGA will not generate any spent fuel by the year 2009.

BAEC TRIGA facility has three Fuel Storage Pits located in the floor of the reactor hall. These pits are meant for temporary storage of spent fuel elements as well as radioactive samples. These storage pits are made of stainless steel having 10 inches diameter and 15 feet depth. Each pit is provided with a lock on the stainless steel cover plate to limit pit access and an M.S. cover plate (with lifting hook) that fits flush with the floor. The storage pits were designed to hold 19 TRIGA fuel elements in each pit. The pit can be water filled to accommodate storage of irradiated fuel elements. There are also 3 fuel storage racks, each capable of holding 10 fuel elements, located along the walls of the reactor tank to provide temporary storage of fuel-moderator of graphite dummy elements.

With regard to the management of the spent fuel from the proposed nuclear power plant, direct disposal after conditioning is envisaged rather than reprocessing. Water pool (wet) storage at the reactor site for a period of more than 30 years is being considered prior to the final disposal in a crystalline rock repository in the northern part of the country. 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 suitability for handling. Engineering details of spent fuel management in general and storage in particular are presently under study.

A Central Waste Processing and Storage Facility (CWPSF) is fast nearing completion in the AERE, Savar campus where the TRIGA Reactor, Isotope Production Laboratory, 14 MeV Neutron Generator, a  $37 \times 10^2$  TBq commercial irradiator and other hot facilities are situated. The country has an on going exploration program for nuclear minerals. A national strategy exists for the management and disposal of various types of radioactive wastes including spent fuel [1].

The authority for the control of all radiological and nuclear practices in Bangladesh is vested on the Bangladesh Atomic Energy Commission (BAEC) as the competent authority. The legal basis for this control are the 'Nuclear Safety and Radiation Control (NSRC) Act 1993 [1] and the Nuclear Safety and the Radiation Control Rules, 1997' which essentially incorporate the requirements of the international Basic Safety Standards. The Act requires a license to carry out any radiological and nuclear practice in the country while the rules prescribe the conditions to be fulfilled and the manner by which a license has to be obtained. The Act 1993, Rules 1997 and Statutory Regulatory Orders of 1996 and 2000 cover all aspects of radiation, waste and transport safety. Disused spent sealed sources are being stored in an interim storage in the CWPSF following short treatment and conditioning.

As to their disposal the currently preferred option is engineered repository. Site investigation work has progressed far enough toward the goal of establishing a demonstration repository at AERE, Savar by the year 2010 [2]. For long-lived highly active problem wastes including spent radium needles and disused radioactive sources, the safe management option is a long-term storage in the CWPSF following treatment and conditioning. But this is not considered as a sustainable solution. The real emphasis is placed on the development of inexpensive disposal methods and availing regional/international repositories [3, 4].

A long-term storage is not a sustainable solution because it has got its own type of risks and uncertainties associated with it. The ideal disposal option for highly active highly toxic 'problem waste' will be a regional or an international geological repository particularly for countries like Bangladesh which has small resource base [3, 4].

Spent fuel storage facilities may be situated at-reactor (RA) OR outside the boundary of a nuclear power station, in other words, away-from-reactor (AFR), possibly at a centralized site. Such facilities may also be classified by the medium of storage, either in the form of 'wet' or 'dry' storage. Wet storage facilities for spent fuel are those facilities which involve the storage of spent fuel in water pools. The spent fuel may be supported within the pool by racks, and/or

contained in canisters placed within the water medium. Dry storage facilities for spent fuel are those facilities which involve storage of spent fuel in a gas environment, such as an inert gas or air. Dry storage includes the storage of spent fuel in casks or vaults. A cask is a massive container which may or may not be transportable.

Vaults consist of above or below-ground reinforced concrete buildings containing arrays of storage cavities suitable for the containment of one or more fuel units.

Various types of wet and dry storage facilities are in operation or are under consideration in different countries. Spent fuel can be safely stored for long periods of time (some spent fuel has already been stored for over thirty years). Nearly all countries operating nuclear power plants are increasing their existing AR storage capacity by re-racking. This can be done by using neutron absorbing materials between the assemblies, or by rod consolidation, or simply by better distribution of fuel in the storage pools. Such modifications have resulted in at least a twofold increase in storage capacity. Further capacity increases may invoke the so-called 'burn up credit' in calculating the criticality of irradiated fuels.

In this paper the current strategy with regard to the safe management and disposal of various waste types, spent fuels, amount of spent fuel stored, storage facilities as well as capacities, storage facilities under construction and planning, national policy for the back end of the fuel cycle and anticipated future trends are highlighted.

The future nuclear power programme and the associated spent fuel management programme in the Peoples Republic Bangladesh is under review. 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 spent fuel management 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 spent fuel management, close international co-operation is desirable to resolve the upcoming challenges.

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## VERIFICATION OF THE SCALE MODULAR CODE SYSTEM FOR CRITICALITY SAFETY AND DEPLETION ANALYSES OF WWER SPENT FUEL FACILITIES

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A verification of the SCALE4.4 modular code system [1] and nuclear data libraries for safety analyses and evaluations of WWER spent fuel transportation and storage facilities has been carried out. The system is developed and applied for licensing evaluation of PWR and BWR spent fuel facilities. The application of this system for licensing evaluations of WWER spent fuel facilities requires its entire verification on the basis of both: data from full-scale critical assembly experiments and experimental data for post decay power and heat transfer for the WWER spent fuel transport and storage casks. Another way for verification is comparison with the computation results from licensed reference codes. The complete verification of SCALE involves verification of the different control modules, such as: criticality control module, depletion control module, heat transfer control module etc. The SCALE applicability for licensing evaluations of WWER spent fuel storage facilities could be evaluate after entire verification.

Two SCALE4.4 control modules have been verified: CSAS6 - for criticality safety analyses of WWER spent fuel facilities and SAS2H - for depletion calculations (full spent fuel inventory).

For CSAS6 verification some results for pin power distribution in full-scale critical assembly experiments, carried out on the LR-0 criticality assembly, have been used. The comparison the calculation results from code systems SCALE4.4 and MCNP [2] and experimental data is performed for 260 fuel pins. In this paper some results of the comparison analysis are presented.

Because of lack of the experimental data, the depletion control module SAS2H has been verified on the basis of comparisons with calculation results, obtained by the NESSEL-NUKO code system [3,4]. This system is especially designed for WWER analysis and is thoroughly tested in international nuclear material safeguards, consisting in determination of the spent fuel characteristics by means of actinides' neutron emission measurements and in different benchmark tasks for determination with reasonable accuracy of the plutonium and americium content in the spent fuel. The verification has been based on calculations of WWER-440 fuel assembly with enrichment 3.6% and geometry, material composition and power history given in the benchmark task [5]. The following spent fuel characteristics have been compared and presented: actinide and fission product concentrations and post decay power at two different final burn up points (30 и 40 MWd/kgU) and at three different cooling time steps – after discharging the spent fuel assembly ( $T_{cool} = 0$ ), after 1 year ( $T_{cool} = 1y.$ ) and after 6 years of cooling time ( $T_{cool} = 6y$ ) at away from the reactor basin.



In the paper some results of the critically calculations of WWER-440 and WWER-1000 spent fuel storage and transportation casks performed by the SCALE4.4 and MCNP code systems

are presented. The calculational models have been developed on the basis of real spent fuel casks, designed by the SKODA Nuclear Machinery and the Izorskie zavody. The calculations have been carried out for fresh fuel assemblies in casks flooded with distilled water. The results obtained by these code systems with appropriate neutron cross section libraries for the considered configurations are in good agreement. The deviations in effective multiplication factor  $K_{\text{eff}}$  do not exceed 0.5%.

The carried out calculations, comparisons and analyses of WWER spent fuel facilities proved that:

- The criticality control module CSAS6 (SCALE-4.4) could be used for safety analyses of WWER spent fuel transportation and storage facilities.
- The code MCNP could be used for calculations need in criticality safety analyses of WWER spent fuel transportation and storage facilities.
- The control module CSAS6 (SCALE-4.4) could be accepted for use in determination of isotope inventory and post decay power of WWER-440 spent fuel assemblies after the detailed comparisons with results from the licensed in Germany code package NESSEL-NUKO.
- The good coincidence between the SCALE and MCNP results show their reliability and applicability for criticality safety analyses of casks for transportation and storage of WWER spent fuel.

The verified computer code packages and nuclear data libraries for control and assessment of reliability and nuclear safety of WWER spent fuel transportation and storage facilities is being offered for the purposes of the Nuclear Regulatory Agency in Bulgaria.

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**REGULATORY APPROVAL OF USED FUEL DRY STORAGE FACILITIES IN CANADA – AN ONTARIO POWER GENERATION CASE STUDY**

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In Canada all used nuclear fuel from commercial power-producing reactors is currently stored at the reactor sites, either in wet storage pools or in dry storage facilities. The resolution of long-term management of used nuclear fuel in Canada is designed to be addressed by the Nuclear Fuel Waste Act which came into force on November 15, 2002. In compliance with this Act, a utility-organized, separately-incorporated Nuclear Waste Management Organization has been established and has initiated a three-year study of possible long-term management approaches, including continued reactor site storage, centralized storage and geologic disposal in the Canadian Shield. Following completion of the study, the federal government will select an approach which the Nuclear Waste Management Organization will then implement.

Until such time as an alternative management approach becomes available, reactor site storage will continue, and thus, dry storage facilities will need to be constructed and expanded. Additional wet storage facilities are not envisaged.

Ontario Power Generation (OPG) has been storing used fuel at the Pickering Used Fuel Dry Storage Facility, which is located within the eight-reactor Pickering Nuclear Generating Station site, since 1996. At the end of 2002 the Pickering facility had 260 dry storage containers in storage containing about 2000 tonnes U. A similar facility has recently started operating at OPG's Western Waste Management Facility site to provide additional storage for used fuel discharged from the eight reactors at Bruce Nuclear Generating Stations A and B. A dry storage facility for the four Darlington reactors is currently undergoing an environmental assessment and has a targeted in-service date of 2007. The OPG-designed dry storage container is a steel lined, steel shelled concrete container weighing about 60 tonnes empty, and designed to contain 384 10-year old Candu fuel bundles (9.5 tonnes). Following filling and draining, the dry storage container lid is welded in place and the container is backfilled with helium.

The main focus of this paper is the regulatory approval process for the recently-licensed Western Used Fuel Dry Storage Facility (WUFDSF). This facility is designed to provide interim storage for up to 1940 DSCs (containing up to 14300 tonne U).

The WUFDSF, which is located within the boundaries of OPG's Western Waste Management Facility (WWMF) near Tiverton, Ontario was given regulatory approval to construct in January 2000, and the approval to operate in August 2002. Each approval was granted by the Canadian Nuclear Safety Commission (CNSC) as an amendment to the WWMF licence, which had previously only allowed storage and processing of Low and Intermediate Level Waste at the WWMF.

An Environmental Assessment (EA) under the Canadian Environmental Assessment Act, a pre-requisite to the Construction Approval, was initiated in 1997. After extensive public consultation and EA reviews by several federal agencies, the federal Minister of Environment gave EA approval in April 1999. A formal public hearing process was not required. Although the Project had wide general support from local elected representatives, one local group that was opposed to the Project from its inception challenged the EA approval in court. This prolonged process ultimately concluded with a favourable outcome for OPG.

The Preliminary Safety Report for the Facility was originally submitted to the CNSC in January 1997. While the licensing process could continue, it could not conclude until the legal challenge to the EA process was resolved.

This paper presents the details of the licensing and approval process for the WUFDSF, including: the regulatory requirements for licensing; an overview of the submissions to CNSC; the significant milestones; the technical (licensing) issues encountered and their resolution; the challenges, some predictable and some unforeseen; and the lessons learned. The paper will also provide a general overview of Ontario Power Generation's dry storage programme.

**SPENT FUEL MANAGEMENT IN CHINA**

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China has made good progress in the development of nuclear power. Qinshan NPP-I has been operated more than 10 years and Daya Bay NPP has been operated near 10 years. This year three new units in China have been put in operation. Other five units will be connected in grid before 2005.

Spent fuel from NPPs in China is in wet-storage on site. Since the capacity of storage facility in Daya Bay Npp is near fully occupied, transportation of spent fuel from Daya Bay to reprocessing plant is planned to be conducted soon.

This article presents briefly the status and trends of spent fuel management in China, the activities carried out and problems faced with. Research programmes established at the China Institute of Atomic Energy, such as the application of burnup credit technology for storage and transport of spent fuel, and R&D on high burnup fuel and MOX fuel, are also introduced in this paper.



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## STATUS AND PROSPECTS FOR SPENT FUEL MANAGEMENT IN CHINA

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The main emphasis of China's nuclear industry has been shifted to peaceful uses of nuclear energy since the 1980s. Up to now, 7 units in 5 NPPs have been constructed and 4 units in 3 NPPs are under construction. The present total installed nuclear capacity is about 5600 MWe. Data of NPPs in Taiwan province is left open for the time being.

With the development of the nuclear power, the amount of spent fuel discharged from NPPs is increasing rapidly in China. Chinese government pays its attention to the security of spent fuel, and carries out the policy of nuclear non-proliferation and guarantees the safety to the public and environment all the time.

### I. Current status of the commercial spent fuel storage

Two commercial nuclear power plants(3 units) are in operation in China, and have discharged spent fuel for several nuclear fuel cycles. All the discharged spent fuel store in the at-reactor (AR) storage pools with wet storage. By the end of December 2002, there are 232 spent fuel assemblies in the pool of Qinshan phase I (CN-1 unit), while 824 assemblies in the pools of NPP Daya Bay (CN-2, CN-3 unit). The accumulated amount of spent fuel generated by NPP Qinshan and NPP Daya Bay is about 445 t HM.

For the limitation of the storage pools capacity, NPP Daya Bay is planning to transport the spent fuel stored at-reactor pools to northwest area for wet storage since 2003. The Everclean Environmental Engineering Corporation(EEEC) will be in charge of this transportation using NAC-STC dual-purpose casks that can be used for both shipping and storage.

It is worth while to note that CANDU reactor in Qinshan phase III has different fuel structure and different discharged ways of spent fuel compared with other reactors we have had ever before in China. We will still pay attention to this problem.

### II. The amount of NPP spent fuel in the near future

Currently, there are approximately 60 t HM spent fuel discharged from NPPs every year in China. The NPPs under construction will start its commercial operation in succession in the near future after their construction stage, thus the amount of discharged spent fuel will be obviously increased from now on. There are 4 units, Qinshan phase II unit 1(CN-5 unit), Ling'ao unit 1 and 2(CN-6, CN-7 units), Qinshan phase III unit 1(CN-8 unit), get to their criticality in the last two years, some of them are in commercial operation now. It is estimated that the annual generation rate of spent fuel for the years 2003 and 2004 will be about 160 t HM and about 250 t HM respectively.

### **III. Spent fuel reprocessing**

In order to use the limited resources more efficiently, China is in favour of a closed fuel cycle including spent fuel reprocessing. A pilot spent fuel reprocessing plant is now under construction in China. China will construct a commercial reprocessing plant as appropriate, taking into account the scale of nuclear power development and the amount of spent fuels in storage.

### **IV. Legislation**

The management of spent fuel is related to many aspects, such as national policies and regulations, technology background, nuclear fuel cycle option, nuclear material control and waste management, and so on. To insure the safety and lawful use of nuclear materials, the State Council passed Regulations on Nuclear Materials Control of the People's Republic of China in 1987. This Regulation requires that every organization holding nuclear material must apply for the license of nuclear material awarded by the CAEA .

Moreover, we are working over some aspects related to the management of spent fuel, such as transportation, waste conditioning and disposal, etc. Certainly the necessity and the form of the spent fuel fund should be involved in.

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**SPENT FUEL STORAGE FACILITIES IN THE CZECH REPUBLIC \***  
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There are six nuclear power reactors operating in the Czech Republic. Four of them type WWER-440 in the Dukovany NPP (put into the commercial operation in 1985-1987), first unit type WWER-1000 in the Temelin NPP (in the eighteen-month-trial operation since 2001) and the second one of the same type is on the start-up stage.

### **Storage Concept**

The concept of the preparation and construction of the spent nuclear fuel storages has been changed several times since the 80s of the last century. The last one was agreed by the government in May 2002 in the document called “Concept of the disposal of radioactive waste and nuclear spent fuel“. This document elaborates the following basic principles of the nuclear fuel cycle back-end strategy:

- Wait and see approach (construction of the spent fuel storage facility enables to get into the waiting position enabling to postpone the decision whether to recycle or to use the spent fuel in transmutation technologies).
- Spent fuel will be stored in dry storage facilities, in storage casks or in dual-purpose casks (transport and storage).
- Priority is to place storage facilities at the nuclear power plants. The locality Skalka will serve as the back up site (underground storage).
- Concurrently with the preparation of the deep repository possibilities of the nuclear spent fuel recycling and adoption of new technologies, aimed at decreasing the nuclear spent fuel volume and toxicity, will be pursued.
- The deep repository will be put into the operation in 2065.

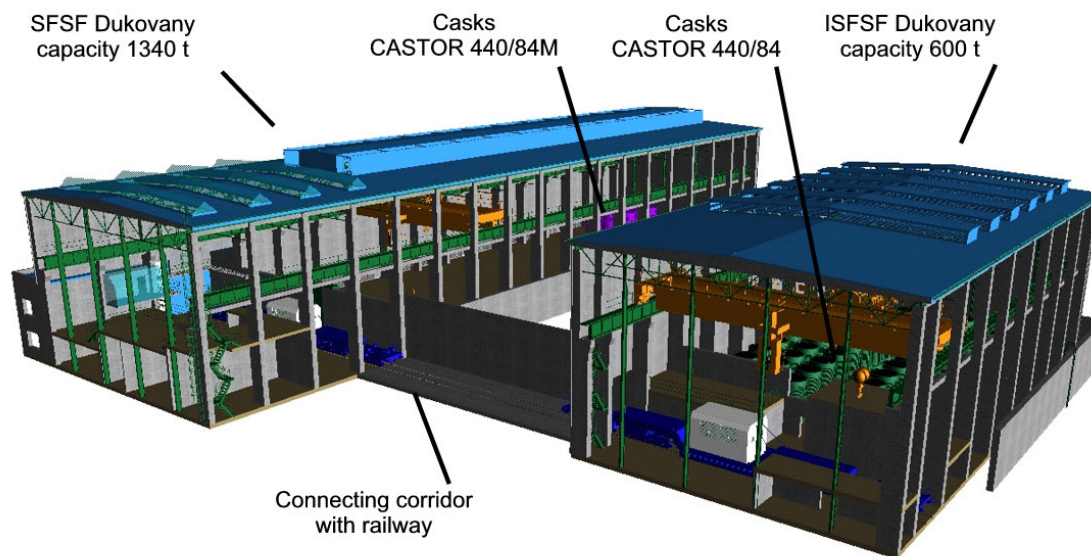
### **Dukovany NPP – Interim Spent Fuel Storage Facility (ISFSF)**

Spent fuel is stored in dual purpose (transport and storage) casks of the CASTOR 440/84 type. The ISFSF has been in the operation since 1995. Its storage capacity of 600t will be exhausted by the end of 2005.

### **Dukovany NPP – Spent Fuel Storage Facility (SFSF)**

The preparation of the 1340 t SFSF was started in 1997. The EIA process took 2 years. CEZ, a. s., got the site permit in 2000. The State Office for Nuclear Safety issued the permission for construction in November 2002, the construction approval is expected in 2003.

Duration of the construction according to the schedule is 23 months in order to start trial operation at the beginning of 2006.



*SFSF and ISFSF Dukovany*

In course of the bidding process the cask type CASTOR 440/84M for 84 fuel assemblies was chosen. The contract for delivery of 25 casks was signed in May 2001. Start of construction of the spent fuel storage building is expected in 2003 after the construction approval enters in force.

### **The spent fuel storage facility (SFSF) in the Temelin NPP site**

The capacity of the pools situated near the reactor enables the operation of the Temelin NPP until 2014. For this reason the preparations for the new SFSF with the capacity of 1370 t began in 2002 by carrying out the feasibility study for various storage of locations in the nuclear power plant site. According to the project plan, the site permit is supposed to be obtained until July 2006, construction approval until September 2008 and the SFSF should be put into the trial operation in February 2014.

### **Skalka Site - back up location for the Temelin SFSF**

Skalka site used to be an underground variant of the former central SFSF. Nowadays it is kept as a back up site for the spent fuel from Temelin NPP, since it is the most suitable location of all, which were screened outside the premises of the nuclear power plants in 1993-1997. The State Office for Nuclear Safety allowed the SFSF to be located in Skalka in January 2000. In March 2001 the authority issued a site permit for the SFSF location in this site. The validity of the site permit expires in 2011.

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## COMPUTER SIMULATION OF 3D STEADY AND 2D TRANSIENT THERMAL LOADING OF CASTOR 440/84 USING FEM

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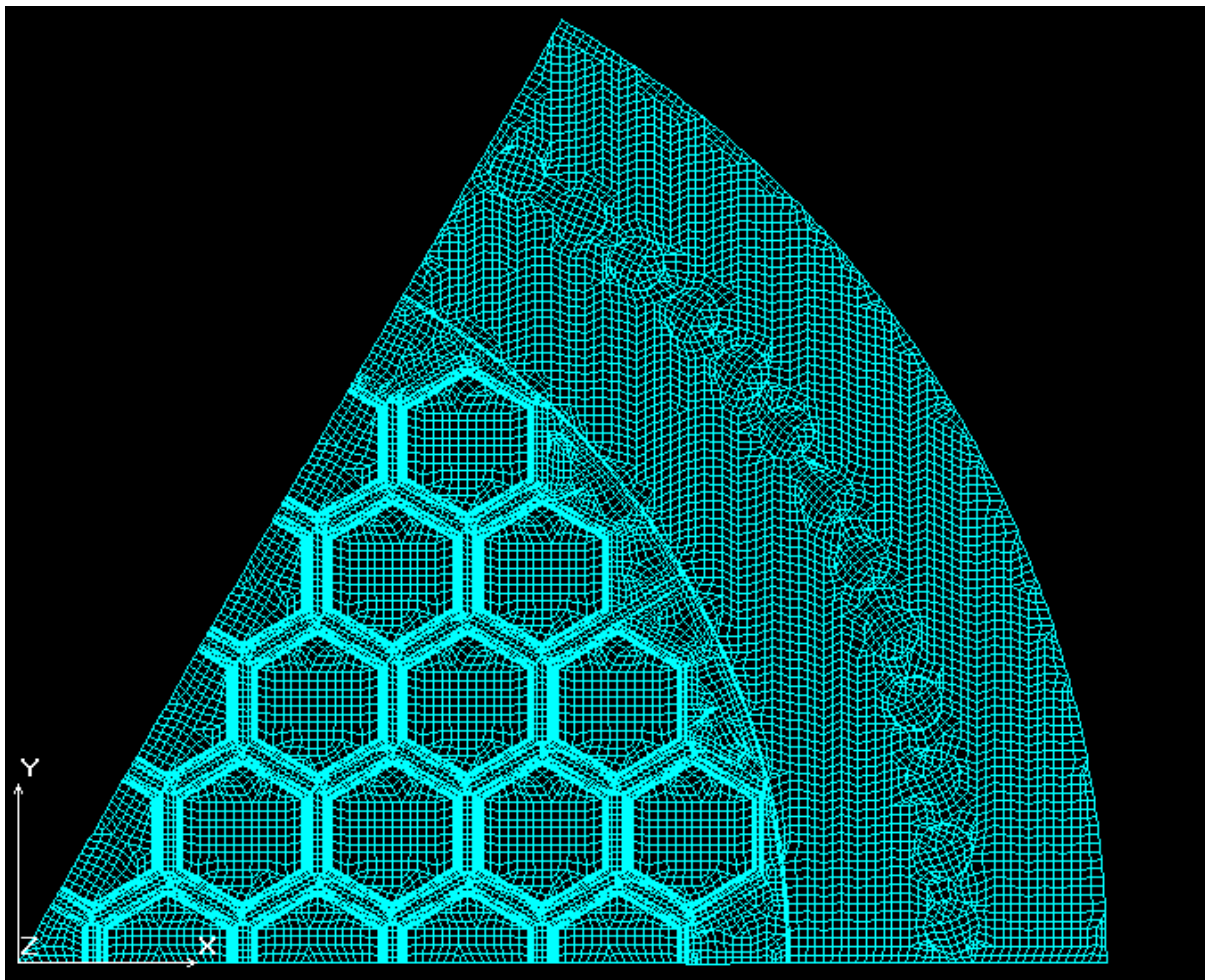
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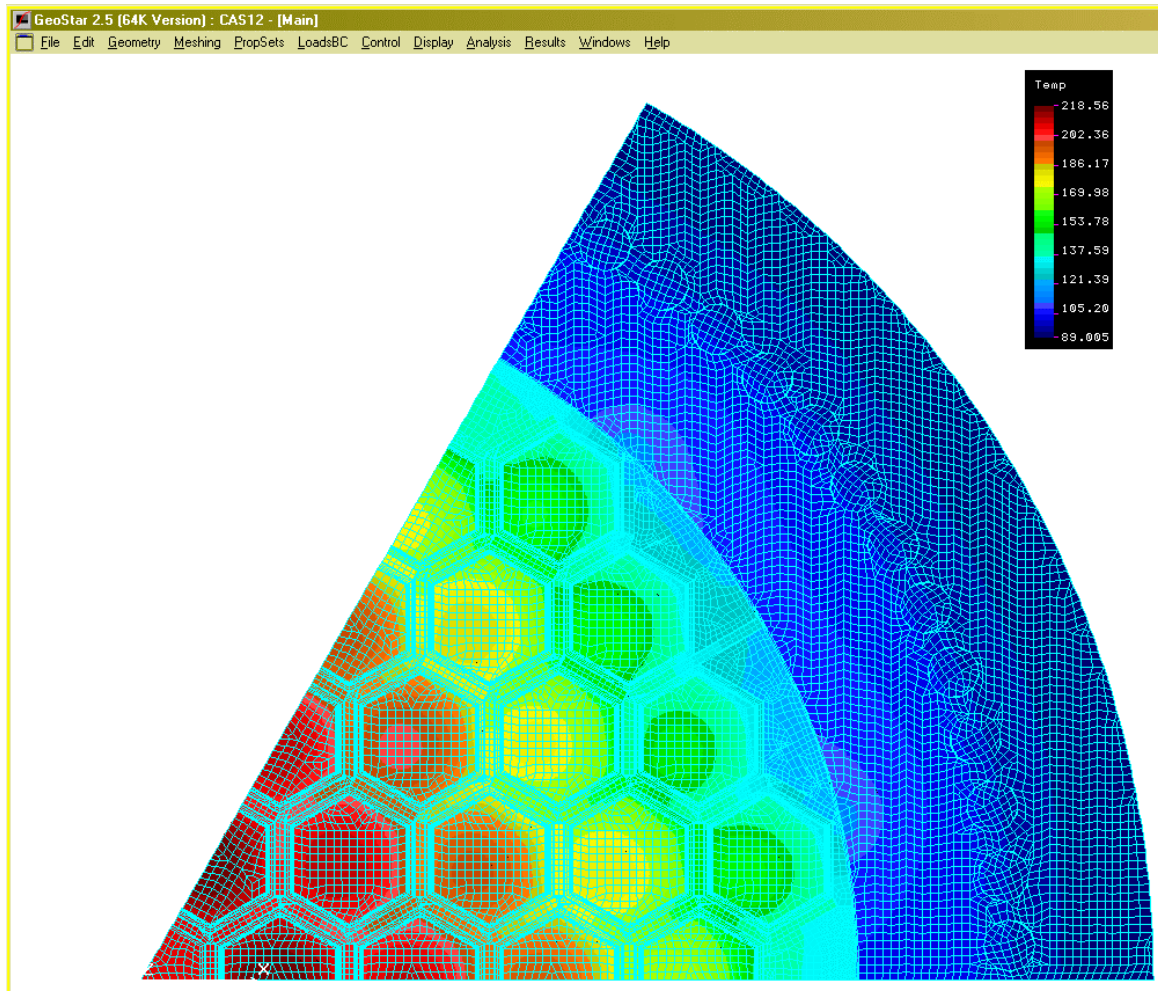
Paper describes system of computer codes developed at the NRI Rez, plc. for the CEZ a.s. company aimed to the realistic best estimate evaluations of the temperature field in the CASTOR 440/84 container, which is used for the Dukovany NPP spent fuel. Brief theoretical description of basic equations and their numerical solution using finite element method creates basis of the paper. Validation of developed computer tools was performed on industrial application and extensive parametric testing. Main validation results are connected with 2D transient modelling of real Dukovany container equipped with thermocouple measurement. The future plans for practical usage and innovations of described analytical tools are presented.

Accompanying pictures:

1. Nodalisation example of the CASTOR.



## 2. Example of steady-state temperature calculation result.



## **CURRENT STATUS OF THE SPENT FUEL MANAGEMENT FROM POWER REACTORS IN THE CZECH REPUBLIC (LICENSING AND OPERATIONAL EXPERIENCE)**

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The paper describes the current situation in the spent fuel (SF) management in the Czech Republic focussed on the licensing and operational experience:

- experience of SUJB gained during the licensing procedures of existing and planned SF storage facilities,
- operational experience with dry spent fuel storage in NPP Dukovany.

In the Czech Republic the use of dry storage technologies is and will be the main option for the SF storage from power reactors. Therefore the national regulatory body (SUJB) gained in recent years extensive experience in the field of the licensing of dry SF storage facilities. The only operational SF storage – the Interim Spent Fuel Storage Facility (ISFSF) at NPP Dukovany was commissioned in January 1997. Due to the requirements of the new Atomic Act, which had been adopted also in January 1997, a new operational license had to be issued in March 2001. Prior the license issue an extensive set of related documents was approved or evaluated including the revision of preoperational safety report. The operational license for ISFSF was issued for the period of 10 years.

As an important part of the whole licensing procedures for ISFSF was the issue of design approval for transport and storage for cask CASTOR 440/84. The design approval for storage is usually issued for 10 years, while the transport design approval for 3 years only.

Since mid 90's of last century it became obvious, that due to the limited capacity of ISFSF (600 tHM) it will be necessary to ensure additional storage capacity for the SF generated during the operation of NPP Dukovany no later than in 2005. After the evaluation of several SF storage strategies it has been decided to build an additional Spent Fuel Storage Facility (SFSF) with capacity of 1340 tHM next to the existing ISFSF at the Dukovany site. SUJB has participated at all stages of licensing procedure required by the Atomic Act; i.e. by the:

- EIA process, which was successfully finished in November 1999,
- issuing of siting license from December 1999,
- issuing of the construction license from October 2001.

The second part of the presentation deals with the operational experience of ISFSF owner – Czech Power Company ČEZ, a. s., NPP Dukovany.



The empty cask is shipped from the manufacturer's facility by railway to the corridor of the reactor building. After rotation from horizontal to vertical position, it is lifted up to the reactor hall to its service place. There several operations starting from the disassembly of the lids and ending with the check of cleanness are performed. Subsequently the cask is transported again using a lifting beam into the loading pit near the reactor and the storage pool. Then the refuelling machine loads the cask with 84 fuel assemblies, inspections including inspection with participation of IAEA inspectors are carried out and the cask is covered under water level with its primary lid. Cask is then lifted from the pit and its surface is fully decontaminated. The cask is transported back to the service place in the reactor hall where all tests of its leaktightness and other control operations are carried out. After completion of all checks, the cask is fully assembled including its third lid and it is secured using IAEA's seals. In such a configuration and following the dosimetry measurement and appropriate decontamination, the cask is transported on railway wagon into the ISFSF. Following the transport to the entrance corridor, the cask is rotated into vertical position and then it is transported directly to its storage position. At this place it is connected to the pressure and surface temperature monitoring system. Detailed description of this monitoring system is presented in the paper as well.

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## KEY ISSUES FROM THE FRENCH R&D PROJECT ON THE LONG-TERM EVOLUTION OF THE SPENT NUCLEAR FUEL IN CONDITIONS OF INTERIM DRY STORAGE

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In France, the interim dry storage (long-term –100 y's– or classical –50 y.) of spent nuclear fuel is studied in the framework of the 1991 Law for the management of spent fuel in the back-end of the fuel cycle. Indeed, only two third of the total annual budget of spent fuel is currently reprocessed in order to balance the Pu annual budget. The other third is stored before further decision concerning its ultimate fate. Several engineering projects are on going in France on long-term storage and lead to address operational questions to R&D. These questions concern the issue of the retrievability of the spent fuel waste package at the end of the storage stage, the monitoring of the fuel packages during the storage and the applicability of potential re-conditioning process. Moreover, the evolution of the radionuclides source term is a major operational question for the safety assessment of the dry storage.

The R&D PRECCI project is performed in CEA with the support of French utilities EDF and FRAMATOME. Some of its objectives are to qualify and quantify the long-term evolution of the spent nuclear fuel (SNF) under conditions of dry storage. This paper will present a synthetic status of the PRECCI project concerning:

- (1) The chemical and physical states evolution of the SNF in a closed system (without external exchange of matter)
- (2) Determination of the long-term cladding mechanical properties after irradiation
- (3) Potential evolution of the SNF in contact with air.

The major outcomes and consequences on the potential design of long-term storage facilities will be enlightened.

- (1) Chemical and physical states of the SNF in a closed system may evolve due to radioactive alpha decays and temperature history, even without any contact with the external medium.

No microscopic swelling due to irradiation damage or chemical evolution of the spent fuel is expected during dry storage in a closed system [1].

Therefore, the evolution of the fuel microstructure and rod internal pressure would essentially depend on the helium behavior in the fuel pellet. In fact, helium which is produced by alpha-decays in the spent fuel, may reach large concentrations particularly in MOX fuels [2]. The few data on helium diffusion available in literature are highly dispersed. Experimental studies were launched in order to investigate helium diffusion mechanisms at low temperatures and the influence of the irradiation defects. First results will be presented.

At the temperature range expected in the dry storage, the mobility of radionuclides in the grains would be governed by diffusion due to alpha self-irradiation [2]. Theoretical and experimental approaches are today developed in order to evaluate the importance of this phenomenon. First assessment of its significance will be given.

- (2) The cladding failure, which controls the release of radionuclides during the dry interim storage, is governed by creep strain induced by the fuel rod internal pressure. New creep law and breaching criteria have been developed in order to model and predict the mechanical properties of the irradiated Zircaloy-4 in the temperature and stress ranges, encountered during dry storage.
- (3) In accidental conditions, corresponding to the loss of integrity of the container during dry storage, the SNF would be exposed to air after the cladding failure. It is well known that  $\text{UO}_2$  oxidises in  $\text{U}_3\text{O}_8$  in air with a significant volume increase (+38%) which transforms pellet into powder. However, very little data is available in the low temperature range ( $< 200^\circ\text{C}$ ) relevant for dry storage and for high burnup UOX or MOX fuels. Experiments are on going in CEA to address these issues. Some results will be presented.

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**DRY STORAGE TECHNOLOGIES: KEYS TO CHOOSING AMONG METAL CASKS, CONCRETE SHIELDED STEEL CANISTER MODULES AND VAULTS**

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The current international trend towards expanding Spent Fuel Interim Dry Storage capabilities goes with an improvement of the performance of the proposed systems which have to accommodate Spent fuel Assemblies characterized by ever increasing burnups, fissile isotopes contents, thermal releases, and total inventory.

Due to heterogeneous worldwide reactor pools and specific local constraints the proposed solutions have also to cope with a wide fuel design variety.

Moreover, the Spent fuel Assemblies stored temporarily for cooling may have to be transported either to reprocessing facilities or to interim storage facilities before direct disposal; it is the reason why the retrievability, including or not transportability of the proposed systems, is often specified by the Utilities for the design of their Storage systems and sometimes by law.

This paper shows on examples developed within companies of AREVA Group the key parameters and elements that can direct toward the selection of a technology in a user specific context.

Some of the constraints are ability to dry store at once a large number of spent fuel assemblies, readily available, on a given site. No urgent need for further move of the fuel is foreseen.

Then clearly a Vault Type Storage system developed and implemented by SGN is an excellent solution : It combines passive safety with immediate large capacity, that allows quick amortization of fuel receiving equipment. In addition the versatile storage position can easily accept in the same facility different fuel types, and also Intermediate and High Level Waste.

This is the reason why a vault system is often a preferred solution for a long-term dry interim centralized storage, for a multiplicity of spent fuel.

It can be also a choice solution when the ISFSI stands on a site that is dedicated permanently to many different nuclear activities.

In most cases, the producers of spent fuel require a large capacity that is cumulated over many years, each reload at a time. Then the key criterion is maximum modularity.

Furthermore, the up front capital costs requirement for this type of solution is minimal, so depending on the chosen discount rate of the investor, they have an additional attraction. That smaller modules allow to change course in back end policy more easily.

Priority of modularity yields two other solutions, dual purpose metal casks of the TN24<sup>TM</sup> family or dual purpose or single purpose concrete shielded welded canisters such as NUHOMS®. These solutions, implemented by COGEMA LOGISTICS, TRANSNUCLEAR Inc. and FRAMATOME-ANP, are very flexible and have been adapted also to quite different fuels.

Among what influences the choice, we can consider :

- In favor of metal casks :
  - Minimal ancillary equipment.
  - Ready to move to final or centralized repository or reprocessing or other ISFSI.
  - Compact systems.
  - Easy rearrangement.
  - Easy handling.
- In favor of concrete shielded canistered based systems :
  - Economics when initial quantity is sufficient to spread out up front equipment.
  - Significant cost – Shielding advantage.
  - Easy local production of the relatively light canisters.

Both approaches, when transportable, are also a factor for public acceptance because of the non-permanent characteristics and because transport licensing refers to internationally recognized rules, standards and methods.

## TRANSPORT AND INTERIM STORAGE CASKS IN SWITZERLAND

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The Swiss utilities have chosen two different ways for the management of their spent fuel after initial on-site cooling:

- either reprocessing at La Hague plant (COGEMA) and Sellafield plant (COGEMA)
- or interim storage at the Central Interim Storage Facility called « Zwischenlager Würenlingen AG » (ZWILAG).

Following international call for tenders, COGEMA LOGISTICS were awarded contracts for the supply of dual-purpose transport and storage casks for the interim storage of various spent fuel assemblies.

All these casks belong to the family of the TN 24<sup>TM</sup> dual purpose spent fuel storage casks in operation in the USA and in Belgium as well. They offer utilities a modular solution for the interim storage of spent fuel in robust metal casks which are fully suitable for off site transports. This flexible product can be readily adapted to suit individual user needs.

The Leibstadt Nuclear Power Plant (KKL) has purchased six licensed dual-purpose TN97L spent fuel casks (97 BWR type fuel assemblies capacity). Three of them are already in operation at ZWILAG. COGEMA LOGISTICS has also delivered a dual-purpose TN52L spent fuel casks (52 BWR type fuel assemblies capacity) presently used for transport of spent fuel for reprocessing.

The Goesgen Nuclear Power Plant (KKG) has purchased four licensed dual-purpose TN24G spent fuel casks (37 PWR type fuel assemblies capacity). They are all in operation at ZWILAG.

The Mühleberg Nuclear Power Plant (BKW/KKM) has purchased 2 TN24BH spent fuel casks (69 BWR type fuel assemblies capacity). At the time of this abstract, cold trials are carried out involving the shuttle transport cask TN9/4 procured by COGEMA LOGISTICS as well.

This paper will present the main features of these casks and the main steps of their development and implementation :

- Main features of the casks :
  - The basic structure is a thick steel cylindrical forging with a welded on forged bottom and two forged steel lids. Containment and gamma shielding features of the cask are mainly provided by this basic structure.
  - 4 or 6 trunnions are attached to this structure for handling, tilting and tie down.

- Inside the cylindrical cavity, a Boron aluminium basket is fitted and provides a structural support for the fuel assemblies and criticality control.
  - Surrounding the cylindrical cavity, a resin layer is encased in an outer shell and provides the neutron shielding features of the cask. Heat conductors ensure the thermal evacuation of the heat from the main shell to the outer shell of the cask.
  - A leak tightness monitoring system and an anti-aircraft crash cover (when needed) are installed during the storage period of the cask.
  - A set of shock absorbing covers is fitted to the flask for transport operation, as well as lateral impact limiters for some cask design.
- Main steps of their development and implementation :
    - manufacturing and licensing (transport and storage)
    - transport and on-site handling cold trials using the ancillary equipment and the real vehicle
    - spent fuel loading and transport to the storage site
    - storage.

## SECONDARY PHASE FORMATION DURING INTERACTION OF SPENT NUCLEAR FUEL AND CLADDING MATERIAL DURING LONG TERM STORAGE AND DISPOSAL LEACHING TESTS

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The long-term storage of spent nuclear fuel in geologic repositories envisages the isolation of the material from environmental factors of influence by a multi-barrier system. The cladding of the material is considered to act as one of the barriers in the near field close to the surface of the material, and therefore plays an important role in the isolation of radionuclides from the geosphere during final storage.

The interaction of cladding and the surface of spent fuel samples (UOX , BU 30 GWd/t and MOX , BU 12 GWd/t) was investigated with static long-term leaching tests. The cladding tubes of spent fuel rodlets were perforated in order to simulate defects. Subsequently, the samples were immersed in water for a period of about 4 years. Analysis of the bulk solution composition was performed with ICP-MS after an immersion period of 3 weeks and 4 years. After 4 years of treatment the fuel elements were separated from the solution and cut into specimen for investigation with optical microscopy and SEM-EDX, as well as measurement of physical parameters (gap width, micro hardness).

A macroscopical optical investigation of the pre-set defects showed the formation of a layer of yellow and brown products on the outer surface of the cladding close to the openings, indicating that the spent fuel matrix was partially oxidized to U(VI) which was leached into the solution.

An investigation of the cut specimens with optical microscopy showed that the formation of alteration products in the fuel-cladding gap proceeded in a different way for the MOX and UOX material sample. In the case of the UOX fuel, a porous layer of deposits is formed on the cladding which results in irregular gap width data. Almost no alteration products were found on the inner cladding surface of the MOX material. The Zircaloy alteration layer was investigated with SEM-EDX.

A qualitative and quantitative assessment of the alteration product formation was performed by thermodynamic equilibrium calculations using the overall solution properties of the system as input. Since the immersion tests were conducted under an N<sub>2</sub> atmosphere with an O<sub>2</sub> content of 4 %, oxidation of the UO<sub>2</sub> matrix to U(VI) is expected. The plutonium present in the MOX sample was initially present in the +IV oxidation state (PuO<sub>2</sub>). The calculations predict that UO<sub>2</sub> is converted to U(VI) (with UO<sub>2,333</sub> as the solubility-controlling phase) and PuO<sub>2</sub> is present as Pu(IV), both in solution and in the solid, under the conditions given. The Zircaloy material is thermodynamically predicted to form the corrosion product ZrO<sub>2</sub> under the experimental conditions. Since ZrO<sub>2</sub> forms a monoclinic phase at ambient temperatures, it



is expected that the material present in the layer will not form binary mixtures of the solid-solution type with the oxides  $\text{PuO}_2$  and  $\text{UO}_2$ , but possibly with the oxides of U(VI). This suggests that  $\text{ZrO}_2$ , once formed on the cladding surface, might show a possible retention capacity for hexavalent U, but not so for Pu and several of the fission product elements in the spent fuel. This hypothesis is coherent with the observed high release rate of fission products in the case of the MOX sample.

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## ADVANCED SPENT FUEL STORAGE POOLS

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Spent fuel from Power Reactors is currently stored either in at-reactor pools or in independent spent fuel storage installations (ISFSI) using wet or dry storage technology. During the past 15 years, storage capacity of at reactor pools was increased using high density spent fuel storage technology. To achieve maximum capacity, storage racks were replaced in many of the power reactors in operation at least once, some of them went through even various reracking cycles.

Independent spent fuel storage installations were established either at the site of power reactors or away from them. They use either storage pools or dry technology, the latter in form of metal casks and concrete silos or vaults.

Storage of spent fuel from power reactors must be safe for the public and must protect the environment from its radioactive content. For this purpose, adequate regulation was developed and is available to be applied. However, advances in fuel and core design as well as the need for extended storage periods require frequent re-assessment of the available spent fuel storage technology.

Improved fuel utilization leads to elevated burn-up resulting in higher heat generation of the spent fuel in the longer term. Also spent MOX fuel generates considerable more heat than spent uranium fuel unloaded from the power reactors 15 years ago. As most of the mechanisms which could endanger fuel integrity are temperature dependent, effective heat removal is one of the challenges spent fuel storage systems have to face.

In order to make most efficient use out of high density storage equipment, designer want to take credit from the actual burn-up of the spent fuel. Existing methodology to analyze burn-up credited spent fuel storage racks for criticality safety is being further developed to be applied in the design of dual-purpose casks or multi-purpose-canisters.

For poisoned high density storage equipment, long term stability of the material as well as efficient neutron absorption is required. Degrading absorbers which contaminate the coolant of fuel pools and primary circuits and fuel assemblies getting stuck in swollen storage cells are well known problems which already caused considerable headaches to many operators and still continue to do so. Although limited in boron content, borated stainless steel has extensively proved as sufficiently effective and extremely stable neutron poison material.

Over and above the basic requirements for maximum safety for operators and the public, logistic in fuel reception is an important aspect to be addressed with the design of independent spent fuel storage facilities. The aspect to keep the operators exposure to radiation as low as reasonably achievable requires expeditious reception of spent fuel and its transfer to the dedicated storage location.

Independent wet storage facilities are known for many years to comply best with most of the expectations as described above. However, the assumption to need active cooling systems and to generate secondary waste had caused vendors and users to look favorably at dry storage systems. Despite the remarkable development achieved in dry storage technology, most of the spent fuel generated up to now is stored in fuel pools, either at the reactors or in independent installations.

One of the latest achievements in wet storage technology is used in Framatome-ANP's wet storage facility design as currently being performed for the new spent fuel pool building to be constructed at the Goesgen Nuclear Power Station in Switzerland. The advanced design of this independent spent fuel storage facility provides a passive cooling system which reliably removes the heat generated by the spent fuel by natural circulation through air cooled heat exchangers. This progressive design makes extensive use of well balanced safety technology with largely passive safety features developed for Framatome-ANP's Boiling Water Reactor SWR1000. Clearly designed to enable the enforcement of strict foreign material exclusion strategies and state of the art pool water purification equipment reduces operational waste generation to negligible minimum.

Due to the passive nature of the operating system, the number of active components which require maintenance is substantially reduced. In addition, due to advanced acquisition methodology of operational data, the frequency of maintenance activities can be determined under consideration of the actual usage. This usually leads to a considerable reduction of human intervention and the time needed to act in radiation areas reducing considerably waste generation and dose burden to personnel.

As a facility dedicated to receive and to store fissionable material, regulatory requirements on access controls as defined by the competent regulators are to be met. The advanced wet storage facility design of Framatome-ANP meets safely with all the requirements for safeguards and physical protection including terrorist activities and actions of sabotage. It provides best for safe and reliable storage of spent uranium and MOX fuel for extended periods. Maintenance and repair concepts are available which allow to predict, if correctly applied, such fuel pools to be operable for periods up to 100 years.

## UO<sub>2</sub> AND MOX FUEL BEHAVIOUR IN LONG TERM DRY STORAGE

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On April 27, 2002, Germany's new nuclear energy act, which replaces the act dated 1959, came into force. Instead of promoting the nuclear energy use its purpose is an orderly termination on the basis of an average plant operational life time of 32 years. Focal points related to fuel assemblies are as follows:

- As of July 1, 2005 the delivery of spent fuel elements for reprocessing will be prohibited and nuclear waste disposal restricted to a final repository.
- The operators of nuclear power plants will be requested to construct on-site storage facilities and to keep spent fuel assemblies until a final repository is available.

Since the new on-site installations will be mainly of the dry type, dry cask storage will gain more and more importance. A central requirement on dry cask storage systems is to meet safety functions. For fuel assemblies, this means integrity of the assembly structure and exclusion of systematic cladding failures for the envisaged storage period.

This paper mainly deals with cladding integrity assessment, since the behaviour of fuel assembly structure is largely determined by the mechanical design and has shown to undergo no adverse timely changes. The cladding behaviour, however, is dependent on material properties and irradiation as well as boundary conditions of reactor operation and storage. Cladding integrity is usually assessed by means of creep laws and rupture behaviour, which can be obtained under various experimental conditions. A stress limit of 120 N/mm<sup>2</sup> and a maximum strain of 1% are used in Germany to avoid systematic cladding degradation. The strain calculations are based on a creep formula of non-irradiated fast creeping cladding, whereas the straining capability is derived from similar irradiated material. The straining capability was determined by short-time creep-burst experiments on high burnup PWR fuel rods with commercial Zry cladding. Burnup was up to 64 MWd/kgU and oxide layer thickness up to 100 µm. The experiments revealed a high straining capability of about 2% and no degradation with regard to increased oxide thickness and hydrogen content of the cladding. The straining capability has been also evaluated for M5™ cladding by creep tests on irradiated material and strains of at least 1 % without defect are guaranteed for a large range of temperatures.

Future fuel cycle developments are featured by an even higher discharge burnup and intensified use of MOX fuel. As a result, rod inner pressures and storage temperatures will increase and a considerable percentage of rods will exceed the 1%-strain limit if calculated by

creep laws based on non-irradiated material. To cope with this more demanding situation, an unnecessary conservatism with regard to the experimental verification, for example short term testing at high stress levels or the definition of a constant strain capability neglecting temperature benefits, has to be replaced by a more realistic methodology. Therefore an extensive program is in progress on M5™ in order to derive more predictive creep laws and rupture criteria for irradiated cladding. On the other hand, variations of the storage period in the range of 20 up to 100 years do not affect integrity assessment methodologies.

## USE OF BURNUP CREDIT IN CRITICALITY SAFETY DESIGN ANALYSIS OF SPENT FUEL STORAGE SYSTEMS

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It is well known that the use of Burnup Credit (BUC) in criticality safety design analysis of spent fuel storage systems significantly impacts the design of the system. BUC is defined as the consideration of the change in the fuel's isotopic composition and hence in its reactivity due to the irradiation of the fuel. Using BUC means to identify that isotopic composition and hence that burnup which just results in the maximum neutron multiplication factor allowable for the system, including all mechanical and calculational uncertainties. This burnup is the minimum burnup necessary for fuel to be loaded in the system. Since the isotopic composition at given burnup depends on the initial enrichment of the fuel, the minimum burnup is usually given as a function of the initial enrichment. The graph of this function is commonly named as "loading curve". Thus, application of BUC to a spent fuel storage system consists in implementation of three key steps:

- Determination of the isotopic composition as a function of burnup and initial enrichment.
- Criticality calculation and evaluation of the loading curve.
- Quantification and verification of the actual burnup of the fuel to be loaded into the system.

The main considerations of the first and the second step will be discussed.

The isotopic composition is predicted by means of depletion calculations. To perform such calculations the parameters describing the fuel design characteristics and the fuel depletion conditions have to be defined. In addition the cooling time that may be credited (e.g., in BUC applications to spent fuel storage/transport cask systems) has to be specified. These parameters will be discussed with particular attention being given to the sensitivity of the neutron multiplication factor of the storage system to variations in the parameters and conditions characterizing the depletion conditions. These parameters and conditions are: Specific power and operating history, fuel temperature, moderator temperature and density, presence of soluble boron in the core (PWR), use of fixed neutron absorbers (control rods, burnable poison rods, axial power shaping rods), use of integral burnable absorbers (gadolinium or erbium bearing fuel rods, IFBA rods). It will be shown how a bounding approach can be obtained for the impact of these parameters on the reactivity of the storage system.

The criticality calculation procedure consists in the following main steps:

- Isotopic selection and validation.
- Validation of the criticality calculation code applied.
- Sensitivity studies on the reactivity effects of axial and horizontal burnup profiles of fuel assemblies.

- Determination of the criticality acceptance criterion (maximum allowable neutron multiplication factor including the impacts of all the mechanical and calculational uncertainties) and determination of the loading curve.

The fundamentals of isotopic selection will be defined, and a survey of the benchmark experiments available for isotopic validation and validation of the criticality calculation code applied will be given. Since the parameters and conditions characterizing the benchmark experiments are usually different from the parameters and conditions describing the spent fuel storage system of interest, a method of checking the applicability of such experiments to the storage system will be briefly described. This method bases the applicability on the similarity of sensitivity coefficients which are defined for the underlying nuclear data characterizing the isotopic compositions and their effect on the spent fuel reactivity [1].

The fact that the axial burnup distribution in a fuel assembly is non-uniform must be considered in the analysis of the storage system. The difference between the system's neutron multiplication factor obtained by using an axially varying burnup profile and the system's neutron multiplication factor obtained by assuming a uniform distribution of the averaged burnup of this profile is known as the "end effect". It has been shown that the end effect may become positive for averaged burnups greater than about 15 MWd/kg U [2]. In view of the vast variety of axial profiles due to the vast variety of irradiation histories that are occurring already in one core alone, the task is given to find a bounding description of the end effect.

It has been shown [2] that non-uniform horizontal burnup profiles resulting from radial variations in the neutron flux in an operating reactor, which are mainly due to leakage at the core periphery and to burnup differences between neighboring assemblies, might lead to an increase of the reactivity of the spent fuel storage system of interest.

The basics of determining a loading curve which bounds the reactivity effects of axial and horizontal burnup profiles will be described.

The validity of the loading curve is tied to the storage system design parameters on which the evaluation of the loading curve is based, the depletion parameters assumed, and the axial and horizontal profiles analyzed. Limitations in the applicability of the loading curve and methods of checking the continued validity of the analysis assumptions will be discussed, therefore.

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## MANAGEMENT OF SPENT FUEL FROM POWER AND RESEARCH REACTORS USING CASTOR AND CONSTOR CASKS AND LICENSING EXPERIENCE IN GERMANY

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During the seventies a first idea of dry storage of spent fuel in casks arose at the GNB mother company GNS. The well-known CASTOR<sup>®</sup> cask design with ductile cast iron (DCI) as cask body base material was developed for the dual purposes of storage and transport. After only five years of developing and testing, the first storage license was granted for a CASTOR<sup>®</sup> cask at the centralized storage facility Gorleben.

Meanwhile CASTOR casks are used at 19 sites on four continents. Spent fuel assemblies of the types PWR, BWR, WWER, RBMK, MTR and THTR as well as vitrified high active waste containers (HAW) are stored in these kinds of casks. Up to now, more than 660 CASTOR casks have been loaded for long-term storage.

The two decades of storage have shown that the basic requirements, which are safe confinement, criticality safety, sufficient shielding and appropriate heat transfer have been fulfilled in each case and, of course, the experience of 20 years has resulted in improvements of the CASTOR cask design.

Starting in the middle of the nineties, the new GNB cask line CONSTOR was developed with special consideration to an economical and effective way of manufacturing by using conventional mechanical engineering technologies and common materials. The cask concept also fulfills all design criteria for transport and storage given by the IAEA recommendations and national authorities. By the end of 2002 forty CONSTOR casks have been delivered and 30 of them have successfully been loaded and stored.

In the past the German disposal concept was mainly related to reprocessing of the spent fuel in France and Britain and further storage of the high active waste (HAW) in the existing central intermediate storage facilities Ahaus and Gorleben. However, in the consensus with the German Government the German utilities declared a change in their spent fuel policy from reprocessing to direct dry storage. In order to minimize the connected number of transports, the erection of decentralized, on-site intermediate dry storage facilities for spent fuel assemblies was agreed.

For the intermediate on-site dry storage different storage facility solutions will be realized. In addition to the several kinds of storage facilities with storage halls, which have been applied for 40 years, a number of utilities applied for a so-called "Interim Storage Facility", this means storage of the casks under mobile shielding covers made of concrete for a maximum of 8 years.



According to the fact that all applicants decided to use CASTOR<sup>®</sup>-casks (CASTOR<sup>®</sup> V/19 for PWR or CASTOR<sup>®</sup> V/52 for BWR) the licencing procedures were able and necessary to be standardized. Thus, the scope of expert assessment can be reduced.

The standardization of the licensing procedure is based on the standardization of the application documents. The structure of the documents and the documents proving the fulfillment of the licence requirements for the cask are the same for all applicants. With additional documents only the differences between the types of storages will be taken into account.

Since the consensus in 2000 three On-site Interim Storage Facilities and three On-site Intermediate Storage Facilities have been licensed. The licences for the other facilities are expected in 2003.

In this paper different types of casks are presented connected with different types of storage solutions. Experiences gained by the large number of cask loadings and more than 4000 cask years of storage will be summarized. The presentation of recent and future development shows the optimization potential of the CASTOR and CONSTOR cask families for safe and economical spent fuel management.

## TRANSURANUS SIMULATION OF WWER CLADDING CREEP UNDER DRY STORAGE CONDITIONS

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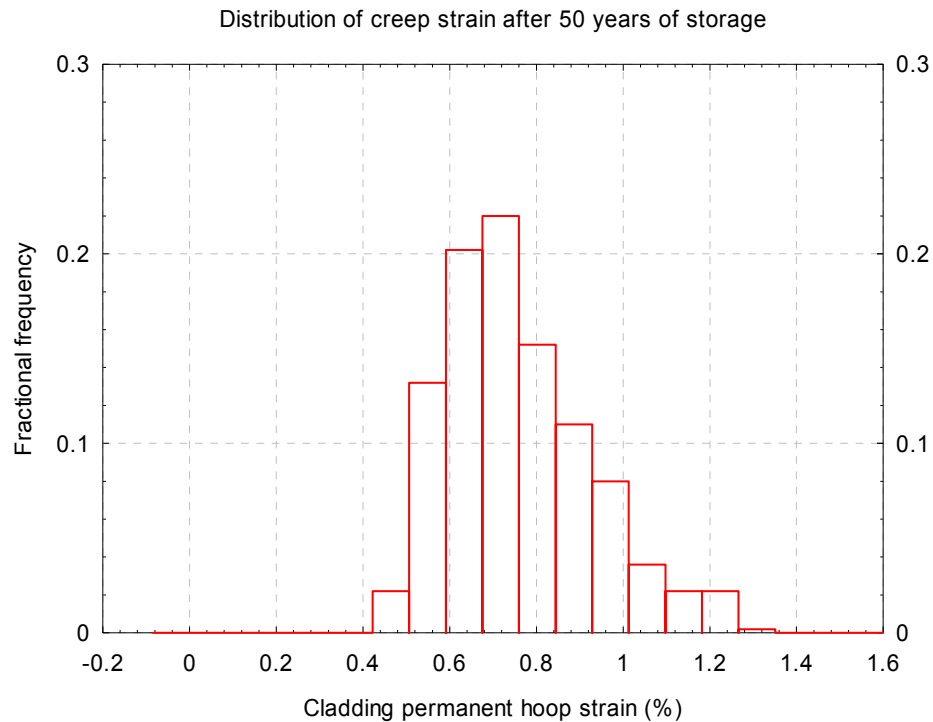
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Limited cladding creep deformation is one of the main safety criteria for the long-term storage of the spent nuclear fuel. To keep the fuel rod hermetically closed the safety limit of the permanent hoop strain is 1%. The fulfilment of this criterion is generally verified by simplified models considering the complex behaviour of a high burnup fuel rod with conservative assumptions. Hence, the extension of the scope of an available fuel analysis code should support a more precise prediction of the claddings' mechanical performance under long-term storage conditions.

In the framework of an IAEA TC project [1] the thermal hydraulic computations for the MVDS (Modular Vault Dry Storage) facility of WWER-440 spent fuels were completed by fuel rod analyses, as well. The TRANSURANUS fuel behaviour code [2] with a new creep rate correlation [3] was considered as the most appropriate tool to define the proper mechanical loading and deformation of the WWER cladding by the end of a 50-year storage period. The extended code is capable to calculate the variation of the internal pressure in the fuel rods and also to determine the distribution of strains and stresses in the cladding during the irradiation in the reactor, during the cooling in the at-reactor (AR) pool and during also the long dry storage period. Hence, the application of the code is supposed to predict more realistic results than the simplified methods applied before to evaluate the cladding creep.

The question of the minimum necessary period of the spent fuel rods' cooling in the at-reactor (AR) water-filled pool before the dry storage was investigated in the study. According to the performed computations the permanent deformation of the cladding in the MVDS tube strongly depends on the wet storage period. It was proved that the 1% strain margin could be exceeded if the spent fuel was transhiped from the AR pool to the MVDS in one year of cooling. However, the creep strain is negligible during the long-term storage if the spent fuel was cooled for at least two years in the water-filled AR pool. As the creep strongly depends on the temperature, the effect of the thermal hydraulic computations' uncertainties produced on the hoop strain of the cladding was investigated by means of Monte Carlo analyses (Figure 1).



**Figure 1. Response distribution of creep strain to cladding temperature variation ( $\sigma = 5\%$ ) at the end of a 50-year dry storage after one year cooling in the AR pool. Results of TRANSURANUS simulations.**

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## SAFETY ANALYSIS OF THE C30 SPENT FUEL CASK FOR THE EXTENDED RANGE OF LOADING PARAMETERS

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At Paks NPP, Hungary, the C30 cask mainly used for the transport of the assemblies from the at-reactor wet service pool to the storage facility, so it primarily serves as an interface to the storage place. Occasionally, it is used for transport from a unit to an other one. Originally, the C30 cask was licensed to the following parameters:

Max. number of assemblies	30
Total heat production	< 15 kW
Average burnup of the cask load	< 33 GWd/tU
Burnup of a single assembly	< 40 GWd/tU
Initial enrichment	< 3.7 %
Cooling time	> 2.5 years

Due to the introduction of the new fuel assemblies, the higher burnup achieved by the old type fuel assemblies and to the need of the lower cooling time the NPP applied for the modification of the license which allow the extension of the parameter range described above. This study examine of the impact of this extension on the subcriticality, on the radiation shielding and on the radiolysis (hydrogen production in the water by the absorption of gamma radiation). The desired range of the parameters extends to 50 GWd/tU burnup of a single assembly and 0.5 years of cooling time for three type of fuel. These types of fuel the old type fuel assemblies (no enrichment zoning, max. enrichment is 3.6 %), the new Russian designed assemblies with enrichment zoning (average enrichment is 3.82 %) and the BNFL designed fuel with enrichment zoning and with gadolinium (average enrichment is 3.9 %) are examined. The main findings of this study can be summarized as follows.

The subcriticality requirements are met by the C30 cask loaded with fuel assemblies listed above for normal and accidental condition as well with fresh fuel assumption. The results of the thermohydraulic analysis were utilized in the analysis.

Based on the constraints from the heat physical analysis, we examined the influence of the extended parameters on the neutron and photon dose rates. The maximum acceptable number of loaded assemblies with extended parameters was investigated. Loading strategies are suggested for different values of burnup and cooling time to meet the radiation protection safety requirements.

The hydrogen production from the water by gamma radiation was examined. The time limit for the explosion hydrogen concentration was determined by a high degree of conservatism. Proposals were made for the case of inadvertent conditions to avoid achieving this concentration.

## SPENT FUEL DRY STORAGE IN HUNGARY

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Safe supply of energy is assured by operation of the Paks NPP, which provides approximately 40% of the total domestic electricity generation. Fresh nuclear fuel has been imported first from the Soviet Union and later from Russia. The subject of spent fuel is a very challenging issue in nuclear power generation. Yearly 40-50 t HM of fresh fuel is loaded in the 4 reactors of Paks NPP, and accordingly, the same amount of spent fuel is unloaded. According to the original fuel strategy the Soviet Union (later Russia) undertook not only to supply new fuel but also accepted of the spent fuel for reprocessing. This arrangement included also that all products of the reprocessing process (all radioactive wastes, plutonium, uranium) were supposed to stay in the Soviet Union. Conditions, laid down in the original concept were changed later: first when spent fuel only after 5 years decay cooling was accepted for shipment instead of 3, and later, when the shipment costs were raised considerably.

Recognizing the fact that the return of spent fuel is becoming uncertain, Paks NPP from the early 1990s, started to promote the construction of an interim spent fuel storage facility, recognizing that this would improve the operational safety of the plant, by eliminating the temporary difficulties caused by delays in the fuel return. The selection of the storage type was made in 1992, which was followed by licensing and construction of the facility. GEC-ALSTHOM's (UK) Modular Vault Dry Storage (MVDS) system was chosen. The commissioning of the first Phase of the facility was finished in 1997, and filling the first vault started at the same time. Last shipment of spent fuel to Russia took place in April 1998.

At the end of 2002, the facility has 11 vaults, thus providing storage space for 4950 assemblies. At the end of December 2002, there are 3017 assemblies in storage. The high number of spent fuel movements did not lead to serious operational events, and doses to the environment and the operators remain below the design values. The facility in its present form is expected to provide space required for the storage of spent fuel generated until the end of 2006. New capacity needs to be constructed until that time. This construction phase means a new stage in licensing also, because except the Site Permit, all other permits were issued for 11 Vaults. Therefore a new licensing process needs to be started.

The development of the strategy for the closure of the fuel cycle is in progress. One basic element of this strategy is the safe solution of spent fuel interim storage. Finalization of the strategy and its acceptance is expected in the near future.

## SPENT FUEL STORAGE IN INDIA

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**1. Introduction:** Indian Nuclear Power Programme has grown from twin BWR reactors at Tarapur to 12 PHWRs of 220 MW each working at various locations. Additionally six PHWR reactors are in advanced stage of construction. India has gone for closed nuclear fuel cycle option to reprocess the spent fuel for recovery of Uranium and Plutonium to meet ever-increasing energy demand. There is a programme to achieve 20,000 MW installed nuclear power capacity by the year 2020. It is planned to construct Spent Fuel Storage Facilities (SFSFs) as the need arises.

Wet storage of Spent Fuel has been the main mode of storage in India pending reprocessing. This paper describes various important issues related to design, construction, licensing and operational experience of spent fuel storage facilities at Tarapur and Kalpakkam.

**2. Design of Spent Fuel Storage Facility (SFSF):** The new SFSFs are located at existing nuclear site to take maximum advantage of existing infrastructure already in place, nearness to reactor and approved site for nuclear facility.

**Layout:** The smooth handling of trailer loaded with shipping cask is ensured by providing two independent airlocks and 7 m wide road with proper turning radius. Location of cask decontamination area, pool water cooling and polishing system and effluent handling system have been suitably decided based on ease of operation and optimum space utilization. The active and in-active services have been suitably located.

**Seismic design:** IAEA TECDOC-250 is followed for seismic design of SFSFs. The independent SFSF are designed for OBE (Operating Basis Earthquake) level of earthquake. The soil-structure interaction has been considered as per ASCE 4-98 standard. The pool structure has been designed for hydrodynamic response during seismic event. The design of various mechanical system and components is carried out as per the respective design codes and standards based on their safety classification and seismic categorization.

**Fuel Pool:** The fuel pool is designed as fully underground structure with single walled construction on hard rock/ strata. Minimum biological shielding of 3m. above a stack of trays has been ensured. The radiation level at water surface and at working level, when fuel pool is filled to its design capacity is less than  $10^{-3}$  mGy/hr (0.1mR/hr).

**Lining & Leak Detection System:** The pool walls and floor are lined with SS 304L plates to avoid ingress/egress of pool water. The SS liner plates are welded to closed channel inserts (embedment) inside concrete wall. The fuel pool is provided with an elaborate leak collection and detection system. Provision has been made for inspection of pool liners.

**Pool water-cooling and polishing System:** The maximum pool water temperature is limited to 42°C in normal condition and 60 °C in accidental conditions. Suitable heat exchangers have been provided to remove the decay heat generated from spent fuel bundles. A polishing system consisting of a cartridge type filter, a cation cartridge (disposable type) and a mixed bed unit (regenerative type) have been provided to remove the fission product impurities like Cs<sup>137</sup>, Sr<sup>92</sup> etc with a turnover time of less than 72 hr. This ensures pool water activity within limits as specified by regulatory authority.

Provision has been made to incorporate disposable type mixed bed unit for polishing of pool water.

**Fuel Handling:** Single failure proof EOT Crane has been provided to handle shipping cask. The reach of the crane has been limited to cask handling area of the pool by layout.

**Other Facilities:** Cask decontamination, handling of low level and intermediate level liquid waste, assorted solid waste, make-up DM water treatment plant are other systems provided for smooth operation. SFSFs are also provided with safety systems such as Ventilation System, Fire detection, fire alarms and fire mitigation system, Access Control System and CCTV monitors, Radiation monitoring system, Class-III and Class-II power supply.

**3. Licensing of the Facility:** Design review and safety review is carried out by the independent expert groups and committee by local and national regulatory bodies. The licenses are given in phases for carrying out construction, commissioning and operation of the facility.

**4. Operational Experience:** The Fuel Pools at reactors and reprocessing plants have been in operation for four decades and their performance is satisfactory. There is no major leakage or failure. Some of the pools have been emptied to clear the muck collected at bottom of the fuel pool and to check integrity of the pool liners. The storage of Zircalloy clad fuel is very safe without any noticeable abnormality under wet storage condition. The storage of old fuel bundles, failed during handling because of extended storage did not pose any problem. The pool water chemistry, temperature and activity have been maintained well within the specified limit.

**5. Conclusion:** India has perfected the technology for design, construction and operation of spent fuel storage facility (wet type) meeting all international safety standards.



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## DESIGN OF SPENT FUEL STORAGE OF PROTOTYPE FAST BREEDER REACTOR

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Prototype Fast Breeder Reactor (PFBR), a pool type sodium cooled 500 MWe fast reactor uses mixed oxides of plutonium and uranium as fuel. The fuel is designed for a peak linear heat rating of 450 W/cm and a peak burnup of 100,000 MWd/t. The average residence time of the fuel in the reactor is 2 years.

The spent fuel is discharged from the reactor off-load at a regular interval of 8 calendar months. About 62 fuel subassemblies (FSA) and 28 blanket subassemblies (BSA) are discharged from the reactor at every fuel handling campaign.

The fuel handling system of the reactor is designed to handle spent subassemblies (SA) of decay heat less than 5 kW/SA. Since the decay of spent FSA is much more than this value immediately after shutdown, it becomes necessary to cool the subassemblies in the reactor itself till the decay heat reduces to 5 kW. Storage of spent FSA for a period of 8 months is necessary to reduce the decay heat to acceptable level. For this purpose 156 location is provided in the reactor to store spent SA with regulated flow.

The spent FSA after a cooling period of 8 months in the reactor are unloaded from the reactor, washed to remove sodium. The failed fuel is identified and the FSA with failed fuel are housed in a partially water – filled leak tight containers. Washed spent FSA and failed SA in containers are then stored in spent fuel storage bay in water.

The bay is designed to hold spent FSA and spent BSA till the decay heat is reduced sufficiently from the point of view of reprocessing. In addition the capacity of the storage accounts for storage of one full core in case of emergency. The design of the bay fulfills safety guidelines prescribed by AERB and also relevant international guidelines. The bay incorporates a single tank in two compartments to facilitate maintenance in case of leak. The concrete tank is lined inside with 304 LN sheet and provision is made to detect leak in the liner at the weld seams. The tank leak is collected by providing a bund around the bottom.

The storage bay has provision for preliminary inspection of selected SA and also for spent fuel loading in a spent subassembly transfer vessel.

## SPENT FUEL MANAGEMENT STRATEGY FOR FUTURE NUCLEAR POWER PLANTS OPERATION IN INDONESIA

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The demand for electricity in Indonesia increases by the years. This increase goes along with the rate of economic development, the rate of population growth and the rapid development in the industrial sector. To fulfill this demand for electricity, it is becoming more difficult to depend on the existing resources which are now getting limited. It is, therefore, very important. that steps should be taken to seek outer resources as alternatives[1].

Based on the thought that a Nuclear Power Plant (NPP) is technically safe, reliable, clean and environmentally-oriented, relatively economical, and supported by our being prepared in respect to the human resources and the infrastructures, including the results of the feasibility studies for NPP development completed in 1996, the option of nuclear power could well be the right solution.

Intensive effort have long been undertaken by BATAN for the introduction of NPPs in Indonesia. The candidate sites, which have long been thoroughly investigated and selected, are located at the Muria Peninsula in Central Java. It is planned that the selected candidate site will accommodate several NPPs with a total power generation of about 7,000 MWe[2].

Indonesia shall choose an open fuel cycle strategy until couple decades after the first NPP operation. Each NPP to be capable of providing 3-5 years of spent fuel storage pool at the NPP building. It is planned that the spent fuels will be stored for further 40-50 years in centralized storage facility at the NPP site. After that, spent-fuels will be stored in long term storage facility, eventually, arrangement must be made for deed geological disposal[2,3]

According to the result of NPP feasibility studies, for achieving 7,000 MWe power generation there are three cases of scenario among many combination of 600 MWe and 900 MWe class NPPs[4]:

- Case I gives the power combination of 600 MWe x 2 Units x 6 Stages.
- Case 2 gives the combination of following power:  
600 MWe x 2 Units x 3 Stages  
900 MWe x 2 Units x 2 Stages
- Case 3 gives the power combination of 900 MWe x 2 Units x 4 Stages

Each case is assumed that contract and construction are proceeded with as a twin unit with an interval of one year.

By assumption that the Light Water Reactor (LWR) is preferred reactor and the quantities average of spent fuel for averaged 1 year operation for LWR (PWR, BWR) are 13 Mtu for 600 MWe and 20 Mtu for 900 MWe, the reactor pond storage capacity for 5 years and the capacity of centralized storage facility for 50 years can be estimated. For case 1, the pond capacity for each reactor on the each stage is 45 Mtu, and the capacity of centralized storage facility for 50 years is 4,320 Mtu. For case 2, the pond capacity for each reactor on the stages of 1, 2, and 3, is 45 Mtu, and on the stages of 4 and 5 the capacity is 100 Mtu, and the capacity of centralized storage facility for 50 years is 3,825 Mtu. For case 3, the pond capacity for each reactor on each stage is 100 Mtu and the capacity of centralized storage facility for 50 year is 7,200 Mtu.

Although the final disposal site is not yet decided, a preliminary study on a final disposal site has, however, been carried out. In the mean time it is expected that new technology on the recycling that proliferation proof will be available and if the economic dictate to do so the open cycle may be converted into the close one.

The spent fuel management strategy for future NPP operation in Indonesia is described.

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**THERMAL CREEP TESTS OF BWR AND PWR SPENT FUEL CLADDING**

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Creep is a major issue of spent fuel integrity for long term dry storage. This paper presents the NUPEC programme of creep tests and some results of it. The first stage of the programme has been started in 1999 and will be carried out until March 2004 under sponsorship of the Ministry of Economic, Trade and Industry(METI). The main objective of the thermal creep tests is to derive a creep calculation formula for BWR and PWR spent fuel cladding.

**[Test Scope]**

Clarifying test of hydrogen effect on cladding creep behaviour using unirradiated specimen  
Creep and creep rupture tests on unirradiated and irradiated claddings

**[Specimen Parameter]**

Fuel and cladding type: BWR 8x8 fuel, Zr-lined Zry-2, RX  
PWR 17x17 fuel, low-Sn Zry-4, SR

Specimen burnup : BWR 0, 47~60 GWd/t  
PWR 0, 34~46 GWd/t

Specimen configuration: 80~100mmL segmented, defueled,

Hydrogen content: BWR 30~140ppm  
PWR 40~310ppm

**[Creep Test Condition]**

Temperature: 330~420 degree C

Stress: 30~300 MPa

Duration time: Max. 4,800 hrs

**[Methodology]**

Pressurization method: Two types  
Welding sealed capsule containing pressurized gas  
Fitting to pressurization system

Cladding diameter measurement: Laser profilometer

**[Test Results]**

Creep rate of the cladding, which contains more hydrogen than solubility limit, is smaller than that of as-received cladding.

Creep rate of the cladding, which contains less hydrogen than solubility limit, is not smaller than that of as-received cladding.

Thermal and stress cyclic condition had no effect on creep rate in steady state condition.

Creep rate of hydride radially reorientated cladding was almost same as that of non-reorientated cladding.

There were no difference of creep rate between claddings with homogeneously precipitated hydride and one with heterogeneously precipitated hydride (hydride rim).

Creep rate of spent fuel claddings was one order smaller than that of unirradiated claddings.

**[Coming Plan]**

Until March 2004, creep tests will be done up to duration time of 560 days.

Creep strain calculation formula for irradiated fuel cladding will be derived using the parametric creep test results considering hydrogen effect.

When creep rupture occurs, strain and failure mode will be observed.

## VERIFICATION OF DUAL-PURPOSE METAL CASK INTEGRITY

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The interim storage is for producing an adjustable time period of the fuel reprocessing. Japanese utilities have plans to start interim storage by 2010. The interim facilities are off site of the reactors and will have no hot-cell. Therefore, it should be important to confirm integrity of cask during storage and transport after storage.

NUPEC has been conducting the dual-purpose metal cask verification test since 1999 under sponsorship of the Ministry of Economic, Trade and Industry (METI). The purpose of this verification test is to confirm integrity of safety functions during interim storage and under transport conditions after the storage. The test is subdivided into two categories, material property tests and system verification tests. The first is to verify degradation of materials of which cask consists. The second is to confirm the integrity of safety functions during storage and transport.

Seal ability of cask lid system during transport after the storage should especially be focused on. Based on 3D-analysis and the results of 1/3-scaled model test, sliding and opening between cask lids and their flanges will be slightly occurred. Thus cask seal ability with aging metal seal should be confirmed by full-scale test. Criteria for opening and sliding distance of metal seal that is assumed degradation should also be established.

This paper presents the general plan and the current status of this verification test. Test name, its goals and current status are summarized on Table-1 and 2. Evaluation for 3D-analysis based on the results of scale model test and establishment of the criteria for seal system will be also discussed in the paper.

TABLE 1: MATERIAL PROPERTY TESTS

TEST	GOALS	CURRENT STATUS
Body & basket metals	To confirm the effects on the strength due to corrosion and SCC	No significant corrosion and SCC by iodine compound in 60 year-storage
	To confirm the effects on the strength due to over ageing and creep	Over ageing treatment for A1 basket materials
Neutron Shieldings	To establish the evaluation method for the degradation of shielding	No significant degradation under closed circumstances
	To confirm chemical changes due to heating and irradiation	No significant changes effecting on the shielding is observed.
Metal seals	To establish the evaluation method for the degradation of sealing	The sealing is estimated with; $\Delta F = C_1 \times T_x (C_2 + \log t) + C_3$ $C_2 = 10-14$ (Al lining)

TABLE 2: SYSTEM VERIFICATION TESTS

TEST	GOALS	CURRENT STATUS
Drop test	Confirmation of the sealing ability with aged metal seals	Designing and manufacturing a full-scale model Drop tests using 1/3scale model
	Verification of assumptions, modeling and methods, which used in the analysis	3D dynamic impact analysis by DYNA3D
Thermal test	Confirmation of the sealing ability with aged metal seals	Design and manufacturing a full-scale lids section model
	Verification of assumptions, modeling and methods, which used in the analysis	3D thermal and structural analysis by ABAQUS

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## POST IRRADIATION EXAMINATIONS OF TWENTY YEARS STORED SPENT FUEL

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The post irradiation examinations (PIE) of stored spent fuels were carried out to evaluate fuel integrity during storage. The spent MOX (Mixed Oxide) fuels irradiated in commercial BWR and the spent PWR-UO<sub>2</sub> fuel irradiated in commercial PWR were used. The burnup of MOX fuels was about 20 MWd/kgHM and that of PWR-UO<sub>2</sub> was about 58MWd/kgHM. Five fuels from MOX fuel assembly were stored under two different storage conditions for twenty years: Three fuel rods of them were stored under wet condition (in water). Other two rods were stored under dry condition (in air) in capsule after cutting to short length segments. For PWR-UO<sub>2</sub> fuel, a fuel pin was stored under dry condition (in air) for twenty years. For MOX fuel, the pre-storage characterization data of fuels, which were symmetry position in the fuel assembly during irradiation, were used to compare with the post-storage PIE results. The following PIE items were carried out for MOX fuel in this study. For wet storage fuel : a) Visual inspection of the cladding outer surface, b) Puncture test. For dry storage fuel : c) Atmosphere gas analysis in capsule, d) Ceramographic examination to observe oxide layer thickness on outside/inside cladding and pellet microstructure such as grain size. For PWR-UO<sub>2</sub> fuel, PIE items including cladding were carried out to evaluate fuel integrity during storage. The results show no marked difference after storage of MOX and PWR-UO<sub>2</sub> fuels. However, further examination is required to conclude on spent fuel integrity during long term storage.



## CURRENT STATUS OF R&D PROGRAM OF SPENT FUEL STORAGE TECHNOLOGY IN CRIEPI

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### 1. Introduction

In 1997, a new study program of demonstrative tests for interim storage of spent fuel had been started, which is mainly related to concrete cask storage technology. Concrete cask storage system is considered to essentially have an economical advantage. This paper introduces the current status of CRIEPI's R&D program of spent fuel storage technology on concrete cask, particularly aiming at the realization of dry storage away from reactor in 2010.

### 2. Demonstration program for qualification of concrete cask performance (FY1998-2003)

To propose "safety standards for concrete cask structures, systems, components", the following demonstration program for qualification of concrete cask performance has been in progress.

Schedule of this program is shown in Table.1.

#### *a. Basic design of Japanese type concrete cask*

Basic design of two types of concrete cask, reinforced-concrete type (RC type) and concrete filled steel type (CFS type) to store the high burn-up spent fuel have been terminated.

#### *b. Fabrication of full-scale concrete cask*

Two types of full-scale concrete cask and multi-purpose canister are being fabricated to apply to the demonstration tests.

#### *c. Demonstration tests*

Heat removal test of concrete cask considering the normal, ab-normal and accidental events, impact test of the metal canister are planned. Seismic test with scale-model cask has been also executed.



## ASSESSMENT OF THE STORAGE CONCEPT FOR CONDITIONED SPENT FUEL

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The spent fuel, the essential by-product of the electricity by the nuclear power reactors, is a highly radioactive waste. Therefore, the development of methods for effective management of this large amount of spent fuel is an important and essential task worldwide. Currently, the Advanced spent fuel Conditioning Process (ACP) is being developed at KAERI as an alternative for effective conditioning of spent fuel for the long-term storage and eventual disposal. This technology involves the process of the reduction of uranium oxide by the lithium metal in a high temperature molten salt bath. In this process, some fission product elements with the high radioactivity and heat load such as cesium and strontium are dissolved in the lithium chloride molten salt. The goals of the ACP is to recover more than 99.8% of the actinide elements and to minimize the radioactivity, heat load and volume of spent fuel to be placed in the interim storage and geological repository.

In order to evaluate the storage characteristics of the conditioned spent fuel, a PWR type spent fuel with its initial enrichment of 4.5 wt% of u-235, discharged burn-up of 48 GWd/tU and 10 years of cooling time was selected as a reference base considering the domestic storage status of spent nuclear fuels. As shown in Table 1, the radioactivity and heat power of conditioned spent fuel decrease to 20.7 % and 26.3 % of those of the unconditioned spent fuel, respectively. The volume of the conditioned spent fuel is decreased to about a quarter of the initial spent fuel by removing the structural materials from the spent fuel assemblies.

Four types of spent fuel storage systems, such as the metal cask, the concrete cask, the horizontal modular system and the modular vault dry system (MVDS), are currently in use world-wide.[1] As described previously, the maximum storage capacity for the conditioned spent fuel would be extended larger than that of the existing spent fuel storage conditions. In order to confirm the adaptabilities of the existing storage system to the storage of the conditioned spent fuels, the safety and compatibility investigations considering the existing systems should be conducted. The storage concepts were established considering the inner basket unit for the conditioned spent fuel as shown in Figure 1.

Assessments of the storage concept were carried out for the four fields of safety analysis as followings. The criticality calculation results show that the existing basket structures satisfy the sub-critical requirements under two hypothetical individual accident conditions. According to the shielding calculation results based on the conditioned spent fuels, the required shielding thickness could be reduced due to the radioactivity decrease. In the point of the temperature evaluation for the normal condition, it is estimated that the maximum cavity temperature of the other storage systems is higher than that of MVDS. Therefore, it is found that MVDS is more advantageous in view of the thermal safety. In the structural evaluations for typical hypothetical conditions such as tip-over, earthquake and drop, the main structures may be needed to reinforce, because the increase of the loaded weight of the storage system would require more structural integrity [2]. In conclusion, the conceptual storage evaluation results show that the MVDS could be adopted in storing the conditioned spent fuels with the extended storage capacities.

TABLE 1. CHARACTERISTICS OF CONDITIONED SPENT FUEL

Metal Storage Cask	Spent Fuel(SF)	Conditioned Spent Fuel (CSF)
Radioactivity (Ci/tHM)	5.461E+05	1.128E+05
Thermal power (W/tHM)	1.724E+03	4.53E+02

\*The basis of PWR spent fuel : enrichment 4.5%, burn-up 48GWd/tU, cooling time 10 years.

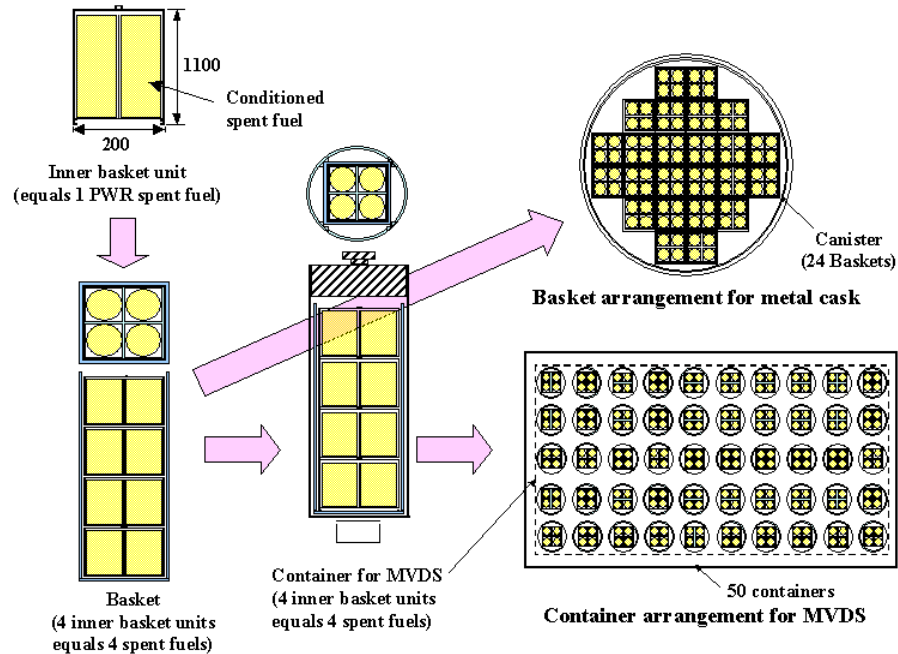


Figure 1. Storage Concept for Conditioned Spent Fuel

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## NEW INTERIM SPENT FUEL STORAGE FACILITY AT IGNALINA NPP

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Lithuania has adopted a cautious to the management of spent fuel. Radwaste Management Strategy was issued and approved by the Lithuanian government in February 2002. The Strategy defines a long-term strategy for management of radwaste including spent nuclear fuel. It is intended to use storage facilities for spent nuclear fuel which dual purpose with would be suitable both for long-term storage and transportation. The Strategy says also that it is necessary to carry out the following:

- the long-term scientific research programme “Feasibility study on disposal of spent nuclear fuel in Lithuania” shall be prepared and implemented;
- possibilities to create a deep geological repository in Lithuania for disposal of spent nuclear fuel shall be analyzed;
- possibilities to create a regional repository using joint efforts of several countries shall be analyzed;
- possibilities to dispose spent nuclear fuel in other countries shall be analyzed and economic justification of such disposal shall be specified;
- possibilities to prolong the storage period for spent nuclear fuel for up to 100 and more years shall be scrutinized.

At the moment INPP does not have a disposal route for its spent nuclear fuel and used fuel assemblies are taken from the existing ponds after not less than 5-year cooling, loaded into storage casks by type CASTOR/CONSTOR RBMK-1500 and taken into interim storage at a site adjoining the power plant. The capacity of the existing store is for 72 casks. It is possible to extend this capacity to 80 casks. But his store has insufficient capacity to contain all of the fuel that will accumulate in the reactor core and storage ponds prior to and at shutdown of the plant and decommissioning of the reactors cannot start until the removal of fuel from the cores is completed. As no long term disposal or reprocessing route is yet available in Lithuania it is proposed to build an additional interim store. The new interim store will contain all of this remaining fuel for a period at least 50 years under adequate containment and biological shielding for fission product decay prior to final disposal. This store will have the capacity to store all of the spent fuel accumulated up to the closure of Unit 2.

These aspects were studied and in accordance with their demands the above mentioned New Interim Spent Fuel Storage at Ignalina NPP shall include following technologies and equipment:

- 
- to remove spent fuel half-assemblies stored in storage/transfer baskets from the cooling ponds;
  - to remove damaged fuel assemblies including fuel debris from the full-assembly cooling ponds, arrange cutting of the damaged assemblies and insertion of the cut assemblies and debris into suitable storage/transfer baskets;
  - to provide an interim storage facility compliant with specified requirements;
  - to provide a safe means of transport of the spent fuel and fuel debris to the interim storage facility;
  - to place the spent fuel and fuel debris into safe and secure interim storage for a period at least 50 years;
  - to arrange for the safe decontamination and/or disposal of the empty storage/transfer baskets if the fuel is removed from these baskets prior to storage;
  - to inspect and identify/record the assemblies prior to storage;
  - to remove external contamination from the surface of the equipment prior to storage;
  - to provide effluent treatment equipment and systems for the treatment of contaminated solid and liquid waste arising at the processing facility;
  - to monitor the storage conditions (including temperature, radiation, gas leakage from the casks, drain water, environment conditions);
  - to provide security arrangements at the fuel store and connections to the existing utilities, railways and roads including fences where necessary;
  - to provide Interim storage facilities for (approximately) 19,000 spent fuel assemblies.

A potential alternative to the interim storage of the spent fuel is reprocessing. At present there are no facilities for the reprocessing of RBMK spent fuel.

The design of the interim storage will anyway permit the retrieval of the spent fuel so that reprocessing would remain an option in the future.

## COMPARISON OF THE MAIN CHARACTERISTICS FOR CASTOR AND CONSTOR CASKS LOADED WITH SPENT RBMK-1500 NUCLEAR FUEL

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There is only one nuclear power plant in Lithuania - the Ignalina NPP (INPP). The INPP possesses two similar units of RBMK-1500 reactors. The INPP reactors were commissioned in December 1983 and August 1987 respectively.

In 1992 decision was made to build an interim dry spent nuclear fuel storage facility at the INPP site. In the early November 1992 tender invitations were extended to nine companies, and finally, the proposal of the German company GNB was accepted to store the spent nuclear fuel outdoors in sealed metal CASTOR RBMK-1500 casks filled with helium. May 12, 1999 the first cask was loaded with the SNF and transported to the storage site. According to the licensing requirements the testing programme for five casks was performed, the report was prepared by operator and approved by VATESI. The license for the operation of the storage facility (from 15 February, 2000) was issued by VATESI February 11, 2000. The licence is issued for 5 years. In total twenty of such cast iron CASTOR RBMK-1500 casks were already loaded with SNF. Agreement for the delivery of 40 metal-concrete CONSTOR RBMK-1500 casks instead of CASTOR RBMK-1500 casks was signed with GNB. Now already about 20 of such casks are loaded with the SNF and delivered to the interim storage facility. The storage site for the 72 casks is the open type facility. To decrease the dose to the public and personal the concrete wall around the casks was built. The CASTOR RBMK-1500 and CONSTOR RBMK-1500 casks are designed for the long-term (up to 50 years) storage of 102 RBMK half-fuel assemblies, which are positioned in the special basket.

The CASTOR RBMK-1500 cask body is made of ductile cast iron. The cask has two lids made of corrosion protected carbon steel. The first lid is provided with a double-barrier sealing system to secure leak tightness. The second lid (guard lid) minimises radiation exposures and ensures the weather and additional corrosion protection of the containment barriers. The body of CONSTOR RBMK-1500 is made of two welded metal cylinders. The space between them is filled with reinforced heavy concrete. The cask has three lids. The first one is fixed to the upper ring and there is a seal there. Two other lids are welded to the body of the cask.

In this paper results on comparison of the main characteristics, such as maximum cladding and surface temperature, dose rate at the surface and at the some distance, and also criticality, for CASTOR and CONSTOR casks loaded with RBMK-1500 spent fuel is presented. Comparison was performed when the cask is just loaded with spent nuclear fuel, and after 50

years of storage (possibly going for disposal) for the same RBMK-1500 SNF: initial enrichment of the fuel – 2 %, burn-up – 20 GWd/MTU, cooling time - 5 years, and the same weather conditions.

It was revealed that maximum cladding temperature is higher for CONSTOR RBMK-1500 cask but surface temperatures are similar for both casks. After 50 year storage in wintertime surface temperature can reach values bellow zero. Analysis of radiation characteristics showed that CONSTOR RBMK-1500 cask has better shielding characteristics than CASTOR RBMK-1500 cask. During 50 years of interim storage period the total dose rate is decreasing from 5 to 20 times in dependence on the location of the point detector and type of the cask. Criticality analysis of CASTOR RBMK-1500 and CONSTOR RBMK-1500 casks demonstrated the effects of the geometrical configuration of fissile material and its distance to the wall of the cask on the value of the effective neutron multiplication factor  $k_{\text{eff}}$  for different density of the water. They also demonstrated that effective neutron multiplication factor  $k_{\text{eff}}$  for an unfavorable operational and hypothetical accident condition is less then allowable value 0.95.



## DRY STORAGE OF SPENT KANUPP-FUEL AND BOOSTER ROD ASSEMBLIES

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Augmentation of the existing spent fuel storage facility at Karachi Nuclear Power Plant (KANUPP)[1] is being planned. For this purpose, an interim dry storage option is under consideration, for which a preliminary design work has been carried out. A dry storage facility comprising of steel-baskets (Fig. 1) in concrete canisters arrangement has been selected[2] for further development. The design of shielding against radiation and decay heat has been based on bundle burn-up and cooling time of 9000 MWD/TU and six years respectively[3]. The design criteria follows the ALARA principle. KANUPP-fuel consists of 19 cylindrical elements assembled into a bundle forming two concentric rings of 6 and 12 elements around a central element. Total bundle length and diameter are 49.609 and 8.140 cm respectively. The fuel element consists of uranium dioxide in the form of pellets sheathed in the Zircaloy-4. The total U-mass per bundle is 13.395 kg.

Two different sequential steps for transferring the fuel from bay to the dry storage facility are under evaluation. The first option is based on the concept of a 36-bundle basket in a flask transferred to the hot cell facility for seal welding and placing in the concrete canister. The second option under consideration is based on the 11-bundle tray flask (Fig.2) transferred to the hot cell facility for loading, seal welding and placing the basket in the concrete canister. Transfer flask weight limit within 15 tonnes is imposed keeping in view the available crane capacity in the services building. Track-one option implementation is thought to be more time consuming at bay and would use the lifting tools already present at KNPC. Track-two requires lesser time at bay but increases the activities at hot cell and number of operations trips from bay to the hot cell.

It is also proposed to shift the Spent KANUPP-booster rod assemblies to dry storage facility in future. A booster assembly consists of four booster bundles containing 2.02 kg/bundle of 10.5% enriched uranium. The essential characterization studies (fission product & actinide inventories, decay heat and gamma spectra variation with cooling time, estimate of neutron source etc) have already been carried out. It has been found that the spent booster rods have considerable abundance of the spontaneous fission neutrons from trans-uranium elements [e.g for a case study of 2000 days cooled booster bundle it is in the order of  $10^4$  neutrons/sec]. Therefore, they would be stored separately, initially it seems to be an appropriate solution to apply the 'dispersion of source' technique to avoid the problem of criticality. As a booster bundle is similar in dimensions to that of a normal fuel bundle, therefore conceptually it seems appropriate to put a few booster bundles in a normal 36-bundle steel basket with unfilled spaces as gaps in between them. This will help to eliminate a number of design, fabrication and handling related problems.

It may be mentioned that we are in the initial stages of addressing spent fuel dry storage problems and many long term solutions have yet to be found. For this purpose efforts are being made for technology development. Our development program strategy is to identify the problems and develop technologies that can provide solutions more quickly, more safely and at a lower cost.

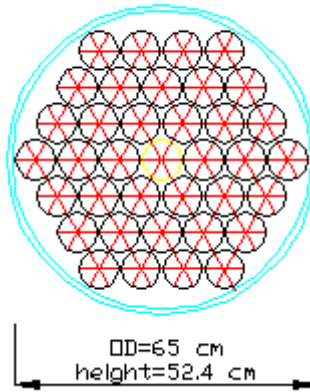


FIG.1. 36-bundle steel basket configuration (8.65cm c/c of bundles)

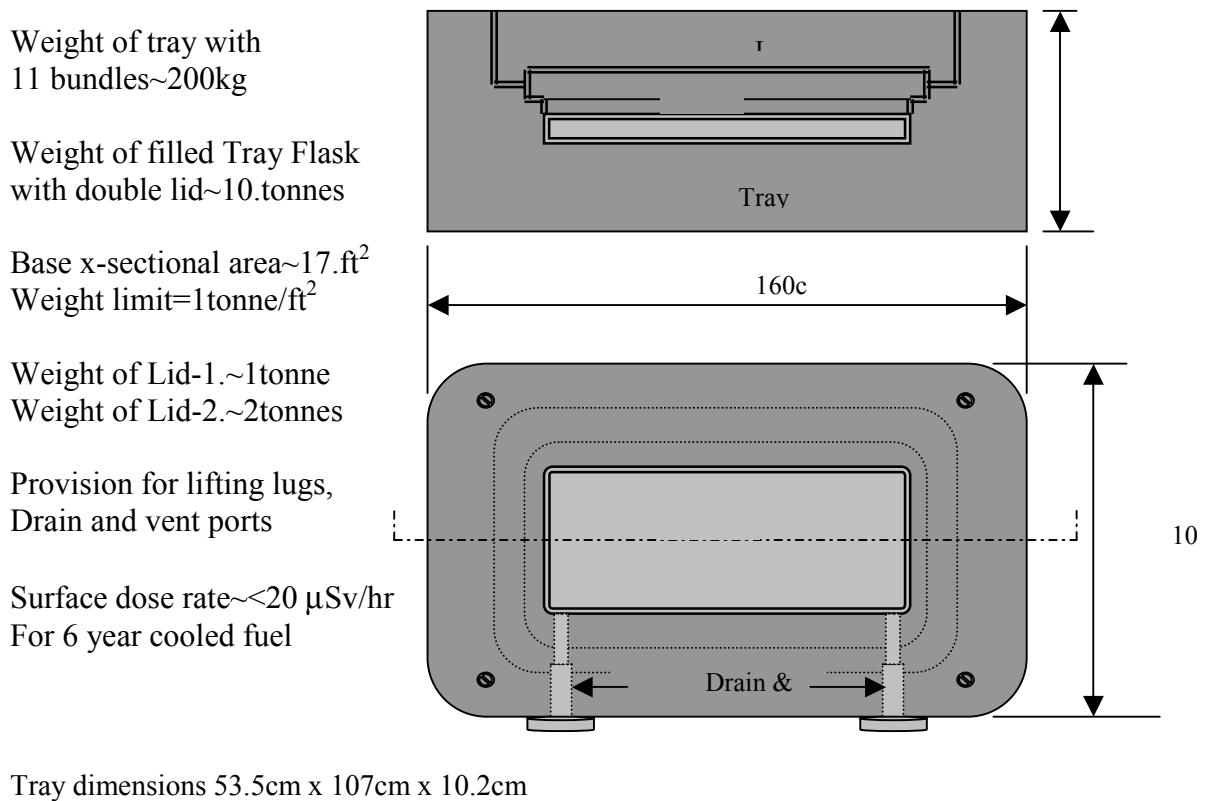


FIG.2. Tray Transfer flask configuration

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## THE LICENSING PROCESS OF CERNAVODA INTERIM SPENT FUEL DRY STORAGE

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For Cernavoda Nuclear Power plant, finding sufficient capacity of storage for spent fuel is essential, if the plant is to be allowed to continue to operate. After the intense initial radioactivity has decayed, the reactor operator would have faced a choice: the fuel can be stored in an extension of the pool (or some other pool) for a longer period, it can be placed in one of several forms of dry storage, or it can be sent to a geologic repository for disposal. Currently, geological disposal is only an objective within the framework of the radioactive waste management policy. Also, maintaining and operating a spent fuel storage pool involves significant operational costs and some modest generation of radioactive wastes. Thus, for spent fuel that is to be stored for a substantial period, the plant operator turned to dry storage. Interim dry storage offers a safe, flexible and cost-effective approach to spent fuel management.

Both experience in countries around the world and a number of regulatory reviews reinforce the conclusion that dry storage is safe. The basic safety goals that must be met are to ensure that: (a) sufficient shielding is provided so that workers at the facility are not exposed to hazardous levels of radiation, and (b) the fuel is contained so that any release of radioactive material to the surrounding environment is reliably prevented. To ensure that dry storage provide adequate shielding and containment, the system design was considered to meet the following regulatory requirements:

- fuel cladding must maintain its integrity while in storage;
- high temperature that could cause fuel degradation must be avoided;
- accidental chain reactions (criticality) must be prevented;
- effective radiation shielding must be provided;
- radioactive releases must be avoided; and
- fuel retrievability must be ensured.

During the autumn of 2000, SNN proceeded with an international tendering process for the supply of a dry storage system to be implemented at the Cernavoda station to store the spent fuel from Unit 1 and from Unit 2 for a minimum period of 50 years. The bidding process resulted in early 2001 in the selection of a technology that uses the best features of dry storage systems in use at similar CANDU 6 reactors in Canada. The Cernavoda storage site will contain MACSTOR storage modules identical to the ones in use at the Gentilly 2 reactor in Canada. The storage site is provided with a physical security system adapted to the storage of spent nuclear fuel that integrates with the station's security system. The facility is designed to meet the stringent Safeguards requirements imposed by the International Atomic Energy

Agency (IAEA) for the control of nuclear fissile materials. The storage module has proved to provide: an excellent shielding of radiation; an excellent structural integrity provided by its massive reinforced concrete construction; an excellent fuel cooling providing safe low fuel temperatures; and a double confinement envelope to the fuel bundles, a configuration believed to offer the best overall protection in the industry. All these features assure a very good behavior of the spent fuel and its integrity in dry storage conditions. During normal operation and following postulated accidents, the proposed system will limit occupational and public doses well within values permitted by the most stringent international standards. The minimum design life of the dry storage facility is 50 years [1].

The licensing processes of Cernavoda interim dry storage was implemented according to the requirements of Romanian Authorities and included the following steps:

Project status	Licenses/permits required
SITING	<ul style="list-style-type: none"> <li>• Siting license issued by Regulatory Body</li> <li>• Siting permit issued by Health Authority</li> </ul>
CONSTRUCTION	<ul style="list-style-type: none"> <li>• Construction license issued by Regulatory Body</li> <li>• Construction permit issued Environmental Protection Authority</li> <li>• Construction permit issued by Health Authority</li> </ul>
COMMISSIONING	<ul style="list-style-type: none"> <li>• Commissioning license issued by Regulatory Body</li> </ul>
OPERATION	<ul style="list-style-type: none"> <li>• Operation license issued by Regulatory Body</li> <li>• Operation license issued by Environmental Protection Authority</li> <li>• Operation permit issued by Health Authority</li> </ul>

Romania does not have specific regulatory requirements and criteria governing nuclear licensing of spent fuel dry storage. For this reason the Regulatory Body imposed the observance of the following American guidance [2]:

- 10 CFR Part 72 – Licensing requirements for the storage of spent fuel and high level waste
- NRC Regulatory Guide 3.48 Standard format and content for the safety analysis report
- ANSI/ANS 57.9-1992 Design criteria for an independent spent fuel storage installation.

The main difficulties experienced in the licensing process are related to:

- lack of specific Romanian regulations for this type of facilities;
- application of NRC guidance to the vault type storage installation;
- meeting some requirements of the environmental authorities.

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## IMPLEMENTATION OF ROMANIAN NPP SPENT FUEL MANAGEMENT STRATEGY - A REGULATORY APPROACH

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NPP Cernvoda Unit 1, started to operate from 1996, producing around 90 t (heavy metal) spent fuel/year. The plant is of CANDU-6 type using natural uranium as fuel, at a mean burn-up of the spent fuel of around 7800 MW\*day/tonne U. The construction of Unit 2 was restarted, and commercial operation of this unit is expected for 2006. Romania intends to restart also the construction of Unit 3.

In establishing the NPP spent fuel strategy, the characteristics of the design of NPP, which allow a capacity of safe storage of the spent fuel in the wet pond for around 7 years and 6 month of operation, were also taken into consideration.

After considering different options, it was found that the best strategy for the management of the NPP spent fuel is the dry storage for at least 50 years, with further possible extension of the dry storage period. This option allows for the time necessary for siting, construction and commissioning of the Romanian geological disposal of the spent fuel, if another more favorable option will not occur meantime.

The selected solution is the Canadian AECL “Monolithic Concrete Module” type MACSTOR. The design of the dry storage covers 30 years of operation of 2 CANDU-6 units. If Unit 3 will enter into operation further capacity will be needed, and a supplementary storage has to be added.

The National Commission for Nuclear Activities Control (CNCAN) has decided that the Initial Nuclear Safety Analysis required for the siting authorization, and the Preliminary Nuclear Safety Report required for the construction authorization, shall observe the structure and the requirements of U.S. Regulatory Guide 3.48 “Standard Format and Content for the Safety Analysis Report (Dry Storage)”, with some modifications related to the characteristics of the CANDU type spent fuel as well as of Romanian regulatory framework. The assessment of the above documents shall be done by CNCAN according to the applicable requirements of U.S. NUREG 1567 “Standard Review Plan for Spent Fuel Storage Facilities” and to the regulatory dispositions issued by CNCAN.

After a relatively complex process, the siting authorization was issued on 12.08.2001, followed by the construction authorization, issued on 05.06.2002. The authorization for operation of the storage is expected in the first part of 2003.

Regarding the work performed at the spent fuel bay at adjacent areas, the regulatory approach is to consider them as modifications under the operational authorization of Unit 1. CNCAN requested, as a main condition for starting the work at spent fuel bay area, the demonstration

of assurance of the nuclear safety during the modification work as well as latter, during the operation of the storage.

The paper details the regulatory process, presenting the main requirements formulated by CNCAN, by regulatory dispositions and authorization conditions, as well as the system of inspections at the site during the construction phase.

The second part of the paper presents some considerations related to geological disposal of spent fuel. Romania is a relatively small country, and subject to earthquakes. Nevertheless the Government considers that sitting a geological repository within the country is necessary for secure implementation of the nuclear program. The foreseen strategy for sitting a deep geological repository for spent fuel considers the assessment of the possible host formations (salt, granite, volcanic tuff, schist, may be even clay), the use of international underground laboratories for developing the concept and finally, after the selection of the site, the construction of an underground confirmatory laboratory on the selected site.

The paper presents the actual status of the activities related to the sitting of the deep geological repository, including the needs for a specialized structure and for funding of spent fuel and radioactive waste management.

CNCAN, taking into account the complexity of the safety issues, and the needs for ensuring flexibility in considering of other options feasible in the future, expects that the program for spent fuel, high-level waste, and long lived waste management will be carried out considering that spent nuclear fuel dry storage for 50 years or longer (100 years) is required to provide a necessary time for implementing the final disposal program.

We expect the sitting of spent fuel disposal by 2030, and commissioning of the repository by 2050.

However, considering the benefits that an international repository can present, Romania would like to keep such an option open, under the condition that international safety and security standards are met and regulatory control at the receiver country is established.



## EXPERIENCE IN PERFORMING THE CERNAVODA SPENT FUEL INTERIM STORAGE FACILITY

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At the level of 1980 – 1989 years, the Cernavoda site was designed and licensed to be erected 5 CANDU 600 Mw Units. Until now, only first unit was put in operation in December 1996 and Unit 2 will be commissioned in the first part of 2006 years. The others units there are erected and completed at different levels. The Romanian Government declared itself favorable for analyzing the possibility to continue the nuclear program with completion of Units from Cernavoda site.

The yearly arising of the design spent fuel quantity in operation of one CANDU 600 MW unit, is about 94 tones Uranium natural (aprox. 4800 CANDU bundles). Each units, includes a Spent Fuel Storage Bay which has a storage capacity for spent fuel produced in approximate 9 years.

Taking in view the increased number of spent fuel bundles obtained in operation the load factor obtained till now was and is still better like the design load factor) and the limited free space from the existing Spent Fuel Bay, at the level of 2000 years, Nuclearelectrica SA, as owner of Cernavoda NPP, issued a Tendering Document for an International Bidding Process.

The Tendering process was based on the results of support studies, including a feasibility study performed in advance under Romanian Research and Development Programs, some years ago. Also, an important support was obtained under the IAEA Technical and Research Programs. The result, it was selected a dry solution and taking in view the schedule for commissioning the N.P.P's Units 3 ÷ 5 is uncertain, for distributing the investments according to the real storage requirements, a modular concept was proposed, placed on the boundary of plant site.

The winner of the bidding process was AECL Company from CANADA for an offer which has as reference Point Lepreau solution for preparing of the spent fuel for dry storage and Gentilly 2 solution for dry storage facility.

In April 2001, a Contract was signed between SNN and AECL for this facility which will use CANSTOR type concrete monolithic modules, already are in operation in Gentilly 2 CANADA, since 1995. The facility has a storage capacity for the spent fuel obtained from 30 years in operation of the Unit 1 and 2. The deadline for commissioning of this facility is May 2003.

Local Romanian companies, SITON for engineering and licensing documentation work and Nuclearmontaj for construction and procurement work were involved in this project (percentage more than 85%) as AECL subcontractors.

During the performing of this facility was gained an important experience, in different fields, more important being engineering work, construction work, procurement activity and in special licensing process.

The licensing process was a good experience from our point of view and has some particular aspects compare with other similar facility process. These could be specified as following:

1. Support documentation for Licensing process fulfilled the US- NRC Regulatory Guide 3.48 "Standard content for the safety analysis report " and US - NRC 10 CFR part 72 "Licensing requirements for an independent storage spent fuel and high level radioactive waste" requirements (although Cernavoda SFIDSF is located inside the boundary of NPP site).
2. Evaluation of an aircraft crash on the CANSTOR module consequences was included into the safety analysis. The results were considered in the emergency plan.
3. Radiological effects on operators and public were evaluated taking into account the multiple contributions of all radiation sources, existing in Cernavoda site (NPP This last aspects

The paper intends to present an overview on this facility and the experience gained by Romanian during the construction of this objective inside the Cernavoda NPP plant.

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## VALIDATION OF DRY STORAGE MODES FOR RBMK-1000 SPENT FUEL ASSEMBLIES (SFA)

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The long-term storage of RBMK-1000 SFA is supposed to carry out in dual-purpose metal-concrete casks and dry storage facilities of a vault type.

For the validation of the SNF dry storage modes, the following researches were carried out:

- The information on SFA condition of RBMK-1000 reactor after operation and wet storage was summarized and analysed.
- The researches of degradation mechanisms of the SFA constructional materials in dry storage conditions in different atmosphere were carried out.
- The mathematical model was developed of SFA fuel rods behaviour in dry storage conditions.
- The calculations of fuel rods condition after 50 years of dry storage were made.
- The mode of dry storage was chosen and the constructive integrity of fuel rods and SFA was validated.

All results of researches were summarized in the calculating model of RBMK fuel rods behaviour during long-term dry storage. This model takes into account all degradation mechanisms of fuel cladding. The calculations within the framework of this model have allowed to determine the SFA final condition after dry storage and to specify the modes of storage.

On the basis of the conducted investigations, the following conditions of RBMK-1000 SFA storage are suggested: the design duration is 50 years, the storage medium is nitrogen, the maximum permissible fuel claddings temperature is 300<sup>0</sup>C.

## ENSURING SAFETY IN HANDLING THE CASKS WITH IRRADIATED NUCLEAR FUEL

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The last decade of the 20<sup>th</sup> century was seeing the beginning of creation of a new technology for management of irradiated nuclear fuel (INF) at the end stage of fuel cycle, which was based on a different type of casks used for the long-term storage and subsequent delivery of INF to the sites of reprocessing, continued storage and final disposal. Concept of using a protective cask is applied, which ensures retention of the content (INF), as well observation of other requirements of safety in storage and transportation of INF. Protection from radioactive emanation and/or spread of radioactivity off the cask limits is ensured by physical barriers, that is, all-metal or composite body, cladding of the body, inner spaces for irradiated fuel assemblies (IFAs), and lids with sealing systems. Natural processes of the emission and convection of ambient air around the cask withdraw the IFAs afterheat to environment.

The double-purpose packing set taken as both a transportation packing unit and a storage facility for INF must come up to requirements for the safety of, respectively, transportation and storage. National and international requirements to transportation packing sets and packages are presented with sufficient details in standards and regulations for safe transportation of radioactive materials, namely the IAEA regulations for safe transportation of radioactive materials, the most complete publication of which was issued in 1966. On national scale of Russia, there exist Rules for safe transportation of radioactive materials, PBTRM-2001, put into effect since the July 1<sup>st</sup>, 2002. In 2000, the “Guiding documents” series included such operational normative documents as:

- “Storage facilities for spent nuclear fuel, using double-purpose casks. General technical requirements and rules to ensure the safety”;
- “Packing sets for transportation and storage of spent nuclear fuel, using double-purpose casks. General technical requirements and rules to ensure the safety”.

This paper considers some key issues involved in requirements of ensuring safety of double-purpose sets for INF, and how such requirements are met at the phases of design, manufacture, test, and commission of a prototype. An example is given of production of the INF transportation and storage packing set.

**RESEARCH OF CORROSION RESISTANCE OF STRUCTURAL MATERIALS OF METAL - CONCRETE CASKS FOR SPENT NUCLEAR FUEL**

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Since 1995 in Russia a family of metal and concrete casks (MBK) is set up for storage and transportation of SNF of different types of nuclear power installations. By now is completed the development and certification of a package set for storage and transportation of SNF of transport nuclear installations under the conditional index TUK-108/1 – one of the MBK family. A batch of 48 items has been produced to resolve the issues of dismantling of nuclear submarines withdrawn from active service and, thus to improve environmental safety in North-West and Far East of Russia.

At the design stage there was performed a big amount of scientific studies to verify and justify safety, reliability and service life of MBK including research in corrosion resistance of structural materials. The intent of studies was the justification of strength of the cask structural materials based on determination of the extent of tendency to intercrystalline corrosion of materials of canisters and shells as well as of the inner lid under interaction of gaseous fission products under conditions of dry cask storage of SNF. In the course of studies the following problems were resolved such as:

- a calculation of fluences of residual neutron and gamma activity of SNF and an assessment of their effects upon the condition of material;
- experimentally, physical and chemical interaction between materials and simulators of SNF fission products was studied;
- the extent of corrosion damage of cask structural materials was evaluated considering temperature, media and duration of service.

The experiments were aimed at the modeling of interaction between the specimens of structural materials and simulators of gaseous fission products under conditions similar to accident state of a cask with unsealed fuel rods characterized by high nuclear fuel burnup. The temperature of tests was chosen as 300 and 500 °C.

In studying the specimens subjected to tests in a medium simulating cask conditions and in determining the compounds generated there were used metallographic methods and qualitative X-ray phase analysis. Metallographic investigations were performed for determination of the corrosion depth of materials and evaluation of material structure. To identify phases generated in the process of interactions between fission product simulators, storage media and steels, there was used the X-ray phase analysis of specimens.

The investigation of steel specimens showed an absence of corrosion defects of all materials at a temperature of 300 °C. At a temperature of 500 °C an area of stainless steel corrosion within 3-4 μ was observed. This area contains cesium, chlorine, iodine, tellurium, oxygen, chromium and iron. A forecast of a potential corrosion depth was made for 09G2SA-A steel at 320-340 °C in the cask environment after 50-year service. It amounted to 55-60 μ. In case of stainless austenite steels the value of damage extent will be smaller even at temperatures to 400-450 °C.

Thus, it may be suggested that the working capacity of cask structural materials under SNF storage conditions will be retained over 50-year period. The report gives the results of comprehensive study in corrosion resistance of structural materials of interior units of the cask for storage and transportation of SNF.

## COMPARISON OF SUBCRITICALITY OF THE INTERIM SPENT FUEL STORAGE BEFORE AND AFTER MODIFICATION

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After political and economical changes in the end of eighties, the utility operating the nuclear power plants in Slovakia decided to change original scheme of the back-end of the nuclear cycle; instead of reprocessing in the USSR/Russia spent fuel will be stored in an interim spent fuel storage facility until the time of the final decision. As the best solution, a modification of the existing interim storage has been realised. In existing interim storage is used new basket with new shape and with compact racks from boron steel.

In Slovakia they are in operations 6 units WWER-440. The original capacity of the interim storage is 5040 assemblies, after modification is 14 112 assemblies.

The original type of fuel (used in operation and stored in the interim storage) has maximal enrichment 3.6%. The current type of fuel (now used only in operation, not in storage) has radial profiled enrichment with maximum 4.0%. The advanced fuel (will be in operation after 2005) has maximal enrichment 4.4%.

The subcriticality both baskets with different fuel by operational conditions:

	C-30	KZ-48
3.6 %	0.85287	0.83384
4.4 %	0.88489	0.87477

In article are compared criticality both baskets by abnormal conditions: mechanical destruction or loss of coolant.

New basket make possible to increase 2,8 times the capacity of interim storage and is more safety then older one.

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## CAPACITY EXTENSION OF THE BOHUNICE STORAGE FACILITY

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Slovenské elektrárne, a.s. (SE) is a major supplier of electricity in Slovakia, generating 85 % of domestically produced electricity. Besides thermal and water power plants, SE currently operates two nuclear power plants — Bohunice (4xWWER-440) and Mochovce (2xWWER-440) — which account for over a half of its generation capacity. The first of the Mochovce units commenced operation in 1998, and the second unit in 1999.

The safe handling of radioactive wastes (from plant operations and the Bohunice A1 decommissioning project), including the long-term storage of spent nuclear fuel (SNF), is a major concern for SE. In 1996 a subsidiary of SE, the Nuclear Installations Decommissioning, Radwaste and Spent Fuel Management (SE-VYZ), was established to handle the decommissioning of nuclear installations, treatment of radioactive wastes and storage of spent nuclear fuel in Slovakia.

### Transports

It was originally assumed that SNF from WWER-440 units would be transported back to the USSR after three years storage in spent fuel ponds. Later, the Russians changed the cooling-down period to 10 years. As a result, the interim storage facility at Bohunice was designed and constructed for SE.

The transport of SNF to the USSR for reprocessing was not accomplished because of political and economical changes in Central and Eastern Europe in the beginning of 1990s. The return of SNF back to Dukovany started in 1995 and was finished in 1997.

### Bohunice store

The interim storage facility (ISF) at Bohunice (see Table 1) is a wet store, where SNF is stored in three storage pools and a fourth storage pool is kept empty for emergency case.

The designed storage capacity of Bohunice ISF was 600t of uranium — 5040 fuel assemblies. SNF is stored in the special storage baskets of T-12 type with 30 fuel assemblies in each. Failed fuel assemblies, which are isolated in special hermetic canisters, are stored in T-13 storage baskets. Fifty six storage baskets, or 1680 fuel assemblies, can be stored in one pool.

The original design enables capacity enlargement by an additional building of 2-3 storage pools.

### Reconstruction of the store

At the end of 1993 and during 1994 the international tender for the supply of a long-term store (or possibly two stores) for SNF from Bohunice and Mochovce was carried out.



The management of SE had decided to cancel this tender in 1994 and look into the possibility of storage capacity enlargement of the existing Bohunice ISF. The decision was taken to upgrade the existing ISF, when it was shown to be technically possible and cheaper. Three main goals were set:

- Seismic upgrading of the ISF – upgrade of building and technological systems to the level of 8 on the Richter scale.
- Increase the storage capacity to cope with all fuel from all the Bohunice units.
- Prolong the lifetime of the ISF for 40 years from the end of the upgrade work.

#### Capacity extension

The need for the higher capacity of the ISF resulted from the amount of existing stored fuel assemblies, as well as from calculated amounts of spent fuel assemblies produced in the future operation of four Bohunice units.

Storage capacity enlargement has not been solved by the construction of additional storage pools but by so called “compactization” which means replacement of the old storage baskets T-12 by new, compact storage baskets KZ-48, together with necessary adjustment of the technological and safety systems.

The old type of the storage basket allows storage of 30 fuel assemblies. One storage pool can store 56 of these baskets. The newly designed storage basket KZ-48 allows for the storage of 48 WWER-440 fuel assemblies where enrichment cannot be higher than 4.4 %. The average burnup of the fuel assemblies in a storage basket cannot be higher than 42MWd/kgU and maximum burn-up of a fuel assembly can be 52.5MWd/kgU. The shape of this basket enables closer storage of these baskets in the storage pool in such a way that there can be as many as 94 KZ-48 baskets in one storage pool.

In this way the previous ISF capacity of 5040 fuel assemblies could be increased to the level of 14,112 fuel assemblies. This increase in capacity also leads to an increase in the heat generation from a maximum level of 516kW to 1990kW. The increase of the heat generation is proportional to the number of the stored fuel assemblies. This requires higher performance of the cooling system, which must be fitted with new plate heat exchangers and pumps. The autonomous cooling water distribution system with its own cooling towers has been constructed in order to ensure independence in the supply of water from the Bohunice distribution system.

Apart from the modifications of the original building design and technology of the ISF, which resulted requirements for seismic upgrading and storage capacity enlargement there has been also other changes and adjustments, which have significantly increased the technical standard of the ISF.

- Installation of MAAP 400 manipulator for fuel handling from T-12 baskets to KZ-48 baskets.
- Constructional adjustments of radiation zone entry at elevation 0.00m.
- Construction of the new radiation zone entry for visitors at elevation of 3.60m.
- Constructional adjustments of the entry hall
- Change of position of the reduction stations of pressurized air and nitrogen together with piping adjustments.
- Air conditioning system adjustment.
- Renewal of the filtration system of pool water.
- Modernization of the decontamination tank for small particles.
- Supplementation of the control system for tightness control of fuel assemblies and corrosion monitoring of storage pools.
- Modernization of radiation control systems.
- Aerosol detection system has been supplemented with detection in more rooms.
- Electricity supply to the ISF is from Bohunice V-1 by two independent 6kV power distributors. Diesel generator has been installed for the event of total black-out of system voltage
- Control system SIMATIC S5 is hierarchic, with a two-level control system with adequate communication software for the operator. The Siemens ETHERNET SINEC L1 allows communication between both levels.

## **STORAGE OF SPENT FUEL IN SLOVAKIA**

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### **Role of NRA SR**

- Performs the state supervision over nuclear safety of nuclear installations in accordance with Act no. 130/1998Supervises over nuclear safety in facilities for storage and transportation of nuclear spent fuel
- Performs the state supervision during designing, construction, commissioning, operation and decommissioning of nuclear installations

### **Legislation**

#### Acts

- Act No. 130/1998 on Peaceful Use of Nuclear Energy
- Act No. 254/ 1994 (am. 78/2000) on State Fund for NF Decommissioning and RW and SNF management

#### Regulations

- Regulation No. 190/2000 on Radwaste and Spent Nuclear Fuel Management
- Regulation No. 284/1999 on Transport of Nuclear Materials and Radwaste
- Other Regulations - on Physical Protection, on Accounting and Control of NM etc.

#### Guides

Guide of NRA SR on Construction and Operation of Spent Nuclear Fuel Storages

### **Spent Fuel Storage Facilities in the Slovak Republic**

- Interim Wet Spent Fuel Storage Facility Jaslovské Bohunice

- Spent Fuel Storage Pools Adjacent to Reactors
- Dry Interim Spent Fuel Storage Facility Mochovce (project preparation)

**Future Development**

- BUC application in the criticality calculation of the WWER-440 fuel assemblies
- Long Term Spent Fuel Storage Facility Development

**THE NPP KRŠKO RERACKING PROJECT**

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NPP Krško

Slovenia

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The NPP Krško has increased the capacity of the spent fuel storage pool by replacement of the existing racks with high-density racks. High density racks with borated stainless steel (boron loading of  $2.44 \text{ mg/cm}^2$ ) in the form of additional neutron absorbing panels have been installed. This will be the second reracking campaign since 1983 when storage was increased from 180 to 828 storage locations. The pool capacity is increased from 828 to 1694 with just partial reracking in the spring 2003. The installed capacity is sufficient for the current design plant lifetime. Complete reracking of the spent fuel pool will additionally increase capacity to 2321 storage locations. Burnup credit methodology, which was approved by the Slovenian Nuclear Safety Administration in previous licensing of an old racks, was implemented with the recent and significant methodology improvements. General design and analysis process as well as criticality safety analysis and results are presented in the paper.

## UPDATE ON SPENT FUEL AND HLW MANAGEMENT IN SPAIN

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According to strategy adopted in the fifth General Radioactive Waste Management Plan (GRWMP) of 1999, ENRESA, the agency responsible for radioactive waste management in Spain, has implemented, together with the NPP operators, different spent fuel storage technologies to manage the overall inventory (around 7,000 tU) expected to be generated in the country.

In the nineties, all the spent fuel NPP pools were re-racked as an initial stage to provide additional At-Reactor capacity. The next steps were, and will be directed, towards using dry spent fuel storage technologies. In this latter respect, the major milestones achieved and plans are the following ones:

- For the first plant requiring additional out of pool capacity, the TRILLO NPP (KWU design), a dual-purpose cask known as the DPT cask, was designed and licensed in Spain. The two first units have already been loaded in 2002 and a campaign of 6 more loads is scheduled to take place in 2003.
- A massive Cask Storage Building has been constructed in the site to host the 80 casks that will be needed during the expected plant's operating lifetime. The startup licence was issued in May 2002.
- The JOSÉ CABRERA (ZORITA) NPP will stop its operation by April 2006, three years ahead of its original schedule. As a part of the decommissioning activities, the spent fuel stored in the pool will be evacuated to a dry storage system. ENRESA is currently studying the possible alternatives to manage the total spent fuel inventory (almost 400 fuel assemblies)

An Away-From-Reactor Spent Fuel and HLW Centralised Storage Installation is foreseen to be available by 2010. It will collect all the spent fuel and some medium and high active wastes generated in the country, together with glasses and other HLW that will come back to Spain from previous reprocessing contracts.

Finally, different studies and demonstration projects are being developed related to the deep geological disposal.

## EVALUATION OF NUCLEAR FUEL CYCLE SCENARIOS WITH RESPECT TO SOME PARAMETERS IMPORTANT FOR SPENT FUEL STORAGE

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LWR fuel cycle scenarios, involving several options regarding fresh fuel composition, irradiation time and back-end processes, have been evaluated from the viewpoint of some parameters important for spent fuel storage. Calculations are performed using MONTEBURNS [1] that couples MCNP4B [2] for neutronic calculations and ORIGEN2 [3] for depletion calculations. Effects of fresh fuel composition, burn-up, and spent fuel storage time on characteristics of spent fuel such as decay heat, radioactivity and composition are analyzed.

LWR fuel cycle scenarios are classified according to the discharge burnup (33 and 50 MWD/kgHM), reprocessing options (without reprocessing, PUREX, COPROCESSING and AIROX), make-up materials (natural U, depleted U, and Th) and fissile material (reactor grade Pu, weapon grade Pu and weapon grade U). 3.25% <sup>235</sup>U enriched UOX fuel with 33 MWD/kgHM discharge burnup is assumed as a reference fuel cycle for neutronic calculations. For each fuel cycle, fresh fuel composition that is necessary to achieve the specified burnup is determined based on the criteria that, batch average core reactivity is the same as that of the reference cycle with three batch fuel management scheme.

Fuel cycle scenarios for a PWR of 1000 MW(e) with a thermal efficiency of 34.5 % and a specific power of 37.5 MW/MT are evaluated in the study. Neutronic calculations are performed for a representative PWR unit cell. A material flowsheet with a base of freshly loaded 1000 kg HM for each scenario is determined.

In COPROCESSING without partial partition and PUREX reprocessing methods, it is estimated that %99.9 of U and Pu is recovered. In COPROCESSING with partial partition, composition of U+Pu product is arranged such that its fissile content is sufficient to achieve the specified burnup. It is assumed that %100 of Iodine, Krypton, Xenon, Tritium and Carbon; %90 of Ruthenium and Cesium; and %75 of Tellurium and Cadmium are removed by the AIROX method [4].

Composition of fresh fuels for each fuel cycle scenario is determined using the methodologies described. Quality of fissile material, make-up material, and reprocessing products have major roles in determining the composition of fresh fuel. More fissile material is needed for the fuel cycles that involve usage of AIROX product as expected. Usage of weapon grade Pu or U reduces the fissile content of fresh fuel needed to achieve the specified burnup.

Characteristics of spent fuel such as decay heat, radioactivity, radiological ingestion and inhalation hazard, and composition for each fuel cycle scenario are analyzed. Amount of trans-uranium elements (Pu, Np, Cm, Am), radioactivity, radiological ingestion and inhalation hazard per tonne initial heavy metal increase as discharge burnup is increased in general. Since increasing the burnup results in reduction of spent fuel arisings, it may be concluded that those quantities per energy generated during irradiation is independent of discharge burnup.

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## MULTI-PURPOSE CANISTER STORAGE OF SPENT NUCLEAR FUEL IN MODULAR VAULT SYSTEM

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The original Modular Vault Dry Storage (MVDS) technology was developed in the early 1980s leading on from the experience gained with the Magnox fuel dry storage facilities at the Wylfa power station in Wales (UK). The Wylfa dry stores were commissioned in 1969 and the MVDS can, therefore, rightly claim to be the only dry storage technology that has an operational and technological background of over thirty years. The MVDS is a very flexible design approach and is suitable for the storage of all types of spent nuclear fuel or high level waste streams. The MVDS design has continued to be updated since the United States Nuclear Regulatory Commission granted the Topical Report license approval in 1988. The MVDS Topical Report covers the interim storage of Light Water Reactor fuels, including both PWR and BWR, at any reactor site in the USA.

The first MVDS constructed in the United States was built at the Fort St Vrain high temperature gas reactor site in 1990, and the MVDS technology has also been applied to the storage of WWER fuel at the Paks nuclear power plant in Hungary. A MVDS facility is currently being licensed by the United States Nuclear Regulatory Commission for storage of United States Department of Energy owned fuels at the Idaho National Engineering and Environmental Laboratory site in Idaho, USA.

The MVDS system was originally designed to store individual fuel assemblies within a Storage Tube or Canister. This system provides maximum flexibility for future off-site transportation as the individual fuel assemblies can be removed from their storage locations and placed into a transportation cask for either road or rail off-site shipment.

A vault storage system based on the proven MVDS technology using a large Multi-Purpose Canister (MPC), is being developed to provide cost-effective interim spent fuel storage system. Integrating the MVDS technology with a large MPC and adapting the cooling, shielding and handling system, allows the new vault storage system to provide high storage efficiency in compact storage buildings suitable for a large spent fuel interim storage facility.

The MVDS vault passive cooling system is capable of rejecting approximately 450kw of heat from each vault module, before either the bulk air temperature causes the vault concrete temperature or the fuel temperature to reach an acceptable temperature limit. It has been possible to re-configure the vault storage array from individual storage tubes to large diameter canisters, and to re-configure the handling equipment transfer larger canisters. By modifying

the design of the MVDS to accept large, multiple assembly, multi-purpose canisters, it has been possible to maintain the technical and operational benefits of the original MVDS, with the additional benefits of multi-purpose canisters. A storage tube is typically 0.2 to 0.4 metre diameter, whereas a multiple assembly canister is typically 1.6 to 1.8 metres outside diameter. In recognizing the difference between the two size of storage vessel, the new MVDS system has been named: Mega-Vault Dry Store.

The Mega-Vault storage facility consists of three main systems:

1. The Storage Vault Modules, where canistered spent fuel is stored.
2. The Canister Handling Machine, which raises and transfers canisters from the cask in the Transfer Tunnel to the storage position in the Storage Vault.
3. The Cask Reception Bay and Transfer Tunnel, where canistered spent fuel is received and transferred to a port under the vault where it can be collected by the Canister Handling Machine. The Transfer Tunnel and Cask Bay are also used to despatch fuel at the end of storage life.

The Mega-Vault MPC has a 1.8m outer diameter and stores up to 76 BWR or 24 PWR fuel assemblies. The MPC is cooled by passive horizontal airflow during storage. The height of the storage vault has been modified to suit the large MPC, structures have been simplified in the storage area, and earthquake protection of canisters has been achieved. The radiation dose at the site boundary is reduced by below ground storage and radiation streaming reduction structures. Also, providing thick concrete shielding minimizes the expected dose of workers. The compact canister handling machine also enables the building size to be minimized.

The paper will describe in more detail the technical and operational features of the Mega-Vault Storage system, together with process flow descriptions and concept drawings.

## NATIONAL POLICY IN THE AREA OF SPENT FUEL MANAGEMENT IN UKRAINE: CURRENT STATUS AND TRENDS (PROSPECTIVE)

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The closed fuel cycle concept in relation to the WWER was adopted in the former USSR.

WWER-440 SFAs were shipped to RT-1 plant (“Mayak” enterprise) for reprocessing.

WWER-1000 SFAs were shipped to Krasnoyarsk-26 (Zheleznogorsk) for storage in the wet away –from-reactor spent fuel storage facility (SFSF) of the prospective reprocessing plant.

RBMK SFAs were transported to the wet away-from-reactor storage located on the NPP site. Reprocessing of RBMK spent fuel was considered inexpedient because of the low content of fissile nuclides.

Presently, Ukraine continuing to ship spent fuel to Russian reprocessing plants is developing the Intermediate Spent Fuel Dry Storage Program (deferred decision).

The forecast of spent fuel generation within 2002 - 2015 year period is shown in figure 1.

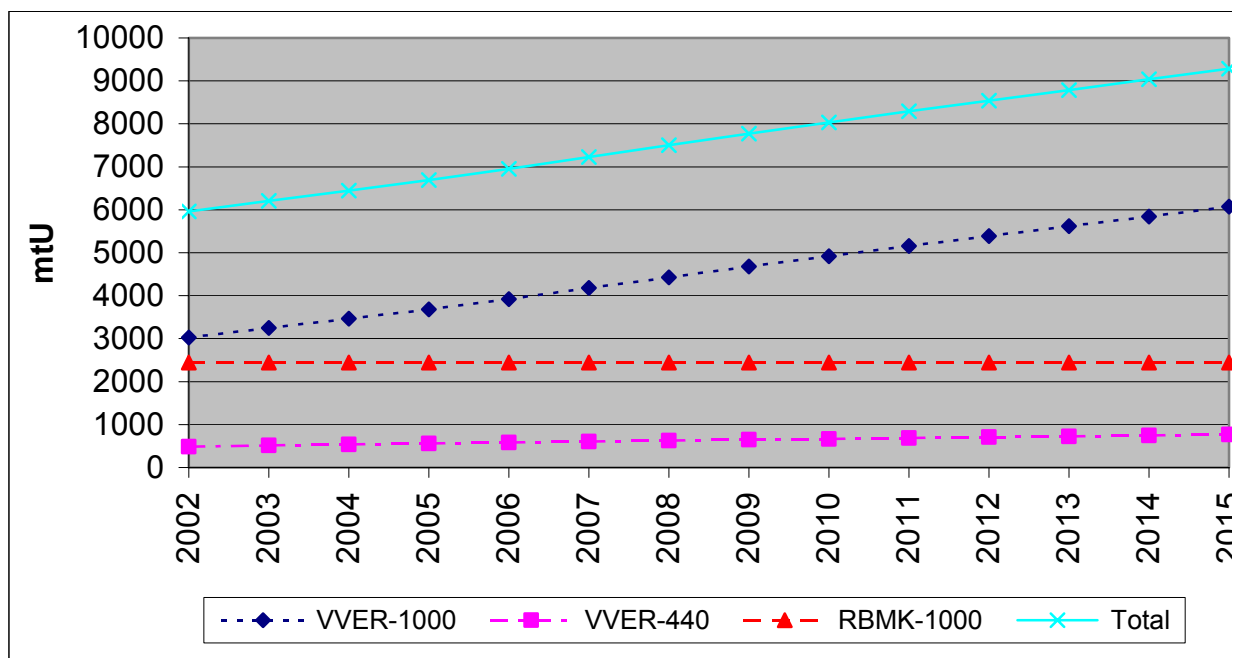


FIG. 1. The forecast of spent fuel generation within 2002 - 2015 year period.

The cost of spent fuel removal services is increasing. The problem of spent fuel dispatch to Russian Federation needs consideration not only from the point of view of cost but also from the point of view of reliable operation and prospects for nuclear energy in Ukraine.

In accordance with the above it should be noted that the construction of RT-2 reprocessing plant is not planned until 2020 that is why Ukraine's WWER-1000 spent fuel will be dispatched to Russia for storage but not for reprocessing in the nearest future.

The capacities of the existing wet SFSF at the RT-2 complex are limited and will fill up by the year 2007 if the current rate of spent nuclear fuel intake from Ukraine, Russia and Bulgaria is kept.

The construction of additional dry storage facility is planned but there is no guarantee that this work will be completed by the year 2007. Thus, spent fuel dispatch to Russia even in the nearest future could be limited due to circumstances beyond Ukraine's control.

The national nuclear utility "EnergoAtom" advances payments for spent fuel reprocessing the results of which it would probably never get because the strategic decision on the structure of the national nuclear fuel cycle is not made.

Several circumstances hinder the decision-making:

- the program of nuclear energy development after the year 2010 and the program of operating reactors life extension are not developed in Ukraine;
- the possibility of the safe operation of WWER-1000 using MOX fuel is not demonstrated and WWER-1000 necessary modernization programs are not developed;
- WWER-1000 MOX fuel manufacturing is absent.

### **RBMK-1000 spent fuel long-term dry storage implementation**

The option of the interim dry SFSF project was taken by means of international tender in 1999. The modular type (horizontal concrete modules «NUHOMS» designed by the Pacific Nuclear, USA and Framatome ATEA) SFSF is being constructed near the Chornobyl NPP site and will be completed by the year 2004. The NUHOMS modules are being built in two parallel lines of 116 modules. Each module contains 1 canister. Each canister contains 196 spent fuel bundle cartridges ( 98 RBMK FAs divided into halves)

The new storage capacity is planned for 21356 RBMK SFAs and approximately for 2000 discharged absorber rods for 100 years. According to the Program, the transport of all spent fuel from the wet SFSF to the dry SFSF should be completed by 2012.

### **Interim dry SFSF implementation on Zaporizhzhya NPP site**

The dry spent fuel storage facility project for Zaporizhzhya NPP is based on VSC-24 cask design by Sierra Nuclear as a prototype. The storage facility project includes 380 VSC-WWER-1000 casks.

The facility at Zaporizhzhya NPP was put into trial operation in September 2001. The first three concrete containers have been loaded. But the design capacity of these containers was reduced from 24 cells to 22 cells according to Ukrainian safety (subcriticality) requirements. Several ways for subcriticality maintenance were considered:

- incomplete loading of the container;
- burnup credit implementation;
- basket improvement due to the use of neutron absorber. (This opportunity will be under consideration after the technology transfer).

The Utility has to obtain the permit of the Ukrainian Regulatory Body for loading of each container.

The facility trial operation was successfully completed and now Zaporizhzhya NPP has to obtain the license for facility commercial operation. One of the conditions for license obtaining is the research of WWER spent fuel behaviour during long-term dry storage.

In the State Scientific Centre of Russia "Research Institute of Atomic Reactors" (RIAR, Dimitrovgrad, Russia) the advancing investigations of three assemblies from Zaporizhzhya NPP) are performed under the contract between ENERGOATOM and RIAR. (FA average burnup is from 44 MWt.day/kgU up to 49 MWt.day/kg U)

### **Central dry WWER SFSF**

Central dry SFSF is apparently the most economically viable decision for Ukraine.

The Ministry of Fuel and Energy of Ukraine and the state nuclear utility "EnergoAtom" are performing preparation works for WWER storage during the period up to 50 - 100 years. One of the most important problems is spent fuel storage technology selection on the basis of option assessment and based on the utilization experience.

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**SPENT FUEL STORAGE FACILITY OF ZAPORIZHZHYA NPP, CREATION, LICENSING, OPERATION**

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The interim spent nuclear fuel facility at Zaporizhzhya Nuclear Power Plant is the very first facility of such type in Ukraine. It is a dry storage facility for storage of spent nuclear fuel of WWER-1000 reactors. The storage facility is designed on the basis of the VSC-24 storage cask. The storage facility is located on the site of Zaporizhzhya NPP. The way of the storage facility creation was not short and not simple. Wide enough experience is obtained by the people involved.

The chronological consistency of the choice of conception, designing, construction, commissioning and operation of the interim spent nuclear fuel storage facility at Zaporizhzhya NPP is described in the Paper. The place of the storage facility in the framework of the “Program of Spent Nuclear Fuel Management of Ukrainian NPPs” is presented.

The process of the storage facility licensing is described. This process chronologically concurs with the development of the licensing system in Ukraine. The problems which arose during the licensing process are described. The causes of the problems and means of their solution are analyzed. The spent nuclear fuel licenses issued by the State Regulatory Authority for Nuclear and Radiation Safety are presented. Licenses conditions are described. Some design modifications and additional measures for the storage facility safety ensuring made during the licensing process are depicted. The reasons for these modifications and measures are clarified.

The storage facility commissioning experience called forth some additional measures. Implemented and scheduled measures for design improvement are presented. The conditions of the storage facility operation and the plans for future modifications of the facility are described.

## THE CONSTRUCTION AND OPERATION EXPERIENCE OF THE INTERIM SPENT FUEL STORAGE FACILITY AT THE ZAPORIZHZHYA NUCLEAR POWER PLANT

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Six nuclear power units of the Zaporizhzhya NPP (ZNPP), which had been put into operation from 1984 to 1995, have not performance capabilities to extend their at-reactor (AR) spent fuel (SF) pools, the same as each and all of WWER-1000 units. Without SF removal [for reprocessing or away-from-reactor (AFR) interim spent fuel storage] SF pools are overburdened with nuclear SF assemblies in 5 (with compact reracking - in 9) years of their operation.

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

In 1994 the official decision was made to build the ISFSF on the basis of a ventilated storage cask (VSC-24) system, which was developed by US 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” (DE&S) to perform the design of the storage containers for the WWER-1000 spent fuel, the technology transfer and the service support of the ISFSF construction. The designing of the remaining components was carried out by Kharkov Design Institute “Energoprojekt”, which performed functions of the general designer for the Zaporizhzhya NPP.

The project implementation came across a great number of various difficulties. The most complicated ones arose with some licensing issues of technology. The creation difficulties of the ZNPP’s ISFSF can be grouped into five areas:

- Occurrence of the most safety-significant issues related to the project quality after weld cracking events in ventilated storage cask systems (VSC-24) at Palisades (March 1995) and Arkansas Nuclear One (December 1996) as well as ignition of hydrogen gas occurred while a worker was welding a closure lid at Palisades (May 1996);
- Lack of data on WWER-1000 spent fuel behavior during long-term storage in the dry environment and dry storage conditions;
- Absence of Ukrainian dry spent fuel storage experience that demanded simultaneous development of the lacking basic standards and resulted in increased conservatism concerning ISFSF’s safety assessment;
- Differences between safety regulatory requirements in country of technology’s origin (USA) and those of the customer country (Ukraine);

- Overcoming preconceived notion and obtaining the public acceptance in neighboring regions of the Zaporizhzhya NPP regarding the ISFSF commissioning. There were some financial and legislative difficulties also etc.

The licensing process of the ZNPP's ISFSF was initiated in March 1995 and was lasting for more than 5 years. In the course of the licensing process the ZNPP prepared and submitted to State Regulatory Body for consideration of the more than 270 documents, based on which about 50 examinations and expert assessments were conducted. Some project failings that were revealed while conducting safety justification, entailed to design modifications of the ISFSF.

Based on the permission issued by State Regulatory Body, on 11 July 1996, the Zaporizhzhya NPP began the spent fuel storage construction. A bed plate construction of the facility was the first step in this line.

On 20 March 1998 the Zaporizhzhya NPP got a license to produce a steel multy-assembly sealed basket (MSB) and a steel concrete cask (SCC), which are safety important units of the ventilated storage cask for WWER-1000 spent fuel (VSC WWER-1000). A manufacturer of the WWER-1000 VSC belongs to the Zaporizhzhya NPP.

The permission to manufacture a pilot batch of WWER-1000 VSC-24 (three casks) was obtained on 24 April 1998.

On 17 September 1999 the safety analysis report (SAR), version 01.2, was approved by the State Regulatory Body on nuclear and radiation safety.

On 17 May 2000 the State Regulatory Body on nuclear and radiation safety approved and published "The report on overall results of the safety analysis of the dry spent fuel storage at ZNPP", which contained positive conclusions.

On 16 July 2001 the NNEGC "Energoatom" obtained the license issued by the State Committee on Nuclear Regulation of Ukraine for commissioning the pilot complex of the ZNPP's ISFSF (3 casks).

According to the license a new nuclear installation started its work after placing the three pilot casks, which were loaded with spent fuel assemblies (SFAs) on 24 August, 30 August and 03 September 2001, respectively. Since then, a limited experimental-industrial operation of ZNPP's ISFSF is on-going.

Substantial difficulties arose from differences between US safety regulatory requirements and those of Ukraine. In the light of this issue the most obvious case is connected with the differences of safety requirements for subcritical state ( $\rho_{\text{subcrit}}=1-k_{\text{eff}}$ ). According to both regulatory requirements the subcrit of the loaded MSB of the WWER-1000 VSC must not be less than 5%. However the Ukrainian nuclear safety regulations require to take simultaneously into consideration two independent improbable events, namely: full loading of MSB fresh nuclear fuel assemblies with highest possible enrichment (4.4%) and water filling (without boric acid) with the water condition (density) of maximum multiplying. In this case, since as the Ukrainian regulations turned out much more



than US regulations for nuclear and radiation safety, the first three pilot WWER-1000 VSC were approved incomplete MSB charge – 22 SFAs instead of 24 SFAs. Moreover this requirement was ensured by means of installing additional poisoner (spent control rod) into every SFA.

In any case, from the conservative point of view, calculations performed do not take into account all actually potential loadings of MSB of WWER-1000 VSC. In this connection, the applicant suggested an approach, according to which a permission of SFAs loading into every MSB would be implemented

on the basis of the approved safety justification. Such an approach is acceptable and does not contradict the safety regulatory requirements. Thereby, the subcrit safety justification and compliance with safety criteria on strength, thermal and physical characteristics should be considered to get the permission for loading each MSB.

Other experience was connected to the welding problems. The ZNPP together with DE&S thoroughly reviewed information regarding the MSB weld cracking at Arkansas Nuclear One, Point Beach, Palisades and implemented modifications to welding procedures and welding materials. This work was realized by means of cooperation with a well-known Paton electrical welding institute (Kyiv).

A very important result is expected to get from a research work on dry long-term storage of WWER SF in collaboration with a Russian SRC “the Research Institute of Atomic Reactors” (RIAR).

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## U.S. NUCLEAR REGULATORY COMMISSION ACCEPTANCE CRITERIA AND CLADDING CONSIDERATIONS FOR THE DRY STORAGE OF SPENT FUEL

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The United States Nuclear Regulatory Commission (NRC) has been evaluating the technical basis for the storage and transportation of spent nuclear fuels over the last few years. The United States (US) regulations for dry storage of spent nuclear fuel require that the fuel be readily retrievable from a storage cask system and that the cladding be protected from degradation that may lead to gross ruptures during the 20 or more years of storage. Historically, the NRC staff has interpreted these regulation to mean that the integrity of the cladding should be preserved under storage conditions.

The spent fuel cladding is the primary structural component that is used to ensure that the spent fuel is contained in a known geometric configuration in storage and transportation casks. To assure that the configuration of the spent fuel is in the as-analyzed condition (via assuring cladding integrity or through the use of specially designed damaged fuel canisters), NRC staff has recently revised two staff guidance documents that identify the requirements that must be followed to assure that the operation and use of certified dry cask storage systems is conducted safely. One of the guidance documents describes the technical justification for prescribing the temperature limits that assure cladding integrity of the spent fuel during storage, while the other provides a means to assure that the configuration of damaged spent fuel is controlled in the storage cask.

Interim Staff Guidance No. 11, Revision 3, (ISG-11) contains guidance and acceptance criteria that are used by the NRC staff when reviewing analyses of the potential for spent fuel reconfiguration during storage operations. The staff's technical analysis, which will be discussed in the paper, demonstrated the need to control creep and hydride reorientation in the cladding during the 20-40 years of dry cask storage by assuring the cladding temperatures are controlled. The guidance prescribes cladding temperature limits of 400°C for normal conditions of storage and vacuum drying and 570°C for off-normal and accident conditions. The 400°C limit will allow licensees to safely store all spent fuel that is currently licensed by the NRC for commercial power plant operations without the use of specially designed equipment (e.g., damaged fuel cans). The use of a single temperature limit for all spent fuel eliminates the need for cask vendors to perform detailed calculations of cladding hoop stress and creep deformation (strains).

The staff anticipates that the majority of the spent fuel inventory placed in dry cask storage facilities will have intact, undamaged cladding. However, from a lessons learned perspective, there will be a small percentage of fuel rods that will have defects ranging from small pinholes, to axial splits, to larger fuel pellet-size openings, to parts/portions of rods due

to the various reactor emistries and operating conditions. Additionally, damaged fuel assemblies could have missing or relocated grid spacers and related hardware. To account for all of these variations of damaged fuel without increasing the burden on licensees, the NRC staff has revised its guidance on damaged fuel.

Interim Staff Guidance No. 1, Revision 1, (ISG-1) provides the definitions for damaged fuel, outlines the method for determining how damaged fuel should be treated in storage or transportation analyses, and provides guidance for classifying spent fuel as either damaged or intact prior to placing the fuel into storage or transportation casks. In general, any fuel that is classified as damaged fuel must be placed in specialized canisters for handling and retrievability or may be treated as intact fuel if an engineering analysis can be performed to show that it meets the regulatory requirements for storage and transportation.

The paper will describe the acceptance criteria for the storage of high burnup and damaged fuel (as they pertain to ISG-11 and ISG-1), the data and information used to support the development of these criteria, and additional NRC-sponsored programs that will be implemented in the future to address the technical issues associated with the storage and transportation of high burnup fuel.

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## INTERNATIONAL EXPERIENCE OF STORING SPENT FUEL IN NUHOMS<sup>®</sup> SYSTEMS

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The NUHOMS<sup>®</sup> system for spent fuel intermediate storage has two main components. The spent fuel is contained in a stainless steel canister that is sealed by welding two stainless steel lids. The canister is stored with its main axis horizontal in a concrete module (HSM). The canister provides a containment boundary and includes an internal basket structure to ensure criticality safety and good heat transfer. The HSM has an access door, an internal support structure for the canister and air vents for the rejection of heat by natural convection. The thick reinforced concrete walls offer excellent radiation shielding properties. After spent fuel loading, canisters are transferred from the spent fuel pool to the storage site using a shielded transfer cask with an integral hydraulic ram to facilitate canister loading into the concrete module.

The first designs were conceived for US commercial light water reactor fuel, and the first storage licenses were obtained from the US Nuclear Regulatory Commission. The initial system for PWR fuel has a capacity of 7 spent fuel assemblies, but this was increased to 24 in the next design. A canister for storing 52 BWR fuel assemblies was soon added to the fleet. As the demand for dry storage systems in the US increased, Transnuclear, Inc identified a market need for higher capacity systems and two further designs were added with capacities of 32 PWR and 61 BWR assemblies. These designs are further evolving to match the trends in increased fuel initial enrichment and burn up.

In addition to meeting the US regulatory requirements for storage, the latest systems are designed to meet US NRC requirements for transportation. This latest development gives NUHOMS<sup>®</sup> users the additional flexibility of a dual-purpose system. Another NUHOMS<sup>®</sup> system was successfully developed to store fuel debris from the Three Mile Island reactor. Solutions have also been developed for safely storing and transporting damaged fuel.

Outside the US, the NUHOMS<sup>®</sup> system has attracted considerable interest. As a licensee for the NUHOMS<sup>®</sup> technology, Framatome has supplied a NUHOMS<sup>®</sup> system for storing WWER fuel assemblies at Metzamor in Armenia. Framatome is also supplying a NUHOMS<sup>®</sup> system for storing RBMK fuel at Chernobyl in the Ukraine.

The worldwide popularity of the NUHOMS<sup>®</sup> system is due to a combination of economic and technical factors. As a system for intermediate dry storage, the modular design allows owners

and operators to defer investment by increasing the installed capacity incrementally. Relatively short fabrication times allow the installation to be timed to meet the operational needs for spent fuel loadings. Local fabrication of the concrete module is another factor that can influence system selection when the use of local labor is desirable. However, the overriding technical advantage of the NUHOMS<sup>®</sup> system is the inherent flexibility of having its two main components designed to fulfill specific technical functions. These components can be individually adapted to suit a specific need without altering the overall system concept. For example, reducing the external radiation dose rates at the storage site is a simple matter of increasing the thickness of the concrete walls of the storage module and this has no direct influence on the canister operations.

International experience in the design, licensing and operation of the NUHOMS<sup>®</sup> system shows it to be a safe and reliable method for both the intermediate storage and off-site transportation of spent fuel. Its flexibility enables a wide range of fuel types to be stored and transported. It is also readily adaptable to local constraints such as handling weight limits, restricted access to fuel pool buildings and a wide range of environmental conditions.

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## THE STORAGE OF SPENT FUEL IN VIETNAM: PRESENT STATUS AND PROSPECTS

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The status of storage of spent fuel in Vietnam is reviewed. It includes the current status at the Dalat Nuclear Research Reactor (DNRR) and the prospects to future nuclear power plant.

At DNRR: since its reconstruction and upgrading, a temporary spent fuel storage was designed and constructed. It's a pool storage with dimension of 2045x1808x3700mm containing distilled water, where 300 fuel assemblies could be stored in.

The method used for calculating the burnup of the Dalat reactor spent fuel is based on the codes of WIMS, HEXAGA, and HEXNOD. The experimental method used for evaluating the burnup of spent fuel is relative method by gamma scanning or measurement of long-lived fission product  $^{137}\text{Cs}$  and activation product  $^{134}\text{Cs}$  [1].

For ensuring the quality of water in the pool storage, the sanitation work has been carried out regularly by circulation pumping and filtering.

The prospects: In the next decade, a nuclear power plant should be introduced in Vietnam. The problem related to the storage of spent fuel will depend on the nuclear power, type of fuel and the policy of supplier. The experience with spent fuel storage of nuclear power plants in the world would be gained and used by the future managers and operators in Vietnam.

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**SFCOMPO: A DATABASE FOR ISOTOPIC COMPOSITION OF NUCLEAR SPENT FUEL; CURRENT STATUS AND FUTURE DEVELOPMENT**

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The isotopic composition of the nuclear fuel, after the irradiation in a power reactor, is one of the most important characteristics of spent nuclear fuel (SNF). It is the basis for the evaluation of several properties of SNF, including decay heat, radiation dose, and the multiplication factor of any configuration containing the spent fuel. Since the determination of this isotopic composition is usually carried out using calculation tools, the validation of these tools is an important aspect to be taken into consideration. Several experiments involving Post Irradiation Examination (PIE) of SNF have been conducted in different countries to obtain experimental data for the isotopic composition of SNF. These experimental data can be used for the validation of the calculation tools.

Based on this background, the Japan Atomic Energy Research Institute (JAERI) developed SFCOMPO [1-4], a database system for the compilation of the isotopic composition of SNF. The database was available for consultation on the Internet. After this initial phase, and given the suitable framework of the NEA for the collection of additional data from member countries, it was agreed that the NEA would maintain and further develop SFCOMPO. This involves both the compilation and the dissemination of the data. Since autumn 2002, this database has been operated on the NEA web server (<http://www.nea.fr/html/science/wpncs/sfcompo/>).

Table 1 summarizes the content of the SFCOMPO database. Measured isotopic composition data from 14 reactors (7 PWRs and 7 BWRs), operated in 4 countries, are contained in the database. The composition of 246 samples is described, including 30 samples from UO<sub>2</sub>-Gd<sub>2</sub>O<sub>3</sub> fuel. The database gives the composition of U, Pu, Am, Cm and several fission products (Nd, Cs, Sr). These data were collected from the open literature.

In view of the importance of these data, the NEA plans to further develop the SFCOMPO database. The aim is to give more details on the operating conditions under which the fuel was irradiated, and to enter data from other reactors. It is hoped that the co-operative framework provided by the NEA Nuclear Science Committee will significantly facilitate this development.

The outline of the SFCOMPO database, as operated on the Internet, will be presented and the plans for its future development will be discussed.

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TABLE 1. LIST OF REACTORS IN SF-COMPO VER.2

Reactor	Country	Reactor Type	Fuel Assembly Type	Fuel Type	Total Number of Samples in Database* (Number UO <sub>2</sub> -Gd <sub>2</sub> O <sub>3</sub> Samples)	Remarks
Obrigheim	Germany	PWR	14×14	UO <sub>2</sub>	23	
Gundremmingen	Germany	BWR	6×6	UO <sub>2</sub>	12	
Trino Vercellese	Italy	PWR	15×15	UO <sub>2</sub>	39	Fuel Assembly is different from modern type.
JPDR	Japan	BWR	6×6	UO <sub>2</sub>	30	Naturally Circulated
Tsuruga-1	Japan	BWR	7×7	UO <sub>2</sub>	10	
Fukushima-Daiichi-3	Japan	BWR	8×8	UO <sub>2</sub> , UO <sub>2</sub> -Gd <sub>2</sub> O <sub>3</sub>	36 (10)	
Fukushima-Daini-2	Japan	BWR	8×8	UO <sub>2</sub> , UO <sub>2</sub> -Gd <sub>2</sub> O <sub>3</sub>	18(10)	
Mihama-3	Japan	PWR	15×15	UO <sub>2</sub>	9	
Genkai-1	Japan	PWR	14×14	UO <sub>2</sub>	2	
Takahama-3	Japan	PWR	17×17	UO <sub>2</sub> , UO <sub>2</sub> -Gd <sub>2</sub> O <sub>3</sub>	16(5)	
Cooper	USA	BWR	7×7	UO <sub>2</sub>	6	
Monticello	USA	BWR	8×8	UO <sub>2</sub> , UO <sub>2</sub> -Gd <sub>2</sub> O <sub>3</sub>	30(5)	
Calvert Cliffs-1	USA	PWR	14×14	UO <sub>2</sub>	9	
H.B.Robinson-2	USA	PWR	15×15	UO <sub>2</sub>	6	

\* Including Samples from UO<sub>2</sub>-Gd<sub>2</sub>O<sub>3</sub> Fuel Rods

## OPTIMIZATION OF CASK CAPACITY FOR LONG TERM SPENT FUEL STORAGE

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Long term storage of spent fuel is a priority topic within the Member States of the IAEA. Long term spent fuel storage was previously addressed in an IAEA Co-ordinated Research Project /1/, which recognized the growing challenge of extending the life of storage facilities. Dry cask storage of spent fuel is playing a steadily increasing role in this regard. Storage practices should comply with IAEA safety requirements “International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources” /2/, including maintaining doses as low as reasonably (taking economic/social/etc aspects. into account) achievable [i.e., the ALARA principle].

Within the framework of the IAEA Subprogramme of Spent Fuel Management, a new project was conceived, focusing on issues associated with the optimisation of cask/container loading (capacity) with respect to long term storage and the related integrity of fuel, see IAEA /3.

Optimization is a part of the design process in which the combination of application objectives, regulatory limits and design margins are innovatively addressed and judiciously balanced in the final design. A primary result of a successful design optimization is a cask of superior assembly and burnup/age capacity that minimizes the total number of required cask loadings. An equally important and parallel benefit is that this process also results in reduced radiation exposure, thereby contributing significantly to maintaining doses as low as reasonably achievable (ALARA objectives). In this sense, both cask designers and regulators have the common ultimate goal of improving cask performance, and thus facilitating optimization.

An initial Consultants Meeting held in November 2002 identified and discussed principal issues regarding the optimization of cask/container assembly capacity and burnup/age capability in the design of systems for long term spent fuel storage and the related integrity of fuel. Working materials developed during that meeting noted that cask designers currently face a number of new challenges including storage of high burnup fuel with correspondingly higher enrichments, the use of mixed oxide (MOX) fuel, and obtaining regulatory approval for the use of burnup credit. Optimization might have different meanings for the cask vendor, the cask owner, the cask operators, and the institution having the ultimate responsibility for the storage, the Licensing and Supervisory Authority. The expected useful life of the fabricated materials that make up the physical cask components is generally expected to be more than 100 years, assuming reasonable maintenance and careful monitoring (e.g., lifting trunnions). Regarding the spent nuclear fuel (SNF) contents of the cask, experience with dry storage is not extensive but gives encouragement that extended periods of dry cask storage without loss of cladding integrity are a realistic expectation. However, considerable additional data is needed concerning cladding performance during longer storage periods and subsequent transport.

A Technical Meeting was held in March 2003 to obtain country-specific views from both regulators and implementers on this topic. After reviewing results of the consultancy and country specific perspectives, participants formed two working groups to focus on implementer/regulator views of the following issues :

- (1) Fuel integrity: Implementers confirmed the need for cladding-related data on creep, cladding absorption of hydrogen, stress corrosion cracking, oxidation, internal gas pressure (helium buildup. Regulators noted “tightness” of potentially damaged (mechanically) fuel rods required further Retrievability: Both groups confirmed the need for early definition of requirements, noting that requirements would vary depending functions (e.g., dual purpose casks).
- (2) Zoning: The potential benefits of zoned cask loading need to be investigated with respect to criticality, shielding, heat removal, operator flexibility.
- (3) Burnup Credit: In order to pursue the storage-related advantages of burnup credit, it is necessary to have good knowledge of spent nuclear fuel characteristics, from both measurement and calculations
- (4) Damaged Fuel: Guidance, including consistent definitions, should be elaborated to ensure that optimisation efforts do not impact protection goals.
- (5) Internal Moderator: Both groups concluded that further development of this topic should be in the context of specific concepts.
- (6) Computer Code Verification: Additional qualification of codes is needed for specific designs. Careful definition of source terms will continue to be important.

In addition to the above issues identified in the consultancy, participants in the March TM identified the following additional topics: Life of cask components, Long term cask maintenance, Long-term performance confirmation, Long term records management.

The above meetings served as key steps toward developing the knowledge base available to IAEA Member States on this subject. Follow-on actions and meetings will be pursued to develop a TECDOC on this subject.

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## SELECTION OF AFR FACILITIES FOR SPENT FUEL STORAGE

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Most countries with growing inventory of spent fuel, being devoid of further destination, have to provide additional capacities for storage which could have been implemented in the past by such simple methods as re-racking in the temporary storage pools at reactor (AR). Technical measures to expand capacities at existing facilities, however, have almost been used up by now and additional capacities have already begun to be provided by new builds of away-from-reactor (AFR) type which can be defined by its independence from the reactor facilities regardless of its locations. With the current trends toward longer term storage of spent fuel in most countries, the demand for AFR facilities for interim storage is expected to continue to increase in the future, for a significant period of time until an eventual breakthrough could have resolved the issues on the endpoint of spent fuel management.

Concerning technical options for spent fuel storage, the successful commercialization of dry storage methods by the industry in the past couple of decades has certainly added some powerful options for AFR storage, with particular advantages for long term storage, and has already begun to dominate since years the new market for AFR storage. The conventional pool storage is still prevalent on global statistics largely for historical reasons, but for some inherent advantages as well in some cases. While it can be said with confidence that competitive services are currently available from the market, it is often not evident how to choose the best option because of the complex factors, especially of non-technical ones, to be considered in the decision. There are a number of factors identified and discussed in this context, with a general consensus that those factors are often unique to the given case and apply in a uniform manner. The selection of AFR storage facilities is in fact a very critical step for successful implementation of a spent fuel storage project, due to a variety of factors with possible consequences in the lifetime of the facilities or even thereafter

The importance of AFR storage in the fuel cycle backend has long been recognized and were dealt in a number of relevant publications. However, some aspects of today's reality associated with AFR storage have significantly evolved from those observed in earlier times giving rise to a need for new deliberations. It should be noted that the focal issues surrounding local, national, or international dimensions are dynamically changing in the present world of globalization, which might emanate some overriding impacts to the choice of option and implementation strategy. An example of recent issue could be the protection of spent fuel storage facilities from terrorism. The recent debate on the retrievability (or reversibility) of spent fuel after disposal at the end of storage could be another exemplary instance of such a circumstance. This paper looks at the various criteria involved in the selection for AFR storage facilities together with approach to the implementation methods.

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## THE GERMAN POLICY AND STRATEGY ON THE STORAGE OF SPENT FUEL

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The policy in the management of spent fuel in Germany has been changed several times. Until 1994 in the Atomic Law there has been included a requirement of reusing the fissile material in the spent fuel elements. This requirement has been changed with the amendment of the Atomic Law in the year 1994 in a way that beginning from this time the operators of nuclear power stations have the possibility to choose the path of reusing the fissile material of the spent fuel by reprocessing them or of the direct disposal of the spent fuel.

After changing of the German Government in 1998 the new Government decided, in accordance with the electricity producers, to phase out of nuclear power. The phasing out was fixed 2002 in a law on basis of an agreement between the Federal Government and the Utility Companies from 2000. In this law for each nuclear power plant a remaining operation time is fixed. In context with the change in energy policy a new waste management concept was developed. Concerning the spent fuel management the separation of Uranium and Plutonium of the spent fuel was not any longer desired by political means with the consequence that only the direct disposal of the spent fuel will be allowed.

A sudden giving up of the reprocessing however was not possible because there have been contracts between the operators of the nuclear power plants and the reprocessing plants in La Hague and Sellafield being granted by changing of notes between the government of the Federal Republic of Germany and the governments of the two reprocessing states. Taken into consideration this situation the abandoning of the reprocessing was demanded in the middle of 2005 by law. After this date no transports to the reprocessing plants will be allowed with the consequence that beginning from this time only a direct disposal of the spent fuel remaining or further arising in Germany will be accepted.

The Federal Government is aiming at the establishment of a repository in deep geological formations until the year 2030 for the disposal of all kinds of waste including the spent fuel. Until the commissioning of the repository the spent fuel have to be stored which has to be practiced at the site of its arising for reasons of avoiding transports and of burden sharing. That should be done in interim storages built at the sites of the nuclear power plants, as required by law, and which are provided only for the spent fuel elements produced at the site. The storage takes place in casks in a dry way. The spent fuel elements of the dismantled nuclear power plants of soviet design in the former GDR shall be stored too in a central storage in Greifswald in casks in a dry way in exceptional cases, if the storage at the site of the nuclear power plants is not possible, there are two central storages at Ahaus and Gorleben which are in operation and can be made available as reserve.

## SPENT FUEL MANAGEMENT STRATEGY IN JAPAN

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Since the start of operation of the first commercial nuclear power plant in 1966, Japan has gradually and steadily increased nuclear power plants. At present, 52 commercial nuclear power plants with the capacity of 45.1TW are being operated and supplying 34% of total electricity generated in FY2001.

To ensure stable and reliable long-term energy supply and to reduce environmental effects such as global warming, the fundamental energy policy is to raise the share of nuclear energy in electricity generation to 42% in 2010. Moreover, all the spent fuel will be reprocessed in order to make efficient use of nuclear material and to enable appropriate management and disposal of radioactive waste. Plutonium recovered from the spent fuel will be loaded in LWRs in the form of Plutonium and Uranium Mixed Oxide (MOX) fuel or utilized for R&D of nuclear fuel cycle such as FBR. Utilities have a program to start loading MOX fuel in 16 to 18 commercial LWRs by the year 2010.

The amount of spent fuel discharged till September 2002 was 18,800tU, and about 8,000tU of them had been transported to JNC's Tokai Reprocessing Plant or foreign reprocessing plants (COGEMA and BNFL) and the other 10,800tU are being stored in pools or metal casks inside nuclear power plants, which total capacity is about 16,000tU .

The Rokkasho Reprocessing Plant of JNFL with the reprocessing capacity of 800tU/y is now under construction and is planned to start commercial operation in July 2005. Its spent fuel storage pool with the capacity of 3,000tU has been already operated since 1999, and is storing spent fuel of 780tU. By the time of start of the said operation, 1,600tU of spent fuel will be stored.

At present, total discharge rate of spent fuel in Japan is 900 to 1,000tU/y, and the rate will increase as further expansion of electricity generation by nuclear power.

For the appropriate storage and management of the increasing spent fuel, it is important to construct interim storage facilities outside nuclear power plants, which will enable time adjustment till reprocessing them. Amount of spent fuel required off-site storage is estimated at 7,100tU by the year 2010. The Law for Regulations of Nuclear Source Material, Nuclear Fuel Material and Reactors was amended in June 1999, to establish a legal basis for private organizations which construct and operate interim spent fuel storage facilities outside nuclear power plants. The Nuclear Safety Commission issued a safety evaluation guideline for the facilities in October 2002 and the utilities have started preparatory activities including the siting efforts for the off-site interim storage facilities.

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## EXPERIENCE FROM EXTENSION AND LICENSING OF THE SWEDISH CENTRAL INTERIM STORAGE FACILITY FOR SPENT FUEL, CLAB, FROM 5000 TO 8000 METRIC TONNES

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Since the operation start of CLAB in 1985, 3880 tonnes of spent fuel has been stored at CLAB. Spent fuel together with radioactive components, stored at CLAB, occupies 92% of the storage pools. The plan for sending spent fuel to CLAB during the next years, indicates that CLAB will be full at the end of 2003. In order to meet the coming demand from the NPP, CLAB capacity will be extended from 5000 to 8000 tonnes of spent fuel.

The project to build storage pools in connection with already existing storage pools began in 1999. The first step was to excavate the rock in parallel with the existing storage pool area at a distance of app. 40 m. During this time the existing CLAB was measured to detect vibration that could harm. The excavation was done with help of explosives.

After the excavation was finished in the beginning of 2001, the building of the storage pools and other concrete structures began. At the beginning of the building period a group of skilled people from both the project and the operation of CLAB was formed in order to detect and analyse potential risks that could jeopardise the integrity of CLAB. The building of the concrete structure is planned to be accomplished in the beginning of 2004.

In parallel with the building of the concrete structure, the upgrading of the systems in operation, which also should serve the new storage pools, began and continues until the new storage pools are connected. This upgrading of the existing system demands a thorough planning to not violate the operation condition set by the regulatory body.

App 6 month before operation of the extended CLAB, no spent fuel or radioactive material will be received though this time will be used to test all system and to show the regulatory body that the extended CLAB meets all regulatory demands.

From the start, the regulatory body through inspections and meetings has followed the project very close. Some experience from the interaction with the regulatory body is:

- The regulator should be active very early
- An open and regular dialogue with the regulator to discuss regulatory requirements and review findings
- Regular meetings between the regulator, operator and the operator's contractors
- On site inspections

## GAMMA-RAY CONTROL OF METAL AND CONCRETE CASK RADIATION PROTECTION

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Metal and concrete cask for durable storage and transportation of the spent fuel is equipped by the remote control device for verification of radiation protection in particular concrete entering and making faults. Operation tenet is irradiation of the cask wall by gamma-rays with an exposure rate at the surface chart design.

### Introduction

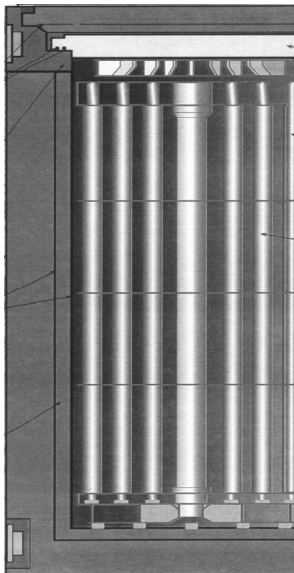


Fig.1. Metal and concrete cask

In compliance with the requirements of national standards and regulations which are valid in nuclear power engineering and also IAEA recommendations [1], transportation packaging modules (TPM) for long-term storage and shipment of the spent nuclear fuel (SNF) have to ensure rated protection against ionizing radiation and withstand emergency impacts while preserving integrity of tightness system and radiation protection.

Special Mechanical Engineering Design Office (SMEDO) has developed and “Ijorskie Zavodi” JSC et al. manufacture such a module on the basis of metal and concrete cask (TPM MCC) for spent nuclear fuel of RBMK-1000 reactors, NPS (nuclear-powered submarines) etc. In general, the structure of

MCC may be presented as three coaxial steel shells the space between them being filled with high-density (4 and

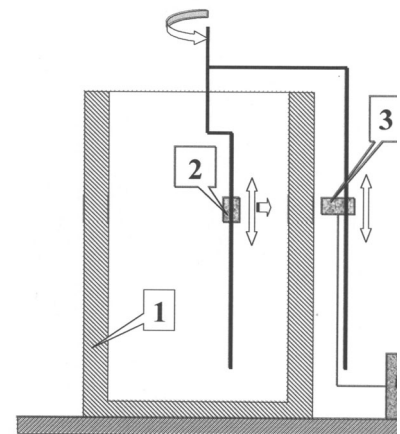


Fig.2. Scheme of the test bench

- 1 – Cask
- 2 – Gamma source
- 3 – Detector
- 4 – Control unit



3.5 g/cm<sup>3</sup>) concrete of high ductility and reinforced with composite grid of bars, clamps and rings (Fig.1).

We have developed a procedure to control radiation protection (RP) of this cask, RP integrity checks after dynamic testing which simulate emergency situation during transportation. Test bench of  $\gamma$ -control [2,3] was designed and constructed as a technical decision of these tasks.

## 2.0. Procedure

The task included assessment of the cask radiation protection parameters, correlation of estimated and pilot data, cask manufacturing quality control as well as assessment of its body

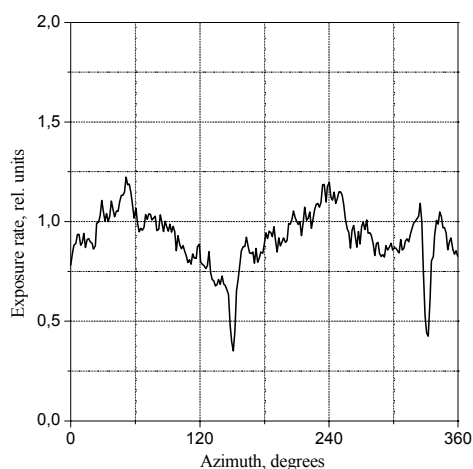


Fig.3. Exposure rate at the MCC surface

concrete filling uniformity and finally, confirmation of RP integrity under applied dynamic loads. RP  $\gamma$ -control method has been selected for this task-solving. This method is based on radiometry of cask walls by irradiation from a radioactive source. So successive investigation and gamma-ray flaw detection of cask wall are being performed in order to determine local values of exposure rate (ER) at its outer surface. This type of control enables to detect inside imperfections (cavities) of body concrete filling and technological deviations of the article manufacturing process: misalignment of shells, ellipsoid shape, deviations of mass thickness etc.

Gamma-control test bench of MCC is an electric mechanical facility fitted with

remote PC-based programmable control (schematically see Fig.2). Its operation is based on synchronous scanning of the cask surfaces provided by a source of ionizing radiation and detection unit. The source-detector couple passes around MCC horizontal perimeter with constant angular speed and during estimated period of time. Thereupon it shifts for a pre-set pitch in vertical plane and run recurs. Thus, the whole cylindrical surface of the cask from top to bottom mark is investigated.

Radioactivity of the radiation source has been specified by optimal selection between adequacy of pulse calculation (ER value) at the cask outer surface and safety operating conditions of personnel. Finally the industrial defectoscope with source of radiation Co-60 on radioactivity of  $10^{11}$  Bq (lines 1.17 MeV and 1.33 MeV, half-life period 5.3 year) is used. Crystal NaJ (Tl)  $\varnothing$  63 mm runs as a detector. Its energy operating range is 20-3300 keV, calculation interval is 300-10000 pulses/s, ER

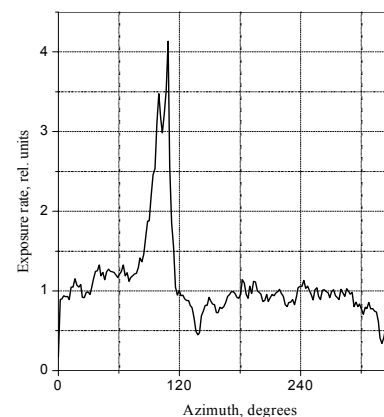


Fig.4. Exposure rate at the MCC surface with defect

calculation error is  $\pm 10\%$ . Information is transmitted via signal conversion circuit and interface and enters PC input both for routine display on the monitor and for accumulation and processing. Operational information is displayed on PC monitor for video control in the form of bar chart: descriptive diagram of ER dependence at outer surface on angular coordinate for one perimeter (Fig. 3). A set of bar charts which have been taken along the whole height of the cask give a complete display of ER values in all points of its surface. Bar chart is a detailed print of exact part of protective wall and for instance, it is acknowledged by its "thin structure", i.e. regular vertical decrements corresponding to bar grid which reinforces concrete. Also being fixed are: tie beams, superposed flanges, other technological elements of article body. One of the main tasks of control is to detect inside cavities in concrete filling (if exists). This task has been already resolved for instance during investigation of global large scale model (1:2.5) when bar charts enabled not only to indicate coordinates of a cavity but also to assess its linear dimensions (Fig.4). Afterwards both method and technique were tested on routine MCC units (lot over 50 ones). The patent for system is available.

To reveal defects the special results processing code SCAN 2D was developed. It allows to indicate flaw coordinate and linear dimensions. The code holds patent registration.

## Results

Estimated assessment of method sensitivity of minimum detected imperfection linear size was satisfactory. It matched with pilot data (calibration) and was accounted for 20 mm.

By testing, display and processing of test results in PC provided buildup of multi-color resultant chart-scanning of ER values on the whole surface of the cask is accumulating. It shows reliable and potentially weak sites of protection, technological elements, bars, hardware and a lot of other elements in compliance with pre-set color scale (24 color hues). Besides, existing portion of this chart is permanently available on the monitor display along with routine bar chart and therefore provides a clear view of probable imperfections for an operator thus enabling to make operational decisions for repeat measurements, local definitions, etc.

A procedure for identification of real imperfections in concrete filling of cask body was created. Control level of imperfection RP is introduced in test bench software support as a threshold signal.

## Conclusions

Selected method of  $\gamma$ -control showed its applicability, reliability and representation during investigation of global model and variety samples of metal and concrete casks for SNF.

A test bench for practical solution of this procedure is designed and developed. Durable operation of test bench had confirmed an accurateness of estimated design, technological and program decisions during its development.

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## EXAMINATION OF THE CREEP RUPTURE PHENOMENON AND THE DEVELOPMENT OF AN ACCEPTANCE CRITERION FOR SPENT FUEL DRY STORAGE

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Significant research has been performed in recent years in the behavior of spent fuel during dry storage in response to regulatory initiatives that had been highly restrictive in establishing conditions for placing high burnup fuel in dry storage. At issue are release and fuel retrieval consequences in the event of cladding failure due to creep, or other potential damage mechanisms, under sustained long-term temperature and pressure. To safeguard against cladding failure, the USNRC issued interim staff guidance that limited the allowable creep strain during the dry-storage license period to 1%, with the additional restriction that zirconium oxide thickness greater than 70 or 80  $\mu\text{m}$  would be limited to less than 3% or 1%, respectively.

In the last several years, a major analytical effort by EPRI, supported by pioneering experimental research by EDF [1], has been undertaken to evaluate spent fuel creep behavior and to examine the technical basis for the 1% strain criterion. These efforts have led to the development of a creep-based methodology for high-burnup cladding behavior in dry storage [2,3], which addresses the phenomenon of cladding creep and creep-rupture in general, including the effects of various damage mechanisms that can have synergistic effects on cladding creep and potential failure. These mechanisms include: (a) the effects of irradiation-damage recovery on the creep rate and the counter-acting effect of hydrogen; (b) the effects of outer-surface corrosion and oxide spallation on cladding stress localization and its consequential effect on the acceleration of creep deformations; and (c) the role of temperature decay on cladding creep during dry storage.

Application of the developed methodology to various testing conditions and dry storage temperature/pressure histories illustrates two important behavioral regimes that are relevant to creep-based licensing criteria. First, creep rupture is brought on by stress-induced plastic instability as a terminal state of tertiary creep, which is possible only if the cladding's yield strength is exceeded, at typically 700 MPa for high-burnup fuel. Such a state of plastic instability is highly unlikely to be reached in dry storage because of the second behavioral regime, namely, that cladding creep deformations under dry storage conditions tend to be self-limiting. This is characterized by a continuously decreasing strain rate asymptotically to zero, which is an inverse behavior to that of tertiary creep. This self-limiting nature of creep deformations is a property of closed pressurized systems, which characterizes the state of fuel rods in dry storage, where the cladding stress is continuously decreasing in proportion to the creep-induced volume expansion and decaying temperature.

These behavioral regimes are illustrated in Figures 1 and 2 respectively. Figure 1 shows the evolution of creep strain under constant pressure for a 1500-hour creep test conducted at Argonne National Laboratory (ANL) [4]. The analysis was carried out well beyond the test duration to illustrate the onset of tertiary creep, which leads to the acceleration of creep deformations and eventually plastic instability followed by rupture. Figure 2 is an analytical simulation of dry storage for a fuel rod with a hydride lens, with similar conditions to the ANL test, showing the self-limiting behavior of cladding creep (even in the local hydride lens region), in contrast to the constant-pressure creep test of Figure 1. Both the hydride lens, penetrating nearly 50% of the cladding thickness, and the zirconium oxide are simulated in the analysis as metal loss.

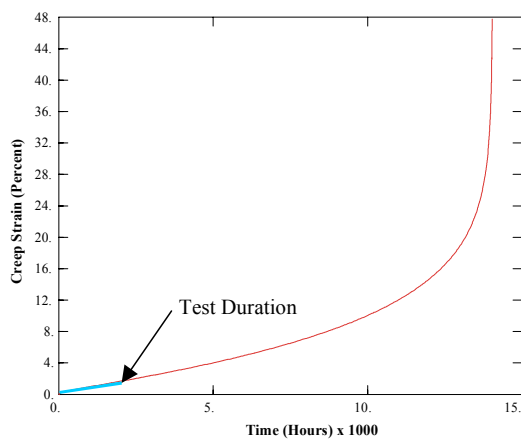


Figure 1 – Constant Pressure Creep Response at 400°C

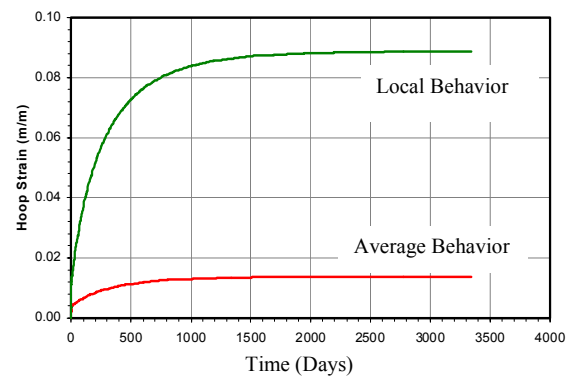


Figure 2 – Response in Dry Storage

The study shows that the 1% strain criterion has no valid basis for judging cladding integrity under creep, and a stress-based criterion is substituted instead as an acceptance criterion for spent fuel dry storage. This is discussed in detail in the paper, considering the role of high burnup effects on cladding performance in dry storage, including corrosion and localized hydrides.

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## EXPERIENCE OF THE OPERATION OF THE INTERIM STORAGE FACILITY FOR SPENT FUEL IN OLKILUOTO

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Finland has two nuclear power plants both of which have two reactor units: Teollisuuden Voima Oy (TVO) owns and operates two 840 MW BWR-type nuclear plant units at Olkiluoto, Eurajoki on the west coast of Finland and Fortum Power and Heat Oy two 510 MW PWR-type units at Hästholmen, Loviisa on the south coast.

Spent nuclear fuel is stored temporarily in water pools at the power plant sites. Both power plants own and operate their own intermediate spent fuel storages at the sites, Loviisa and Olkiluoto, from where the spent fuel will be subsequently transferred for final disposal in a geological repository deep in the bedrock. The power plants concerned are responsible for the implementation and costs of spent nuclear fuel management. In 1995, TVO and Fortum Power and Heat Oy established Posiva Oy to be responsible for final disposal of the spent nuclear fuel generated both at the Loviisa and the Olkiluoto power plant units and for a host of other expert tasks relating to spent nuclear fuel management.

In May 2001, the Finnish Parliament ratified the decision in principle concerning the final disposal of spent nuclear fuel, accumulating from the four power plant units, in the bedrock at Olkiluoto. In May 2002, the Parliament ratified a decision in principle concerning the final disposal of spent nuclear fuel from the planned new nuclear power plant unit also in Olkiluoto, regardless from where the new power plant will be located, Loviisa or Olkiluoto.

In the next few years, Posiva will start the construction of an underground characterization facility on the island of Olkiluoto at Eurajoki, which has been selected as the site for the final disposal facility for spent nuclear fuel as mentioned above. This characterization facility is called ONKALO, and it will be used to acquire detailed information about the bedrock at Olkiluoto, to be utilised in the planning of the final disposal facility. ONKALO will finally be used as part of the final disposal facility. According to the programme, the construction of the final repository will begin in 2010 and the disposal will be started in 2020.

The interim storage facility for spent fuel at the Olkiluoto plant site, the KPA Store, is a wet AWR storage located close to the power plant units. The storage has been in operation since September 1987. The storage was designed by the Finnish company Imatran Voima Oy (IVO) nowadays Fortum Power and Heat, with the Finnish building company YIT as the main construction contractor. Consequently, the construction and design of the storage facility was over 90% Finnish, and only the transfer cask, the storage racks and a few valves came from other countries.

The storage comprises the storage building and a separate seawater pumping station. The storage building is built into the bedrock and is partly underground. The storage building is divided according to function into a reception section, a storage section, a process section and a control section. The storage contains a cooling system that is used to transfer the heat in the fuel to the sea. The cooling system comprises two cooling chains, each of which contains three cooling circuits. One of the cooling chains alone is sufficient to cool the fuel. The storage building is connected with two tunnels to Olkiluoto 1 and with pipelines in the ground to the seawater pumping station. The necessary process and tap water is conducted through one of the tunnels from Olkiluoto 1. Used resins and the water to be purified are pumped to the waste building at Olkiluoto 1. Radiation control, monitoring of air and pool water is conducted in the same way as in the power plant units.

The storage has three storage pools and one evacuation pool, a storage capacity of totally 1,220 tU and is designed to store the spent fuel generated from both power units Olkiluoto 1 and 2 during an operational time of 40 years. The storage can be enlarged with extra pools if the operational time of the power plants will be continued, as TVO expects, and also in case the fifth reactor is going to be constructed at the Olkiluoto site. The cooling capacity of the storage ponds will be increased accordingly, if necessary.

After a cooling time of two to five years the spent fuel is transferred from the power unit to the storage in a CASTOR spent fuel transport cask, which has space for 41 BWR fuel assemblies (7.3 tU) and weighs filled about 93 tons. During transfer, the cask is water filled. The residual heat max allowed is 20.8 kW. 104 transfers have successfully been made and there are for the moment 4,264 fuel assemblies, 59 % of the capacity, stored in the storage pools.

No leaking fuel is transferred to the storage. Leaking fuel rods are discharged from the leaking fuel assembly and put into hermetically sealed capsules, stored in a rod rack, which has the dimension of and is handled like a fuel assembly.

The storage is supervised and controlled by the control room personnel of the Olkiluoto 1 unit. All alarms are routed to its control room, and the control room staff makes daily inspection rounds at the storage. There is currently no permanent staff at the storage facility. The group that takes care of the waste and fuel handling is also responsible for the transportation and handling of spent fuel, as well as the operation of the storage facility.

One of the main goals of the storage was that the handling of fuel assemblies and the operation of the storage processes must be easy enough not to require employing permanent staff for these purposes. This goal has been achieved. All together the operational experience of the KPA Store has showed that the expectations have been met.



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## CANADA'S NATIONAL POLICY ON THE LONG-TERM MANAGEMENT OF NUCLEAR FUEL WASTE

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Nuclear energy is an important part of Canada's diversified energy mix. There are 22 CANDU reactors in Canada located in the provinces of Ontario, New Brunswick, and Québec. Like any other industry, nuclear fuel cycle operations produce some waste, and for this paper, we will focus on nuclear fuel waste, i.e., the irradiated fuel taken out of nuclear reactors at the end of their useful life. Canada has no plans to reprocess and recycle this used nuclear fuel, so current plans are based on direct long-term management. Although nuclear fuel wastes is currently in safe storage, steps are now underway to develop and proceed effectively with the implementation of long-term management solutions.

A cornerstone of Canada's approach to addressing radioactive waste management is the Government of Canada's 1996 Policy Framework for Radioactive Waste, which has set general policy for dealing with a all radioactive waste from the nuclear fuel cycle (nuclear fuel waste, low level radioactive waste, and uranium mine and mill waste). The Framework clearly indicates that the federal government will ensure safe, environmentally sound, comprehensive, cost-effective and integrated waste management, including disposal; that it will develop policy, regulate and oversee the waste owners to ensure compliance with legal and financial requirement in accordance with approved disposal plans; and that the waste owners are responsible for the funding, organization, management and operation of long term management, including disposal, facilities.

With respect to the long-term management of nuclear fuel waste, a deep geological disposal concept was developed by the federal crown corporation Atomic Energy of Canada Limited (AECL) and Ontario Hydro, and, in October 1988, it was referred by the government for review by an independent Federal Environmental Assessment Panel. AECL submitted the Environmental Impact Statement to the Panel in 1994. The Panel reported its conclusions and recommendations on the acceptability of the concept in March 1998. It found that *"from a technical perspective, safety of the AECL concept has been on balance adequately demonstrated for a conceptual stage of development, but from a social perspective, it is not. As it stands, the AECL concept for deep geological disposal has not been demonstrated to have broad public support. The concept in its current form does not have the required level of acceptability to be adopted as Canada's approach for managing nuclear fuel waste"*. Thus it was recommended that Canada should increase public confidence before proceeding with any general approach on the long-term management. With the Panel's recommendations in mind, and with further consultations with stakeholders, including the public, the Government of Canada developed the *Nuclear Fuel Waste Act (NFW)* which came into force on November 15, 2002.

The *NFW Act* is a stand-alone piece of legislation with some 30 articles and without regulations. The *NFW Act* deals essentially with social, financial and socio-economic considerations of the long-term management of nuclear fuel waste. It complements the health, environment, safety and security requirements under the *Nuclear Safety and Control Act*. The *NFW Act* incorporates at the legislative level requirements which establish a process for due effort in addressing social impacts; these impacts are to be addressed on the same footing as technical matters throughout the development and implementation of a solution for the long-term management of nuclear fuel waste. The *NFW Act* provides for 1) the nuclear industry to set up a waste management organization to manage the long-term waste management activities related to nuclear fuel waste 2) the owners of the waste to establish trust funds to finance long-term waste management responsibilities and 3) the waste management organization to submit a report containing options for government decision long-term waste management, within three years of the coming into force of the *NFW Act*.

The *Nuclear Fuel Waste Act* list three options which must be included in the study, namely; the deep geological disposal concept, long-term storage at nuclear reactor sites, and centralized long-term storage, either above or below ground. The waste management organization may study and present additional options for consideration if it wishes to do so.

Canada has now adopted a legislative framework to move effectively towards the implementation of a solution for the long-term management of its nuclear fuel waste. By November 2005, the “Nuclear Waste Management Organization”, established by the nuclear industry on October 24, 2002, will submit its report on proposed options. This will be followed by a decision by the Government of Canada. The requirement and characteristics of storage of nuclear fuel waste over the long-term in Canada will then become more certain. Details of any future storage plan need to await the Government decision on the approach to the long-term management of nuclear fuel waste in Canada.

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**TECHNIQUE OF MONITORING CLADDING INTEGRITY OF RBMK-1000 SPENT FUEL ASSEMBLIES AFTER LONG STORAGE**

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At present time spent nuclear fuel (SNF) from RBMK-1000 reactors is stored in AR-cooling pools and wet AFR facilities at the NPP sites. A total amount of 9,000 tU in SNF has been stored at the storage facilities at NPPs for the operating period of RBMK-1000 power units. The capacities of operating storage facilities with provision for densification of stored SNF are close to exhaustion.

The period of fuel wet storage reaches 28 years, i.e. approximates the maximum allowable time, which in accordance with existing assessments is about 30 years. FAs are stored in one-seat cans filled with water, which is not purified or replaced. Constantly accumulating water impurities, salts and hydrogen peroxide accelerate corrosion processes of FAs and fuel rods structural materials and may cause loss of fuel rod cladding integrity. In this connection, as well as in compliance with Russian conventional concept of management of SNF from RBMK-1000 reactors it is proposed to shift SNF to dry storage, which as the international experience suggests, provides FA cladding integrity for a long period, especially when using inert gas, in which atmosphere structural material is practically free of corrosion.

When shifting SNF to dry storage it is suggested to cut FAs into two bundles. Such cutting will be performed in hot cells being an integral part of the department of preparation and storage of SNF in dual-purpose metal concrete casks (MCC). At the present time these departments are being constructed at the Leningrad and Kursk NPPs. At Leningrad NPP the construction is to be finished by the end of 2003. At the Smolensk NPP the preparation for construction is underway.

In 2002 dynamic tests were successfully performed to show compliance of MCC for storage and transportation of SNF – TUK-MBK-RBMK (TUK-104/109) with requirements of IAEA rules.

Only intact FAs will be stored in MCC. The technology of dry storage of leaking FAs is at the stage of development. It sets the problem of monitoring FA cladding integrity prior to cutting into bundles and loading into MCC. In accordance with the design technology of SNF wet storage the leaking FAs, discharged from reactor, are stored in sealed cans in AR-cooling pools. A part of the leaking FAs (predominantly not quite protected from gas penetration) was transferred to AFR storage facilities in TK-18 with the aim to empty AR-cooling pools.

The existing techniques of monitoring FAs cladding integrity, in particular for RBMK-100 FAs, are intended for usage at an operating reactor or during discharge. Techniques of monitoring FAs cladding integrity during or after a long wet storage are at the stage of development.

This report presents the results of developing a technique of monitoring FAs RBMK-1000 cladding integrity after a long wet storage and its general postulates. The main requirement when developing this technique was to monitor integrity prior to transfer of FA to hot cells for cutting into bundles and loading into MCC.

The technique of monitoring FAs cladding integrity is intended for reliable determination of a degree of fuel rods integrity after long storage in cans, which water environment is not purified.

## ON-SITE INTERMEDIATE STORAGE FACILITIES IN GERMANY

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In 2002 German utilities and the federal government agreed on the future of nuclear power in Germany. Part of this “consensus” are site specific phase out periods, termination of reprocessing and the erection of on-site storage facilities. Shipment of spent fuel to reprocessing plants is only allowed until 2005, shipments to the German joint storage facilities of Ahaus and Gorleben have to be minimized. Each nuclear power plant has to create its own spent fuel storage with appropriate capacity.

The necessary storage capacity is about 20 to 40 t/a and per plant, corresponding to 40 to 80 PWR- and 140 to 180 BWR-fuel bundles, depending on plant power and discharge burnup. Wet storage capacity inside the reactor building of most German NPPs was increased (compact storage) when shipments of spent fuel were ceased, following discussions about surface contaminations on casks. To avoid shut-down some plants even built short term storage facilities.

Whereas short-term storage is to guarantee undisturbed operation of plants on an intermediate time scale, long term storage – until final nuclear waste disposal after 2030 is available - is also driven by economical considerations.

Of the existing storage techniques wet and dry storage were analyzed with respect to costs and time for design, licensing and construction. Both techniques fulfill the safety requirements, but wet storage turns out to be more costly because of the necessary heat removing devices and operating expense.

Total costs for construction, operation and decommissioning of a wet storage facility – as it was built in Obrigheim in 1998 on a smaller scale - rank from 80 to 250 million Euro, 1,4 million € annually and 45 million € for decommissioning (price index of 2003), respectively. Compared to that, dry storage needs about 26 million € for construction, annually 0,3 million € plus 2 – 3 casks (e.g. of the CASTOR type) with around 1,2 million € per flask and about 12 million € for decommissioning.

Both scenarios avoid shipment costs to central storage facilities of about 0,25 million € per cask. The costs for purchasing the casks are evenly disturbed over the residual lifetime of the NPP, which is advantageous compared to the higher initial costs of wet storage. The timeframes for design, licensing and construction are 6,5 years for wet- and 4 years for dry storage.

Considering the risk associated with any public licensing procedure beside the above stated economical facts makes dry storage the favorable choice.

The first on-site long term storage facility at the NPP Emsland, Lingen, was consequently designed as a dry storage facility, applied for license in 1998 and started operation in 2002. The sound construction accounts for major accidents, such as earth quake, explosions and aircraft impact. The willful attack with a commercial airliner was also considered.

The same design will be realized at the northern German plant sites and comparable constructions are under way in the four NPPs in southern Germany. Another, more site specific tunnel-construction is currently being designed for Neckarwestheim.

The plants limited individual phase out periods require the possibility to operate the storages stand-alone. Currently NPP operators consider the auxiliary systems, as emergency power, waste management systems and maintenance facilities for casks, avoidable.

In total on-site dry storage facilities are the best solution considering economical, ecological and safety aspects, to answer politically motivated requests for minimizing shipments of spent nuclear fuel in Germany.

## CONCEPT OF BN-350 REACTOR SPENT FUEL HANDLING DURING ITS STORAGE AFTER SHUTDOWN

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According to the Kazakhstan Government Decree (#456; Apr 22, 1999), the fast BN-350 reactor has been shutdown in 1999th and the plan of its decommissioning has been started. By a decision of the Kazakhstan Government is defined that its spent fuel must be placed into 50 year's long-term safe storage with following dismantling and final disposal. This plan most important part is the concept of the spent fuel handling during its storage after shutdown.

Next main stages of this concept are discussed here: discharge of the fuel, spent fuel packaging into special canisters, temporary storage of these canisters in the reactor pool, to choose the site for long-term spent fuel storage and transport cask, making a choice of safety technique for canisters dry storage during 50 years.

At first stage the fuel has been removed from the core and placing into reactor pool. The second stage lasted practically about two years. During this period all spent fuel has been packaged into special sealed canisters filled with inert gas.

On next stage the site outside of the reactor one for long-term storage of these canisters and the way to ship them into the casks were chosen. The selected site is placed within the territory of former Semipalatinsk nuclear site. But the casks for spent fuel ought to be shipping as by rail and trailer vehicles too. Until its shipping will be started the canisters have been temporary stored under the water into BN-350 reactor pool.

The following step is to define the transport cask design and the way to store the canisters at the site during long-term period about 50 years. There are two projects, which have been under consideration.

The first is to use single-canister iron cask designed for shipping only. In this case the canisters with spent fuel ought to be transporting to the site into these casks and then transshipped into special "silo"-type storage. Each canister has to be placed into individual near surface silo. The goal of this way is to assure physical protection of nuclear materials and radiation safety but also to provide required technical conditions during the whole storage period. Nowadays the design and technological developments of this cask are implemented. Also the programme of it testing is developed too.

The second is to use seven canisters iron-concrete cask designed both for shipping and storing the canisters inside it during 50 years. In this case intermediate operation to transship canisters from cask into storage is missed. The main technical parameters of it are defined today. Just its design and engineering developments are carrying out now.

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**STORAGE OF SPENT FUEL FROM THE NUCLEAR POWER PLANTS  
GREIFSWALD AND RHEINSBERG IN THE INTERIM STORAGE NORTH**

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The nuclear power plants of the former GDR - NPP Greifswald and NPP Rheinsberg - were shut down in the year 1990 to decommission all reactors. After the shut down of the 5 units in Greifswald (Russian reactors type WWER 440) and 1 unit in Rheinsberg (Russian reactor type WWER 70) was the storage of the spent fuel the priority assignment.

The units are in operation since:

unit 1	1974
unit 2	1975
unit 3	1978
unit 4	1979
unit 5	1989 (putting into operation)

In 1990 in NPP Greifswald existed 5037 spent fuel elements. At that time 1011 elements contained in the reactors, 1628 elements in the cooling ponds of the units and 2398 elements were stored in a wet interim storage (WIS) at the Greifswald site. At the Rheinsberg site 220 spent fuel elements existed.

For starting the decommissioning the units should be free of nuclear fuel. Therefore the fuel elements from the reactors and the cooling ponds of the NPP Greifswald were transferred from 1994 to 1996 to the WIS. A part of the low burn – up fuel elements from the putting in operation of unit 5 had been sold to the Hungarian NPP Paks.

A final depository for spent nuclear fuel in Germany doesn't exist. For a long time the wet storage of nuclear fuel elements is disadvantageously. Therefore a new interim storage was built at the Greifswald site. In the Interim Storage North (ISN) the spent fuel elements are stored in CASTOR<sup>®</sup> 440/84 casks.

The ISN allows the conditioning and storage of the radioactive materials from the dismantling of the nuclear power plants Greifswald and Rheinsberg. The ISN consists of 7 halls for storage of radioactive residues and 1 hall for storage of spent fuel in CASTOR<sup>®</sup> 440/84 casks.

Currently the spent fuel elements from the WIS were transferred - via a special equipment built in unit 3 - into CASTOR<sup>®</sup> 440/84 casks and transported to the ISN for further storage. The spent fuel elements from the NPP Rheinsberg were loaded into 3 CASTOR<sup>®</sup> 440/84 casks and transported to the ISN in May 2001.

In hall 8 of the ISN in December 2002 are stored 26 CASTOR cask with spent fuel elements.



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## LICENSING EXPERIENCE AND TECHNICAL ISSUES RELATED TO WET STORAGE OF SPENT FUEL IN THE UNITED STATES

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Wet fuel storage facilities in the United States were originally designed for short-term storage of spent fuel. However, early technical challenges with fuel reprocessing and government policy changes resulted in wet storage becoming, by default, the principal method for long-term storage of spent fuel.

To accommodate the increasing inventory of spent fuel, the storage racks have been reconfigured to higher-density storage patterns one or more times during the lifetime of each plant. These high-density storage racks employ fixed neutron poisons to maintain a large margin to criticality, but some polymer-based neutron poison materials have shown evidence of degradation under certain wet storage environments. As an alternative to replacement neutron poison inserts, the NRC has found credit for soluble boron to be an acceptable means of mitigating accidents that increase reactivity. However, this credit is limited to applications where the nominal storage configuration remains subcritical without credit for soluble boron and undetected boron dilution is considered an incredible event.

In addition to changes in storage density, changes in United States reactor operating practices have posed additional challenges to spent fuel pool support systems. Efforts to manage risk in shutdown reactors have increased the frequency with which the full core has been transferred to the spent fuel pool during refueling. Power uprates and progress toward ever shorter refueling outages have further increased the challenge to maintain adequate decay heat removal. To maintain safe spent fuel pool temperatures during refueling periods, the NRC has been shifting licensing analyses from a fixed design scenario to administrative controls on minimum decay time that balance the decay heat load with the available heat removal capacity. A measure of defense-in-depth is maintained by ensuring availability of forced cooling after a single component failure and by verifying that makeup water capacity exceeds the maximum rate of loss due to evaporation.

Lastly, evaluations of beyond design-basis accidents in spent fuel pools have indicated potential off-site consequences comparable to those of a reactor accident, with the exception that immediate health effects are much lower. These accidents originate with very unlikely but energetic events, such as cask drops, that damage the pool structure and drain the cooling water. The estimated frequency of these random events is very low and the NRC has concluded that even with conservative estimates of potential consequences, the overall risk is acceptable. Nevertheless, the NRC has begun development of more realistic modeling of accident progression to better estimate consequences.

Although wet storage of spent fuel is a mature technology, the licensing activities have remained dynamic in the United States. However, further expansion of dry storage options may reduce or eliminate some of the licensing challenges described above.

## METHODS FOR WWER-1000 FUEL TESTING UNDER DRY STORAGE CONDITIONS

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Transient and emergency conditions of dry storage differ from the standard ones in the elevated temperature of the fuel rod claddings and are the major hazard from the viewpoint of fuel rod unsealing.

The thermal testing of re-fabricated and full-size rods was carried out at electrically heated test facilities and included several series with interim inspection of the fuel rods.

At FSUE SCC RF RIAR a complex of test facilities for thermal testing of the irradiated WWER-1000 fuel pins within the temperature range of 300-600<sup>0</sup>C in different gaseous atmospheres (inert gas, air) was created. Both shortened re-fabricated fuel rods, and full-size WWER-1000 fuel rods removed from fuel assemblies spent at NPP can be tested.

The procedure for fuel rod testing incorporates the following:

- verification of the fuel rods by non-destructive methods;
- thermal testing of fuel rids;
- interim non-destructive assay of the fuel rods between several successive tests (if performed);
- material testing of fuel rods on completion of the testing.
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The facility for testing of irradiated full-size WWER-1000 fuel rods consists of three stand-alone electrically heated modular units 5m high.

Capabilities of the facility and its main characteristics:

- remote reloading of fuel rods;
- testing temperature range of 300-600<sup>0</sup>C;
- simultaneous testing of 18 irradiated full-size WWER-100 fuel rods in three stand-alone modular units;
- simultaneous simulation of several gaseous and temperature conditions of SNF storage;
- regular gas sampling from any of the units;
- profiling of the temperature area throughout the fuel rod;

- thermocycling – simulation of daily or seasonal fluctuations in temperature of the environment;
- non-uniformity of the temperature axial distribution throughout the fuel rods does not exceed  $\pm 3\%$ ;
- non-uniformity of the temperature radial distribution in the facility does not exceed  $\pm 2\%$ .

The average burn-up of the fuel rods selected and verified for the testing fell within the range of 39-49MW\*day/kgU. The parameters of the fuel rods corresponded the typical condition of the WWER-1000 fuel rods after 3 or 4 fuel cycles of the standard operation in the WWER-1000 reactor and follow-on storage in the cooling pool for 3 years.

The fuel rods were tested in helium medium at the temperature of 400-550°C. Hoop stress in the fuel rod claddings were 36-110MPa, the period of testing of different fuel rods at different temperatures lasted 2.5-50 hours. All the fuel rods retained integrity during the thermal testing.

The dependence of hoop and axial strains of the fuel rod cladding on time, temperature, and stress was plotted.

The maximum hoop strain of the cladding accounted for 2.6% at the testing temperature of 550°C and hoop stress of ~110MPa.

Comparison of the testing results with the assessments of long strength of irradiated Zr+1%Nb claddings (Larson-Miller parametric dependence) revealed the strength margin of the claddings prior to their failure.

During the testing of the fuel rods gaseous fission products were detected to release under the cladding, the release depended on the fuel burn-up and the testing temperature.

The fuel rod cladding strain has a complicated nature and is determined not only by thermal creep, but also other processes that might be connected with thermo-mechanical fuel-cladding interaction at fuel rod heating during the testing.

At present the testing of fuel rods under emergency conditions in the air atmosphere are being accomplished, and long-term testing of the fuel rods under standard storage conditions have been started.

**THE IDAHO SPENT FUEL PROJECT**R. ROBERTS<sup>A</sup>, D. TULBERG<sup>A</sup>, C. CARTER<sup>B</sup>

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The Department of Energy awarded a privatized contract to TTFWI (Tetra Tech Foster Wheeler Inc.) in May 2000 for the design, licensing, construction and operation of a spent nuclear fuel repackaging and storage facility. The TTFWI Team consists TTFWI (the primary contractor), Alstec, RWE-Nukem, RIO Technical Services, Winston and Strawn, and Utility Engineering.

The Idaho Spent Fuel (ISF) facility is an integral part of the DOE-EM approach to accelerating SNF disposition at the Idaho National Engineering and Environmental Laboratory (INEEL). Construction of this facility is also important in helping DOE to meet the provisions of the Idaho Settlement Agreement. The ISF Facility is a substantial facility with heavy shielding walls in the repackaging and storage bays and state-of-the-art features required to meet the provisions of 10 CRF 72 requirements. The facility is designed for a 40-year life.

## SOME ASPECTS OF THE RUSSIAN NUCLEAR FUEL CYCLE DEVELOPMENT

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### 1. Management of Spent Fuel in Russia

The existing in the Russian Federation scheme for handling SNF from power reactors, research and transportation units allows safe storage and partial reprocessing of irradiated fuel. The present report illustrates the role of enterprises affiliated to Minatom's Department of Nuclear Fuel Cycle:

- The RT-1 complex of the "Mayak" Production Association carries out radiochemical reprocessing of SNF from WWER-440 reactors in Russia and Ukraine, BN-600 reactor, SNF from research reactors and nuclear power plants of sea vessels;
- The Mining and Chemical Complex carries out centralized intermediate storage of SNF from WWER-1000 reactors in Russia and Ukraine; some SNF from WWER-1000 and RBMK-1000 is currently stored in water medium in on-site pools.

Therefore we could state that the Russian Federation has actually implemented two nuclear fuel cycles for different types of reactors:

- Closed nuclear fuel cycle for SNF from WWER-440, BN-600, research reactors and ship's nuclear power plants – the cycle includes radiochemical reprocessing of fuel and partial use of the recovered products (uranium, plutonium and other nuclides) and different techniques for waste treatment and storage;
- Deferred nuclear fuel cycle for SNF from WWER-1000, RBMK-1000 and some other activities.

The annual volume of SNF unloaded from a single NPP unit and cumulative volume of SNF generated by Russian NPP's are shown in Table 1. It is clear that the total volume of SNF generated in the Russian Federation is relatively small (below 10% of the world SNF). The capacity of the existing repositories for SNF from WWER-1000 will be sufficient for 6 to 8 years, and the capacity of the repositories for SNF from RBMK will be exhausted in 3 to 5 years.

Some fuel from research reactors and criticality test beds is located on site of large Russian nuclear centers, e.g. Research Institute of Nuclear Reactors (NIIAR), Physics Energy Institute (FEI) and Kurchatov Institute.

TABLE 1. SNF GENERATED IN THE RUSSIAN FEDERATION

Reactor type	Units in operation	Total generation of SNF, HMt/year	Estimated SNF storage, HMt/year (early 2003)
BN-600	1	16	Under reprocessing
WWER-440	6	70	Under reprocessing
WWER-1000	8	170	~ 6 500
RBMK-1000	11	450	~ 10 000

It is worthy to note that Minatom has created a reliable industrial framework for cost-effective and on the whole environmentally safe implementation of the licensed activities related to SNF management. All Minatom's enterprises involved in back-end nuclear cycle are licensed by the State Committee on Surveillance of Nuclear Safety (Gosatomnadzor) – the main supervisory body in compliance with the existing legislation.

“The Strategy of Nuclear Power Development in Russia in the first half of the 21<sup>st</sup> century” approved by the RF Government in May 2001 and the concept for the nuclear sector advancement up to 2010 stipulates development of enterprises for SNF management on site of generation and fuel cycle plants, a gradual transition to completely closed cycle in terms of recycled uranium and power plutonium with actinide and long-lived fission products transmutation in fast breeder reactors.

Prospective close-up of the nuclear cycle will allow saving uranium as a result of using recycle fission materials and reducing toxicity of radioactive waste subject to disposal.

Minatom carries out preparatory work along various trends of promoting services and SNF infrastructure related to nuclear fuel cycle.

#### **Priority targets for improving the SNF management system at Minatom's enterprises**

Since its creation in the 1970—1980-ties the Russian industrial SNF infrastructure for SNF transportation, storage and reprocessing was oriented exclusively at management of SNF from reactors constructed in the USSR and COMECON member-states (WWER-440, WWER-1000 and research reactors).

Entering the world market of services related to storage and reprocessing of SNF from foreign NPP's will require additional expenditures for development and improvement of industrial SNF infrastructure of nuclear cycle back-end.

The Minatom's concept of management of SNF from Russian power and research reactors as well as ship's nuclear power plants in the period up to 2010 stipulates:

- Modernization of the existing RT-1 complex at the “Mayak” Production Association and creation of a state-of-art scheme for management of all types of radioactive waste;

- Increasing the capacity of the existing centralized SNF repository for SNF from WWER-1000 at the Mining and Chemical Complex from 6000 to 8400 HMt;
- Construction at the Mining and Chemical Complex of a vault-type dry long-term repository for SNF from WWER-1000 and RBMK-1000 reactors with the design capacity of 33 000 HMt and manufacture of the necessary means for SNF transportation.
- In the long-term perspective – construction of RT-2 complex at the Mining and Chemical Complex for reprocessing of SNF from WWER-1000 with a set of installations for conditioning and disposal of radioactive waste.

The first stage of implementing the concept of a closed fuel cycle (up to 2010) is related to modernization of the existing RT-1 complex, expanding the capacity of “wet” repository of SNF at the RT-2 and dry storage of SNF from RBMK-1000 reactors. It is considered reasonable to postpone construction of the RT-2 complex till commissioning of the new generation fast breeders (estimated up to 2020). These trends for improving nuclear fuel cycle are described in the “Sectorial concept for SNF management” approved by the Minatom’s Board in January 2001.

### **Modernization of the SNF reprocessing plant at the “Mayak” Production Association**

The first Russian plant on radiochemical reprocessing of SNF from NPP’s – the RT-1 complex at the “Mayak” Production Association - was based on the military facility with inadequate scheme of waste management. Today Minatom takes extensive efforts to rehabilitate water bodies in which large volumes of high active waste were discharged (Karachai, Techa river cascade). Similar to other radiochemical plants the RT-1 complex uses a variety of Purex process that inevitably results in formation of large volumes of solid and liquid waste. Comparison with foreign plants shows that the RT-1 complex is unique in terms of reprocessing various SNF as well as potential for industrial extraction and fractioning of nuclear fuel components for further use. The adjacent radioisotope plant is capable of reprocessing the extracted radionuclides for manufacturing different sources of ionizing radiation.

A facility for vitrification of liquid radioactive waste (EP-500/3 ceramic melter) allows solidification of 300 to 310 liters of liquid radioactive waste per hour with vitrified waste stored in a vault-type dry repository.

It should be noted that throughout the operation of the RT-1 complex (over 25 years) not a single serious accident was reported at the “Mayak” Production Association that was classified above one unit on the International Nuclear Emergency Scale (INES).

Today the amount of fuel sent for reprocessing at the RT-1 complex of the “Mayak” Production Association is considerably below the design output.

**Design output of the RT-I plant: 400 t/year**

**Limitation of SNF reprocessing volume by the local authorities: 250 t/year**

**Actual reprocessing volume: 120-150 t/year.**

In order to improve technical and economic performance of SNF regeneration at the RT-1 complex an integrated investment project has been prepared with the aim of partial modernization of the production, particularly:

- **To create capabilities for reprocessing of SNF from WWER-1000 (and PWR) aimed at increasing the load of the reprocessing complex.**
- **To improve technological flowchart for reducing specific volume of the liquid radioactive waste formation.**
- **To construct installations for waste reprocessing and conditioning.**

The current technological capabilities of the RT-1 complex are shown in Table 1. Following modernization (by 2008) the plant will reprocess up to 300t/year of SNF with environmentally acceptable parameters of emissions and discharges.

#### **Enrichment of recovered uranium**

The “Mayak” Production Association carries out reenrichment of recovered uranium up to 2.6% content of U-235 for fabrication of fuel for RBMK-1000 through mixing of solutions of uranyl nitrate obtained through reprocessing of different types of irradiated fuel assemblies including those filled with medium enriched uranium (~20%). No concentration of even uranium isotopes is performed.

As is known, a higher burnup of SNF in the reactor lowers the quality of unburnt uranium due to an increased content of U-232 and U-236 isotopes.

Given the planned increase of reprocessed SNF volume at the RT-1 complex and future incorporation of recycled uranium in the WWER fuel, the Minatom enterprises are faced with the target of elaborating a cost-effective technology for direct enrichment of recovered uranium. The most suitable for this purpose is the Siberian Chemical Plant (Seversk, Tomsk region) having the industrial potential for radiological purification, conversion of uranyl nitrate into hexafluoride, state-of-art installations for isotope separation, and also a unique geological repository for liquid waste disposal for hundreds of years.

On request by Cogema in the 1990-ties the Siberian Chemical Plant successfully implemented a classical scheme for direct enrichment of recycled uranium obtaining enriched UF<sub>6</sub> of the following composition: U-235 up to 4.95%; U-232 up to 10-6% and U-236 up to 1.4%. The plant has technological schemes that allow a substantial reduction of U-232 and U-236 isotopes in the low-enriched marketable product. In our opinion these options are of interest to foreign users having inventories of reprocessed uranium for use in nuclear power.



### **Creation of a dry repository at the Mining and Chemical Complex**

Russian planning and design organizations prepared a design of SNF repository in 2002. Today the project documentation is at the stage of approval and expertise.

Construction of Stage 1 will start when the supervisory bodies finish the project expertise and issue a license for construction. Commissioning of Stage 1 is expected in 2005-2006 together with auxiliary structures and buildings, heat supply network and railroads.

During construction of Stage 1 it is planned to strengthen physical protection of the repository, modernize the SNF infrastructure (railroads and motorways, power supply, etc.) and launch production of transportation means for delivery of SNF from RBMK-1000 at the NPP.

The investor in the repository construction is “Rosenergoatom” Concern responsible for operation of all NPP’s in the Russian Federation. With the total cost of the repository of ~12.5 billion rubles, the Concern’s investment in construction will amount to about 950 million rubles (over US\$ 30 million). The design capacity of the repository is slightly above the requirements of the Russian NPP’s (up to 38 000 tons of SNF) that would allow in perspective to offer the interested foreign partners services related to SNF storage at the Mining and Chemical Complex.

The above-described activities are supported by the Government of the Russian Federation and included in the Federal Target Program “Energy Effective Economy” that envisages a framework for the Government support of investment projects, particularly R&D. As a rule, construction of new facilities is financed by profits of enterprises or target assignments incorporated in product price.

