WM2008 Conference, February 24 - 28, 2008, Phoenix, AZ
Abstract #8243

Spent Fuel Storage Operational Experience With Increased Crud Activities

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ABSTRACT

A significant part of the electricity production in Hungary is provided by 4 units of VVER 440 nuclear reactors at the Paks Nuclear Power Plant. Interim dry storage of the spent fuel assemblies that are generated during the operation of the reactors is provided in a Modular Vault Dry Storage (MVDS) facility that is located in the immediate vicinity of the Paks Nuclear Power Plant. The storage capacity of the MVDS is being continuously extended in accordance with spent the fuel production rate from the four reactors.

An accident occurred at unit 2 of the Paks Nuclear Power Plant in 2003, when thirty irradiated fuel assemblies were damaged during a cleaning process. The fuel assemblies were not inside the reactor at the time of the accident, but in a separate tank within the adjacent fuel decay pool. As a result of this accident, contamination from the badly damaged fuel assemblies spread to the decay pool water and also became deposited onto the surface of (hermetic) spent fuel assemblies within the decay pool. Therefore, it was necessary to review the design basis of the MVDS and assess the effects of taking the surface contaminated spent fuel assemblies into dry storage.

The contaminated hermetic assemblies were transferred from the unit 2 pool to the interim storage facility in the period between 2005 and 2007. Continuous inspection and measurement was carried out during the transfer of these fuel assemblies. On the basis of the design assessments and measurement of the results during the fuel transfer, it was shown that radiological activity values increased due to the consequences of the accident but that these levels did not compromise the release and radiation dose limits for the storage facility.

The aim of this paper is to show the effect on the operation of the MVDS interim storage facility as a result of the increased activity values due to the accident that occurred in 2003, as well as to describe the measurements that were taken, and their results and experience gained.

INTRODUCTION

Hungary is situated in Central Europe and is a member of the European Union since 2004. Having limited natural resources, the nuclear industry plays an important role in the nation's electricity supply. Currently, there is one commercial Nuclear Power Plant at Paks with four units and an Interim Spent Fuel Storage Installation (ISFSI) at the same site adjacent to the NPP.

Paks Nuclear Power Plant with its four VVER-440 type Pressurized Water Reactor provides around 40 % of the total electricity generated in Hungary. The new fuel has been imported from Russia, and spent fuel was shipped back to the country of origin until the beginning of the 1990s. Since then, some of the spent fuel shipments were delayed, and some of them were completely cancelled thus creating a backlog of spent fuel filling all storage pool positions at the plant. In order to assure the continuous and reliable operation of the nuclear reactors, Paks NPP's management decided to implement an independent spent fuel storage facility and chose GEC-Alsthom's Modular Vault Dry Storage design in 1992. Following an extensive licensing procedure the ISFSI was commissioned in 1997.

INTERIM SPENT FUEL STORAGE AT PAKS SITE

The MVDS is illustrated on Fig.1 and provides for at least 50 years of interim storage for VVER-440 fuel assemblies in a contained and shielded arrangement. The fuel assemblies are stored vertically in individual fuel storage tubes, the matrix of storage tubes being housed within a concrete vault module that provides shielding. To prevent the development of eventual corrosion processes, the fuel assemblies are in an inert nitrogen environment inside the storage tubes.

Decay heat is removed by a once-through buoyancy driven ambient air flow across the exterior of the fuel storage tubes, through the vault and the outlet stack. There is no direct contact between the fuel assemblies and the cooling airflow.

A similar type facility has been in operation at Fort St. Vrain in Colorado, USA since 1991. Another storage facility where the design of the Canister Handling Machine was based on the Paks MVDS Fuel Handling Machine is in operation at the Hanford Site in Washington, USA since 1999.

The Paks storage facility can be divided functionally into three major structural units. The first one is the vault module where the spent fuel assemblies are stored in the vertical tubes. These vault modules include a minimum of three or maximum five vaults depending on the geometrical arrangement. The vaults are capable of accommodating 450 spent fuel assemblies at the existing vaults and 527 spent fuel assemblies in the vaults to be constructed in future.

The second major structural unit is known as the charge hall where the fuel handling machine travels during the fuel handling operations. The charge hall is bordered by the reinforced concrete wall of the ventilation stack on the one side and by a steel structure with steel sheeting on the other side.

The third major unit is the Transfer Cask Reception Building (TCRB) in which the reception, preparation, unloading and loading of the transfer cask takes place. The fuel handling system and other auxiliary systems are installed in the TCRB.

The fuel assemblies are transported to the MVDS from the at-reactor pool using the C-30 transfer cask and its railway wagon. The transfer cask is received in the TCRB where it is removed from the railway wagon and prepared for fuel assembly unloading. The fuel is raised into a drying tube directly above the cask where the fuel assembly is dried prior to being lifted into the fuel handling machine. The fuel assemblies are transferred, within the fuel handling machine, to the vertical fuel storage tubes in the vaults.

Once the fuel handling machine has moved away from the storage tube the air is evacuated from the tube and replaced with nitrogen gas. Then the tube is connected to the built-in nitrogen system that monitors the storage environment of the spent fuel assemblies.

Construction and extensions

Due to its modular nature the MVDS facility has been constructed according to the operational needs of the Paks NPP. The TCRB and the first three storage vaults were constructed in 1996. Commissioning of the facility took place in 1997 by filling the first storage vault with 450 spent fuel assemblies.

The total capacity of fully extended ISFSI is planned to be 33 vaults. The construction of the storage vaults are divided into three different stages. The first stage covered the first three vault modules accommodating altogether 4950 storage positions in 11 vaults. This capacity was based upon the amount of the spent fuel arising from the Paks reactors over an operating period of 10 years. Construction of the second and third vault modules, each with four storage vaults was finished in 1999 and 2002 respectively.

Having constructed the TCRB and the first 11 vault a new licensing process was necessary for the further enlargement in 2004. Based on the existing operational experience and cost reduction, some technical modifications were proposed although the basis of the storage technology has not changed (MVDS). The elastomer “O” ring seals on the storage tubes and the connecting monitoring system were replaced by metal sealing and a simplified leakage monitoring system. As part of the storage capacity enlargement the number of the storage positions in the vaults starting from the vault 17 are to be increased. Instead of the triangular pitch pattern the tubes will be arranged in a square pitch. This will enable each vault to contain 527 storage tubes instead of 450 within the same plan area. With this modification which is in accordance with the ongoing NPP life extension the total storage capacity of the 33 vault ISFSI will increase from the original 14 850 to 16 159 fuel assembly.

Based on the implementation license that was issued by the Hungarian Atomic Energy Authority construction of a new vault module (vaults 12-16) was finished by 2007 as stage II. Further enlargement as Stage III is going to be continued on the other side of the reception building.

Operation of the Paks ISFSI

The first license to operate the Paks MVDS was issued to the operator of the Paks NPP as the owner and licensee of the new IFSFI.

According to the modified Hungarian Atomic Energy Act that became effective parallel with the original commissioning of the ISFSI, a new agency called Public Agency for Radioactive Waste

Management (PURAM) was established in 1998. PURAM was designated to carry out the multilevel tasks in the field of the radioactive waste management including the interim storage of the spent fuel. A multi-step take-over process was started following the establishment of PURAM. In spite of this takeover, the NPP still has tasks regarding the facility. The NPP staff operate the ISFSI in a contractual arrangement with PURAM. By end of 2007 altogether 5112 spent fuel assemblies are stored in the Paks ISFSI.

EXPERIENCE OF STORAGE OF CONTAMINATED SPENT FUEL

Spent fuel assemblies with increased surface contamination

An accident occurred at unit 2 of the Paks Nuclear Power Plant in 2003, when thirty irradiated fuel assemblies were damaged during a cleaning process. The fuel assemblies were not inside the reactor at the time of the accident, but in a separate tank within the adjacent fuel decay pool. As a result of this accident, contamination from the badly damaged fuel assemblies spread to the decay pool water and also became deposited onto the surface of (hermetic) spent fuel assemblies within the decay pool.

Therefore, it was necessary to review the design basis of the MVDS and assess the effects of taking the surface contaminated spent fuel assemblies into dry storage.

Review of the MVDS design bases

Radiation source terms of the Paks MVDS can be divided into following groups:

- Direct and scattered radiation from spent fuel assemblies
- Release of the crud
- Release of cask water

The radiation of the spent fuel assemblies are minimized by bulk and labyrinth shielding. The spent fuel assemblies are stored in hermetically sealed tubes to prevent the spread of contamination. During the spent fuel handling the spread of contamination is prevented by ventilation system including HEPA filters.

Due to the bulk concrete shielding the dose rates around the facility are very low. Being a dry storage facility, the liquid effluents are not very relevant. The airborne releases are dominated by the crud and end evaporated cask water during fuel handling and drying.

Cask water design values

Cask water activity values for the health physics calculations were derived from the decay pool isotope specific data as the cask is loaded with spent fuel assemblies in a common water volume with the decay pool. Before the loaded cask is released from the NPP the cask water content is replaced. Based on this procedure a dilution factor gained from measurements of the boron concentration dilution in the cask water was used to define activity data for H3, C14, Cr51, Mn54, Co58, Co60, Fe59, Sr89, Sr90, Ag110m, Cs134, Cs137, Pu238, Pu239, Pu240, Am241, Cm242, Cm244.

Crud design values

During reactor operation, products of erosion and corrosion within the reactor primary circuit become mobile and are activated by neutron bombardment within the core. The attachment of this "crud" to the external surfaces of the fuel assemblies forms an additional source of radiation to be considered in the MVDS radiation protection design.

The physical form and activity of fuel crud that is specific to Paks VVER-440 fuel assemblies, in the absence of measured data, was difficult to quantify. Paks NPP early experience of this activity source had established that the crud is well bound to the surface of the fuel assembly.

For the purpose of safety assessment, a judgement of the possible fuel crud inventory per fuel assemblies had been made. The activity values have been taken from the typical fuel crud activities described in USA report (SAND-90-0365, May 1990. "Literature Review of Crud Spallation Source with Application to Nuclear Waste Repository". K G Adams.), which details a literature review of nearly 9000 spent fuel rod crud experience of US PWR NPP.

The initial impression provided by report was that the presence and properties of PWR fuel crud vary considerably from reactor to reactor, with the largest influence on crud properties being the chemical composition of the water within the primary reactor circuit. It was also concluded that basing the MVDS design on the maximum crud activity concentrations would be grossly pessimistic, as only a small proportion (< 2%) of US PWR fuel rods have fuel crud deposits in excess of peak values at discharge. The assessment of the radiological consequences in normal operations of Paks NPP fuel crud has therefore been based on the average activity concentrations measured from the US PWR experience.

Modifications to USA spent fuel crud activity data were made to account for the relative surface area of the VVER-440 fuel assembly. All the measured parameters were normalized to an activity level of fuel crud at the time of fuel discharge.

The total fuel crud inventory that was used for the radiological consequence analysis in normal operations was therefore based on Co60, Mn54, and Ag110m at 3 years decay. For the purpose of accident analyses, a conservative upper bound activity for fuel crud was used, derived from the maximum values of the activities. The activity values for accident cases are around two order of magnitude higher than the values corresponding to the normal operations.

While it remains fixed to the fuel assembly, the fuel crud does not concern the radiation protection design of the MVDS, because the fuel assembly radiation source levels dominate. However, possible detachment of the fuel crud and its subsequent separation from the fuel assembly could result in fuel crud particulate being the predominant radiation hazard in some plant areas.

Release limits

According to the isotope specific activity data explicit annual release limits were defined for the Paks MVDS. They were derived from the 10 microSv/year dose restriction for the off-site doses, both for the airborne releases and for the liquid effluents. Operational experience showed that

the real release values were always well below the derived release limits. The radioactive releases during the first six years of operation and the corresponding limits for the selected isotopes (Co-60, Mn-54, Ag110m, H3) are summarized in Tables I and II.

Table I Airborne release in Bq from Paks MVDS

Year	1997	1998	1999	2000	2001	2002	Limit
Number of loaded fuel assemblies	450	497	400	500	750	420	
Co60	1.6E4	2.3E4	1.4E4	2.0E4	2.3E4	1.8E4	7.8E09
Mn54	1.4E4	2.2E4	1.2E4	1.8E4	2.1E4	1.8E4	1.1E11
Ag110m	1.4E4	4.1E4	1.3E4	1.9E4	2.1E4	1.6E4	7.1E09
H3	1.7E8	2.3E7	2.8E7	3.1E8	1.6E8	1.6E8	2.1E14

Table II Liquid effluents in Bq from Paks MVDS

Year	1997	1998	1999	2000	2001	2002	Limit
Number of loaded fuel assemblies	450	497	400	500	750	420	
Co60	3.1E5	3.1E5	9.9E4	4.9E+05	3.0E5	4.1E5	3.1E10
Mn54	9.3E4	1.2E5	3.3E4	1.2E+05	5.7E4	9.2E4	1.8E12
Ag110m	8.8E5	2.4E5	5.1E4	4.2E+05	6.5E4	3.4E4	1.3E12
H3	6.3E6	2.3E6	2.3E6	8.4E+06	3.1E6	4.8E6	1.6E15

Crud and cask water activity extension

To assess the effects of taking the surface contaminated spent fuel assemblies into dry storage the above described cask water and crud activity data were reviewed and extended. It has covered the following results:

- measurements from the successfully cleaned fuel assemblies,
- decay pool water activity values affected by the accident,
- surface samples taken from the fresh fuel assemblies being prepared in the pond before the accident to load into the reactor as part of the refuelling
- measurements on the samples merged into decay pool water samples to evaluate the adsorption, desorption effects.

Based on these result the crud input data for the spent fuel assemblies were extended with the activity values for Ce141, Ce144, Pu238, Pu239, Pu240, Cm242, Cm244 and the cask water activity values with U238, U234, Cs136, Ce141, Ce144, Zr95, Nb95. Release limits were also defined where it was necessary.

All the affected safety assessment were reviewed and extended as part of the FSAR and submitted to the authority for approval. The result of the calculation showed that the dose restriction will not be compromised.

Modification of the design bases on the experience gained from the measurements

The contaminated hermetic assemblies were transferred from the unit 2 pool to the Paks MVDS in the period between 2005 and 2007. Continuous inspection and measurement was carried out during the transfer of these fuel assemblies.

Based on these measurement results the design base approach for the health physics calculations was modified for the crud related data.

Taking into consideration of the adsorption, desorption effects, all isotopes that were measurable in the decay pond water were considered as contamination sources on the surface of the fuel assemblies. This meant that the list of isotopes was identical for the cask water and for the fuel assembly surface contamination.

Measurement results also showed that the specific activity values for the original crud isotopes (Co60, Mn54, Ag110m) were conservative enough, so there were no need to modify those.

There were only two new isotopes (Sb125 and Ru106) that were not in the isotope specific list earlier. They were added both to the fuel assembly surface contamination and to the cask water inventory.

Separate activity values were defined for the fuel assemblies coming from the Unit 2 and for the other three units to make difference according to the pond water and fuel assembly surface contamination activities.

The modified spent fuel surface contamination values has been derived from the sum of the adsorption and original crud activity data, while the cask water activities are the sum of the dilution of the pond water, the desorption and 0,1% dilution of the crud.

Based on these modified design values the dose rates arising from the airborne releases and effluent were recalculated. The results have showed the summarized annual values are still well below the dose restriction. Assuming 500 fuel assemblies are transported annually from the NPP to the ISFSI the used portion of the dose restriction for the airborne releases will be 4,54E-3, while the same value for the effluents 1,13E-2.

Review of the release and exposure data

The increased level of the cask water activity and surface contamination of the fuel assemblies transferred from the NPP has resulted in more frequent HEPA filter changes in the ventilation system of the fuel preparation room in the Transfer Cask Reception Building. In spite of this, the evaluations of the radiological protection plan show good results. Both the radiation releases and the personal exposures are low compared to the radiation protection limits.

The radioactive releases between 2003-2006 for the same isotopes shown in Table I and II (Co-60, Mn-54, Ag110m, H3) are summarized in Tables III and IV.

Table III Airborne release in Bq from Paks MVDS between 2003-2006

Year	2003	2004	2005	2006	Limit
Number of loaded fuel assembly	480	270	500	480	
Co60	2.4E+04	1.4 E+04	2.3E+04	5.3E+04	7.8E09
Mn54	2.0E+04	1.3 E+04	2.0E+04	4.9E+04	1.1E11
Ag110m	1.8E+04	1.2 E+04	1.9E+04	4.8E+04	7.1E09
H3	1.9E+08	1.0 E+08	7.6E+07	2.8E+08	2.1E14

Table IV Liquid effluents in Bq from Paks MVDS between 2003-2006

Year	2003	2004	2005	2006	Limit
Number of loaded fuel assemblies	480	270	500	480	
Co60	2.1E+05	1.2E+05	2.9E+05	1.5E+06	3.1E10
Mn54	6.9E+04	6.4E+04	6.5E+04	2.1E+05	1.8E12
Ag110m	3.0E+04	3.4E+04	2.5E+04	3.1E+04	1.3E12
H3	2.2E+06	7.1E+05	8.8E+05	3.3E+06	1.6E15

Comparing the isotope specific limits and the measured values it can be seen that the measured data are still far below the release limits. The same conclusion can be drawn for all the isotopes determined in the radiation source terms.

The doses arising from the airborne releases and effluent are summarized in the Table V.

Table V Used percentage of the dose restriction

Year	Effluents		Airborne releases		Summary	
	nSv	%	nSv	%	nSv	%
2000	0.17	0.0017	0.10	0.0010	0.27	0.0027
2001	0.10	0.0010	0.11	0.0011	0.21	0.0021
2002	0.14	0.0014	0.08	0.0008	0.22	0.0022
2003	0.08	0.0008	0.09	0.0009	0.17	0.0017
2004	0.05	0.0005	0.07	0.0007	0.12	0.0012
2005	0.13	0.0013	0.20	0.0020	0.33	0.0033
2006	0.54	0.0054	0.44	0.0044	0.98	0.0098

Comparing the above described values with the use of the dose restriction derived from the modified design values a conclusion can be made that the design values used for the release calculation are conservative as they predict higher values than the measured ones.

The individual and collective doses of workers are shown in Table VI.

Table VI Radiation exposure of workers

Year	2000	2001	2002	2003	2004	2005	2006
Number of loaded fuel assy.	500	750	420	480	270	500	480
Collective dose (man* mSv)	2.1	2.1	8.6	6.4	3.7	6.3	7.6
Average individual dose (mSv)	0.002	0.002	0.007	0.006	0.003	0.004	0.005
Highest individual dose (mSv)	0.071	0.035	0.582	0.849	0.168	0.298	0.198

The higher value in 2002 was partly due to the increased crud level on the fuel handling machine primary filter. A continuous activity monitor was installed on the filter to reduce the high dose rate from the filter exchange activity.

The collective dose from 2003 shows a significant increase compared to the values corresponding to the first operational years of the Paks ISFSI. This increase is due to the increased crud amount on the fuel assemblies and the accident that happened in 2003. However, the individual dose values are well below the dose limit for the operators (20 mSv/year).

SUMMARY

On the basis of the design assessments and measurement of the results during the fuel transfer operations, it was shown that radiological activity values increased due to the consequences of the 2003 accident but that these levels did not compromise the release and dose limits for the fuel storage facility.

In the environment there was no measurable radioactivity as a result of the operation of the Paks ISFSI. The exposure of the surrounding population was calculated on measured releases and meteorological data. The calculations show negligible doses until 2004.

Due to the increased surface contamination on the spent fuel assemblies the dose rate increased almost 5 times compared to the least annual value, but still less then 0.01 percent of the allowed dose restriction.

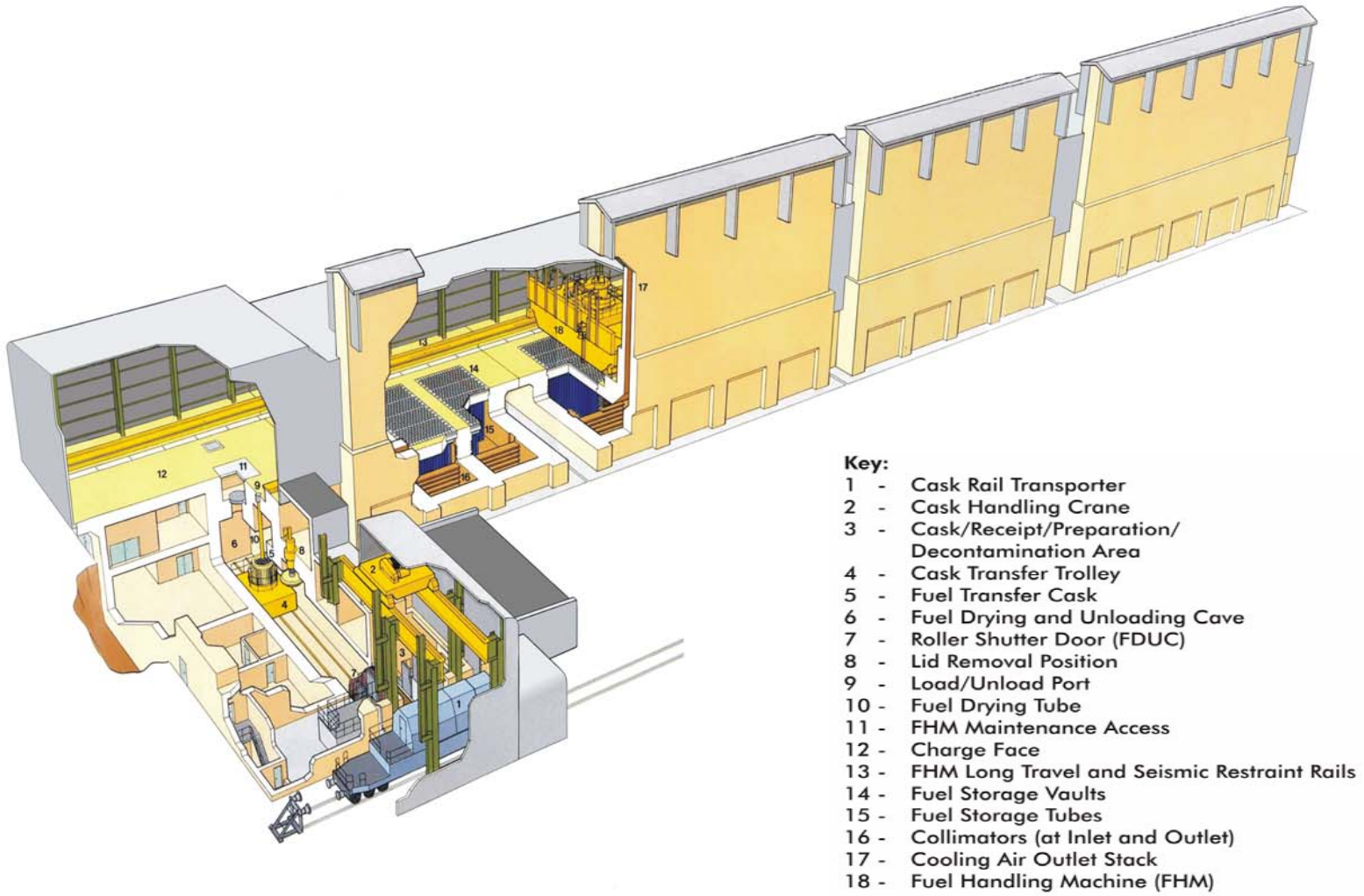

Fig 1. Paks ISFSI