CH0300040

MANAGEMENT AND INSPECTION OF INTEGRITY OF SPENT FUEL FROM IRT MEPHI RESEARCH REACTOR

V.G.ADEN, S.Yu.BULKIN, A.V.SOKOLOV
Research and Development Institute of Power Engineering,
P.O.Box 788, 101000 Moscow, Russia

and

A.V.BUSHUEV, A.F.REDKIN, A.A.PORTNOV
Moscow State Engineering Physics Institute - Technical University (MEPhI),
31, av. Kashirskoe P.O. 115409 Moscow, Russia

ABSTRACT

The information on wet storage and dry storage of the spent nuclear fuel (SNF) of the IRT MEPhI reactor and experience from SNF shipment for reprocessing are presented. The procedure and a facility for nondestructive inspection of local power density fields and the burnup of fuel assemblies based on studying the ϒ-activity of some fission products generated in U^{235} and procedure for inspection of the fuel element cladding leaktightness are described.

1.Introduction

The IRT MEPhI research reactor is a water-cooled water-moderated pool reactor with the power of 2.5 MW. It has been successfully operated at Moscow State Engineering Physics Institute since 1967. The first criticality of the reactor was attained on May 26, 1967 and its power was 500 kW in October 1967, 1 MW in 1970, 1.7 MW in 1971 and 2.5 MW in 1972. A facility for inspection of the fuel cladding leaktightness was installed in the interim SNF storage in 1975.
In the course of reactor operation 77 fuel assemblies (FAs) with EK-10 fuel elements and 102 IRT-2M, -3M FAs were used: until 1975, FAs with EK-10 fuel rods with meat of UO_2 in a magnesium matrix and with an aluminium alloy cladding, and afterwards IRT-2M and IRT-3M FAs with annular fuel elements having a square cross-section and containing the meat of U-Al (or UO_2) in an aluminium matrix and the aluminium alloy (SAV-1) cladding (the FAs contain three, four or eight fuel elements of such type).
In 1989, 48 IRT-2M spent fuel assemblies (SFAs) were transported to reprocessing plant RT-1.
The transportation for reprocessing of IRT-2M SFAs rather than older EK-10 SFAs has been caused by the fact that RT-1 does not reprocess EK-10 fuel elements because of their small amount, low enrichment and the necessity of using a process somewhat different from the regular process.
The SNF transportation was later stopped because of economic difficulties.
Currently the SNF of the IRT MEPhI reactor is stored in two different ways:

- wet storage (in a storage pool filled with chemically desalinated water (CDW) – 54 SFAs with annular-type fuel elements;
- dry storage (in leaktight containers with an air environment) – 77 rod-type SFAs.

The Russian spent nuclear fuel handling concept provides that the strategic direction of nuclear power development is the nuclear fuel cycle closing that is to ensure a more complete use of natural nuclear fuel and artificial fissile materials produced by nuclear reactors (plutonium, etc.), minimization of the radioactive waste amounts from SNF reprocessing and approximation to the radiation migration equivalence of the disposed waste and withdrawn natural fuel.
In this connection, the SNF of the IRT MEPhI reactor as well as the fuel of other Russian research reactors should be reprocessed based on the extraction technology used by RT-1 (PUREX-process).

2.Experimental and calculational research of fuel burnup

Short-term and long-term loads experienced by FAs in the core are monitored for efficient and safe operation of the IRT MEPhI reactor. The reactor power is monitored based on measuring the heatup of the coolant flowing through the core. The contributions of different FAs to the overall power are determined by calculations. The fuel burnup in each FA is determined based on results of these calculations proceeding from their total power generation. A neutronic code TIGR [1] is currently used to calculate neutron fields, power density and fuel burnup values. The TIGR code was verified by comparing calculation results with experimental operation data. Integral characteristics of the reactor (reactivity, critical position of the control rods) were compared for different core loads. It is known that such measurements characterise the reactor as a whole. So quite an accurate and detailed experimental information is required for a more detailed verification of the calculations.

As direct measurements of the U^{235} content in the irradiated FAs using U^{235} own gamma radiation are not feasible, then the measurement of the burnt-up U^{235} in the irradiated FA was based on measuring the gamma activity of a particular fission product formed as the result of the U^{235} burnup.

The characteristics of the widely used fission products for analysing the irradiated nuclear fuel burnup are presented in Table 1 [2, 3].

Table 1

Fission product	$T_{1/2}$	Yield at U^{235} fission (%)	Gamma radiation energy (keV)	Quanta yield (%)
Zr^{95}	64.0 days	6.50	724.2 756.7	43.1 54.6
La^{140}	1.68 days	6.27	1596.17	0.844
Nb^{95}	35.0 days	6.50	765.8	99.8
Cs^{137}	30.17 years	6.22	661.6	85.1

A measuring scanning type facility consisting of a regular transfer container, a scanner, a collimator system and a spectrometric system was constructed in the reactor hall for studying irradiated FAs. Spectrometric systems with germanium detectors of different types and designs were used for measurements.

The reactor was shut down, all FAs were withdrawn from the core and put into the interim storage by the start of the experiment. For carrying out measurements, the FAs were alternately withdrawn from the interim storage, put into the transfer vessel and moved over a vertical channel to a lead thick-wall transfer container.

The experimental values were compared with the respective calculated data. And the calculated value of the relative burnt-up U^{235} amount was compared with the experimental value of the relative amount of the accumulated fission product. Experimental and calculated values of the power density distribution over the FA height were mostly close and major differences are observed only for FAs with control rods. The calculated fuel burnup distribution over the height of FAs mostly correlate with the ^{137}Cs measurement results. Differences beyond the experimental error boundaties are observed for some of the six-tube low-burnup FAs containing control rods. The experimental and calculated results correlate better for eight-tube high-burnup FAs.

The produced spectra were processed using modern Russian and foreign codes. The obtained data on the initial fuel distribution in FAs, neutron field in the reactor core, fuel power generation and burnup in FAs were compared with the FA certificate data and calculation results produced using TIGR and GETERA codes being regular IRT MEPhI reactor neutronic calculation codes. Differences between the experiments and calculations have been found out and analyzed. General conclusions have been made with respect to the correctness of the calculation results and judgements were made regarding the errors of the calculated values.

Some new procedural approaches that have been developed during the work (loading indicators into the operating core FAs and use of a special irradiation time mode for getting information on processes in the fuel during different periods) expand the capabilities of in-pile experiments and may be used in other reactors.

3.Spent fuel inventory and ways of SFAs storage

Some of the IRT reactor SFAs characteristics are presented in Table 2.

Table 2

No.	Description of characteristic	SFA type			
		IRT-2M (90%)	IRT-2M (36%)	IRT-2M KS*) (36%)	IRT-3M (90%)
1	Residual enrichment in terms of U^{235}, %	40 – 50	20	20 – 22	40 – 50
2	Estimated residual U^{235} content, g	90	130	230	150

*) – IRT-2M-KS FAs are experimental FAs developed at the Kurchatov Atomic Energy Institute and they differ from the regular IRT-2M FAs is a higher U^{235} load (from 230 g to 390 g).

15 IRT-2M KS FAs were tested in IRT MEPhI reactor. The tests have shown that such FAs are not reliable enough – according to the results of fuel cladding integrity detection in 1984-1986, 4 FAs with the fuel burnup of 10-15% were withdrawn from the core with the initial gas non-leaktightness and removed to the storage pool for regular storaging and there was no degradation of the storage pool water radiation characteristics.

Spent fuel inventory is summarised in the Table 3.

Table 3

Location	SFA type	SFA quantity
Dry storage facility	SFAs with EK-10 fuel elements	77
Wet storage facility	IRT-2M (90%)	22
	IRT-2M (36%)	3
	IRT-2M KS (36%)	15 (4 of them are leaky)
	IRT-3M (90%) – 8 tube ones	14

Configuration of dry storage and wet storage is presented in Fig.1. The dry storage facility represents dry channels with the diameter of 180 mm made in a concrete body for storage of irradiated experimental devices. SFAs are located in leaktight canisters in the dry channels, 3 SFAs in each canister. The channels are closed with a shielding plug made of heavy concrete in a steel casing.

SFAs were transferred to the dry storage facility after a 5 year cooling period in the storage pool. SFAs were put into the canisters and the canisters were sealed in a hot cell. The environment inside the canisters is air. The SFAs were preliminarily dried with hot air blowout. From 5 to 6 canisters are installed in the channel. Dry channels with the SFAs are ventilated forcedly during the reactor operation. The canisters are cooled by natural convection during long-term reactor shutdown periods.

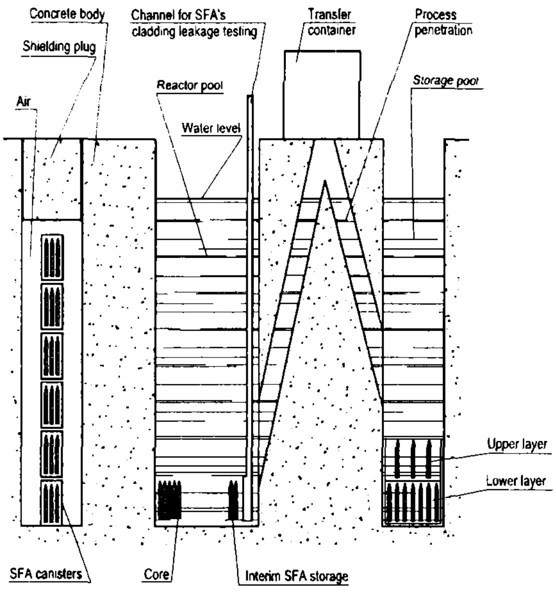

Fig.1.

A separate pool filled with CDW is used for wet storaging. The storage pool has a system of ion-exchange filters responsible for water chemistry and a system of water quality and temperature control. The water temperature is 30-40°C.
54 SFAs are placed in a special fuel storage rack in two layers – there are 48 SFAs in the lower layer and 6 SFAs in the upper one. Prior to the installation for storage, SFAs were cooled in the reactor pool outside the core where they were monitored for leaktightness based on water sipping in an isolated canister using the Te-I procedure of detecting leaky SFAs.

4.Procedure for detecting SFAs cladding leakage

The SFA cladding leakage extent is determined using the method of measuring the FA cladding surface contamination with fission-produced Te^{132}.
Due to high sorption capacity and short path length of Te^{132} nuclei in the cladding material, the appearance of tellurium on the cladding surface is caused by the fission-produced tellurium release from beneath the cladding via its defects (cracks, pits, etc.) and the amount of the absorbed Te is proportional to the fission-produced Te release rate (the proportionality factor depends on the reactor operation mode). With the period of $T_{1/2}$ = 77 hours, Te^{132} undergoes a β-decay with generation of daughter I^{132} ($T_{1/2}$ = 2.3 hour). As iodine has lower sorption capacity, it is released from the cladding to the coolant that flows around the FAs. Thus, the Te^{132} amount on the FA cladding surface can be determined by installing the FAs from the reactor core 40-50 hours after the shutdown, when the absence of fission-produced I^{132} on the FA cladding surface is guaranteed, in a leaktight accumulator with very pure water and by measuring the Te^{132} amount in the accumulator. The I^{132} concentration is measured based on the area of $E\gamma$=670 keV and 770 keV photopeaks in the γ-radiation spectrum of the water sample taken from the accumulator.

5.Equipment and technology for SFA shipment for reprocessing

In 1989, 48 IRT-2M SFAs were transported to reprocessing plant RT-1. A TK-5M shipping cask intended for these purposes was used for transportation. The weight of one shipping cask is not more than 5500 kg. Casks (in batches of 4 pieces) were transported by a trailer truck to a railway track to the Kurchatov Institute, Moscow, and put onto a TK-5 container car.
One of the basic conditions of putting the SFA into the shipping casks for SFA shipment for reprocessing was a gamma radiation exposure dose rate that was not to exceed 100 R/h at a distance of 1 m off the FA. The SFAs shipped for reprocessing were cooled in the storage pool for 3.5 to 11 years and the gamma radiation exposure dose rate was 20 to 80 R/h. The SFAs were put into casks using regular transfer equipment – rods with collet grips and a transfer container. Prior to being loaded into casks, the SFAs were dried by hot air blowout.
Russian regulations prescribe [4] that spent FAs to be delivered for reprocessing should not have leaky fuel elements causing contacts of the fuel composition with the SNF storage facility water (microdefects of the fuel cladding corresponding to the «gas non-leaktightness» of the cladding are permitted).
The SFAs to be shipped were:

- checked for leaktightness;
- visually examined;
- checked for passability in a gage.

The (average) burnup of the SFAs in this batch was more than 40%.
The leaktightness of the fuel cladding was checked by each FA sampling in the storage pool based on detection of I^{132} isotope technique.
There were no comments with respect to the SFAs condition after the checking operations being completed.
The SFA quantity coming annually to the storage pool is small (some 4 pieces) and the storage facility volume will be enough for approximately another 10 years. But according to the available information [5], the condition of similar SFAs (the fuel composition of U-Al alloy in an aluminium matrix, the

thickness of the fuel cladding made of an aluminium alloy is 0.9 mm) stored in the storage pool water and having the burnup of around 15 MW·day, was normal for the first 15 to 20 years of storage after which problems started caused by more fission fragments released to the water as the result of the fuel cladding corrosion. The SFAs of the IRT MEPhI reactor that have been accumulated to date still have a time reserve of 10 to 15 years till the start of the potential loss of the fuel cladding leaktightness after which it is not possible to store them in the existing conditions. In this connection, it is necessary to consider possible variants of the subsequent SFA handling – their transfer to the dry storage facility or for reprocessing. Most of the accumulated SFAs (52 pieces) have been cooled for more than 5 years after their withdrawal from the core, which enables their transportation to reprocessing plant RT-1. It is rational to simultaneously transport 77 SFAs with EK-10 fuel elements stored in the dry storage for accumulating the body of such assemblies on the RT-1 site with their further reprocessing.

6.Conclusions

The IRT MEPhI reactor has been the basis for research activities and training of specialists in the industry throughout the period of its service. Most of the developments have been practically implemented in the field of nuclear engineering and technology. Thus, for example, verification work has been completed based on obtained experimental data of the neutronic TIGR code and the regular Te-I procedure for monitoring the fuel cladding leaktightness has been introduced.

Long-term water storage of SFAs in the storage pool has not lead to a visible degradation in the condition of the SFA structural materials, but if storage is continued in these conditions, the integrity of the fuel cladding is not guaranteed. The only possible solution of this problem both for the IRT MEPhI reactor and other Russian research reactors is transportation of SNF for reprocessing in near future as it is more expensive to set up a dry storage facility on the reactor site and to transfer the fuel to a more safe dry storage facility.

References

[1] M.F.Shchurovskaya, V.P.Alferov, A.A.Pinegin, V.V.Khromov. Development and Verification of the IRT MEPhI reactor Calculation Model. *Atomnaya Energiya. 1996. Vol.81. Issue 5.*

[2] D.Reily, N.Ensslin, H.Smith Jr., S.Kreiner. Passive Nondestructive Assay of Nuclear Materials. *NUREG/CR-5550, LA-UR-90-732, 1991.*

[3] JEF-PC Version 2. A PC program for displaying data from the joint evaluated file library.

[4] OST 95 10297-95 Industry Standard «Spent Fuel Assemblies of Nuclear Research Reactors. General Requirements for Supply».

[5] S. Chwaszczewski, W. Czajkowski, E. Borek-Kruszewska, T. Matysiak, M. Madry. "Management and physical assessment of the spent fuel from research reactors in Poland". *ENS RRFM'2001, Transactions of 5th International Topical Meeting on Research Reactor Fuel Management, April 1 to 3, 2001, Aachen, Germany.*