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Technical Note

Innovative technologies for spent fuel safe management at Ignalina channel-type reactors

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ABSTRACT

In Lithuania, all spent nuclear fuel (SNF) resulted from the operation of the Ignalina Nuclear Power Plant (INPP), which had two Russian Acronym for “Channelized Large Power Reactor”-type reactors. After the final shutdown, the total amount of SNF at the INPP was approximately 22,000 fuel assemblies. All these assemblies will be stored for about 50 years and disposed of after that. The decision to shut down and decommission both reactors in Lithuania before termination of design period raises a significant challenge for the treatment of accumulated SNF. Therefore, various techniques and technologies for SNF management were developed and justified for that specific case, and a set of special equipment was installed at the INPP, the effectiveness of which was demonstrated during its operation. This article presents unique techniques related to the management of SNF adopted and commissioned at the INPP after its operation shutdown, namely fuel rod cladding leak tightness control system and special equipment for collection of possible spillage during handling of SNF assembly in the hot cell. The operational experience and measurement results of fuel rod cladding leak tightness control system are presented.

1. Introduction

Decommissioning of nuclear reactors involves many activities; one of the important activities is spent nuclear fuel (SNF) management. SNF needs to be managed in a safe, responsible, and effective way [1,2]. It is very important to prevent the release of fission products into the environment during reactor operation, SNF storage, and also during the handling and reloading of fuel assemblies. Failure of fuel assembly can occur due to embrittlement of the fuel cladding material, corrosion of the cladding, and mechanical damage from handling accidents. Hence, the integrity of fuel elements is the major issue to be checked before the fuel assembly undergoes any further handling operation. To enable this, an effective leak detection system is necessary.

Sipping is the most common technique used to detect leaking fuels; it is based on the measurement of fission product activity. This technology is divided into vacuum sipping, dry sipping, wet sipping, or in-mast sipping, depending on physical phenomena and the state of the fission products to be detected [3]. The measured radioisotopes are Xe, Kr, Cs, and I. In-mast sipping systems, installed on the refueling machine in the reactor building, are designed to identify irradiated leaking fuel assemblies during core unloading or SNF management operations.

Spent fuel assemblies (SFAs) are usually not suitable for direct treatment in terms of size, weight, and form for later processing and encapsulating [4]. Russian Acronym for “Channelized Large Power Reactor” (RBMK) SFAs are normally cut into half-lengths for packing into storage/transport containers, baskets, etc., for further management. During this kind of SNF mechanical treatment, various loose debris and spent fuel spillage are generated in hot cells of nuclear power plants (NPPs) when using various cutting machines. Remote equipment (e.g., manipulators with high degrees of freedom) is normally used for picking up of fuel debris arising from treated operations, but not all devices are effective or adaptable in specific cases.

Lithuania's particularity in the nuclear field is that there are two Soviet-designed RMBK-1500 reactors at the Ignalina Nuclear Power

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Plant (INPP), originally the most powerful reactors in the world when they started operation and the only ones of such design in the market. The units were put into operation in 1983 and 1987, and the reactors are identical in terms of core design. As a result of the political dialog leading to the European Union enlargement, Lithuania agreed to the early shutdown of its reactors: Unit 1 was shut down in 2004, and Unit 2 was shut down in 2009. After final shutdown, the total amount of SNF at the INPP was approximately 22,000 fuel assemblies (about 2,500 tons). The spent fuel was fully removed from the Unit 1 and was transferred to fuel storage pools. On 1 February 2011, defueling of SNF from Unit 2 and its transfer to pools was started. The INPP Unit 2 reactor still has about two-third of its fuel inside (data at the beginning of 2017).

The decision to shut down and decommission both reactors in Lithuania before termination of the design period raises the significant challenge for the treatment of accumulated SNF. At the time when the decommissioning strategy was approved, the existing dry storage facility for SNF was almost fully loaded, and construction of a new facility had not started yet. Upcoming significantly increased activities related to SNF management require adaptation of additional dedicated facilities and techniques for further processing of SNF. A fuel element cladding leak detection system for SFA at the INPP was developed. This system has a set of equipment that enables remote tests of fuel cladding integrity during unloading of SFA from the reactor core and transfer from unit transport-packing sets to storage casks using a refueling machine; system enables processing and recording of test data results on a personal computer (PC). A special system for collection of debris during cutting of SFA in hot cell was developed as well. A set of such equipment has been produced, supplied, installed, and commissioned at the INPP. It should be mentioned that such systems were designed as additional emergency systems that protect the environment and operators from ionizing radiation during different operations with SNF.

This article presents technologies and techniques related to the management of SNF adopted and commissioned at the INPP after the closure of the plant.

2. Detection of leaking SFA

Fuel failure is an important issue for all nuclear utilities. Nuclear energy specialists devote considerable effort to developing effective and efficient means of detecting damaged/leaking fuel assemblies during the operation of a nuclear installation. After analysis, the monitoring and/or measurement data are used to decide how to deal with the spent fuel.

2.1. Fuel leakage mechanism

The fuel failure rate differs by the type of reactor. Based on statistical data on fuel failure [3], the highest world average fuel failure rate belongs to Water-Water Energetic Reactor (VVER) (~15 leaking SFAs per 1000 discharged FAs), followed by ~14 for Pressurized Water Reactor (PWR), ~4 for Boiling Water Reactor (BWR), and~ 0.4 for CANada Deuterium Uranium (CANDU) [3]. The major mechanisms that cause leak failure or damage of fuel rods are grid to rod fretting (PWR), crud/corrosion (BWR), and debris damage (CANDU). In the case of VVER fuel, the dominant failure causes are debris damage, fretting wear, and deposits and displacement of fuel rods during transportation. Other defects are related to technological nonconformity during manufacturing/fabrication of fuel rods or are due to fuel and/or cladding behavior during operation. Release of fission gas products from fuel and spreading under cladding of PWR, BWR, VVER, and CANDU fuel rods have been fairly well studied, and there is a wide set of experimental and statistical data on fuel failures that cannot be said for RBMK fuel.

Currently, there are no reliable statistically significant data on the distribution of leaking RBMK fuel rods for the four classes of defects, i.e., microdefects, gas leaks, developed gas leaks, and macrodefects. Such classification is used in the Russian regulations [5] depending on the degree of leak. The results of studies on failure phenomena in RBMK fuel rods of different design loaded with different types of fuel (uranium and uranium–erbium) in the range of fuel burnup from 6 MWd/kgU to 26 MWd/kgU [6] showed that the main reason for the depressurization of the investigated spent fuel rods is cladding mechanical fretting by extraneous objects in the coolant (debris fretting). The size of these defects can be attributed to the class of macrodefects. In this case, water enters under the fuel cladding, where water vapor is segregated into oxygen and hydrogen due to high temperatures and radiolysis. Further hydriding and oxidation occur on the inner surface of the cladding with the formation of secondary defects [7].

2.2. Review of cladding leak detection methods

Based on worldwide experience of the development and application of various leak detection techniques, two classes of methods are most effective and reliable, i.e., in-mast sipping during fuel assembly unloading from the core or shuffling operations using a refueling machine and fuel assembly leak tests in special cells [3]. Gas sipping is used in these cases, allowing an increase in detection sensitivity in comparison with that obtained by water sipping in SFA pools.

During unloading operation, owing to a decrease in the hydrostatic pressure (elevation of SFA), the fission products, located under the cladding of detective fuel rods, release and enter the refueling machine mast. To increase the efficiency of the method, an air or argon stream is injected at the bottom part of the machine mast, which entrains the gaseous products and increases the release of radioactive fission products from the coolant to the gas sampling system. Xe-133, with a half-life of ~5 days, is used as the indication nuclide for qualitative leak detection test of each fuel assembly during refueling operation above the reactor.

Leak detection tests in special cells (sipping in cells) are performed usually either to refine results obtained using on-line in-mast sipping or to assess the type and size of the defect in the leaking fuel rod. One of the most sophisticated systems is the Fragema sipping test cell stand [8]. The examined SFA is placed in a close-fitting vessel, inside which the water temperature is quickly increased, thereby increasing the internal pressure in the rod. Detection is based on the measurement of fission gas products and fission products of nuclear fuel in the water when release occurs in the case of cladding failure. Using a special software program, the leak size is characterized by an “equivalent diameter” determined by the fission product release kinetics. Using this Fragema sipping test cell, the leaking fuel rods are grouped into two classes, i.e., with defect sizes smaller or larger than 35 μm [8].

In addition to these two classes of sipping methods, an ultrasonic technique is used [3], which enables the presence of water under cladding of the leaking fuel rods to be detected. These methods are based on the effect of attenuation of ultrasonic waves propagating in fuel cladding in the presence of water in the pellet–cladding gap.

Until the shutdown of the INPP, the measurement of activity of fission gas products during leaking fuel assembly handling was performed during vacuum drying of assemblies or during the unloading of the SFA from the core, when reactors were on power. Leak detection of SFA, discharged from the reactor core after more than 5 years of cooling, was performed by measuring the activity of

Kr-85 in a 5-m^3 container before dispatching SFA for interim dry storage. In the presence of leaking SFA, the activity of Kr-85 varied from 8,000 Bq/L to 56,000 Bq/L during the vacuum drying process. How many leaking fuel rods were present and the level of burnup were unknown. Measurement was based on the registration of the gamma activity of fission product releases through a defect in nontight cladding within the tested fuel assembly, which was placed in a refueling machine bar (in-mast sipping). Fission gas products are released as a result of a decrease in coolant external pressure when a tested fuel assembly is raised to the top transportation position. The resultant drop between internal and external pressures leads to gas and fission gas products being dissolved from the nontight cladding into the surrounding coolant. Level of gamma radiation in SFA unloading process varied in the range of 1–5 Mr/h.

The described technologies do not address all the requirements for detecting fuel cladding leak tightness; therefore, we developed the new technologies described below.

2.3. *In-mast sipping using refueling machine at the INPP*

After shutdown of Unit 1 in December 2004, a large number of fuel assemblies, which had not exceeded design burnup limit, were left in the reactor core. After reactor cooling, these irradiated fuel assemblies were reused in reactor Unit 2 [9], and that gave tangible economic benefit. Before loading them into Unit 2, it was necessary to ensure that they were leak tight. For this purpose, a leak detection system was developed and designed to detect leaking fuel assemblies in the process of

- unloading SFA from the Unit 1 reactor core after at least 1.5 years of cooling;
- reloading SFA from transport-packing cask into storage cask at Unit 2.

This in-mast sipping system belongs to a class of indicators and provides separation of inspected SFA into the two classes—tight and nontight SFA. The SFA is nontight if the assembly has at least one depressurized fuel rod with cladding failure, ranging from defects such as *gas leak* to defects that occur when direct contact between nuclear fuel and coolant takes place. The method is based on detection of Kr-85 which is released from the nuclear fuel matrix, which diffuses into the free volume of the fuel rod and then through a cladding defect into the local environment. Kr-85 detection is carried out during loading of the SFA from the core into the refueling machine, or from the transport-packaging cask into the refueling machine, by measurement of beta activity of the gas environment in the machine main mast (5 m^3). The measurement is performed on-line without significant loss of time.

SFA unloading was performed from the shutdown, cooled RBMK-1500 reactor core or when the reactor is running on power; SFA reloading from transport-packing cask to storage case is performed using the refueling machine. These actions are performed using in-house equipment according to plant procedures and proven experience. In-mast sipping detection equipment is designed so that the installation invokes minimal changes in the technological scheme of the plant (Fig. 1). Gas sampling in the telescope mast is performed from the gas line through which gas is discharged to the plant's special venting system. When the SFA is raised from the core to an upper position inside the machine mast, the differential pressure caused by the change in elevation promotes the release of fission products from the defective rods. A vacuum is created inside the mast, resulting in additional output of gas from the leaking fuel rod(s). Therefore, the amount of Kr-85 in the measured sample increases, and this increases the sensitivity of the method. Fission gas products are pumped from the mast of the refueling machine using the vacuum pump. Gas passes through the detecting unit of the beta radiometer before being discharged to a special venting system. The signal from the detection unit enters the registration box of the beta radiometer and, after primary treatment, is transmitted to a personal computer. Radiation safety in the process of detection of leaking SFA is achieved by the presence of a sufficient level of radiation monitoring. The testing process is controlled from the remote operator desk. Both automatic and manual control modes can be used.

This type of in-mast sipping system was installed in both Units 1 and 2 in 2006. Systems are integrated into the plant procedure for SFA discharge from reactor, i.e., all fuel rods undergo leak tests using these systems. As modernization of the system was not performed during its operation, the shortcomings have not been detected. The burnup range of tested SFA is 100–2600 MWd/FA. Based on accumulated experience, it can be said that by ensuring the vacuum level of 0.5 P_{atm} in the mast of the refueling machine and employing a radiometer with a threshold of sensitivity ~10^4 Bq/m^3, all leaking SFAs are being detected (Fig. 2). A total of 1,659 SFAs have undergone testing at Unit 1; four nontight SFAs were detected in the burnup range of 1800–2300 MWd/FA. A total of 1,507 SFAs were tested at Unit 2, and two nontight SFAs with burnup of 2,077 and 2,021 MWd/FA were detected.

2.4. *Cladding leak detection system in hot cell at the INPP*

Based on sufficient experience and knowledge accumulated at the Ignalina plant and on the basis of established in-mast sipping methodology, an essentially similar cladding leak detection system for use in the hot cell was developed, designed, installed, and commissioned. Detection is based on the registration of Kr-85 beta activity inside the hot cell. According to the current standard INPP procedures, after SFAs were retained in storage pool SFAs, they were transported to the hot cell for dismantling. Only tight fuel bundles can be placed in a transport-packing cask. Therefore, it is necessary to monitor the SFA integrity throughout all activities in the hot cell, i.e., at the time of discharge of SFA from storage pool and during the cutting process. During unloading from the pool, SFA goes into the hot cell, inside which a ventilation system maintains a constant vacuum level of about 196 Pa to prevent the spread of contamination to other premises. The temperature in the hot cell is 40°C. Gas sampling in the two zones inside the cell is carried out, and it enables

- to test the cladding integrity of fuel rods before cutting, while SFA is lifted from the case of transfer equipment;
- to test the cladding integrity of fuel rods during the most potentially dangerous operation, i.e., cutting SFA into two bundles.

The fuel rod(s) may be damaged by the cutter during the process of cutting activities. Obviously, if this were to occur, a greater quantity of Kr-85 from the damaged fuel rods would be released than the case in which depressurization may occur in the early stages of the technological process. The output of fission gas products from defective fuel rods will depend on the size of the defects. Macrodefect or, at least, developed *gas leaks* will be established in the case of destruction of a fuel rod by a cutter.

The SFA cladding leak test system was installed in the hot cells of Unit 1 (2009) and Unit 2 (2008). Modernization of the system was not performed during its operation, and therefore, shortcomings have not been detected. The leak detection system is a complete set of parts and components for the remote monitoring of cladding integrity when SFA is moved into the hot cell for cutting. The

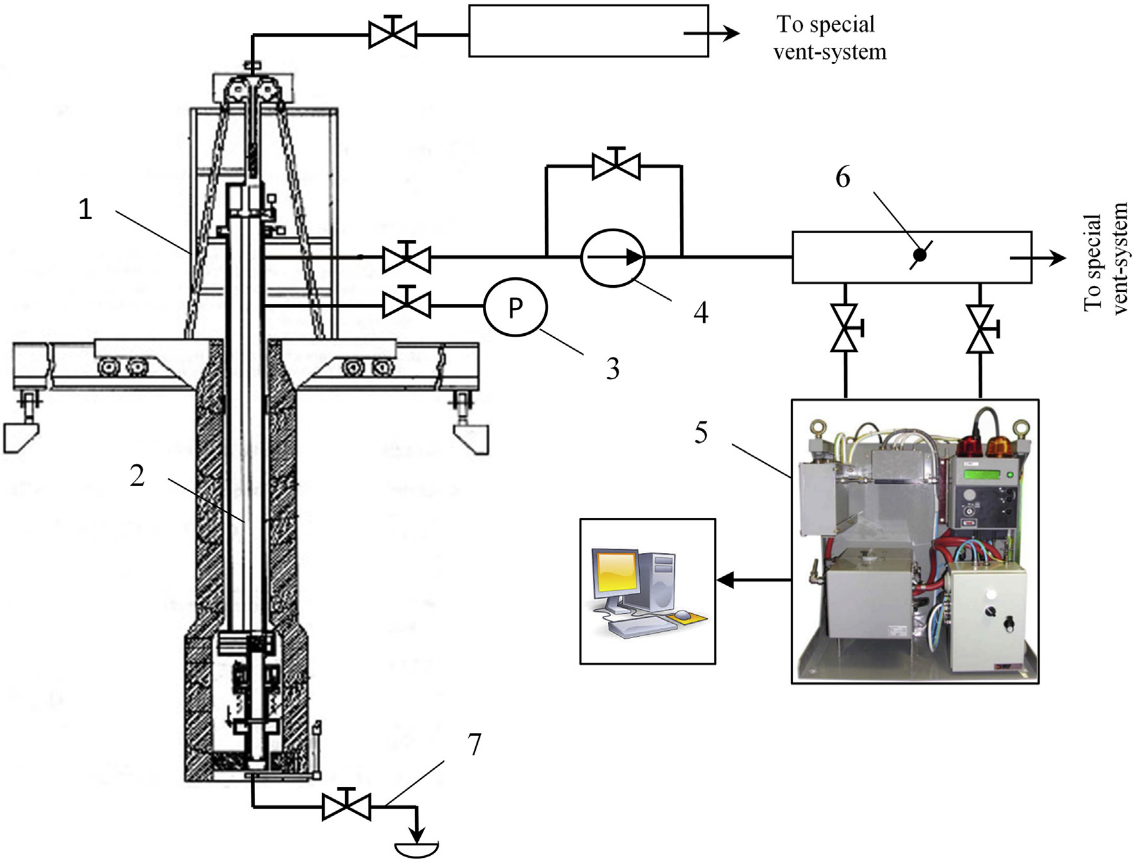

Fig. 1. Block scheme of in-mast sipping system at the Ignalina NPP: 1, refueling machine; 2, coupling unit; 3, vacuum gauge; 4, vacuum pump; 5, detecting unit (NGM 204L); 6, choke valve; 7, drainage.
NPP, nuclear power plant

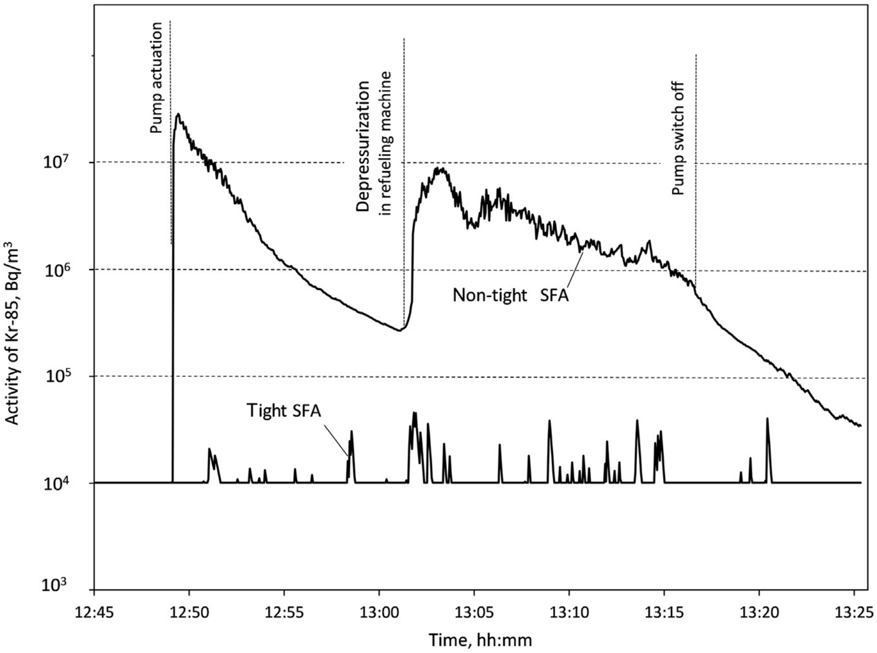

Fig. 2. Detection of nontight SFA using the Ignalina NPP in-mast sipping system.
NPP, nuclear power plant; SFA, spent fuel assembly.

system is integrated into the plant procedure of SFA dismantling. All fuel rods undergo leak tests during dismantling and the tests results are recorded on the PC. The existing Ignalina NPP vacuum system is used for pumping gas probes. SFA leak detection system is constantly under working conditions; sampling of gas probes for the control is carried out simultaneously in both controlled zones of the hot cell. A block scheme of the system is given in Fig. 3. Device NGM-204L is used as the measuring part of the control equipment.

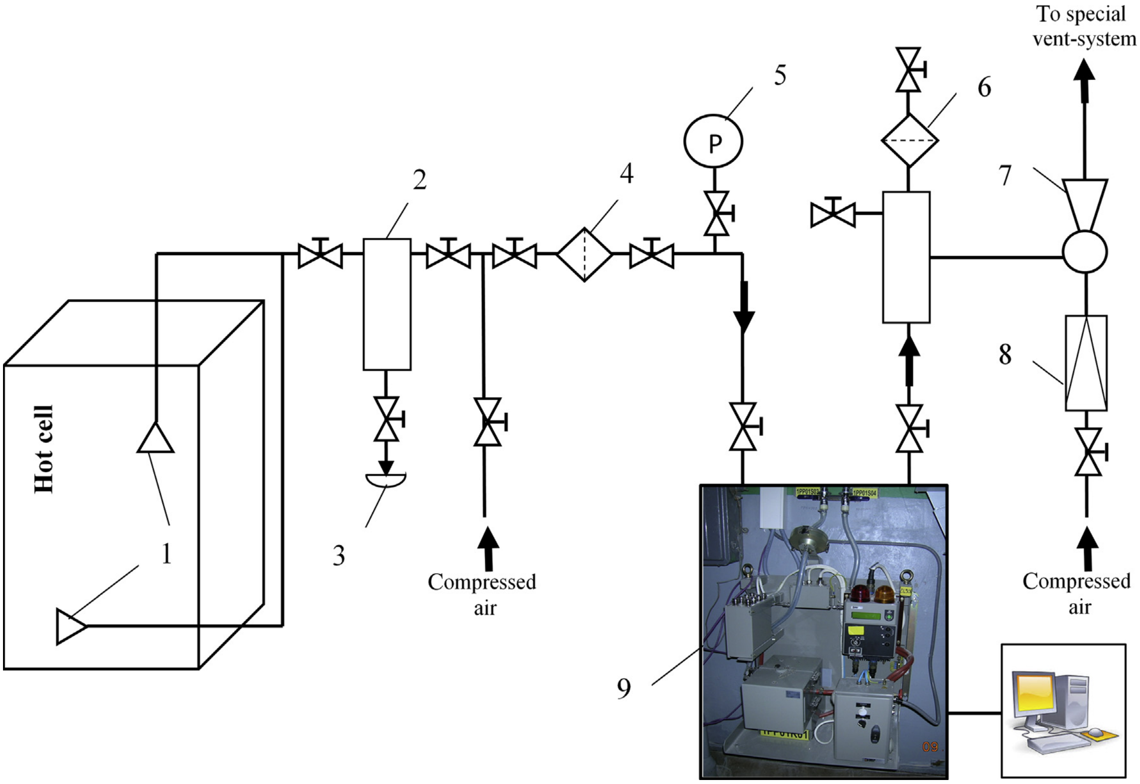

Fig. 3. Block scheme of the SFA cladding leak detection system in the hot cell: 1, air sampling zones; 2, dehumidifier; 3, drainage; 4 and 6, aerosol filters; 5, vacuum gauge; 7, ejector; 8, pressure regulator; 9, detecting unit (NGM 204L).
SFA, spent fuel assembly.

Results of justification calculations and accumulated operational experience confirmed that by creating special stagnant zones for sampling in the hot cell (~20 L), by using a radiometer with a threshold of sensitivity 10^4 Bq/m^3, and by following the standard conditions in the hot cell, all leaking SFAs are detected in the burnup range of 1,500–2,800 MWd/FA, while SFA is being extracted from the storage pool and put into the hot cell during cutting operations, if an SFA is damaged (Fig. 4). A total of 51 and 482 SFAs underwent testing in the hot cell of Unit 1 and Unit 2, respectively; only one nontight SFA with burnup 2572 MWd/FA was detected at Unit 2 before cutting. Installed cladding leak detection system in hot cells of the INPP ensures radiation protection when SFAs are stored longer in the fuel storage pools or during SFA loading into transport-storage casks and storage at dry spent fuel facility.

3. SNF spillage collection

The equipment for SNF spillage collection in the hot cells of the INPP is designed to eliminate the consequences of an accident associated with the violation of an element of the system of barriers to the spread of ionizing radiation and radioactive substances into the environment, e.g., the cladding of fuel elements.

The initial design in all RBMK power plants does not include the separation of SFAs by cutting (with the exception of the INPP). Hot cells for cutting fuel assemblies into bundles of fuel rods and loading fuel bundles into dry-type storage containers were installed later at the Leningrad NPP, Kursk NPP, and Smolensk NPP. Any information on equipping of the hot cells with equipment for SNF spillage collection is missing. The INPP is the first power plant with RBMK reactors, for which the technical design was originally intended for the separation of SFAs by cutting.

3.1. SNF management in hot cells of the INPP

The technological process for the preparation of SFA for intermediate storage at the INPP involves cutting the fuel assembly into two pieces, thus separating the upper and lower fuel rod bundles and removing the central tube or central rod from the fuel assembly. To perform this operation, a hot cell with a set of necessary equipment is provided. However, the hot cell design did not provide for the collection and disposal of waste in the form of SNF spillage, which may form during SFA cutting. Spillage of nuclear material can occur by carrying out operations in hot cells with SFA: for UO_2 fuel from 2% to 2.8% initial enrichment of U-235, the maximum burnup is 3200 MWd/FA.

Cutting in the hot cells is only carried out for leak-tight SFAs. Formation of spillage in the hot cells is possible in the following cases:

- dropping of an SFA or bundle of fuel elements during handling operations;
- fuel bundles are cut as a result of operator error during separation of the SFA;
- erroneous transfer to dismantling process of damaged SFA and spillage of fuel element pellets during handling operations.

Therefore, a set of equipment for SNF spillage collection in the hot cells of the INPP was designed to perform a technological process that ensures

- safe collection of all spilled nuclear material within the hot cells into special capsules compatible with the storage basket;
- sealing of capsules (restoring the containment barrier);

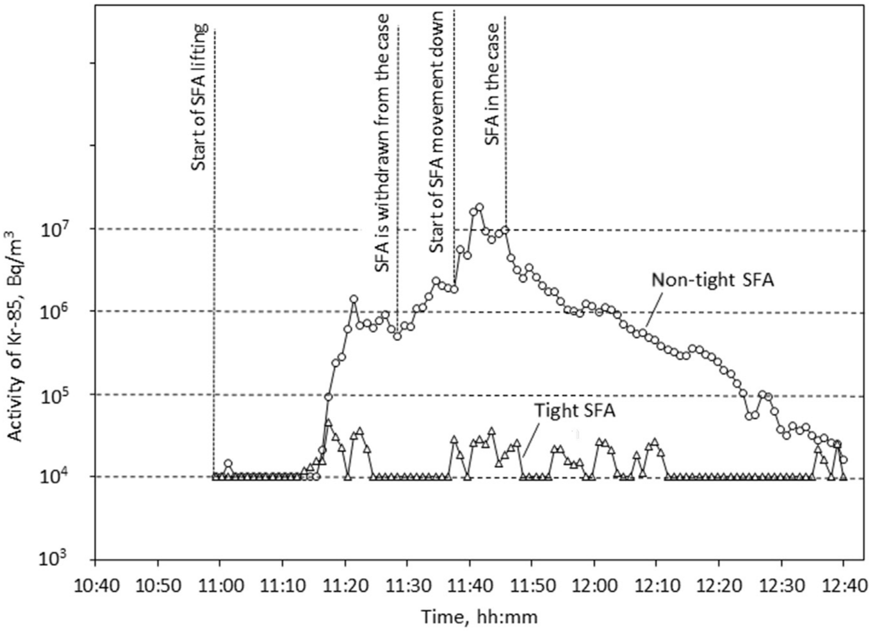

Fig. 4. Detection of nontight SFA using cladding leak detection system in the INPP hot cell during SFA elevation from the case.
INPP, Ignalina Nuclear Power Plant; SFA, spent fuel assembly.

- placement of capsules in storage baskets for further intermediate storage.

Maximum volume of collected nuclear material in the hot cell is 50 kg of uranium dioxide spilling, the equivalent density of which is $3.79\ g/cm^3$. The minimum size of the collected SNF particles is $d = 0.05\ \mu m$; meanwhile, the maximal size is one fuel pellet.

There are two possible ways to determine fuel element damage: visual, when the hot cell operator directly sees damage of fuel rods, and instrumental, when the signs of fuel element damage are identified by the system of automatic monitoring of radiation safety or leak detection system.

3.2. Locations and volume of spent fuel spillages

Determination of location and volume of SNF spillage is performed visually, through the protective glass of the hot cell camera, using a regular periscope and an industrial television system. This television system is based on video surveillance equipment supplied with a color high-resolution dome camera (Fig. 5) and a high-performance PC. The video system is equipped with software for the camera control, with a recording function and an installed program for measuring the sizes of objects.

Fig. 5. Video camera (N190-PTZ) used for visual control in the hot cells of the Ignalina NPP.

A cartogram of the hot cell is drawn up; the camera is installed in the penetration and entered into the hot cell. The floor in the hot cell is inspected in all possible ways. The images of the computer system are recorded on digital media. The cartogram of the hot cell lists the locations of fuel pellets and their fragments. At the same time, the digital sizing software provides the geometric dimensions of fuel fragments. The total volume of spillage of nuclear material is determined by summing the volume of spillage units on the cartogram. After determining the locations of spillage and drawing up of a cartogram, all operations in the hot cell, except for the collection of spillage, are prohibited.

The radiation monitoring system in the hot cell is intended for monitoring, recording, displaying, collecting, processing, and issuing reporting information on the monitored parameters characterizing the radiation state of the complex. The radiation monitoring system provides dose rate, information on gamma and neutron radiation in the workplaces, monitoring of volumetric activity of gases and aerosols in the hot cell, measurement of radioactive contamination of the surfaces of equipment and compartments, and individual monitoring of external and internal exposure of the personnel.

3.3. Collection of nuclear material spillage

The collection of spillage is performed by pneumatic and hydraulic systems. The main element of the pneumatic system is a gas jet air ejector (0.6 MPa), which is used to create a vacuum in the suction nozzle, capturing particles, spilling, and transporting them into the capsule through the coupling hose (Fig. 6).

Using the crane manipulators and other remote equipment, the operator collects fragments of fuel pellets in the capsule. Collected SNF fragments are marked on the cartogram, and work continues until all the fragments noted on the cartogram are collected.

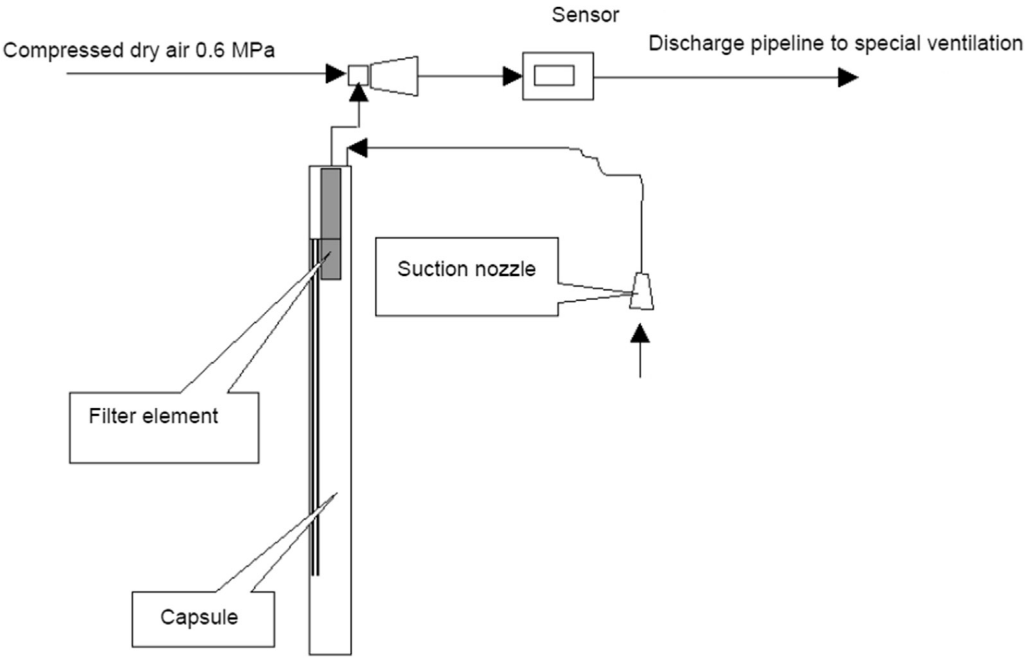

Fig. 6. General operating scheme of equipment for SNF spillage collection.
SNF, spent nuclear fuel.

Verification of the completeness of the collection of spillage is performed visually through the protective glass of the hot cell camera using a periscope and a television system.

After collecting spillage with the pneumatic system, small SNF fragments can remain on the floor of the hot cell, on the equipment, and in the drainage trap. The final collection of SNF spillage is carried out by a hydraulic system. The operator flushes (using chemically desalinated water 0.6 MPa) the particles of spillage into the trap of the hot cell using remote equipment. First, flushing of visible particles of spillage, which cannot be recovered by the pneumatic system, is performed. Then, the equipment is washed. Equipment having a complex spatial configuration, in which the potential for accumulation of particles of spillage is highest, is most carefully washed. A special box is provided for the placement of capsules of pneumatic and hydraulic systems.

After completion of work using the pneumatic and hydraulic systems, the capsules are dried. For this drying, compressed, dry air is supplied through the conduit duct and the perforated tube inside the capsule. Air is supplied to the bottom of the capsule through a perforated tube. This ensures the drying of the entire volume of particles of spillage in the capsule. The humidity level of the air withdrawn from the capsule is monitored by a sensor installed in the humidity-measuring cavity. When the control value of the humidity level of 4–5 g/m^3 is reached, the air supply to the capsule is terminated.

3.4. Sealing of capsules and loading into the storage basket

Sealing of capsules with SNF spillage using pneumatic and hydraulic systems is performed to ensure high level of containment of spillage for subsequent storage. After drying, the capsules are sealed by welding. Sealing of capsules with SNF spillage is performed at the welding stand using manual argon-arc welding. The capsule is lowered into a box by the crane of the pool hall and transferred to a hot cell. According to the performed thermal calculations [10], the storage of capsules with spillage does not affect the temperature regime in the storage container. In the case of depressurization of the capsules in a container, only a partial release of gaseous fission products from the capsule into the inner volume of the container is possible. Such a release process is in general analogous to the decompression of the fuel element. Depressurization of the capsules does not have a negative impact on the storage conditions. The design of the capsules is compatible with the design of the standard basket (type 32M) and the storage container; this did not require changes in the handling technology and SNF storage conditions at the INPP.

Capsules with a filter element and with fragments of nuclear fuel are loaded into the storage basket using the hot cell crane. Inside the capsule, there is a filter designed to prevent the breakthrough of SNF particles entering the capsule together with air, back to the volume of the hot cell. The filter is a porous cylindrical element; the filtering part provides precise air purification and particle retention of up to 0.05 microns in size, with an efficiency of up to 99.999%. Capsules with spillage are loaded only in the peripheral cells of the basket, whereas capsules with filters can be loaded in any free cell. Temporary storage of the filled capsules in the hot cell is possible when the fuel cutting operation is suspended.

3.5. Radiation monitoring system

The system of radiation monitoring within the SNF spillage collection set in the hot cells is designed for monitoring, recording, displaying, collecting, processing, and issuing reporting information on the monitored parameters characterizing the radiation state of the whole equipment. This system provides the following types of radiation monitoring:

- monitoring of the gamma dose rate and neutron radiation at workplaces of personnel engaged in collection of spillages;
- monitoring of volumetric activity of gases and aerosols in the volume of the hot cells and in the ducts of the ventilation system;
- monitoring of radioactive contamination on the surfaces of the equipment and the premises in which it is located;

- individual monitoring of external and internal exposure of personnel.

The equipment for SNF spillage collection in the hot cells of the INPP was designed as an emergency system for the case of SFA damage incident. For the whole period of operation of the hot cells, no such incidents were observed. The main safety-related tasks of the system, both in normal operation and in the event of an incident, are to prevent criticality and ensure adequate radiation protection for personnel and the public.

4. Conclusions

Unique technologies related to SNF management, designed and commissioned at the INPP after its shutdown, namely a fuel rod cladding leak detection system and special equipment for collection of nuclear material spillages in the hot cell, are described in the article.

An in-mast sipping system, installed on the refueling machine in the reactor building, is designed to identify irradiated leaking fuel assemblies during core unloading or shuffling operations. This system belongs to a class of indicators and provides a separation of inspected SFAs into the two classes—tight and nontight SFA—in the process of unloading SFAs from a cooled-down reactor core 1.5 years or more after its shutdown and during reloading of SFAs from transport-packing casks into storage cases. The detection method is based on Kr-85 registration. The operational experience obtained shows that by ensuring a vacuum level of 0.5 P_{atm} in the mast of the refueling machine and employing a radiometer with a threshold of sensitivity ~10^4 Bq/m^3, all leaking SFAs are detected. The installed system was successfully used during reloading of irradiated FA (~1000) from Unit 1 to Unit 2. About 3,200 FAs underwent leak tests in total, and six FAs were detected as nontight. An essentially similar cladding leak detection system for use in the hot cells of the INPP was installed and employed; during reactor dismantling, more than 500 SFAs were tested, and only one SFA was detected as nontight. This means that fuel failure rate for the Ignalina RBMK are about two leaking SFAs per 1,000 discharged fuel assemblies.

The equipment for collection of SNF spillage that was installed in the hot cells of the INPP is devoted to safe collection of all spilled nuclear material within the hot cells into special capsules, sealing of capsules, and their placement in storage baskets for further intermediate storage. Capsules are compatible with existing storage basket design, and further SFA management follows standard storage procedures at the INPP. The installed system eliminates the consequences of an accident associated with the breach of the fuel cladding barrier, e.g., the spread of ionizing radiation and radioactive substances into the environment. Incidents of SFA damage (formation of spillage) were not observed for the whole period of operation of the hot cells at the INPP.

Conflicts of interest

The authors have no conflicts of interest to disclose.

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