

Handling and behaviour of AVR fuel elements for interim storage

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Abstract

Between 1966 and 1984, nearly 290000 spherical fuel elements with several types of coated particles have been irradiated in the Juelich Experimental Power Plant (AVR). The spent fuel elements have been discharged from the reactor into small stainless steel canisters with a capacity of 50 pebbles each (AVR can). The AVR cans were received and racked in a pool facility of the hot cell laboratory and in the DIDO reactor with the aim for interim storage. After several years of storage in the pool, the small AVR cans were decanted in stainless steel storage canisters with a capacity of 950 fuel elements maximum (dry storage can). In addition an interim storage facility with 158 transport/storage casks (CASTOR-AVR), made of nodular cast iron, was installed. Each cask is suitable for the storage of two dry storage cans.

Complementing the storage of AVR fuel, a measuring program was initiated to demonstrate the safety of dry storage of spent HTR fuel elements and to provide data for licensing commercial storage casks. It was found, that under interim storage conditions only the radioactive gaseous nuclides H-3, Kr-85 and C-14 (CO₂) can be released, and as a consequence these are the principal nuclides measured in this programme. Additional gamma- and neutron dose rate measurements at the surface of the transport-/storage casks were carried out.

Introduction

From August 1966 to December 1988, the experimental power plant AVR was operated by the Arbeitsgemeinschaft Versuchs-Reaktor GmbH on the site of Forschungszentrum Jülich (FZJ). The AVR is a gas cooled high temperature reactor with a power of 50 MWth. The core consists of approximately 100 000 spherical fuel elements of 60 mm diameter with coated particles embedded in a graphite matrix. During the years of operation 50 fuel elements were unloaded every day after reaching the maximum burn-up (20% FIMA).

During the operational years, nearly 290 000 spherical fuel elements with several types of coated particles have been irradiated in the reactor. After shutdown of the reactor, the fuel elements have been removed from the core. This procedure was finished in 1998 but preparation for interim storage has not been finished due to licensing reasons. For the handling of fuel elements on their way to the interim storage plant, special procedures have been developed at FZJ. These procedures shall be described below.

Unloading of the AVR core and decanting of fuel elements in the hot cells

During unloading of the AVR core, the fuel elements were stored in so-called AVR cans. Each of these cans has a capacity of 50 pebbles. They are made from stainless steel and closed by means of a plug. The AVR cans were transported to the hot cell laboratory, where they are stored in a water pool serving as a buffer storage (Fig. 1). The maximum capacity of this pool is 68 300 fuel elements corresponding to 1365 cans.

From the water pool, the cans are transported to a hot cell where they are opened and unloaded. When the plug is removed from the can, special attention is paid for indications of water penetration. Such a water penetration may have happened due to leaky sealings during the preceding storage in the pool. In case of indications of water penetration the concerned fuel elements are treated in a special way (s. below).

Dry pebbles are decanted in the hot cell into so called "dry storage cans" made from stainless steel (Fig. 2). A plug with a Viton elastomer sealing closes these cans. Each can has a capacity of 950 fuel elements.

After transportation to the waste cell laboratory, the dry storage cans are placed in a large storage cell with a capacity of 140 storage cans. Here they are placed in transport/storage casks (CASTOR-THTR/AVR), made of nodular cast iron (fig. 3). One CASTOR-THTR/AVR cask can take up two dry

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storage cans, corresponding to 1900 fuel elements. Since 1997 the license has been extended to 140 additional cans.

Treatment of wet fuel elements

As mentioned above, some of the fuel elements took up moisture during storage of the AVR cans in the water pool of the hot cell laboratory. These wet fuel elements are identified by visual inspection and weight control. After separating them from the dry ones, they are filled in separate dry storage cans that are closed leak tight by welding (Fig. 4). This welding is performed in a hot cell with the welding head and the dry storage can inside and the welding aggregate outside. The leak tightness is required for two reasons:

- moisture in the CASTOR AVR cask may cause corrosion on the unprotected cast iron walls and the sealing surfaces,
- avoidance of gaseous radio nuclides (H-3, Kr-85, C-14).

In order to guarantee leak tightness after welding, the weld is tested by means of a helium leakage test. But in order to determine the helium leakage rate for a welded dry storage can, some preparatory measures have to be taken. First of all, part of the air volume in the can has to be exchanged against helium. This is carried out by means of a so-called un-plugging system. The un-plugging apparatus is connected in a gas-tight way to the neck of the dry storage can. After unplugging, the can is flushed by helium, while 20% of the air from the can (24 l approximately) is stored in a flask from where it can be analysed for H-3, C-14 and Kr-85. The dry storage can is filled with helium at 1 bar. Next the plug with a Viton sealing is exchanged against a plug with an O-ring made from silicone. This exchange of sealings is carried out because the high density of the Viton ring would make helium leak testing after welding impossible. Due to the higher permeability of the silicone ring for helium, short time after welding a concentration of helium is accumulated below the weld which is high enough to allow detection of leakage rates of $< 1 \times 10^{-5}$ mbar l/s.

After welding and leak testing, two dry storage cans are placed in a Castor AVR cask. These casks are transported to the AVR interim storage site.

Loading of the CASTOR THTR/AVR Cask

The loading of the CASTOR THTR/AVR cask is carried out in the waste-cell area of the interim storage site. Each cask is filled with two dry storage cans (Fig. 5). The cask is closed by a double lid system. Before loading, the sealing surfaces of the lids are inspected and prepared outside the cell. After inspection of the sealing surfaces the primary lid is placed on the empty cask, which is moved into the loading cell.

Here the primary lid of the cask is removed by means of a mounting tool. In order to guarantee the correct replacement of the lid, all positions are controlled by a laser equipment. During the loading process of the cask with the two dry storage cans, segmented cover plates protect the sealing surfaces.

After replacement of the primary lid, the door of the hot cell can be opened, and a first in-situ radiation protection measurement is carried out. Now four screws of the primary-lid are fixed by hand, and after another surface radiation measurement the cask is driven out of the cell for further treatment.

Now, the other screws of the primary-lid are installed and tightened by means of a torque key. Helium leak testing controls the leak tightness of the primary-lid and the result is documented. In order to control the pressure between the two lids, a pressure gauge is mounted to the secondary lid.

After installation of the secondary lid, its leak tightness is also tested with helium, and the function of the pressure switch is checked. Finally the CASTOR THTR/AVR cask is transported to the AVR interim storage (Fig. 6), which is located in the same building.

Release of gaseous radionuclides from fuel elements

Complementing the storage of AVR fuel, a measuring program was initiated to demonstrate the safety of dry storage of spent HTR fuel elements and to provide data for licensing commercial storage casks.

With respect to interim storage of graphitic fuel elements of up to 50 years, only few gaseous radionuclides have to be considered. These are H-3 (as HT, HTO or T₂), C-14 (as ¹⁴CO₂) and Kr-85. Different from final storage conditions, Ra-226 has not to be considered under interim storage conditions. The main sources for tritium release under storage conditions are (p-n)-reactions with He-3 isotopes of the cooling gas and (n-α)-reactions with Li-6 pollutions in graphite. Tritium from ternary fissions is well included in the coated particles. 95% of the tritium inventory exists in the form of HTO. Each so-called dry fuel element is able to take up 50 mg of water; for wet fuel elements this value may rise up to several grams. This water may exchange its hydrogen atom with HT or T₂ gas and HTO is generated. If the water content in the cans is high enough (fluid water instead of moisture), radiolysis gas may be generated. Overpressures of 200 mbar were measured in closed cans.

Due to the high cross section for (n-p)-reactions in nitrogen, the main contribution of the C-14 generation comes from the air contact of graphite during the fuel element production. The nitrogen penetrates into the pores of the graphite and is not released at the high operating temperatures of the fuel elements. Hence C-14 is mainly found in the surface areas of the fuel elements. It can be released as CO₂ during oxidation processes forced by radiation.

Kr-85 is generated as a pure fission product. This means the inventory of a fuel element is dependent on the burn-up and the type of fuel. Most of the Kr-85 is included in the coated particles only a small amount is generated from heavy metal pollutions in the matrix material and from particular fuel particles with damaged coatings (< 10⁻⁴ of the total inventory).

In order to investigate the release of gaseous nuclides, two dry storage cans were quipped with valves (Fig. 7) and connected to a flushing gas circuit (Fig. 8). In a closed circuit, the gas is pumped through a pre-heated sampler. The pre-heating to 30°C is required to avoid condensation of HTO. For analysis, 100 ml were taken out of the total gas volume of 120 l and analysed by means of a proportional counter that was able to separate low energy (H-3) and high-energy beta radiation. Kr-85 and ¹⁴CO₂ were separated by means of a trap filled with soda lime as a CO₂ absorber.

Additional gamma- and neutron dose rate measurements at the surface of the transport-/storage casks were carried out after 1, 6 and 10 years. After one year the main contribution of the γ-dose is due to Ce¹⁴⁴Pr with a half life time of 285 d and photon energies of > 1.4 MeV. This means fuel elements with a low burn-up and a high fuel inventory require more screening efforts. After longer ease off time CS-137 (T_{1/2} = 30 a) gives the main contribution to the irradiation, and fuel elements with low burn-up require less screening.

Neutron irradiation is mainly due to Pu-238 and increases more than proportionally with the burn-up.

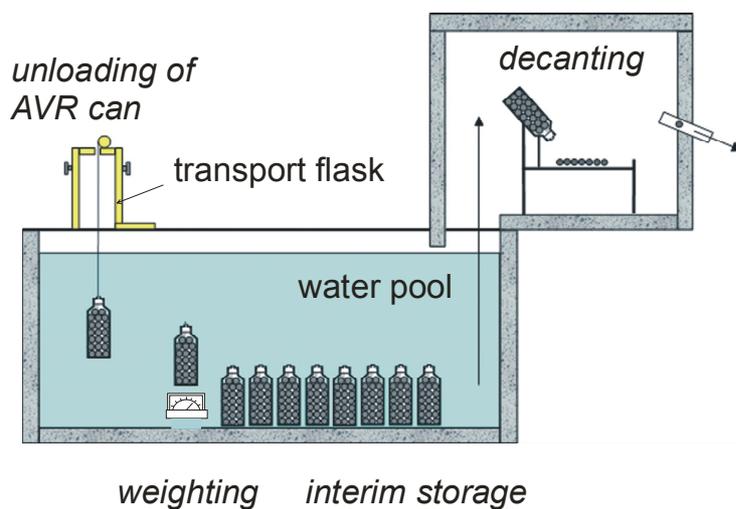


Fig. 1 Decanting station

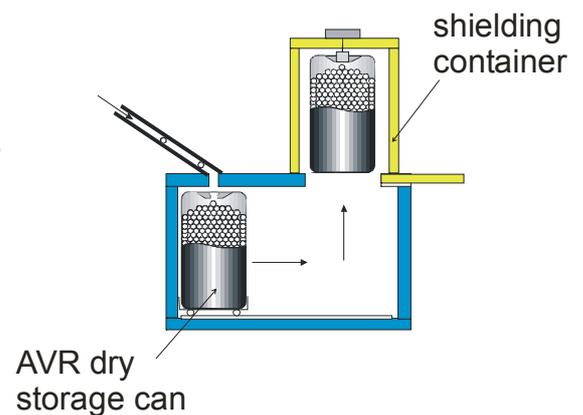


Fig. 2 Filling of dry storage cans



Fig. 3 CASTOR-AVR transport and storage cask

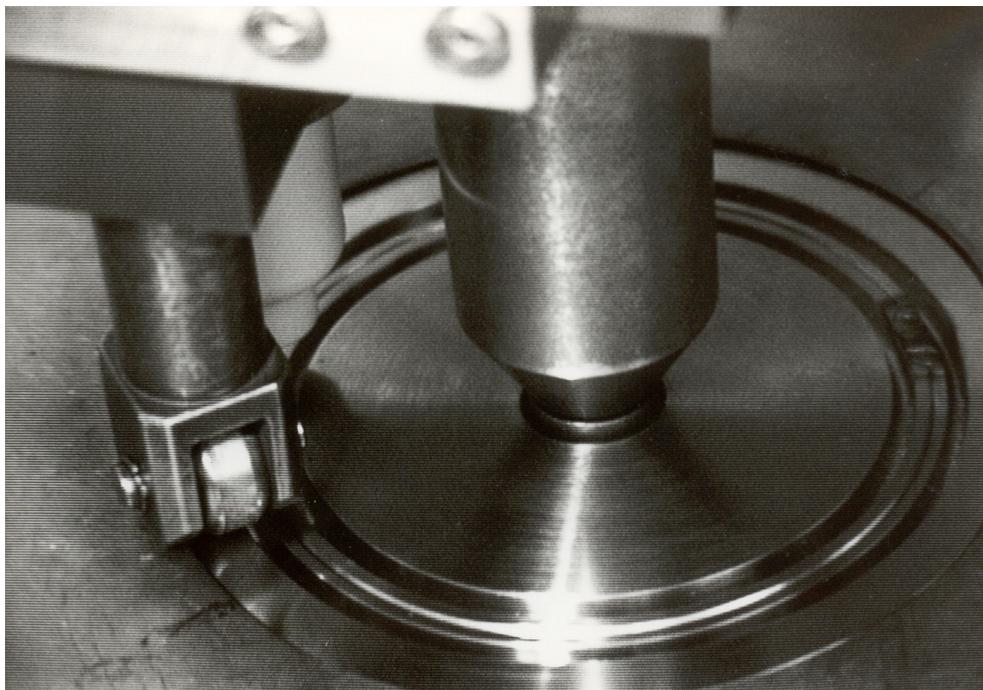


Fig. 4 Welding machine for dry storage cans

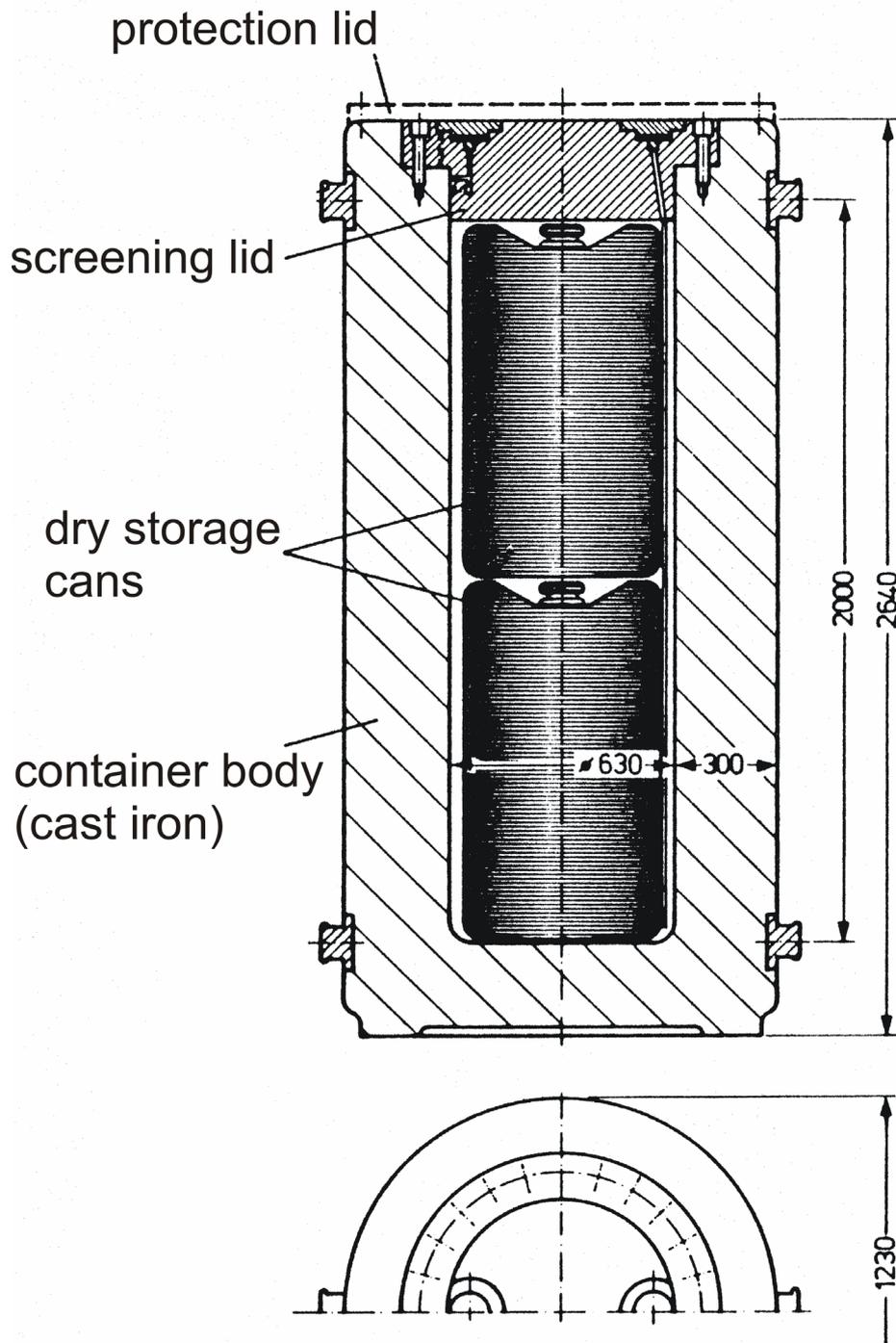


Fig. 5 CASTOR-THTR/AVR transport and storage cask (schematic diagram)



Fig. 6 AVR interim storage plant



Fig. 7 AVR dry storage can equipped with valves

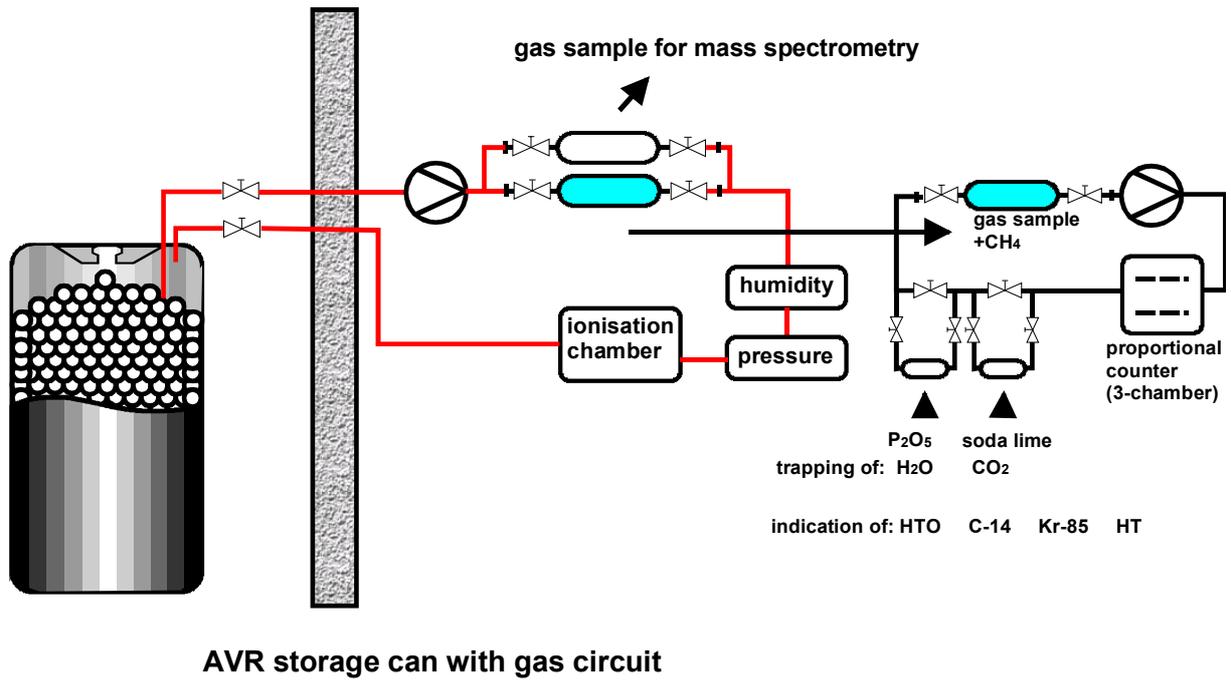


Fig. 8 AVR dry storage can with gas circuit