Fuel Activities Related to ADS development at the SCK·CEN: (application to MYRRHA)

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Abstract

In the framework of the MYRRHA project, devoted to the design of a multipurpose experimental accelerator-driven system (ADS), an R&D program for ADS fuel development, studies, qualification and management is under preparation at SCK·CEN. Different kinds of fuels are expected to be used and tested in the MYRRHA ADS. MOX fuel with *20-30 % Pu* will be used as a basic fuel for the subcritical core. Some targets with minor actinides (MA) will be irradiated close to the spallation target, to study the MA-incineration in the ADS. Furthermore, the fuel of commercial nuclear reactors can be tested in the reflector zone. Finally, some experimental assemblies with uranium-free fuels containing MA could be loaded in the subcritical core as well. The development, qualification, handling and management of these fuels pose a lot of challenges that should be resolved before starting the ADS construction. The needed techniques, tools, infrastructure and hot-cell capabilities should be foreseen and developed in the meantime. This paper provides an overview of the related activities and achievements at SCK•CEN in collaboration with Belgonucleaire.

1. Introduction

About 2,300 tons of spent fuel are produced every year only in the OECD countries and few tens thousand tons of nuclear waste are now in intermediate storages around the world. Through partitioning and transmutation (P&T) of the transuranium (TRU) actinides and some long-live fission products (LLFP), the radiotoxicity of the high-level waste (HLW) going to geological disposal can be reduced by a factor of 100 compared with the current once-through approach [1]. Different strategies are under consideration in different countries for incineration of TRU and transmutation of LLFP. Using accelerator driven systems (ADS) is considered to be one of the effective and safe solutions. Interest to the development of such ADS is continuing to increase. Projects for development of a demonstration ADS are under study over the world [2-4]. The first step to a prototype DEMO-ADS-burner is the design of a small experimental ADS. The Belgian Nuclear Research Centre SCK·CEN is now developing such type of ADS which is named MYRRHA (**M**ultipurpose h**Y**brid **R**esearch **R**eactor for **H**igh-tech **A**pplication) [4]. This paper presents the framework and the actual status of the MYRRHA project, its concept and main challenges related to the MYRRHA fuel development and handling.

2. MYRRHA - a multipurpose experimental ADS

The applications considered presently in the MYRRHA project focus primarily on the ADS concept demonstration, waste transmutation exploration, radioisotope production and safety research on sub-critical systems. The MYRRHA concept, as it is today, is based on the coupling of a proton accelerator with a spallation target surrounded by a subcritical neutron-multiplying medium. The protons hit a liquid Pb-Bi spallation target and produce the neutrons needed to sustain the chain reaction in the subcritical core surrounding the spallation target (Fig. 1.).

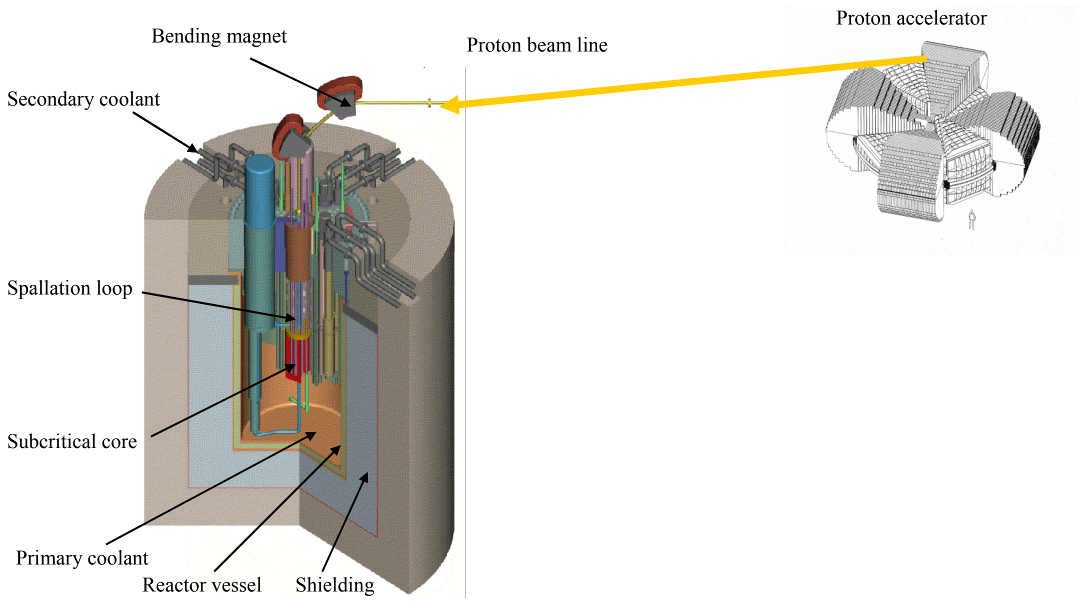
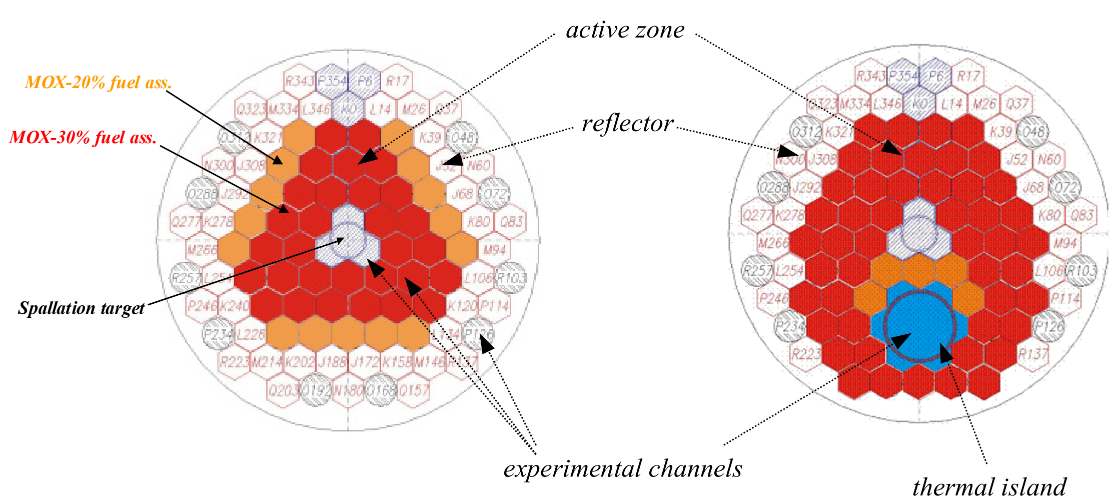


Fig. 1. General schematic view of the MYRRHA ADS [1].

In the framework of the MYRRHA project, the pre-design stage is currently being finalised and the associated R&D programs, to assess the most challenging issues of the present design, are in progress or under development.

To meet the aimed research goals and to assure a large experimental flexibility, four different neutron spectrum zones – thermal, resonant, fast and quasi-spallation – are foreseen in the different configurations of the current core design. In the basic design, the central annular fast spectrum zone is composed of the hexagonal assemblies of the fuel pins with MOX containing 20-30 wt.% of the plutonium dioxide (Fig. 2 a). This zone is very suitable for reactor material testing and transmutation studies due to the high fast neutron flux attainable there (~10^{15} n/cm²s). The thermal spectrum zone represents an inserted in-pile section at the core periphery, which is named in the relevant technical documents "thermal island" and can be realised by using a special core configuration (Fig. 2 b). This thermal island will enable transmutation experiments with LLFP and MA, irradiation experiments with fuel of LWR, the production of radioisotopes.



(a) (b)

Fig. 2. Typical core configurations of the MYRRHA ADS:
a) basic configuration; b) configuration with the thermal island [5].

Whereas MOX fuel is expected to be used as driver fuel in the subcritical core at the first stage of the MYRRHA operation, the different types of experimental fuel rods containing TRU and LLFP will be tested in the experimental channels disposed nearby the spallation target, in the fast core, in the thermal island and in the reflector.

3. MYRRHA fuel R&D program

One of the critical issues is the subcritical core design and related research and development (R&D) of both the driver fuel elements and the experimental fuel target-rods with TRU or with LLFP. Significant (R&D) efforts will be required to optimise the current fuel design and to demonstrate its robustness under all expected operation conditions. The current fuel R&D program has the following objectives to be addressed:

- optimisation of the fuel element design;
- modelling of the long term and accidental fuel behaviour;
- qualification of the core materials and components;
- fuel supply, fabrication, and licensing.

<u>Optimisation of the fuel design</u>. In order to accelerate the fuel pin design procedure, the fuel fabrication process and the licensing procedure we started with existing fuel designs for fast reactor cores such as SuperPhenix (SPX) [6], except that a low-swelling martensitic stainless steel T91 was chosen as the cladding material, which has good mechanic parameters and corrosion resistance in LBE environment [7]. The preliminary design of the fuel assembly is presented in Fig. 3.

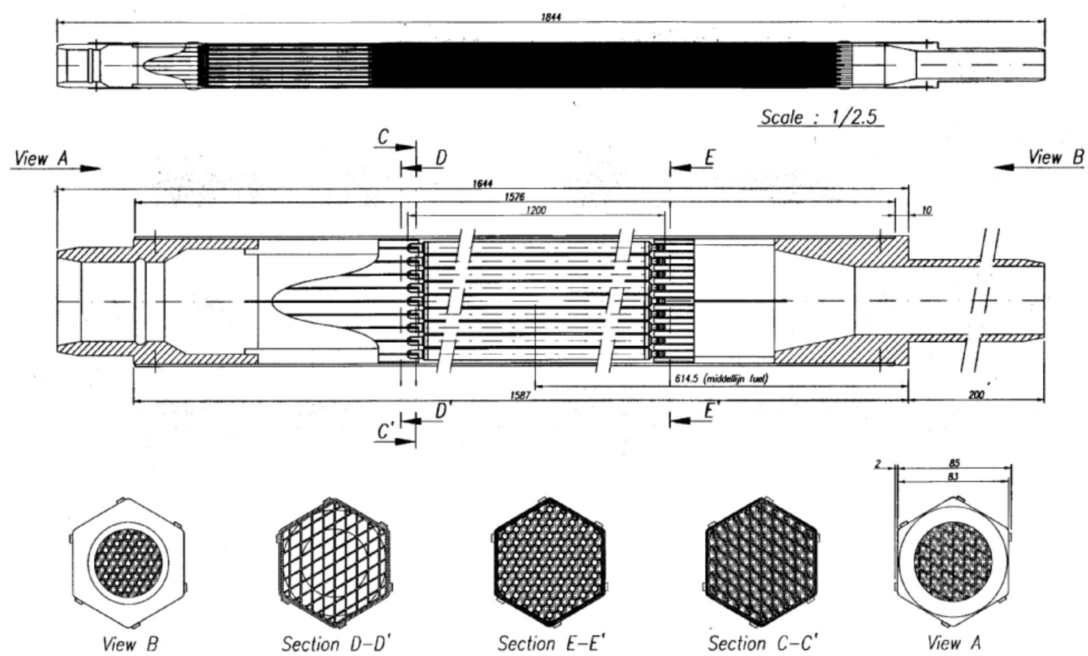


Fig. 3. Preliminary design of the MYRRHA fuel assembly [8].

Although SPX type fuel has been chosen as starting point in the preliminary fuel design, continuous performance assessments are being performed to optimise all parameters of the fuel pin (Pu-vector, enrichment, pellet density and dimensions, cladding diameter and thickness, gas plenum dimensions, reflector segments, ...) and fuel assembly (dimensions, grids design, coolant distribution and mixing, fixing the assembly in the support plate, assembly handling), aiming at better performances in normal operation and resistance to off-normal transients. Alternative designs of a fuel pin (solid pellet, filling with liquid metal) are also not discarded.

The fuel handling from the bottom of the reactor vessel is a special feature of the MYRRHA reactor core. The assemblies are consequently embedded into an upper plate and pushed upwards by the buoyancy force, as the density of the coolant is larger than that of the assembly. This introduces very specific behaviours, if compared with the classical lower plate embedment found in the Fast Breeder Reactors (FBR). For instance, in operation, the net pressure on the wrapper pushes it towards the pin bundle. In FBR the net pressure pushes the wrapper outwards and, due to neutron creep, a significant expansion may raise problems. It is also noticed that in the MYRRHA core, the coolant flow tends to push the assemblies upwards, against the support plate, with an intensity that is even much larger than the buoyancy force acting in the same direction. In a classical FBR, the coolant tends to blow the fuel upwards and a special device is necessary to prevent the assemblies from moving away. Such issues lead to shaping a MYRRHA assembly that is no simple scaling down of an existing FBR assembly, as other degrees of freedom and constraints have to be taken into account.

Modelling of the fuel behaviour under normal and design base accidental (DBA) conditions will be performed before the decision about the final design of the fuel element will be taken. The most important issues to be analysed are the thermo-mechanical and corrosion behaviour of the fuel rod, its resource (life time) determination, resistance to the multiple proton beam trips and prognosis of the allowable operational limits. As a regular reloading and reconfiguration of the core is expected every three months, a very important aspect is the prognosis of the fuel assembly bowing and deflection.
The core coolant chemistry 3D thermal hydraulics and heat removal under typical Loss Of Flow (LOF), Loss Of Heat Sink (LOHS) and Transient OverPower (TOP) accidental conditions will be considered as well.

Qualification of materials and core components. Despite the fact that the SPX fuel was very detailed qualified and a large experience exists in its production and operation, the use of the T91 martensitic steel - in Pb-Bi environment and in presence of the high energy (> 10 MeV) neutrons and protons field close to the spallation target - requires supplementary qualification tests of the core materials and components. Some of these tests are being performed in the framework of the EURATOM FP5 and FP6 EC Programmes (SPIRE, TECLA, TETRA, MEGAPIE), bilateral cooperation between SCK·CEN / ENEA / ITU-FZK, and own SCK·CEN research projects (mainly MYRRHA). Three kinds of clad evaluation experiments will be carried out: irradiation tests without Pb-Bi (or in stagnant Pb-Bi), out-of-pile corrosion experiments, synergy of the exposure to radiation and to Pb-Bi.
After qualification tests and final selection of the core materials, some experimental fuel rods will be prepared and tested in a material testing reactor. A mock-up fuel assembly, containing seven mock-up fuel pins will be tested at out-of-pile conditions in one of the existing large Pb-Bi loops. The preliminary planning for the development of the specifications and programs for these tests and subsequent examinations is under development.

Fuel supply, fabrication and licensing. Several issues have to be resolved concerning fuel production. A preliminary study has started in the framework of a bilateral cooperation between SCK•CEN and Belgonucleaire in order to identify and analyze the possible sources of plutonium, the challenges to be expected at fuel production, the equipment upgrades needed at the fuel manufacture, selection of the fuel production option, assessment of the cost of fuel production, etc. These aspects could not yet be thoroughly examined at this stage of the fuel design. However, some trends and constraints have been identified.
It is clear that plutonium is rather abundant in Western Europe and some of it has a negative economical value. It was first envisaged to use existing highly enriched MOX pins from "orphan assemblies" constructed for reactors that are now dismantled. Unfortunately these assemblies are too long for a small reactor like MYRRHA. In addition, the cladding of these pins is made of stainless steel, which is not the preferred material. It was thus deemed unreasonable to bias the reactor design to just enable using second hand fuel pins.
On the other hand, opening the pins to use the pellets as a raw material is a difficult operation. This requires a specific workshop, the cost of which can hardly be supported by the MYRRHA project alone.
The reference option is now to produce the highly enriched MOX pins from freshly separated civilian plutonium from LWR, featuring low americium content. Appropriate amounts of such a product are physically available in Western Europe, as less than 200 kg of Pu is necessary for a first core. Belgonucleaire's MOX production plant at Dessel is able to produce that kind of high Pu enriched pellets, as its facilities have initially been build to produce high Pu content fast breeder reactor fuel and at first produced part of the SNR MOX fuel core. These pellets however, did not feature a central hole. If such a hole appears to be necessary for MYRRHA - as it was for SuperPhenix - production tests and quality control might require more time and effort. Fixing a wire on the pin, assembling the bundle and

introducing it into a wrapper is not the most difficult part of the job but the know how to perform these tasks has to be reactivated. A practical issue to be taken into account is finding a steel supplier who would accept producing very small high quality batches of T91 tubes and plates.
It is premature to talk about production costs of finished assemblies but it can already be emphasized that the fixed costs will be huge in comparison with the incremental costs.
It might thus be wise to produce one core and several reloads during the first production campaign. Apart form the financial and some materials resource aspects, there is no doubt today about the technical capability of Belgonucleaire to produce highly enriched MOX assemblies for MYRRHA.
Concerning the licensing of the fuel, it is clear that lead test assemblies should be tested in FBR to demonstrate the validity of the design and to benchmark the computer codes that will be used to extrapolate to ADS situations. This is a classical R&D topic that is, above all, time consuming and that could therefore have a major impact on the planning of the project.

4. Fuel handling

An important issue that should be resolved during the design phase of ADS MYRRHA is the scenario and means of the fuel handling.
In the current prelimenary design of MYRRHA the fuel handling system consits of the following components [9]:

- in-vessel fuel handling machine;
- in-vessel spent fuel storage;
- ex-vessel fuel transfer machine;
- cooling device;
- cooling device pool;
- long-term storage pool;
- transfer device (between the cooling device pool and the long-term storage pool).

<u>In-vessel fuel transfer.</u>

The traditional approach of fuel loading from the top is very difficult to apply in the research ADS MYRRHA as due to the presence of the proton beam, coming from above into the reactor centre, and the presence of irradiation devices inserted in penetrations in the reactor cover. Because these last irradiation devices will stay several cycles in the reactor, and because their handling will be an arduous process, it is desirable to keep the irradiation devices into location while reloading the core. This is possible with the fuel-handling machine installed at the periphery of the core with an offset arm for fuel handling from underneath. The spallation loop, running through the centre of the core, makes a fuel assembly difficult to be reached for a single handling machine installed at the opposite side of the reactor. So, this requires a double fuel handling machine system, installed opposite of each other with respect to the core axis.

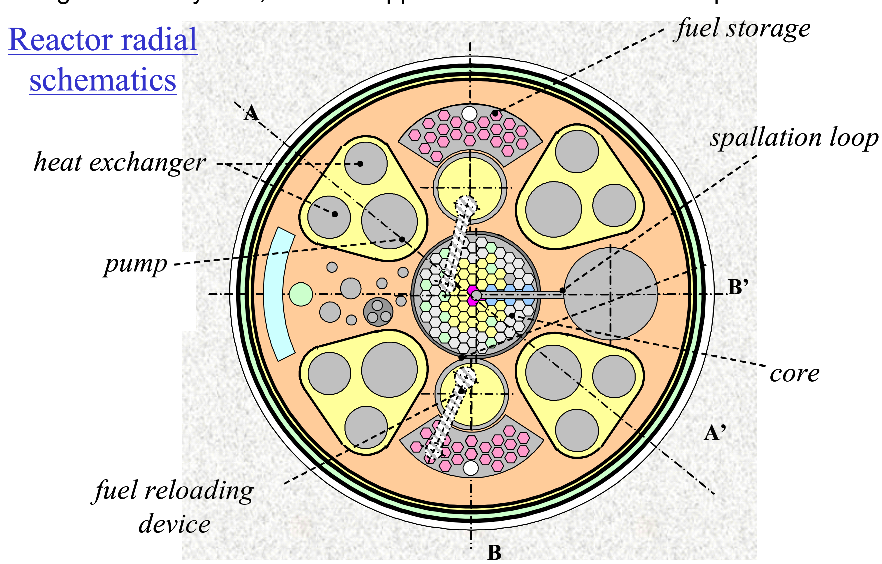


Fig. 4. Reactor radial schematics [9].

A double fuel handling machine avoids the placement of an elbow in the offset arm. Such an elbow would prevent the direct actuation of the gripper from the outside and would require sensors and actuators installed close to the gripper, which will be a source of difficulty in operation. Therefore there are two identical machines on both sides of the core, inserted at 180° of each other and at 90° to the spallation loop (Fig. 4).

The task of the in-vessel fuel handling machines is to take fuel assemblies from one position inside the vessel to another position, more or less in the same horizontal plane. Because of the presence of the spallation loop, there is no direct connection between the two halves of the reactor. Fig. 5 shows the axial schematic and the general working principle of the system.

A fuel handling machine is made of a rotating plug, inserted in a penetration of the reactor cover. The rotating plug has a penetration at its periphery where the handling arm is inserted. The rotating plug is part of the reactor shielding and has therefore the same thickness as the cover.

There are basically three possible types of positions: the subcritical core, the in-vessel fuel storage (where fuel assemblies are put in a waiting position before being loaded into the core or unloaded from the core), the loading/unloading stations (where the fuel assemblies are handed over by the in-vessel to the ex-vessel fuel handling machines and vice-versa).

The handling arm is split into two components: the vertical main handling arm and the horizontal offset arm. The main handling arm passes through the reactor cover down to the bottom of the core. It must be long enough to allow the extraction of a fuel assembly. The offset arm is shorter and bears the gripper at its extremity. Both arms are hollow and contain the actuation rods for the gripper.

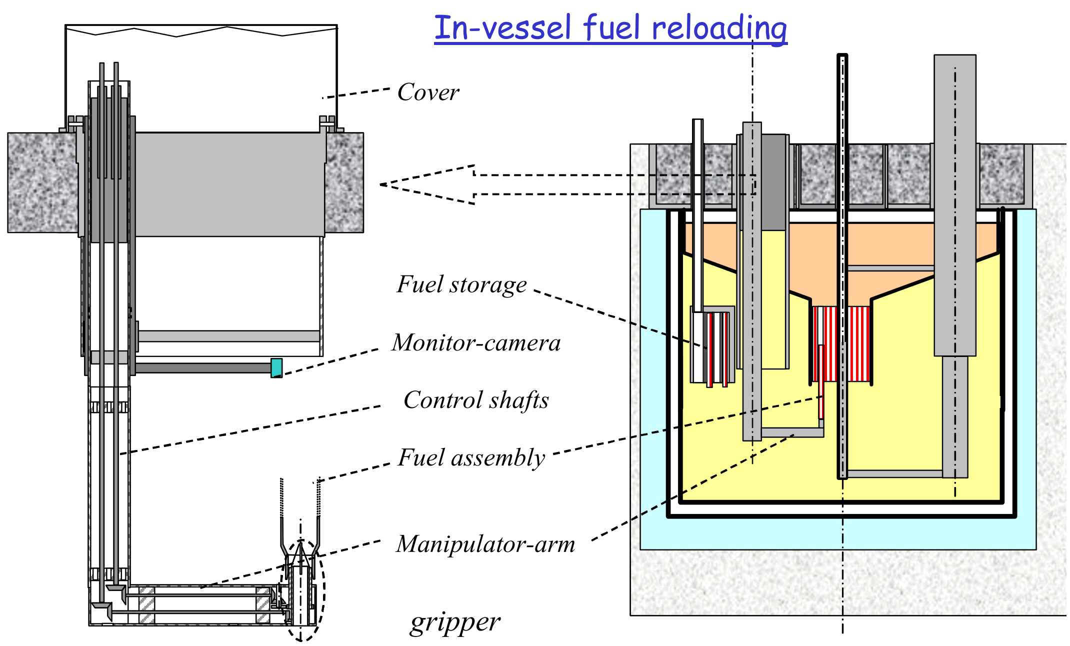


Fig. 5. In-vessel fuel handling [9].

The gripper is an annular piece at the extremity of the offset handling arm, to be inserted in the lower nozzle of a fuel assembly. The gripper is hollow, to avoid any cooling interruption during the operations – at least by natural convection. It allows to catch a fuel assemble when properly inserted in its lower nozzle. The gripper can also rotate the fuel assembly to the correct orientation, before insertion in the core. The amplitude of the rotation should allow turning the assembly by 180° or more. The correct orientation of the fuel assembly is verified by a notch of adequate depth in the nozzle, in which a pin of the gripper must insert. Once the assembly is positioned, the gripper can be disengaged without pulling it.

The actual relative positions of the gripper and the nozzle, and the correct orientation of the latter will be detected by an ultrasonic camera installed on an auxiliary arm, mounted on the main handling arm. The purpose of the camera is to locate the actual position of the lower nozzles of the fuel assemblies before grabbing them.

The fuel handling system presents may different leak paths for the activity inside the reactor to be released outside. Therefore, it is found convenient to install the whole machinery under a cover. It will prevent possible leaks to contaminate the reactor hall. Of course, the cover will have to be removed before any maintenance work can be undertaken.

All motions during the fuel handling are electrically actuated to facilitate flexible connections. The position of the arm is measured by sensors placed above the reactor cover, to keep them out of radiation fields and allow “easy” periodic maintenance. Those are angular and linear displacement sensors. In addition, force feedback, under the form of strain gages or similar, should detect an interference of the arm with the structures or other anomalies such as the inadequate insertion of the gripper in the fuel assembly nozzle.

Periodic maintenance is facilitated by the relative accessibility of the actuators and of the sensors. Nevertheless, interventions on the arms or on the grippers might be necessary. Those components are permanently submerged in the primary PbBi and are exposed to the neutron flux. Hence, they will be activated to some extent – and on top of that Po contaminated - and their maintenance can only be performed in a shielded and isolated environment. The plug and its handling machine can be removed from the reactor as a single assembly. The machine can be removed from the plug if all actuators and systems of the upper stage are removed beforehand. It is quite clear that this will be a laborious operation. A spare handling arm should be kept in reserve to avoid the long shutdown caused by the unavailability of such an important component.

Ex-vessel fuel transfer.

The fuel is taken in the reactor vessel, at a temperature of at least 150 °C, transferred out of the reactor in a cooling machine, in which the fuel is extracted from the PbBi and cooled down to a temperature at which it can be stored in a water pool. The concept presented below provides the following features [9]:
a fuel assembly is always under liquid during the whole process;
active cooling of the assembly can be maintained during the whole process;
PbBi and the structures are in contact only with demineralised water - no intermediate liquid - so that there is almost no contamination of the primary coolant;
a fuel assembly is never dropped in an intermediate position and remains constantly attached to the transfer machine until it is released to water;
the transfer and cooling are performed in a closed vessel, so that, in case of leaking fuel element, or failure of the cladding during the transfer, the amount of activity release to the reactor hall is limited;
the transfer is being made in a bucket filled with PbBi, providing some shielding against gamma radiations emitted by the assembly (depending on the amount of PbBi present in the bucket, of course). Interventions on the transfer machine could be envisaged if the machine is blocked during a transfer;
because of the transfer under liquid and the active cooling, a transfer could be envisaged very shortly after a reactor stop. This is particularly important if irradiation targets are to be recovered rapidly or must continue their irradiation whereas the fuel assembly has to be unloaded.

The fuel transfer machine can handle one assembly at a time. But it could be used during the operation of the reactor. A dual-assembly concept is proposed, which needs a larger vessel penetration and a larger cooling machine. Two identical machines are needed (transfer and cooling).

The ex-vessel fuel transfer machine is a device mounted on a trolley (Fig. 6). The trolley travels from a position just above the unloading position of the reactor to the cooling machine pool.

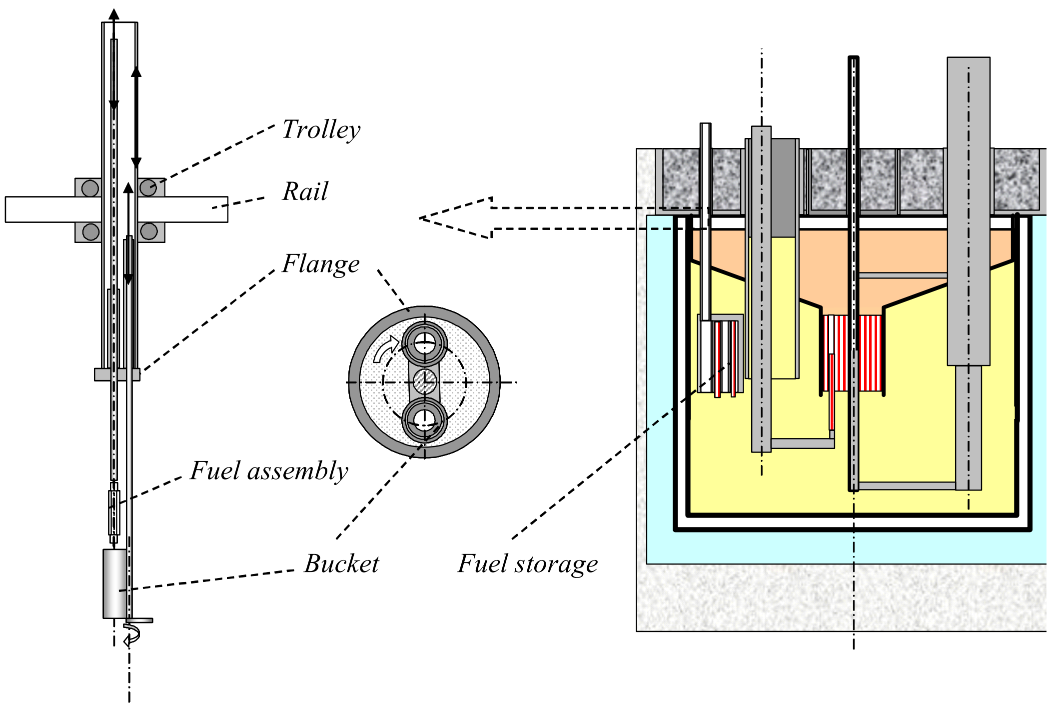


Fig. 6. Fuel withdrawal from the in-vessel storage [9].

The trolley supports three systems able to travel vertically, independently of each other:

1) the flange support tube - this flange is close the reactor vessel during most of the unloading process - it has two tight penetrations in which the two other systems are inserted;
2) the grabbing tool, i.e. a long tube, able to pick up the assembly deep in the reactor vessel and bring it back close to the flange;
3) the bucket, fitted with a cooling coil to prevent overheating of the assembly during the transfer or in the event the transfer is blocked half-way for any reason - it has also a vertical motion, plus the possibility to rotate by 180°.

After positioning of the fuel transfer machine at the unloading position of the reactor vessel, the flange is lowered to fill the penetration of the vessel. The bucket and the grabbing tool are further lowered through the thick reactor cover. At the adequate depth (corresponding to at least one fuel length), the bucket bottom is rotated by 180°, to give way to the grabbing tool. The grabbing tool is lowered to the position at which it can grab the fuel assembly from the in-vessel fuel handling machine. Then, the grabbing tool is raised so that the lower end of the assembly is inside the bucket. The bucket bottom is then turned by 180°, so that the bucket is closed. The bucket can now be raised up to contact with the flange. At that moment, the assembly is in a closed transfer container. The flange is raised and the trolley transfers the system above the cooling device.

Cooling device

The cooling device is a vessel, closed by the flange of the fuel transfer machine. The vessel rests at the bottom of a dedicated pool. The machine is connected to a water cooling loop, operating between 20 and 200 °C.

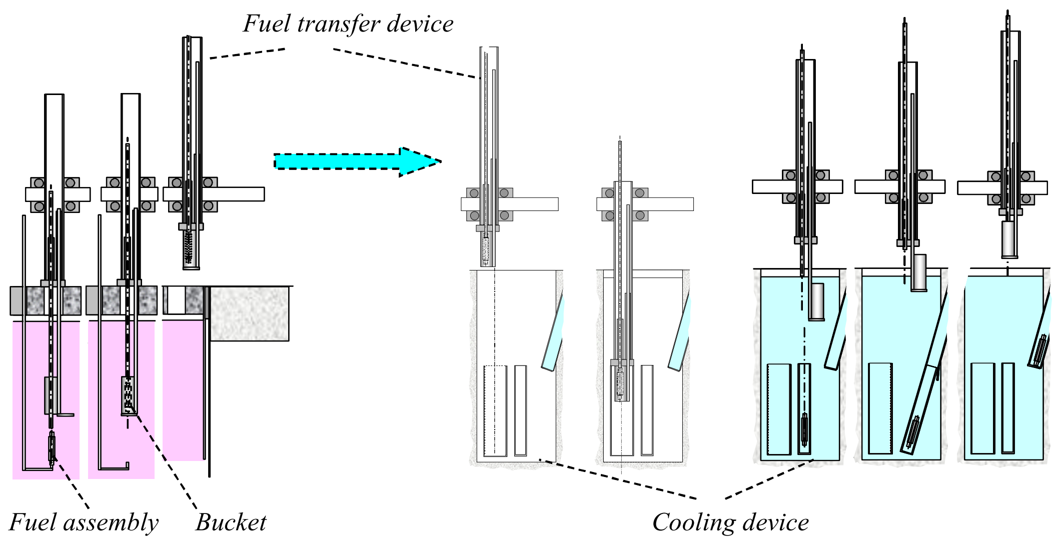


Fig. 7. Fuel removal and transfer [9].

During the start phase of the transfer procedure, the cooling machine and its pool are empty. The transfer machine lowers the flange – and the bucket containing the assembly - and closes the cooling machine (Fig. 7). The bucket is further lowered on the bottom of the machine. After a tightness test, the machine is filled with water at about the same temperature as the PbBi in the bucket. The assembly can then be pulled out of the bucket, in the water, without thermal shock.

The grabbing tool can be fitted with a water connection to flush the assembly, in order to remove the liquid Pb-Bi trapped inside, as much as possible. The cooling of the water can begin, at a controlled rate, down to room temperature. During the process, the PbBi contained in the bucket will freeze, but the assembly is out. Once at room temperature, the pool is filled with water. The pressure is released in the machine and the flange is raised. The assembly is now under water and the rest of the transfer can be made according to water reactor technology.

Then the trolley brings the assembly above the transfer position. The bucket is rotated by 180° to give way to the assembly. The assembly is lowered in the transfer tube and released there. The grabbing tool, the bucket and the flange are retracted and sent to a re-initialisation position. The transfer tube is tilted and put into communication with the long-term storage pool where the assembly is transferred.

Parts wetted by water could be dried by heating above or close to 100°C, before returning to the reactor and bringing possible contamination inside. The bucket should be unfreezed. It must be done out of the reactor if the system must be used for the loading of fresh fuel. If fresh fuel is to be loaded, the fresh fuel must be heated (e.g. by air circulation) before being inserted in the bucket.

The masses involved in the system are heavy, so the motions will have to be slow. As a first estimate, a cycle time of 5-6 hours per fuel assembly was obtained. It must be born in mind that there are two identical machines. To further speed up the process, a double, or possibly triple bucket could be envisaged.

5. Hot-cell needs

ADS related fuel developments ask for hot cell research well beyond the current LWR fuel research. Both, the qualification of the basic driver fuel (20-30 % Pu MOX fuel compatible with LBE environment) and the assessment of the experimental (MA containing) fuel targets/rods/assemblies, call for additional requirements to be fulfilled by the hot laboratories.

- The hot lab should be licensed to deal with high-Pu content fuel and fuel containing minor actinides well beyond the trace amounts as present in the classical fuel for which they are currently licensed.
- The hot cells should be shielded appropriately to cope with the radiation hazards of the high burn up fuels and the highly activated steels and Pb-Bi. The β/γ-shielding of hot-cells that are currently in use for research on (high burn up) classical nuclear fuel obviously is sufficient. However additional specific neutron shielding might be necessary to cope with fuels containing elevated amounts of the neutron emitting Cm-isotopes.
- Additional precautions might be necessary to cope with the high residual heating stemming from the transuranium elements and fission products.
- In view of the high (radio-)toxicity of Pu, Am and Cm (the last one being characterized by an annual limit of intake two orders of magnitude lower than Pu and Am, two elements showing similar already high radiotoxicity) [10], hot cells equipped with leak-tight inner glove boxes will be necessary. Appropriate venting of the hot cells will be necessary as well to cope with the activated Pb-Bi (fuel contamination), exhibiting ~109 Bq/kg, mainly caused by the α-emitter ^{210}Po which is volatile in the form of PoH_2 and PoO. Due to its restricted half-life of 138 days, appropriate cooling times of the fuel before being submitted to hot cell examinations could be envisaged as well.

6. Conclusions

The preliminary design of the multipurpose experimental ADS system MYRRHA is about completed. Its conception incorporates an R&D mission both on the sub-critical system itself and on its application for waste transmutation, radioisotope production and safety research. Optimal driver fuel design is being developed and concurrent R&D programs on the fabrication and qualification of the driver fuel are under preparation. Innovative driver fuel loading/unloading systems have been designed for the typical ADS core environment. The fuel research, needed both for the qualification of the driver fuel and the assessment of the experimental (MA containing) fuel targets, asks for hot cells being able to cope with fuels loaded with elevated amounts of transuranium elements and contaminated with activated Pb-Bi alloy.

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