STATUS OF THE AVR DECOMMISSIONING PROJECT WITH SPECIAL REGARD TO THE INSPECTION OF THE CORE CAVITY FOR RESIDUAL FUEL

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ABSTRACT

The Arbeitsgemeinschaft Versuchsreaktor AVR GmbH at Juelich in Germany is decommissioning the 15 MWe experimental pebble bed high temperature gas-cooled reactor (HTGR) plant, operated from 1967 to 1988. In 1994 Safestore decommissioning started with defueling the about 100,000 sphere-shaped fuel elements. In 1998 began the inspection of the reactor for residual fuel. The major task, a radially directed drilling into the core cavity was very ambitious both mechanically and radiologically. The contractor designed the 280 mm diameter drilling to be carried out in one cut. Stringent protection measures were required since the AVR is heavily contaminated with mainly Sr 90 bound to fine graphite dust. During the task a number of serious problems occured.

The TV inspection result was completely unexpected. By cracks in and shifting of bottom reflector graphite blocks coolant penetration slits have widened so that fuel pebbles could sink into these slits and not roll off into the fuel discharge pipe during defueling. Some pebbles could be removed by the manipulator but most of them still remain.

During the drilling and later from inside the core cavity samples of near-core materials were taken to update the site characterization data.

INTRODUCTION

The Arbeitsgemeinschaft Versuchsreaktor AVR GmbH at Juelich in Germany is decommissioning the 15 MWe experimental pebble bed high temperature gas-cooled reactor (HTGR) plant, one of the most unusual nuclear power plants in the world. The AVR reactor was very successfully operated from 1967 to 1988.

Figure 1 gives a quick overview of the plant: The pebble bed core, the helium circulators and the steam generator are integrated in the inner reactor vessel. Between this and the second reactor vessel the first biological shield (ore granulate) is provided. The reactor is concentrically positioned in the containment vessel. The cylindrical reactor building, a concrete tower of 1.5 m wall thickness, forms the second biological shield. The reactor building is in its lower part surrounded by ring buildings. Reactor auxiliary systems were located in the containment and the ring buildings. Those in the ring buildings contained only little or no radioactivity. The reactor building is connected to the turbine hall.

For more information about design and achievements, former publications like /1/, should be referred to.

In March 1994 Safestore decommissioning started with defueling the about 100,000 sphere-shaped fuel elements (6 cm diameter). After many delays, this task could eventually be terminated in June 1998. However, the fuel is still a major concern.

STATUS OF PROJECT

The core is defueled, and the fuel elements have been transported to the neighboring Juelich Research Center where they are intermediately dry stored in CASTOR casks.

Starting parallel to defueling, a number of other components and systems have been addressed (status Nov. 1999):

- With the exception of the turbine, the turbine hall installations are dismantled. Dismantling outside of the buildings is fully terminated; the cooling towers are demolished.
- The ring buildings have been cleared from all reactor auxiliary systems and been equipped with new service infrastructure to better cope with the coming dismantling tasks in the containment.
- In the containment a large amount of shielding material (bricks, lead) has been removed, the insulation of secondary circuit components been stripped, and the deep-temperature insulation of the gas purification system been sucked off. The 120 reactor vessel pipe penetrations of the steam generator have been cut above the outer vessel and closed by welding.
- A large amount of radioactive waste could be sent to the ERAM final disposal facility before it was closed, and most of the contaminated steel scrap went for melting.

It is planned to dismantle all auxiliary systems in the containment and to seal the outer reactor vessel as the later Safestore border. The Safestore license requires that these tasks cannot start until a close checkup for residual fuel has been performed. These checks comprise the core cavity and all of the fuel handling system. The checkup of the fuel handling system could not be terminated yet, as of Nov. 1999. The major task within these checks, however, the inspection of the core cavity, in combination with sample taking of near-core materials, was successfully terminated. The conduct and findings of this task are so interesting that they are presented here in more detail.

CORE CAVITY INSPECTION

The reflector bottom of the core cavity is funnel shaped, with an inclination of 30°, ending in the 0.5 m wide vertical fuel discharge tube. In that geometrical setting it could simply be assumed that all pebbles must have rolled into the discharge tube and that there was no need for an inspection. The reflector bottom, however, contains many slits for coolant penetration, 3.4 cm wide and in radial orientation. A piece of broken pebble could have stuck in such a slit, and, standing out, one or more pebbles might lay behind it guided by the slit like by a rail. Thus, it was concluded, that whatever inspection tool would be used, it should offer the possibility to give such pebbles a slight push to "derail" them so that they could roll off. Poking-out of stuck pieces was not considered necessary because of their small size.

Possible access to the core cavity
Three possible ways to bring inspection equipment to the core cavity were identified:

- by the pebble feeding pipe system,
- by the fuel discharge tube and
- by a radial bore.

The first two ways would use existing openings, and access into these would have to be made by removing the respective component of the fuel handling system. The pebble feeding pipe system had already been used for an inspection of the top reflector in 1984 /2/, and all the equipment was still with AVR. But only a very narrow and flexible manipulator could have been inserted, and it seemed hardly possible that with such a manipulator, reaching down over the full height of the core cavity (~ 4 m), a pebble could have been touched and derailed.

Access by the fuel discharge tube was also long, but with its much larger diameter sturdy enough equipment could have been pushed upward into the core cavity. Disadvantages here would have been that the tube is tilted at its lower end, that the normal pebble discharge path would have been blocked, and that the workers would have come nearest to eventually found pebbles.

The new creation of a radial access meant a high expenditure but it would offer the best access for both cavity floor inspection and possible pebble management, and the decision for this method was taken. Supportive in the decision was the possibility of taking samples from the various materials encountered during the drilling, and the possible use of already existing openings in the outer reactor vessel in combination with hollow spaces in the first biological shield between the two reactor vessels. Making use of these spaces would, in particular, avoid the necessity to drill through 0.75 m of loose ore granulate filling of the biological shield.

Three different materials would be encountered during the drilling: mild steel, carbon and graphite. Pretests had revealed that the carbon material could be dry cut with acceptable tool lifetime. This also supported the decision.

The largest of the hollow spaces, in which the neutron flux measurement instruments had been located, would have been most suitable as starting point for the drilling, but at that position the carbon insulation (0.5 m tick) and the thermal shield (steel, 150 mm) are interrupted by graphite windows. Hence, no respective samples would have been obtained.

So, finally, the smaller hollow space at about core midheight and a reactor angle of 0°, designed to house a neutron generator, was chosen to start the drilling. Unfortunately the material transport shaft in the containment is located at 0° directly adjacent to the outer reactor vessel, and the drilling equipment had to be installed across the shaft.

Radiological situation

From its operational time, the AVR is heavily contaminated with Sr 90 and Cs 137. While Sr 90 is bound to very fine graphite dust, and is thus very volatile, Cs 137 has entered the ceramic materials graphite and carbon by diffusion and has plated out on metal surfaces and partly also penetrated into these surfaces by diffusion. The opening of the primary system leads to an immediate dust release and opened surfaces show very high β dose rates of ≤ 1 Sv/h.

In terms of γ-radiation, the unshielded borehole would be a major source. During the drilling process, a release of Cs 137 by evaporation could not be ruled out.

AVR has most probably the strongest β-contamination, and in the worst form, of all nuclear installations world wide. This unique "record" has its reason in the reactor's particular operational history: When in 1974 the average coolant outlet temperature was risen from 850°C to 950°C and fuel temperatures in certain areas of the core went near to 1300°C coated particles

of the early fuel element charges with carbidic fuel showed a high release of strontium, barium and europium. Dust filtered out of the coolant showed peak specific activity values for Sr 90 of 2.4 GBq/g-C in 1976 /3/. The carbidic fuel elements had at that time reached high burnups of up to 20 % fima. During reactor operation the pebbles were cycled, measured individually, and those with highest burnup discharged. Many of the high-burnup pebbles had escaped their intended discharge because of the limited precision of the then used burnup measurement device. Interestingly, in post examinations no broken coated particles were found.

Dust was continuously removed from the primary circuit in the coolant purification system. However, only a very small portion of the dust was carried by the coolant. The bulk of it is deposited on surfaces. Thus, this dust in the primary system, henceforth, represented a permanent and virtually undepletable source of serious contamination.

The second major radiological concern when opening the primary system is the release of tritium in form of HTO. The source for the tritium are the ceramic internals. The tritium concentration in the reactor vessel atmosphere strongly depends on the moisture there. Fortunately, this is very low, only of few µbars. Measures would have to be taken to prevent ambient air, and ambient moisture, from entering the vessels.

Protection measures
Measures to protect the workers from dose uptake and the containment from contamination spreading comprised:

- The workers exclusively used ventilated suits.
- A larger glovebox was provided between the driving unit and the outer reactor vessel that tightened on the vessel. In it, the tools were manipulated and the samples handled.
- The inner and outer reactor vessel – both connected – were kept at a reduced pressure of about 2 mbar on the average with plant-own pumping facilities. The reduced pressure extended also into the glove box. However, the box pressure could be treated independently.
- Dry nitrogen could be let into the glove box to keep the moisture in the reactor vessel atmosphere as low as possible. By this measure, the amount of tritium in the gas pumped from the vessels could be kept minimal.
- The working area was enclosed by a ventilated tent. The use of such tents has been a standard procedure at AVR for many years.

The dust and chips created when dry cutting the ceramic material meant a potential fire hazard. To exclude any risk it was regulated that the oxygen content of the reactor vessel atmosphere ought not to exceed 10 %. After the final reactor shut-down the reactor vessels had been depressurized and filled with nitrogen. In the course of the years some air had entered the vessels and at the beginning of the drilling task the oxygen content was about 5 %. Plant-own measurement facilities could be used to check both oxygen and moisture.

During the task the oxygen content indeed rose close to the 10 % limit. In not exceeding it the above mentioned nitrogen supply provision for the box also proved useful.

Conduct of core cavity drilling
The radial drilling being an ambitious task, AVR involved a contractor for both planning and execution. The contract also included the taking of samples (above).

The task was extraordinary from both the mechanical and radiological point of view. The contractor designed the 280 mm diameter drilling to be carried out dry in one cut in both steel and ceramic material. Accordingly heavy and bulky was the machine that had to be installed in rather narrow space. The overall drilling depth was 2.3 m and the following shells had to be penetrated (Fig. 2):

- first biological shield, adaptation of neutron generator hollow space,
- inner shroud of biological shield, steel, 15 mm,
- inner reactor vessel, steel, 40 mm,
- thermal shield, steel, 150 mm,
- reactor shroud, steel, 20 mm,
- carbon insulation, 0.5 m,
- graphite reflector, 0.5 m.

As usual in nuclear technology, the whole task was tested on a mock-up at the contractor's site. Already here, and also later in situ, various unanticipated problems arose that in sum delayed the project for about half a year. At first, the contractor decided widely on his own, and only the overall safety and health physics was provided by AVR. When the various detailed technical problems also led to radiological problems it was mainly by the initiative of the regulatory side that the whole organizational structure was changed. Further-on the contractor had to discuss each detailed step with staff specialists, they had the "last word", and AVR took up full responsibility, also in detail.

The in-situ installation of equipment began in June 1998, and on 6 Jan. 1999 the drilling was completed. In its course, the following two major problems had to be solved:

- When cutting larger the neutron generator space in the biological shield bio-shield granulate emerged and flowed into the box. It turned out that the planning-based drawing did not represent the true situation, and that an outer tube supporting the granulate did not exist. The problem was solved by pressing-in such a tube. This procedure was also pre-tested at a gravel-filled mock-up. (The event had a positive side result: It became clear that the granulate has not coagulated but is still very fluid. Its later removal can easily be done, e.g. by sucking.)
- It was not taken into account, that at the reactor angle of the drilling, 0°, one of the vertical-running pebble feeding pipes to the core is located. It is fixed on the reactor shroud but not so rigid that it could have simply been cut away with the used drilling tool. The centre drill of the tool had already penetrated the pipe, entered the shroud, and broken off there. The problem was eventually solved, after extensive pre-testing, by employing an especially designed cup saw in connection with a kind of bajonett tube designed to sufficiently fix the pebble pipe for the cutting procedure. To be able to cut out a sufficiently long piece of the pipe the actual drilling had been widened preliminarily. The remaining centre drill point, fully immersed in the shroud, was successfully removed also by an especially adapted tool.

The drilling of the ceramic structure went on smoothly without any problems. A liner tube designed to house the manipulator was then pressed into the borehole. This procedure again caused some difficulties, and also the bearing of the manipulator in the liner tube had to be modified so that the manipulator could be handled satisfactorily.

Some <u>health physics records</u>: At the unshielded borehole a γ dose rate of 9 mSv/h was measured, and 1.1 Sv/h in the core cavity centre. In the end, the whole task including the sample-taking, led to a collective whole-body dose of 24.1 mSv (hands 231.6 mSv), for an in-situ occupation of 1900 man-hours.

Results of cavity inspection
After the manipulator holding a TV camera had been inserted, a completely unexpected situation on the cavity floor was revealed: There was still quite a number of pebbles visible that had sunken into various coolant penetration slits of the graphite bottom reflector structure and could in this way not roll off into the fuel discharge tube. The respective slits had widened by cracks in various bottom reflector blocks and also by block shifting (Fig. 3) so that the 60 mm pebbles could fit in.

The whole of the cavity floor was closely inspected, and all in all 77 pebbles were counted. The video pictures suggest, however, that more pebbles are lying in deeper positions. With a simple tool at the front of the manipulator, 33 of the visible pebbles could be made to roll off. The other pebbles are lying too deep to be simply "derailed", and a modified and stronger manipulator would have been necessary. The expenditure for this was seen to be too high by AVR in both time need and additional dose uptake, and it would have been in no way certain that all pebbles could have been removed.

On the other hand, AVR is sure that an according amount of residual fuel in the reactor vessels is neither a problem for safety nor for Safeguards in further Safestore decommisssioning and the later dormancy period, and has asked the licensing authority to extend the Safestore license accordingly. An upper limit of 197 fuel pebbles, respectively fuel pebble equivalent in form of broken pieces, has been evaluated to be residual in the reactor vessels containing an upper limit amount of 98 g of fissile material.

SAMPLING FOR SITE CHARACTERIZATION

An extensive post-operational R & D program on the materials used in the AVR was turned down years ago for cost reasons, and also due to the lack of interest in HTGR technology, and nuclear technology on a whole, in Germany. However, a sound site characterization for the later dismantling of the plant down to green field is indispensable.

For the near-core structures activation calculations have been carried out. The flux history for various locations was assessed by computer model calculation and the activation of relevant nuclides calculated for the unit of concentration (1 ppm) of the according impurities. However, the impurity values themselves are rather uncertain. There is little information in the old records and for the few available inactive material samples the question remains if their impurity levels are sufficiently representative for the built-in materials.

The core cavity drilling, therefore, offered an excellent opportunity to validate the existing data by sample taking (Fig. 4). Furthermore, information on contamination levels and spreading could be obtained.

The techniques for the sample-taking during the drilling process consited in sucking some of the cutting chips and dust created by the drilling tool. The dust was directly sucked into a filter cartridge, whereas the metal chips were first powdered in an electric mill. Later inside the core cavity samples from the graphite reflector surfaces were taken by using an ordinary

drill (12 mm) down to a depth of about 100 mm. Here also, the drilling dust was sucked via a tube line into filter cartridges. In the glove box the sample material was weighted and partially portioned before it was locked out. To avoid too much heat-up of the sample material, the drilling went at very low speed (10 – 20 rpm).

During the drilling, seven samples of the metallic structures were taken and five each in the carbon and graphite. The in-cavity sampling comprised 9 different locations mainly oriented to validate axial flux distribution data. Each sample taken is actually a mixing-together of various sub-samples to increase its representativeness.

The samples are closely analyzed at laboratories outside of AVR. Preliminary measurements, on a whole, confirm the existing theoretical data in their order of magnitude. For single nuclides, however, there are also some larger deviations.
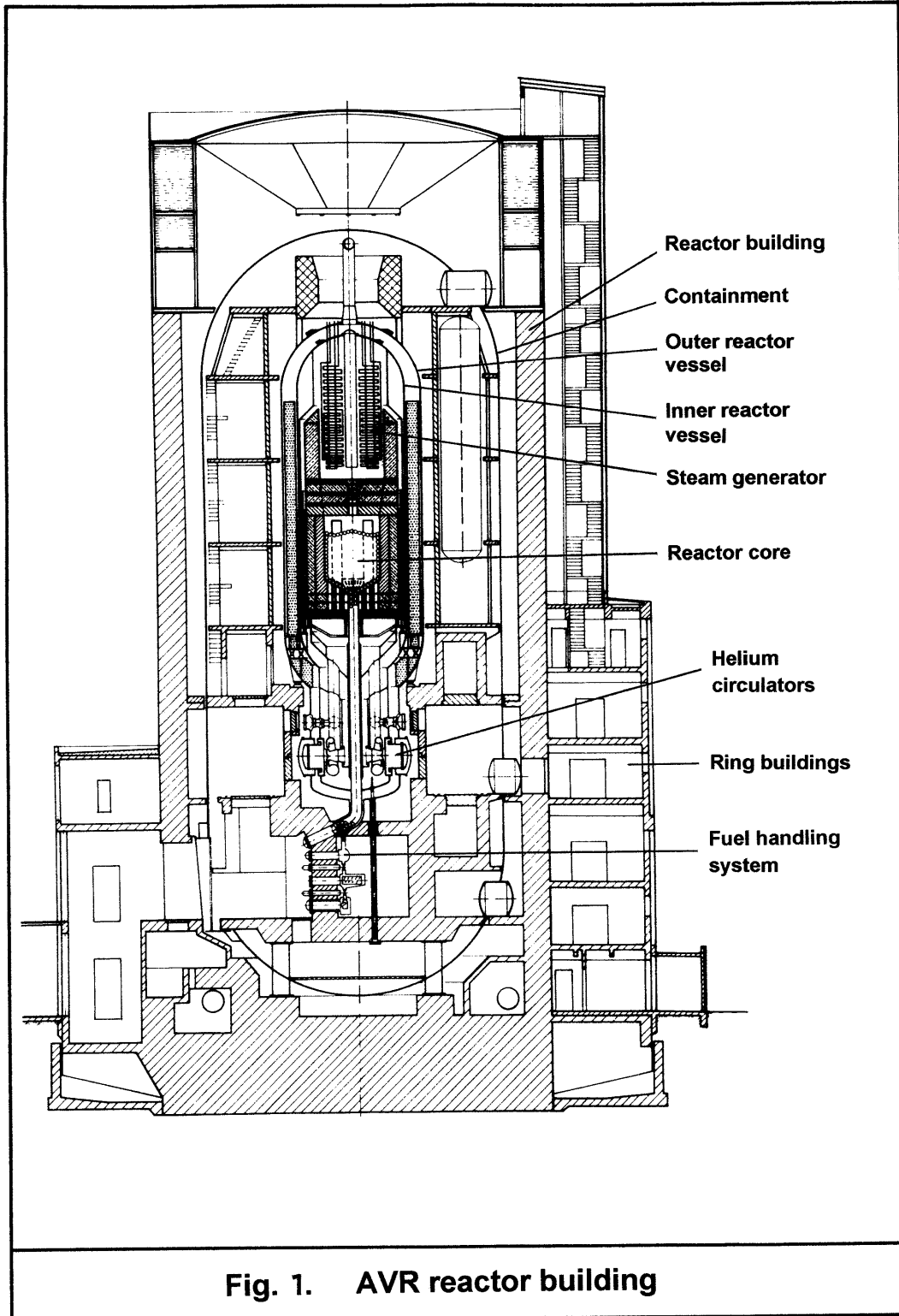
CONCLUSIONS

In most reactor decommissioning projects the fuel is of little or no concern because defueling has already been completed in the operational phase. In old gas-cooled reactors there is sometimes a problem with residual fuel, especially when there was some fuel accident. At AVR there was no accident of that sort. However, the fuel has blown to pieces all decommissioning planning so often optimistically presented at former meetings. The regular defueling took four years instead of the originally scheduled 19 months. And the inspection for residual fuel, thought to be done in half a year, seems rather to consume two. We refrain, therefore, from updating any progress predictions here. If there is one lesson learned so far then maybe this: Concentrate your decommissioning efforts on the on-going critical-path task and be content with any little progress achieved. Do not be nervous about abviously good progress of other "comparable" projects. The odds are not equally distributed, and the final bill is made up in the end.

REFERENCES

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/2/   W. Delle et al., Extended Abstracts 17th Biennial Conference on Carbon, 1985, pages 354 – 355, and Reports of Kernforschungsanlage Juelich GmbH, Juel-Conf.-52, 1985, pages 8 – 14
/3/   Ivens, Wimmers, Operational Experience with HTR Fuel in the AVR Experimental Power Station, IAEA Meeting, Moscow, Oct. 1983

LIST OF FIGURES

Reactor building

Containment

Outer reactor vessel

Inner reactor vessel

Steam generator

Reactor core

Helium circulators

Ring buildings

Fuel handling system

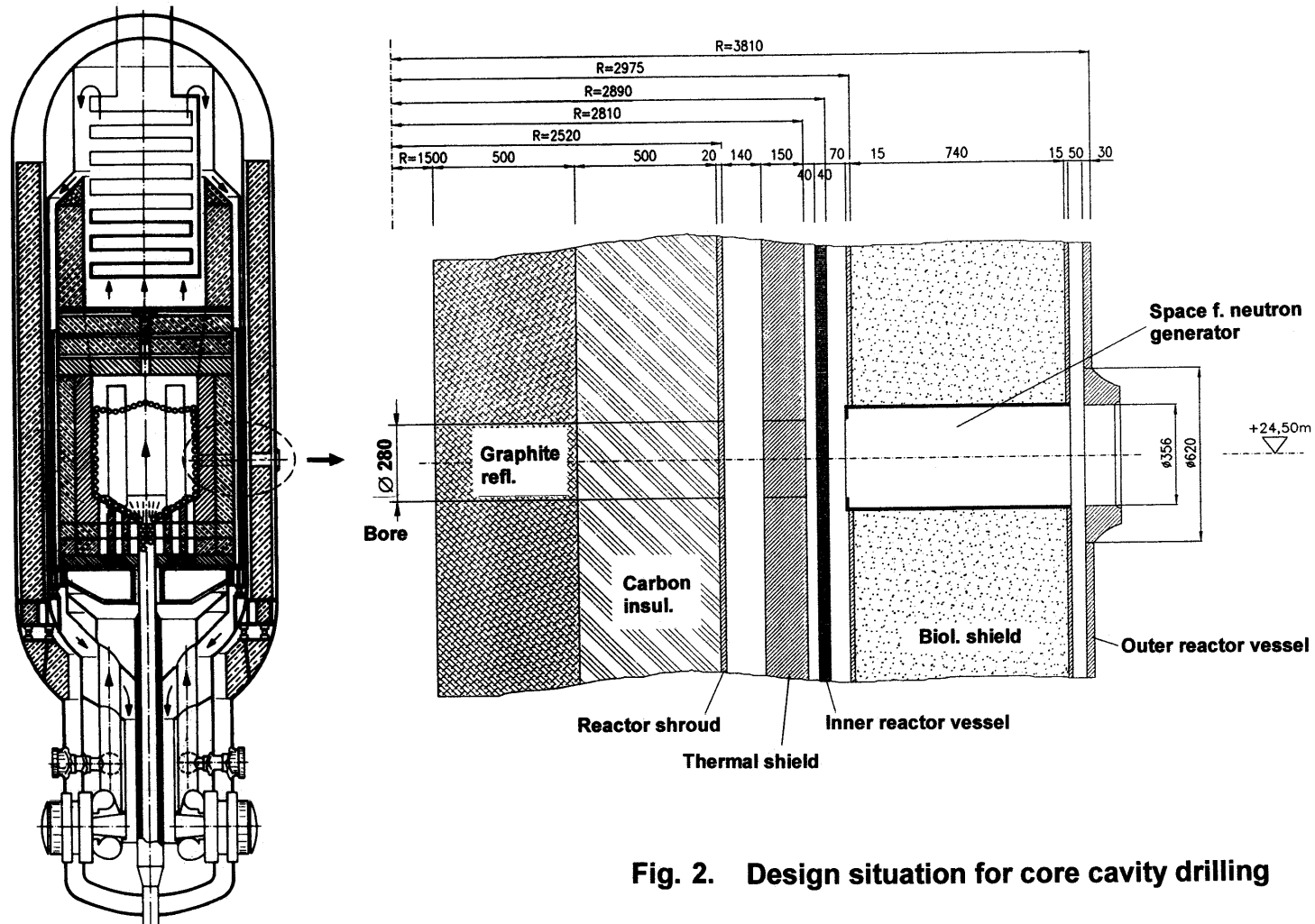Fig. 1.   AVR reactor building

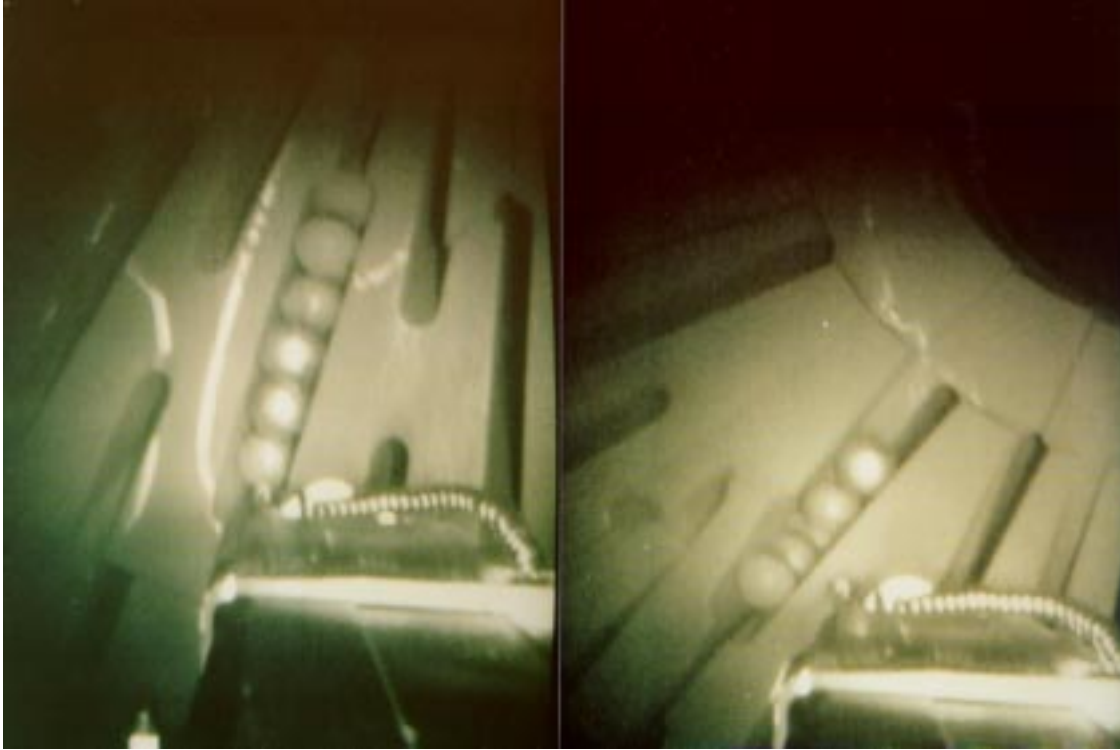**Fig. 2. Design situation for core cavity drilling**

Fig. 3. Cracks and residual fuel pebbles in the AVR bottom reflector (inspection Feb. 1999)

**Fig. 4. Drilling site installation and
locking out of a sample**