Planning for the Decommissioning of the ASTRA-Reactor

Konrad Mück, Jörg Casta
Austrian Research Center Seibersdorf

Introduction

The ASTRA Reactor, a 10 MW multipurpose MTR research reactor at the Austrian Research Center Seibersdorf is to be decommissioned after 39 years of successful operation. The reactor had reached first criticality in September 1960 and was operated at a power level of 100 kW until April 1962. In May 1962 the power level was raised to 1 MW and in August 1962 to 5 MW. In 1969 the power level was further increased to 6 MW and three years later to 7 MW. Since January 1975 the reactor has been operating on a maximum power level of 8 MW and in the last decade of 9.5 MW. The average operating period per year was 500 - 900 MWd/year since 1966. From this rather intensive utilization an activity inventory is to be expected which requires a thorough planning of the decommissioning procedure and stages.

To perform a safe and environmentally compatible decommissioning of the reactor, the possible options and the required stages for decommissioning and removal of the radioactive components were evaluated in a decommissioning study. It was decided to follow the stages proposed by IAEA (1). One of the major questions to be answered in the study was the optimum time scale of reaching each stage and the optimum intervals between each stage.

To support the decisions at each stage, an estimate of the activity inventory in the various parts of the reactor and the waste volume to be expected was performed. Measurements of various materials as far as accessible and numerical evaluations where not accessible were carried out.

Another important issue with regard to dismantling and demolitioning are the methods and procedures to be employed to result in a minimal radiation exposure of the employed staff, minimize the arising amounts of waste and minimize the impact on the environment (2). The choice and deployment of most of these methods is based on long experience at the ASTRA-reactor with the removal of highly-radioactive structures, experimental facilities and beam tubes. As far as foreseeable at present, they will be described.

Characteristics of reactor

The ASTRA-Reactor is a 10 MW multipurpose research reactor. A view of the reactor is given in fig. 1. The reactor was designed for irradiation of samples from the top through the pool water, by irradiation loops and rabbit systems, by irradiation devices adjacent to the core and by a thermal column and beam tubes. Some of the irradiation facilities and devices were installed at a later phase. This was possible due to the high flexibility of the design of the reactor. These later installed features include a fast neutron irradiation facility for seed irradiation, large volume irradiation chambers and irradiation facilities for NTD-silicon production.

Table 1 Relevant parameters for ASTRA-Reactor (3)

type of reactor	adapted pool type reactor, light water cooled and moderated
fuel enrichment fuel type	U_xSi_y-Al 19.85 % MTR fuel element with Al-cladding 23 fuel plates in standard element (412 g ^{235}U) 17 fuel plates in control element (304 g ^{235}U)
reflector	Beryllium and water reflected reflector consisting of Beryllium-elements
thermal power	10 MW
neutron fluence, thermal fast	$1{,}0 \cdot 10^{14}$ n/cm²s $0{,}7 \cdot 10^{14}$ n/cm²s
coolant flow	840 $m^3 h^{-1}$ (velocity: 3.5 m s^{-1})
primary system, volume material	190 m^3 aluminum

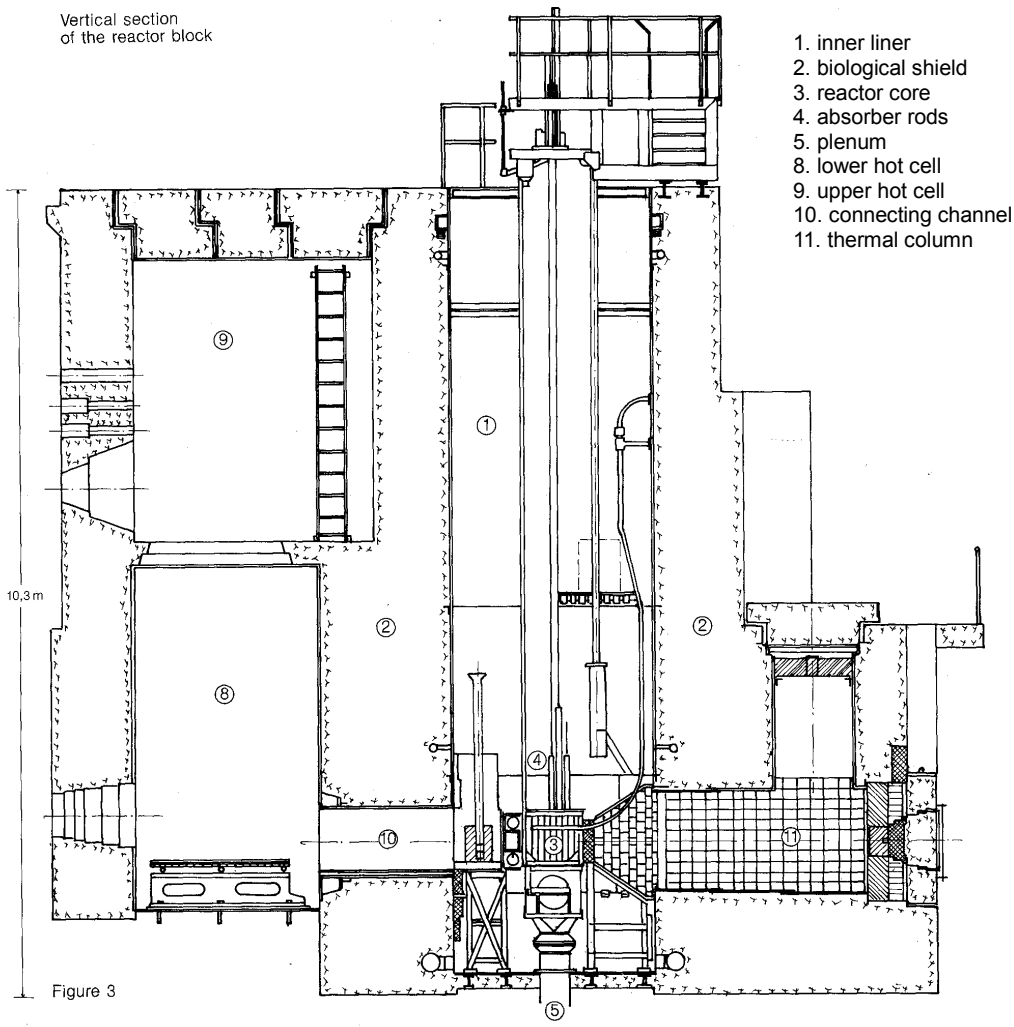

Figure 1 Cross section view of ASTRA reactor

The reactor originally possessed 10 horizontal beam tubes of which 7 were removed 15 years ago and the rest 3 years ago (4,5). From this manipulation of higher radioactive parts (in the range of some TBq) with structure difficult to handle, some of the experience of the staff in manipulating higher radioactive, bulky parts results.

The reactor is a light water moderated and light water cooled, pool-type reactor. Its core consists of typically 20 - 24 MTR fuel elements of high U-content (412 g ^{235}U per fuel element). The neutron balance and thus the burn-up of the core is optimized by a Be-reflector around the core which consists of 25 beryllium elements of similar size as the fuel elements. More details are given in table 1.

ESTIMATE OF ACTIVITY INVENTORY AND WASTE VOLUMES

For an estimate of the total activity content and the waste volumes to be expected an evaluation of the activity and waste volume had to be performed on a component by component basis. Each component, device or structural element was surveyed, its chemical composition estimated and the approximate total neutron fluence at its site in the reactor reconstructed. Of particular importance in this context were trace elements with high neutron activation cross sections. From long-term experience with most of these materials, from activation analysis performed before some of the materials had been deployed in the reactor and from measurements performed in the context of the study in a few cases where no data were available, the activity of these components was estimated. Where no measurements were available, activities were calculated by activation tables as given by Erdtmann et al. (6). This is particularly true for the main chemical elements of a specific device.

For the experimental facilities in the ASTRA-reactor this is given in table 2. Experimental facilities in

this context mean all facilities which had been used for irradiation purposes and are rather easily removable. Due to their utilization in the reactor, they were exposed to the highest irradiation fluences and therefore have the highest specific activities.

Table 2 Estimated activity inventory in the experimental facilities

Radio-nuclide	Half-life	Radiation, γ-energy [MeV]	maximum specific activity [Bq/g]	Material
^{3}H	12.3 y	β, no γ	$1\ 10^{5}$	beryllium
^{14}C	5736 y	β, no γ	2	graphite
^{51}Cr	21.7 d	ε, γ (0.32)	$1\ 10^{10}$	stainless steel
^{46}Sc	83.3 d	β, γ (0.89,1.12)	$2\ 10^{5}$	aluminium
^{54}Mn	312.5 d	β, γ (0.83)	$2\ 10^{7}$	steel
^{55}Fe	2.7 y	ε, no γ	$5\ 10^{9}$	steel
^{59}Fe	44.6 d	β, γ (1.09,1.29)	$2\ 10^{8}$	steel
^{58}Co	70.8 d	ε,β^{+}, γ (0.51,0.81)	$1\ 10^{9}$	nickel, stainless steel
^{60}Co	5.27 y	β, γ (1.17,1.33)	$1\ 10^{8}$	Be, Ni, Al, stainless steel
^{59}Ni	75000 y	ε,β^{+}, no γ	$5\ 10^{6}$	nickel
^{63}Ni	100 y	β, no γ	$6\ 10^{8}$	nickel

The major activity in these components is ^{51}Cr which is activated in stainless steel. With a half-life of 21.7 d, however, its activity is virtually gone one-year after final shutdown. Among the longer-lived radionuclides only ^{60}Co with a maximum specific activity of 1.10^{11} Bq kg^{-1} is of significance. This nuclide is also of relevance with regard to its penetrating gamma-radiation and requires special precaution and protection in dismantling procedures. Another major radionuclide is ^{63}Ni in the nickel-liners of irradiation devices for Si-doping. But since ^{63}Ni is a pure emitter, the major contribution to the exposure of the staff during decommissioning comes also in these irradiation facilities from ^{60}Co.

The amount of waste to be expected from these experimental facilities is listed in table 3. Although efforts had been undertaken throughout the history of the reactor to deploy aluminum wherever possible, a major fraction of this waste is stainless steel. This comes mainly from beam tube experiments. Aluminum parts are not only easier to disintegrate, they also contain usually much less activity. The majority of induced activities have short half-lives and are sufficiently decayed until the begin of dismantling. However, in aluminum also ^{58}Co and ^{60}Co is produced from impurities. Therefore, aluminum from experimental facilities near the core usually contains too much activity to be classified as low-level-waste.

Table 3 Estimated waste amount from experimental facilities

Material	waste volume [kg]
aluminium	2153
graphite	283
lead	180
beryllium	50
nickel	38
stainless steel	2528
boron carbide	2
sum	5234

Table 4 shows the estimated maximum specific activity in the various devices installed in the pool. This comprises the grid plate, the plenum, beam tubes, the thermal column, control rods, the inner liner of the pool and the lead plates to shield the concrete pool wall against gamma radiation during the life-time of the reactor. The major activity contributor is ^{175}Hf with a half-life of 70 days. This activity is to 100 % contained in the Hafnium absorber rods which were used in the reactor in the last 15 years. The rather short half-life of 70 days and the low specific activity of ^{178}Hf results in an easier handling when the core is dismantled after a one-year cooling-down period for the fuel elements. Apart from the absorber rods, the most important activities are ^{51}Cr and the Co-isotopes, similar to the experimental facilities.

Another major radionuclide in that context is the ^{63}Ni in the silicon irradiation facilities and in stainless steel parts of some devices. Not very relevant to the total activity in the waste, but relevant with regard to amount, lead which was used on the outer side of the pool to protect the concrete from heating and irradiation

Table 4 Estimated activity inventory in the devices fix-installed in the pool

Radio-nuclide	Half-life	Radiation γ-energy [MeV]	maximum specific activity [Bq/g]	Material
^{3}H	12.3 y	β, no γ	$1\ 10^{5}$	beryllium
^{14}C	5736 y	β, no γ	2	graphite
^{51}Cr	21.7 d	ε, γ (0.32)	$3\ 10^{9}$	stainless steel
^{46}Sc	83.3 d	β, γ (0.89,1.12)	$2\ 10^{5}$	aluminum
^{54}Mn	312.5 d	ε, γ (0.83)	$5\ 10^{6}$	stainless steel
^{55}Fe	2.7 y	ε, no γ	$2\ 10^{9}$	stainless steel
^{59}Fe	44.6 d	β, γ (1.09,1.29)	$6\ 10^{7}$	stainless steel
^{58}Co	70.8 d	ε,β^{+}, γ (0.51,0.81)	$1\ 10^{9}$	stainless steel
^{60}Co	5.27 y	β, γ (1.17,1.33)	$1\ 10^{8}$	Be, Al, stainless steel
^{59}Ni	75000 y	ε,β^{+}, no γ	$5\ 10^{6}$	stainless steel
^{63}Ni	100 y	β, no γ	$6\ 10^{8}$	stainless steel
^{175}Hf	70 d	β, γ (0.34)	$2\ 10^{10}$	hafnium
^{178}Hf	31 y	γ (0.32,0.57)	$6\ 10^{2}$	hafnium

Table 5 Estimated waste amount from devices fix-installed in the pool

Material	waste volume [kg] activated	waste volume [kg] contaminated
aluminium	2732	3162
graphite	10300	0
lead	10635	0
concrete	6246	0
stainless steel	722	0
cast iron	708	0
paraffin	765	0
glass	0	461
beryllium	130	0
hafnium	18	0
plastics	0	25
boron	17	0
sum [kg]	32273	3648

brittling, contributes to the waste volume (table 5). This and graphite in the thermal column and graphite-irradiation channels are the major contributors to the waste volume from fix-installed devices in the pool.

Another major part of the waste comes from contaminated surfaces in the pool and the primary system of the reactor. These are basically aluminum pipes which show a rather uniform contamination throughout the whole primary system. Measurements by shielded gamma-spectroscopy on-site in which background gamma rays were subtracted, showed a contamination of inner surfaces of the primary system of about 170 Bq cm^{-2}. This consists of 100 Bq cm^{-2} ^{124}Sb, 30 Bq cm^{-2} ^{46}Sc and 40 Bq cm^{-2} ^{60}Co. While ^{46}Sc and ^{124}Sb have rather short half-lives of 83 and 60 days respectively, the half-life of ^{60}Co is too long for adequate decay before removal of the component. After a decay of 14 months (until transport of spent fuel elements) the activity of ^{124}Sb will have decreased to 2.5 Bq cm^{-2} and that of ^{46}Sc to 0.2 Bq cm^{-2}, together less than the limit for laboratory surfaces, but ^{60}Co will have decayed only to 34 Bq cm^{-2}. Solutions for this rather voluminous part of waste (roughly 18 t aluminum) are currently under investigation.

Table 6 Estimated waste amount and surface areas of materials from the primary system

Material	Mass [kg]	Surface [m²]
aluminum	17942	1622.05
stainless steel	8212	905.18
steel and cast iron	4832	82.12
PVC	154	27.63
PE	185	44.80
polyester	22	0.47
sum	31347	2682.25

The second largest fraction of the surface contaminated waste is the heat exchanger built of stainless steel. It contributes about 8 t weight and since its surface amounts to about 34 % of the total inner surface of the primary loop, it is expected to have about 34 % of the surface activity.

The last feature at the end of the decommissioning usually concerns the biological shield. It consists of

baryte concrete and a very dense iron framework. An exact measurement of the activity concentration in the concrete shield and its depth penetration is only possible when all the fuel elements and the major part of radioactive material have been removed from the pool and the pool water drained. However, an estimate of the activity concentration in the baryte concrete was performed by taking small concrete samples of the baryte concrete at the outside of the biological shield and activating them before final shutdown. In this way the main activation products in the concrete were determined.

Results are given in table 7. They represent the data determined 83 days after sample irradiation, characteristic for a shutdown time of 83 days, and after a cooling period of 3 years, a period typical for the removal of the main radioactive components and possible demolition of the concrete block. The derived results correspond to a content of 40 % Ba in the concrete which agrees with typical Ba-concentrations in biological shields. It also

Table 7 Average activity inventory in concrete samples of the biological shield

Radio-nuclide	Half-life	Radiation, γ-energy [MeV]	activity [kBq/g] decay time: 83 days	decay time: 3 years
^{46}Sc	83.8 d	β, γ (0.8, 1.12)	2.2	0.0
^{54}Mn	312.2 d	ε,γ (0.83)	0.2	0.0
^{59}Fe	44.5 d	β, γ (1.09,1.29)	2.9	0.0
^{60}Co	5.272 y	β, γ (1.17, 1.33)	0.3	0.2
^{131}Ba	11.5 d	ε,β^+, γ (0.37, 0.49)	8.7	0.0
^{133}Ba	10.5 y	ε, γ (0.3, 0.36)	2.7	2.2
^{152}Eu	12.7 y	ε,β^+, γ (0.34, 1.41)	0.2	0.1

agrees well with measurements on bore hole samples taken at the FMRB of Braunschweig, a reactor of 1 MW power level (7), when upgraded to 9.5 MW power level and the neutron flux at the inner side of the pool liner is integrated over the life-time of the reactor. From measurements at the FMRB (7) a penetration of thermal neutrons into the concrete shield by an exponential function $e^{-0.05\ d}$ was assumed. This results in a penetration depth of ~ 1 m if a an exemption limit of 500 Bq/kg is assumed.

In table 1 - 6 the waste volume and the maximum activity concentration of each waste category are described. From these data a balance of the average activity concentrations in each waste group and component type was derived. Summing over all these component categories gives the total amount of waste and the approximate activity arising. This was grouped according to the usual waste categories of high-, medium- and low-level waste in table 8.

As known from previous dismantling experience, the by far greatest activity is contained in 300 kg material (spent fuel elements). After the transport of these spent fuel elements to the reprocessing and storage facility, the remaining activity is about two orders of magnitude lower. Also here, almost 100 % of the activity is contained in just about 320 kg of medium-level waste, while the largest part of the waste volume, 99.8 % of the volume, contains only about 6 GBq of activity. The activity concentration in this later part of the waste is in most parts very low, and the amount of waste produced will significantly depend on parameters for unrestricted use given by the licensing body.

Table 8 Waste amount and activity inventory after 1 year decay time

type of waste	amount [kg]	activity [GBq]	relevant nuclide
high-level	308	$1\ 10^7$	^{137}Cs
medium-level	320	$2\ 10^5$	^{60}Co
low-level			
activated	101000	5	^{133}Ba
contaminated	60000	1	^{60}Co

DISMANTLING PROCEDURES AND STRATEGIES

For experimental devices, irradiation capsules and devices in most cases simple cutting and sawing techniques will be applied before conditioning. There is long experience with cutting procedures with higher

radioactive substances in which the exposure of the staff never exceeded very low levels. If required, cutting is performed under water to adequately shield the radioactive components. Also special shielded configurations may be applied, e.g. for cutting of the beam tubes and beam tube inserts a special device for transport and shielding is available by which already in the past several beam tube experiments were removed from their position to a storage site. With the same device beam tubes were cut and removed from the pool later when these irradiation facilities were replaced by irradiation facilities for silicon doping (4,5).

For some experimental devices cutting techniques with a special machine designed for cutting the head and bottom of fuel elements before shipment to reprocessing may be used. As there are several options at the ASTRA-reactor available, the optimum choice will be feasible according to the device to be removed.

By these techniques all experimental and irradiation devices as well as fixed installed systems such as the grid plate, the plenum and the protective lead shields will be removed. For the later a precise cutting design will be decided on at a later stage.

Dismantling procedures for the primary system are simple since the activity levels are low and consist of only surface contamination at levels slightly above limits for laboratory surfaces. Further treatment of the pipes, e.g. surface decontamination or bulk pressing, will be discussed further in the future to minimize the waste.

Procedures and techniques to dismantle the biological shield are evaluated at present. A decision on the optimum procedure will not be taken before the time when such a decision is required. In about three years the decision is pending and the most advanced techniques available at that time (8,9,10) will be applied.

According to IAEA recommendations (1) a three stage process in dismantling is foreseen. Of the possible options an immediate dismantling to stage 1 of IAEA technical guide-lines (storage with surveillance) after the final shipment of spent fuel and complete removal of high-level waste from the site, succeeded by an immediately following, continued dismantling to stage 2 of these guide-lines (restricted site use) was identified as the most reasonable and under present auspices optimum choice. The reasons are that the majority of radionuclides possess either half-lives up to about 80 days which decay sufficiently to permit a continuing dismantling after stage 1, or of half-lives so long that waiting periods of more than 50 years would be required to substantially reduce exposure levels. Since these later nuclides show rather low activity levels, the handling of most contaminated and slightly activated components may be performed without much complication at an early stage.

CONCLUSIONS

To perform an environmentally safe and to the dismantling staff least exposing decommissioning of the reactor, the possible options and the required stages for decommissioning and removal of the radioactive components were evaluated. It was decided to follow the stages proposed by IAEA (1) with virtually no interval between stage 1 and stage 2. Also stage 3 may follow immediately after stage 2 if adequately prepared.

The methods of choice for cutting and removal of the major radioactive components and devices are more or less the same as deployed in the past with the removal of experimental facilities not required anymore. A long-standing experience at the ASTRA-reactor is available for these techniques.

A preliminary evaluation of the arising radioactive waste was performed. The estimates amount to approximately 320 kg of medium-level radioactive waste and about 60 t of contaminated and 100 t of activated low level radioactive waste. The activity inventory is roughly 200 TBq in the medium-level waste and 6 GBq in the low-level waste

REFERENCES

1. IAEA. *Planning and Management for the Decommissioning of Research Reactors and Other Small Nuclear Facilities*, Technical Reports Series Nr. 351, Vienna (1993)
2. IAEA. *Safety in Decommissioning of Research Reactors*, Safety Series Nr.74, Vienna 1986
3. Safety Analysis Report for ASTRA-Reactor, Seibersdorf, Austria (1983)
4. K. Mueck. *Verkürzungen der Strahlrohrkammern am ASTRA* (in German). Report RBS-32 (1985)
5. J. Casta. *Reparatur der Abschlußklappe/Ablauf und Verkürzung der Strahlrohrkammern G und H* (in German). Report RBS-41 of Research Center Seibersdorf (1996)
6. G. Erdtmann. *Neutron Activation Tables*, Verlag Chemie Weinheim, New York (1976)
7. W. Hajek. *Bericht über Kernbohrungen am biologischen Schild des FMRB*, PTB 6.301 (1998)
8. W. Hajek. *Dismantling and Decommissioning Technologies and Techniques for Research Reactors*, 25th anniversary WM99, 28 Feb. - 4 March 1999, Tucson, Arizona
9. H. Kistmacher, H. Haferkamp, D. Seebaum, A. Feld, Fr.-W. Bach. *Einsatzmöglichkeiten des Lasers in der Stillegungstechnik*, Stillegungskolloquium Hannover, Juni 1997
10. IAEA. *Decommissioning Techniques for Research Reactors,* Technical report series No. 373, Vienna (1994)