

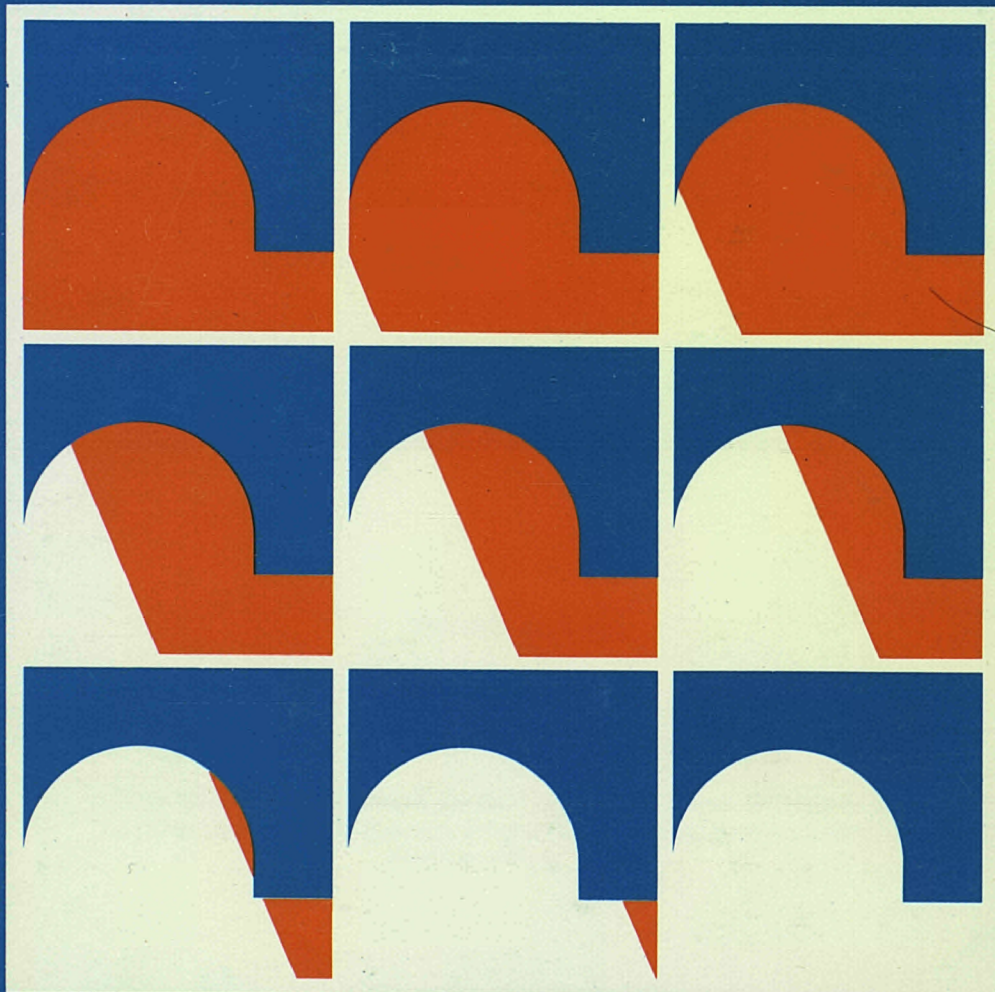
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Commission of the European Communities

# nuclear science and technology

**State-of-the-art review on technology for measuring and controlling  
very low level radioactivity in relation to the decommissioning  
of nuclear power plants**



**Report**

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## State-of-the-art review on technology for measuring and controlling very low level radioactivity in relation to the decommissioning of nuclear power plants

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### Final report

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SUMMARY

The written information in relation with the measurement devices for very low radioactivities, aimed at the decommissioning of electronuclear power plants, is scarce and much scattered. However, some realistic information can be obtained from the specialists working on nuclear plant dismantling sites.

The major radionuclides with a half-life superior to 1 year found in activated or contaminated materials, can be listed from activation and fission calculations and spectrometries performed on the various effluents and wastes released from nuclear plants. The full list of the radiations emitted by such radionuclides shows the problems arising from their measurements.

In practice, the in-situ measurements deal with two or three radionuclides only. The other contaminants must be determined on some samples in laboratories by means of radioanalysis and spectrometry technologies. These technologies are briefly described.

One of the problems concerns the detection threshold of the measuring instruments. This threshold varies in relation with the detector efficiency, the background and the measurement duration. The application of this concept of threshold to various types of instrument is studied. Tables and graphs are given that sum up the various possibilities of detection for the usual detectors according to their effective area.

The possibility to reduce the number of monitoring points using statistical methods is presented.

The difficulties met by the persons responsible for the monitoring underline the limitations attached to the existing instruments and give an idea of the methods or the instruments that would be necessary.

The utilization of passive integrating doseimeters with delayed reading such as track detectors for alpha radiations and radio-thermoluminescent crystals for beta and gamma radiations appears to be useful for the decommissioning and dismantling controls.

The use of very small size and sufficiently sensitive detecting probes would facilitate monitoring in places which are difficult to reach.

The industrial manufacturing of large shielded cells, easily decontaminable and adapted to handling would facilitate the work.



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## I. PURPOSE

The object of this study is to consider the technical characteristics related to the measurements of very low radioactivities necessary to monitor the decommissioned nuclear power plants in order to :

- reuse the buildings or the lands with or without restrictive conditions, after decontamination and partial or complete dismantling,
- recover the very low level radioactive materials for reusing or recycling, with or without restrictive conditions,
- release materials in ordinary landfill.

Geometry of materials and components and often inhomogeneous activity distribution prevent application of simple direct measuring procedures. Furthermore, elevated background irradiation may impede employment of direct on-site measurements even for flat components.

Therefore, after the choice of instruments to comply with various measurement requirements, the development of measuring procedures and strategies is important.

Choice of detectors and measuring procedures depend greatly on the activity limits set for unrestricted release, and on the procedures required by the competent authorities. These regulations do not yet exist in all countries. The existing regulations or specific authorizations are based on surface and/or mass activities. Surface activity limits of the order of one Bq/cm<sup>2</sup> for beta emitters have often been used in different countries. Limits for alpha emitters are usually lower.

## II. GENERAL DISMANTLING CONSIDERATIONS REGARDING THE RADIOACTIVITY MONITORING

The final state of the lands and maintained buildings of a former nuclear plant, as well as the various components and equipment, widely depends on the way the decontamination and the dismantling were carried out.

The checking for unrestricted use or release must be performed during dismantling monitoring. It would be unrealistic to foresee a detailed final checking. The meticulous recording of the residual radioactivity is progressively performed during dismantling. The control authorities must check the seriousness of this monitoring and examine rapidly, for safety reasons, the entire area, the relatively few measurement points being randomly spread. However the crucial areas must be carefully checked.

Professional personnel leave the site rather quickly after plant shutdown. It is very important that plans are carefully updated and registered with the reactor "history" (operation, incidents, accidents, different interventions). Thus the radiological monitoring as well as the choice of the best solutions for the preparation of the dismantling works will be made easier. It is often useful (when possible) to let the plant's former users dismantle some of the components or to associate them with the management of the dismantling contractors. From this point of view it seems advisable to keep the time between the reactor shutdown and its dismantling as short as possible. The cost/benefit analysis for the choice of the waiting-time before dismantling should take into account risks due to forgotten incidents that occurred during nuclear plant operation and due to the diminishing availability of the workers who built, maintained, repaired and used the plant.

Dismantling activities are themselves contaminating. Cutting materials and demolishing result in the dispersion of radioactive dust and aerosols. Radioactivity measurements are only correct at the time when they are made. The radiological situation of the different parts of the site change with time. It is important to select dismantling techniques that scatter least the radioactivity and to proceed to a local decontamination after each operation. Contamination maps must be checked sometimes at intervals of a few days after each working step. For example, it is necessary to perform new measurements after the use of explosives for the dismantling. The dismantling engineering must be carefully and methodically planned.

The use of important mechanical and human means reduces the cost of a dismantling site. But every work interruption is expensive. It is therefore important to define precisely the decommissioning objectives before beginning the works. In case of an unforeseen situation, decisions must be taken very quickly, which implies the permanent availability of the radioactivity monitoring technicians.

The radioactivity measurements must be scheduled together with the preparation of the dismantling operations, not only to determine a better site management, but also to use opportunities. For example, the monitoring technicians should therefore measure deep ground activity at the time of excavation and take samples of the cleaned land before it is covered up or surface contaminated.

The on site measurement of low activities is often difficult or impossible. In fact, the practical and economical organization of the dismantling site implies a demolition and a dismantling order which does not always allow the most radioactive elements to be removed first. For example, valves used for the isolation of circuits having contained active fluids are not removed. These valves are generally highly contaminated, especially by cobalt 60.

The classification of materials according to their radioactivity must be carried out before their removal, during demolition and dismantling. The checking should be performed without delaying works on the site. For example, materials may not be evacuated immediately. In this case, they are temporarily stored in a special place until the results of the samples analysis, deciding their future destination, are known.

The radioactive waste quantities can be reduced by some simple operations (pre-sorting, decontamination, precautions to prevent mixing, dust trapping, elimination of the highest levels of contamination before demolishing buildings and so on). However certain procedures or a too fine separation could delay works, stop the use of big machines and would be more expensive than the partial mixing of wastes.

The activity limits set by the authorities must be applicable and translated into meter readings and into a real work organization.

The following remarks can be made on the various measurements undertaken during the shutdown, decommissioning or dismantling of nuclear plants :

- Some samplings for laboratory spectrometric analysis are absolutely necessary, the contaminants spectrum changing from one plant to the other and in the various parts of a single plant. This spectrum can also vary after some decontamination. Each radionuclide has its own penetrating depth, fixation strength and decontamination factor.

Most of the radionuclides which have been measured in the year following plant shutdown have short half-lives, and have thus disappeared at the dismantling time. However, their measurement is useful to assess the possible contribution of long half-life radionuclides, difficult or impossible to measure, and which could determine the choice of the final dispatching of some materials. The two major radionuclides detected and measured on low activity parts are Co 60 and Cs 137. The latter only appearing significantly in plants where substantial cladding rupture have occurred.

Very low level measurements are difficult and often impossible to carry out on site because of environmental background. Such background is due to the most active parts which have not yet been cut into smaller parts, or which are going to be left on the site if the site is to be used again in the nuclear field.

Therefore, these measurements usually required either to create a restricted area with a low background inside the plant, or to transport materials to an outside storage area or a hall, or to collect samples to be measured in laboratories.

Nevertheless, an on-site monitoring is necessary to remove the most active parts which do not need low level measurements.

- The monitoring of concrete surfaces is often carried out by a wipe test.

- The global measurement, after calibration, is the only way to undertake the monitoring of pipes which have a small diameter. This technique enables us to determine the total activity which is contained in each part. This is possible when dealing with gamma-emitters such as Co 60 and Cs 137. With these methods, a spectrum of the contaminants must be checked by sampling every representative area of the circuits. This control helps in evaluating the homogeneity or the heterogeneity of the surface contamination.

- Sweating-out after decontamination means the progressive rise to the surface of radionuclides from the inside of a material. It depends on a great number of parameters such as : material composition, granulometry, surface condition,...

The phenomenon is often difficult to predict but never very important because the activity remaining inside a decontaminated material is low compared to the activity on the surface. It was very low or non existant in the French plants investigated.

- Controls of flat or in slightly curved surfaces are carried out by moving slowly the measuring head over the surface. Depending on the individual case considered, the surfaces which are checked at every run are either contiguous or slightly separated. The distance can be equal to the detector width. Thus, the monitoring duration can be decreased without risking overlooking a major contaminated spot, gamma sensors being also sensitive to lateral emissions.

The dose rate measured is increased by a high activity area located near-by. Therefore, the user has to locate the contamination spot and measure it.

The speed at which the detector is moved, and the distance between the various runs or the number and duration of each measurement are chosen according to the synthesis of information concerning :

- the environmental background and the activity limit to be measured,
- the features of the part to check : its history : possible incidents, water projections, contamination ; its position in relation with the ground, the verticality, the flowing direction of the contaminating fluids ; its physico-chemical characteristics that may have an influence on the contamination range : areas with different temperature, welding line, oxidation or corrosion spots, junction between different materials, ruggedness, pipe knees, discontinuities, eroded protective sheets,
- the contaminant features : dust, gas, liquid, chemical composition.

Generally, there is a greater number of measurements carried out than actually recorded. The recorded values are either chosen arbitrarily (geometric distribution for example) or defined according to the result of measurements : maximum values and minimum values when there are superior to the background.

### III. CHARACTERISTICS OF THE MATERIALS TO CHECK

The control procedures, the conception of the measuring instruments, their sensitivity, their spectral response must be adapted to the materials to be checked and to the characteristics of radionuclides.

The radionuclides are found in two types of materials :

1) the activated materials located near the core and irradiated by neutrons. This activation is relatively homogeneous inside the material. The reactor parts submitted to relatively low neutron fluences and with a low specific activity are essentially the biological shield made of reinforced concrete, and steel parts,

2) the contaminated materials. This contamination comes from the activation of the corrosion products conveyed by the primary water and from the dispersion of the irradiated fuel through cladding ruptures. It results from leakages in the primary circuit, processing and storage of radioactive effluents and wastes, maintenance and repairing works, fuel discharging operations and working incidents. Gases and aerosols also give rise to a deposit of radioactive materials on walls, ceilings and air shafts.

This contamination mainly occurs in the reactor building, near the primary circuit, especially on the floor, and near the fuel discharging facilities, the storage pools, the processing and storage facilities of radioactive effluents and wastes.

It also occurs in the secondary circuit and in the turbine in case of leakages from the steam generator.

Contamination generally accumulates on the material's surface and does not enter very deeply. But the surface distribution is not homogeneous. Gathering may occur on the walls or on the floors behind or below equipment which has been removed (such as motors, cables, sheaths, pipes, vessels), particularly when leakages or flows may have occurred. In pipes and tubes, contamination is more important at the discontinuity points of the fluid flow (bends, fans, knees ...). Contamination settles preferentially on rough surfaces, for example on the oxidized parts of metals where these faults are noticeable.

Contamination greatly varies in different concretes because of their porosity. Penetration and fixation of contamination depend also on the radionuclide's chemical nature. For example, cesium enters more deeply into concrete than cobalt.

Table 1 lists the various radionuclides with a half-life superior to one year contributing to the surface contamination of the materials of a nuclear plant or which can be found in materials submitted to the neutron flux (reinforced concrete, steel) /1/. The proportion of radiations emitted in significant quantities and their energies are indicated. We have not mentioned the very soft X-rays (some keV) following the electronic captures because they are not measurable in the working zone.

Taking into account the decay and daughter products of the radioactive materials, the major radionuclides at the time of dismantling or after it are not the major ones at the reactor shutdown. Their presence may be hidden by radionuclides with a shorter half-life.

But dismantling standards, as well as radioactive waste classification standards take into account these radionuclides with long half-lives which may play the leading part in the long term. The monitoring operations must then establish the contamination spectrum at the various areas of the decommissioned plant.



#### IV. FRENCH EXPERIENCE

Large power reactors have not yet been dismantled in France. The French experience comes from the dismantling of laboratories, radiochemistry and nuclear fuel reprocessing facilities, experimental and low-power reactors, and uranium ore processing plants. Engineering studies prior to the dismantling of the first nuclear power stations - natural uranium - graphite - gas type reactors - are under way. Contamination and activation measurements have been made on the following decommissioned reactors : Marcoule G1 (2 MWe), Marcoule G2 (38 MWe), Chinon 1 (70 MWe), Cadarache Pégase.

Written information directly concerning very low level radioactivity measurements is scattered and relatively scarce. Practical information such as : monitoring procedures, know-how, description and use of measurements devices is rare and not yet detailed. Interviews with specialists gave us interesting and realistic information.

According to the dismantling specialists, each nuclear plant presents particular problems that require a specific detailed study, a meticulous preparation and rigorous organization of the working place.

Plants of similar type or in which similar processes took place are never identical, did not work under the same conditions and were not subjected to the same deteriorations and modifications. Activation and contamination may occur in different ways.

##### IV.1. French regulations

A council directive of 1/6/1976 from the Commission of the European Communities, based on the ICRP works and adopted by France applies to the production, processing, handling, use, holding, storage, transport and disposal of natural and artificial radioactive substances and to any other activity which involves a hazard arising from ionizing radiation.

It proposed that the requirements for reporting and obtaining prior authorization need not be applied to activities involving :

a) radioactive substances when the quantities do not exceed in total the values of :

- 0.1 microcurie for nuclides of very high radiotoxicity
- 1 microcurie for nuclides of high radiotoxicity
- 10 microcuries for nuclides of moderate radiotoxicity
- 100 microcuries for nuclides of low radiotoxicity

b) radioactive substances of a concentration of less than 0.002  $\mu\text{Ci/g}$  (0.01  $\mu\text{Ci/g}$  for solid natural radioactive substances).

The 80/836 Directive modified the previous one and changed the values of the limits which became 5000 Bq instead of 0.1 microcurie and 100 Bq/g instead of 0.002  $\mu\text{Ci/g}$ .

This new directive is not yet applied in France. Although these limits have not been set to allow unrestricted release of waste, they have been used in few cases, to dispose of or to recycle a few batches of materials.

Regulation concerning waste that can be released without control has not yet been established. Recommendations do, however, exist for the disposal of radioactive waste of very low activity by small scale users, such as hospitals and laboratories. These recommendations contain daily limits for discharges in surface waters and sewers and to the atmosphere, and for transfer to incinerators and ordinary landfill. These limits do not apply to materials from the dismantling of nuclear installations. For the latter, a case-by-case authorization procedure is followed, involving a radiological impact assessment, the competent authority being the Ministry of Health.

The first permit for unrestricted release under this procedure was issued in 1968 and concerned ferrous and non-ferrous metals from the Atomic Energy Commission Installations. The limits concerned surface activities and were  $3.7 \text{ Bq/cm}^2$  for beta-emitters and  $0.037 \text{ Bq/cm}^2$  for alpha-emitters.

These values have been set taking into account the regulation on radioactive materials transportation (decree of the 1<sup>st</sup> July 1966) which specified these values as limits for the non-fixed contamination, and after a risk analysis studying the possible exposure in the following cases :

- external exposure from scrap piles,
- internal exposure due to dust inhalation during scrap handling,
- internal exposure through a wound during scrap handling,
- external exposure from a truck made with recycled steel.

The maximum individual doses, calculated with pessimistic parameters, ranged from  $1 \mu\text{Sv}$  to  $1 \text{ mSv}$  per year.

The second permit was granted in 1972, for materials from the dismantling of a uranium ore processing plant with a surface activity limit of  $5.E-5 \mu\text{Ci/cm}^2$  for alpha-emitters. This permit was renewed in 1980 for two other plants. To this surfacic activity limit was added dose rate limits :  $0.75 \text{ mrem per hour}$  for beta-rays and  $0.125 \text{ mrem/h}$  for gamma-rays at 30 cm from the surfaces, and for the outside of parts, easy to touch, a contact dose rate of  $1.5 \text{ mrem/h}$  for beta-rays. For outdoor grounds the limits were  $100 \mu\text{rad/h}$  at a distance of 15 cm and  $50 \mu\text{rad/h}$  at 1 meter.

In 1981 an authorization was given with the same surfacic activity limit of  $5.E-5 \mu\text{Ci/cm}^2$  for the melting of drums used for uranate transport. The melting was carried out in an ordinary steel-works without any restrictive conditions for the steel recycling.

At present, a simplified procedure is being considered in order to allow the recycling of metals through melting-works with limits that could be set at  $1 \text{ Bq/g}$  for beta-emitters.

#### IV.2. Examples of French decommissioning

##### IV.2.1. The experimental reactor PEGASE

PEGASE was a 35 MW thermal swimming pool reactor situated at Cadarache and designed to test on a true scale, in 18 gas loops, the fuel elements of natural uranium-graphite-gas or heavy water-gas reactors. These tests were extended to the fuel elements of HTR and fast reactors. PEGASE was in operation from 1963 to 1975 /2/.

In May 1964 there was a fuel cladding failure occurred with perforation and a partial melt down of several slabs. In the loops, studies and long-term tests were carried out on the evolution of the failed claddings of NUGG fuels. The internal parts of the plant were thus contaminated by the fission and activated products. The loops include force tubes intended to receive the fuel elements for testing, which were immersed and therefore activated. They were separated from the rest of the loop in order to be repaired, or replaced by tubes of different diameter or different materials and then stored in the open air. The non-immersed circuit in the loops included a cooling circuit, a pump and an exchanger controlled by a valve. Depending on the tests, these loops were made of stainless steel or aluminium alloy.

The dismantling of the plant was carried out in several stages from 1976, when the core and the experimental fuels were removed, to 1984. A few checks and some dismantling remain to be carried out on materials which were transported outside the plant, in particular on the external parts of the loops.

None of the buildings were demolished. The outlying buildings are being reused for non-nuclear work. The reactor building, which includes the pool, the storage tank and the stripping tank (dismantling loops), is being utilized for the temporary storage of irradiated fuels in tanks.

The tanks were coated with 18/8 stainless steel, and some were painted. Contamination of the pool, measured by wipe test shortly after shutdown of the reactor, was estimated at  $0.2 \mu\text{Ci}/\text{cm}^2$ . The spectrum of contaminants was determined on dusts suspended in the pool water. The main contaminants were as follows : Ce 144 + Pr 144 : 20%, Cr 51 : 16%, Ce 141 : 11%, Co 60 : 10%, Ru 106 + Rh 106 : 9%. Decontamination tests were carried out on samples taken from the material in the pool. Following decontamination the surface activity was of the order of  $1.E-5 \mu\text{Ci}/\text{cm}^2$ , the contaminants of the aluminium samples being Sb 124 : 96%, Co 60 : 4% and those of the steel samples Cr 51 : 91%, Co 60 : 9%.

The pool underwent decontamination by water jets under pressure and by codine. The residual surface activity was between  $1.8 E-6$  and  $6.3 E-5 \mu\text{Ci}/\text{cm}^2$ , allowing access to workers and then reutilization of the pool.

In the building itself the floors and walls were monitored by smear tests, and the highest surface activity was between  $1.E-6$  and  $1.E-7 \mu\text{Ci}/\text{cm}^2$ .

All the materials in the plant were dismantled and classified in different categories according to their activity :

1. High activity : dose rate higher than 200 mrad/h ; 1200 kg including 3270 Ci. These parts were cut up under water in the stripping tank, while activity was measured with the aid of an ionization chamber by removing the parts from the water. The preponderant radioactive element was cobalt 60. 800 kg were sent to the Centre de la Manche for permanent surface disposal. 400 kg, with too high activity, were placed in temporary storage.

2. Medium activity : waste which cannot be decontaminated ; 25 000 kg including 168 mCi.  
This involves either parts activated in the pool or parts which cannot be decontaminated because of their shape. They were disposed of as radioactive waste.
3. Low activity : waste which can be decontaminated ; 24 700 kg of stainless steel and 2 900 kg of aluminium. Following decontamination these materials could be classified in four categories :
  - 3 280 kg containing 1.5 mCi were disposed of as radioactive waste.
  - 10 965 kg of stainless steel and 1 785 kg of aluminium containing 40  $\mu$ Ci and 80  $\mu$ Ci respectively, as well as the deactivation torus of 4 700 kg containing about 0.5 mCi, to be melted. The corresponding limits of specific activity were  $1.E-4 \mu$ Ci/g and  $1.E-5 \mu$ Ci/cm<sup>2</sup>.
  - 6 780 kg which contained no detectable activity were sent to the scrap heap.The decontamination treatment produced 23 m<sup>3</sup> of liquid effluents containing 4.6 mCi.
4. Non active materials : 75 700 kg.  
These materials are considered to be neither irradiated nor contaminated as a result of their previous utilization. After verifying the absence of activity, measured with a highly sensitive apparatus used at a site selected for the low level of background, they were sent to the scrap heap.

Methods of measurement : A substantial proportion of the materials consisted of tubes, pipes and ventilation shafts and therefore presented problems in monitoring the surface activity. On the deactivation torus of 97 m<sup>2</sup> in area, analyses were carried out by spectrometry on 20 samples taken from inside and outside, followed by the measurement of surface activity inside (684 measurement points) and outside (650 points). On the loops, of smaller diameter, it was not possible to measure interior surface contamination directly. The spectrum of contaminants, the decontamination factor and the interior and exterior surface activity of the pipes were determined on the basis of samples taken before and after decontamination. After decontamination, cobalt 60 was the principal radioactive nuclide. Interior surface contamination was estimated by means of dose rate measurements carried out with the aid of a highly sensitive detector.

Tests carried out on pipes and tubes - smear tests, aerosol samples taken from the circulating air following repeated shocks and vibrations on the circuits - showed that the contamination remaining after decontamination was fixed. There was no evidence of any sweating-out phenomenon.

#### IV.2.2. Le Bouchet's ore processing plant

The ore processing plant in Le Bouchet worked from 1953 to 1970. 5 800 t of French ores with high contents of uranium (6 to 20%) were treated from 1953 to 1956. Urano-thorianite concentrates and uranium already separated from its daughters were then treated. The facility comprised 50 buildings over 80 000 m<sup>2</sup>. The decommissioning and dismantling realized from 1972 to 1980 required :

- The removal of radioactive wastes stored on the site,
- The decontamination and removal of treatment equipment,
- The decontamination of the buildings, twenty four of them being demolished,
- The removal of contaminated soil : decontamination has not been implemented, in some place the soil was excavated up to a depth of three meters.

The limit adopted for this decommissioning and authorized by the Health Ministry were fixed :

For materials and inside buildings

For alpha emitters :

- . non fixed contamination :  $5.E-6 \mu\text{Ci}/\text{cm}^2$ .
- . fixed contamination :  $5.E-4 \mu\text{Ci}/\text{cm}^2$  on local spot,  
average of  $5.E-5 \mu\text{Ci}/\text{cm}^2$  on  $1 \text{ m}^2$ .

For beta rays :

- . Dose rate at 30 cm : 0.75 mrem/h.
- . Contact dose rate on the outside of easily accessible parts :  
1.5 mrem/h.

For gamma rays :

- . Dose rate at 30 cm : 0.125 mrem/h.

Items with internal parts impossible to measure were considered as radioactive.

For outside buildings :

The limits were fixed at ten times higher than inside.

- For the site -

Soil between 0.5 and 3 m deep : 10 mCi/t for uranium 238  
0.5 mCi/t for uranium 226 and 228

Soil between the surface and 0.5 m :

- . gamma dose rate at 15 cm : 100  $\mu\text{rad}/\text{h}$  for each  $1 \text{ m}^2$  area
- . gamma dose rate at 1 m : 50  $\mu\text{rad}/\text{h}$  for each  $100 \text{ m}^2$  area

These limits have been set taking into account possible exposure in the following cases : radon daughters inhalation, external exposure outside and inside houses built on the site, internal exposure due to dust inhalation, internal exposure due to ingestion of crops from the site and fish from the river.

Characteristics, quantities and destination of materials were as follows :

900 t of equipment were re-used in similar plants.  
1400 t of iron scrap were sold.  
40 000 t of scrap, soil and rubble with activities lower than  $0.2 \mu\text{Ci}/\text{kg}$  were disposed of in a landfill.

135 t of lead sulfate with 25 mCi/kg of radium 226 are stored, waiting for an underground disposal.

2000 t of radioactive waste were disposed of in the shallow land burial Site de la Manche.

Alpha measurements were carried out using a plastic scintillator detector with the following characteristics : effective area 33 cm<sup>2</sup>, efficiency 30 %, background 0.02 pulses/s. The surface activity limit of 5 E-5 µCi/cm<sup>2</sup> corresponds to 10 pulses/s.

A collimated gamma sensor was used for soil measurements. The square grid was materialized with string above the soil. For each square meter five measurements were carried out, in each corner and in the middle. Only the average value is recorded on the map.

#### IV.2.3. The 40 MW reactor G1 in Marcoule

This natural uranium-graphite-air cooled reactor was in operation from 1956 to 1968. The building and the interior equipment are being reused as a testing shop. The experience concerns the primary cooling circuit, made of carbon steel, which was first measured by scanning using a sensitive beta-gamma detector. Part of this circuit was slightly irradiated by neutrons. These pipes were then cut into pieces and decontaminated.

The decontamination process included a preheating of the pieces with warm high pressure water, spraying with a decontaminant solution, brushing, rinsing with hot water and a high pressure nozzle.

After decontamination the non fixed contamination was checked through wipe tests made with cotton saturated with phosphoric acid and measured with a sensitive beta-gamma detector.

23.7 tons with non fixed contamination, even after two decontamination processes were stored as radioactive waste.

The other 1760 parts without non-fixed contamination were numbered, weighed and measured. The majority of the parts weighed about 50 kg. Most of them weighed between 10 and 400 kg, with the lightest around 2 kg and the two heaviest 1300 and 1800 kg.

Spectrometry carried out on samples showed that Co 60 was the major and often unique contaminant.

The final measurements were made with a sensitive beta-gamma detector with a detection threshold corresponding to 0.5 Bq/g for parts with neutron induced activity and 2.5 Bq/cm<sup>2</sup> for surface contamination. The background was about 10 pulses/second.

Table 2 summarizes the measurements made on these parts.

The parts were sorted out into two categories :

43.4 tons, with activities lower than 0.5 Bq/g or 2.5 Bq/cm<sup>2</sup>, corresponding to the second column of Table 2 were stored in the scrap yard for further recycling.

101 tons with induced activity and without non-fixed contamination were considered as radioactive waste.

## V. GERMAN EXPERIENCE

### V.1. German Regulations

In the Federal Republic of Germany the 'Atomgesetz' (AtG-Atomic Law) /3/ and the 'Strahlenschutzverordnung (StrlSchV - Radiation Protection Regulation) /4/ regulate the management of radio-active material and especially the management of material used, produced or to be disposed off as radio-active waste during operation of nuclear power plants.

The existing law does not comprise explicit regulations for the decommissioning of nuclear power plants and for unrestricted waste disposal or recycling of low-activity material.

Such criteria can only be derived from regulations for the protection of the public and the environment against the hazards of ionizing radiation ( §§ 44 to 48 StrlSchV) or from regulations specifying limit values for the unrestricted use of formerly contaminated laboratories, workshops, tools or containers for radioactive material (§§ 64 and 35 StrlSchV).

According to § 44 StrlSchV the maximum allowable dose in the outer control areas of nuclear power plants due to direct irradiation caused by plant operation and incorporation of radio-active material as referred to in § 45 StrlSchV, is limited to 150 mrem per year.

According to § 45 StrlSchV the maximum allowable dose of the common public due to nuclear installations has to be 'as low as possible' and is limited to 30 mrem per year (30 mrem-concept).

Concentration limit values for waste air, water and solid waste material are defined in § 46 and § 47 StrlSchV with reference to maximum allowable rates of annual inhalation or ingestion of various radioactive nuclides tabulated in appendix IV tbl. IV 1 col. 4 StrlSchV. These values meet the requirements of the 30 mrem-concept.

The applicability of concentration limits defined in §§ 46 and 47 StrlSchV is furthermore restricted by the principle of § 28 StrlSchV 'as low as possible'. Therefore the definition of release limits is subject to separate decisions of the licensing authorities in each case.

As mentioned in a comment to § 28 StrlSchV, licensing authorities will have to consider technical feasibility as well as economical aspects.

Definitions of emission concentration limits for waste air and water with respect to § 46 StrlSchV are routinely made in the licensing procedure for plant operation. However, at present there is little experience for unrestricted release of solid waste materials from decommissioning operations.

Relevant values are defined in § 47 and § 4 StrlSchV.

All these regulations, ruling the management and disposal of radio-active waste and the general management of radio-active material have to be taken into account in order to determine acceptable release limits.



The final definition of concentration limits according to § 4 and § 47 StrlSchV ( $10^{-4}$  of the limits for unrestricted release) again is subject to the regulations of § 28 StrlSchV.

The applicability of these values to the unrestricted release of material deriving from decommissioning of nuclear power plants is strictly opposed in a circular letter of the Bundesminister des Inneren (RS II 1-511831/11; Sept. 20, 1979) commenting on § 4 StrlSchV. The comment emphasizes that all materials from decommissioning are subject to restricted disposal (§ 47 StrlSchV) or to restricted management (§ 3 StrlSchV). Exceptions may be approved by licensing authorities, even for concentrations above these limits.

Exceptional approval of unrestricted disposal, handling or recycling may be considered only if the residual activity is insignificant and no precautions have to be taken additionally to reduce hazards to the public and the environment.

In order to state insignificance of residual activity of materials or waste produced by restricted management or authorized processes (decommissioning), the nature of activity (nuclide compound) as well as the nature and quantity of waste material itself has to be considered.

If unrestricted disposal or handling is approved, the material is no longer to be considered 'radio-active' as defined in § 2 AtG and is not allowed to be marked according to § 35 StrlSchV.

The main measuring problem during decommissioning operations on the basis discussed is the evaluation of activity concentration values. Additionally an analysis of the nuclide compound of contamination and the calculation of total activity inventory is necessary.

The surface contamination limits given in §§ 35 and 64 StrlSchV refer to transport containers respectively laboratories, workshops, working sites, tools and devices to be released for unrestricted disposal or utilization.

Reference to these values can be made for the release of structures with plain geometry.

The acceptance of surface contamination limits as given in §§ 35 and 64 StrlSchV (for alpha-emitters :  $1 * 10^{-6} \mu\text{Ci}/\text{cm}^2$  ( $0,037 \text{ Bq}/\text{cm}^2$ ), for other nuclides :  $1 * 10^{-5} \mu\text{Ci}/\text{cm}^2$  ( $0,37 \text{ Bq}/\text{cm}^2$ ) is subject to § 28 StrlSchV and thus depends on the individual case.

The main measuring problem for release procedures based on the regulations of § 35 and § 64 StrlSchV is the evaluation of specific surface activity averaged over an area of  $100 \text{ cm}^2$ .

The nuclide compound of contamination and the activity inventory of the material to be released will have to be analyzed resp. calculated.

Doserate measurements of plant structures and components and especially doserate monitoring in buildings and compartments can support release-decisions and procedures. For relevant limit values § 44 StrlSchV may be referred to.

Because of the 30 mrem-concept the maximum dose rate has to be less than 0,003 mrem/h above natural background, exposure by inhalation or ingestion included.

If exposure by inhalation or ingestion, e.g. caused by exudation of radioactive gases and aerosols, can not be excluded, the above limit for buildings, compartments and components has to be defined even more restrictively.

## V.2. Decommissioning of the Nuclear Ship OTTO HAHN

One of the most typical cases of dismantling was that of the first European marine propulsion reactor in the ship NS Otto Hahn. This reactor was shut down in 1979 after 10 1/2 years in operation. Following a study carried out in liaison with NIS (Nuklear-Ingenieur-Service) on the various options open, it was decided to dismantle the reactor and the nuclear part of the plant completely.

About 1 200 tonnes of material were removed, including 480 tonnes for the vessel and its protection, and 300 tonnes of decontaminated material which was transported to the GKSS site and placed in a special room for further study. The interior structures of the reactor, representing 1.5 % of the mass of dismantled material, contained almost all the activity, mainly arising from cobalt 60.

Public release and the unconditional reutilization of the structures of the ship and part of the dismantled components (370 t) was fixed, in accordance with the Regulation on protection against ionizing radiation of 13 October 1976, at a surface activity limit of  $10^{-5} \mu\text{Ci.cm}^{-2}$  (beta-emitters).

An independent expert (PTB = Physikalisch-Technische Bundesanstalt, Braunschweig) verified conformity with this regulation by sampling and measuring with the aid of a highly sensitive dose rate meter developed by PTB.

The walls and ceiling of the nuclear zone were made of steel. The external protection of the containment vessels and the protection of the fuel tank were in reinforced concrete. These installations of relatively simple geometric shape, involving relatively low non-ingrained contamination, were decontaminated. This work was monitored by the authorities, who laid down that, for the final monitoring measurements, the universal probe should be replaced by:

- a sensitive alpha - beta probe for the components of surface activity, with slightly higher level background
- a sensitive beta - gamma probe for painted or rusted components, with low level background.

The dismantable components, amounting to 3 600 m<sup>2</sup>, were measured in a low level background zone. The fixed parts were measured on the spot, after reducing the ambient background by removing all the radioactive sources; 14 400 m<sup>2</sup> were measured in this way.

28 % of this surface area underwent a second decontamination  
5.5 % underwent three decontaminations  
3 % underwent four decontaminations.

There were a total of 1 440 000 measurements, including 1 350 000 using the Minicont with the universal probe, requiring 11 500 hours of work. 90 000 final monitoring measurements carried out by the authorities (Service for Protection against Radiation and the independent expert) took 1 250 hours.

The activity of components of complicated geometric shape was measured indirectly on samples whose surface activity was determined after gamma-spectrometry analysis.



Decontamination procedures were monitored by means of electrolytic sampling and gamma-spectrometry measurements on 60 samples of slightly higher activity, about  $10^{-4} \mu\text{Ci}\cdot\text{cm}^{-5}$ , and decontamination was assessed with the aid of the Minicont detector.

Many measurements were carried out on the more active parts of the plant, in particular using Ge-Li spectrometry, in order to obtain a good knowledge of the contaminations and their distribution and to apply this to the strategy of unconditional release.

Ambient and contact dose rate measurements carried out by the independent PTB expert confirmed the care with which the decontaminations were executed and the accuracy of the measurements. Where the average surface activity obtained after decontamination ( $= 6\cdot 10^{-6} \mu\text{Ci}\cdot\text{cm}^{-2}$ ) was exceeded slightly this was discovered quickly and surely with the selected dose rate meter.

### V.3. Decommissioning of the Nuclear Power Plant Gundremmingen

The first decommissioning of a land-site nuclear power plant set to work in the Federal Republic of Germany is that of the Gundremmingen plant.

The 238 MW (el.) BWR was shut down in 1976 because of an operational disturbance of the turbine-circuit. Decision to put the plant out of duty finally and to dismantle less contaminated sections step-by-step, providing the reactor-building for safe-enclosure, was made in 1980 after various alternative decommissioning concepts had been thoroughly studied.

First of all the power-house components, mainly consisting of the components of the turbine-circuit, should be dismantled and decontaminated. The amount of the material to be released for unrestricted reutilization or disposal should be as large as possible.

By means of a large-scale experiment, in the course of which approx. 100 tons of material, dismantled from different sections of the turbine-circuit and representing the various grades and kinds of contamination as well as the different component-categories, were to be processed, feasibility of concept had to be demonstrated. Furthermore the experiment should render data for the assessment of costs, required man-power, secondary-waste quantities and the hazards to the environment and the staff in charge of decommissioning.

With reference to Strahlenschutzverordnung (StrlSchV - Radiation Protection Regulations) the limit-value of surface-contamination for components to be released for unrestricted reutilization or disposal was fixed at  $0.22 \text{ Bq/cm}^2$  by decree of the authorities, the more restrictive limitation of the value expressively named in the regulations ( $0.37 \text{ Bq/cm}^2$ ) allowing for non-detectable low-energy beta- and gamma-emitters

During the large-scale experiment approx. 110 tons of steel-structures, representing the component-categories pipes (diameter  $> 200 \text{ mm}$ ), valves, mantle-sheets (tanks, apparatus) and machinery-parts were dismantled from turbine-circuit sections with various operational conditions. The initial-state surface-contamination varied from 5 to  $1800 \text{ Bq/cm}^2$ . These components should be decontaminated by electropolishing mainly.

Release measurements were carried out by employment of alpha-beta sensitive contamination monitors. In order to confirm the accuracy of the contamination measurements and constancy of the nuclide-compound of the residual contamination, additional measurements employing different measuring procedures had to be performed especially for components which at minor areas (bore-holes etc.) could not be measured by direct monitoring procedures. From the areas solid-samples had to be taken at random to be evaluated and analyzed by single nuclide GeLi-detection.

Meanwhile for approx. 106 tons of material, dismantled in the course of the experiment, unrestricted release has been stated by official decree.

Because of the favourable results of the experiment, authorities held out prospects to approve the decommissioning concept and to authorize the complete decommissioning of the power-house.

## VI. METHODOLOGY

Measurement methodology depends on many considerations, for example :

- The activity limits can be fixed for surfaces or masses.
- The authorities may require the activity measurement of all the surfaces or masses.
- The limits may be fixed for each radionuclide or for groups of radionuclides (for example : alpha emitters, beta-gamma emitters, pure beta emitters, etc...).
- The regulations may fix the maximum size of the surface or mass over which measurement are averaged.
- Possibility of indirect measurements.
- Characteristics of apparatus (background, detection limit, efficiency,...) in relation to activity limits.
- Distribution of activity : activation and/or contamination.

### VI.1. Limitations on the measurements possibilities

To check activities and surface contaminations, detectors such as gas flow counters, GEIGER-MULLER counters or scintillation detectors are used. These detectors give a number of pulses which is proportional to the number of photons or particles reaching the detector.

The measurements consist of :

- either counting over a given period and, if necessary, repeating measurements when the average value and the statistical error are assessed. The detector has to be fixed during the measuring time,
- or a continuous measurement of the average counting rate. The advantage of a counting-ratemeter, also called integrator, is to give the distribution of the activity, as the sensor is slowly moving over the surfaces being checked. Therefore, the contamination points can be easily localized.

The average counting rate is set over on a previously determined period and is defined by the time constant of the integration circuit. For these reasons, the sensor should remain stationary during a time twice as long as the time constant (the statistical error of this measurement is equal to 1 % when the measurement duration is five times this constant).

When the pulses from the detector have a magnitude proportional to the radiation energy (gas flow counters, scintillation counters, semi-conductors) the instrument can be supplied with a device selecting or discriminating the various pulse magnitudes. It is possible to detect a particular radionuclide inside a mixture and to remove partially the radioactive background or the sensor's and electronic components' background

This last possibility is convenient, for example, when :

- a detector is sensitive to different radiations (beta and gamma sensors).
- a radionuclide is considered as critical regarding the standards of radioactive waste disposals, or the re-use conditions,
- a radionuclide is used as a contaminant indicator in a mixture previously identified by spectrography.

Appendix C presents the detection thresholds' definition for different types of apparatus and give the values for some commonly used detectors.

## VI.2. Difficulties arising from alpha radiation measurements

The alpha contamination measurement of low activity parts does not seem to be necessary in reactors where only a few cladding ruptures have occurred. But this issue may appear in future dismantlings.

It would be illusory to proceed to some direct alpha measurements on grounds or rubbish : the surfaces are rough or uneven. It is difficult to bring the detector sufficiently close to the source and the thin detector windows can be quickly damaged. The alpha radiation direct measurements is only possible with fragile thin window detectors on smooth surfaces.

Moreover, we can only measure the activity on material surfaces, because alpha radiations are absorbed by very thin layers of matter.

Particular attention has to be paid to areas difficult to reach : corners, angles, cavities, sheath, pipes, cables, etc. So very small-sized alpha sensors must often be used, for example Si junctions. Their sensitivity is unfortunately limited.

Some elements that would be difficult to check because of their location, may be systematically considered as radioactive waste.

The portable instruments used in-situ, use air proportional counters to detect alpha particles. Stationary instruments are often supplied with proportional gas flow counters. The magnitude of the pulses is proportional to the energy produced by the radiation. So we can differentiate alpha radiations from beta-gamma radiations with a pulse height discriminator or by changing the counter voltage. Large window proportional counters (800 cm<sup>2</sup> and more) are convenient to check smooth and wide surfaces.

Scintillation counter with a thin layer of zinc sulphide are also used to detect alpha radiations.

To decrease the alpha particle absorption, detector windows are very thin ( $\approx$  0.7 milligram per square centimeters). They are therefore fragile.

Semi-conductors with silicon surface barriers are used for alpha spectrometry. The real thinness of the crystal (100 to 200 micrometers) reduces the interference of the gamma or beta radiations. The deposited layer (semi-conductor junction and contact electrode) is extremely fragile. It must be protected from any frictions. Although silicon can be used at the ambient temperature, the background together with the limit of detection can be reduced by cooling the detector. These measurements are carried out under vacuum conditions.

Grid ionization chambers filled with argon and pressurized under 1.5 to 2 bars are also used in the alpha spectrometry laboratory. The magnitude of the pulses given by the chamber is proportional to the alpha particles energy. The background measured over  $2\pi$  between 4 and 6 MeV is less than two pulses per hour, the detection threshold is very low.



A sample which has been contaminated by alpha emitters should be prepared as a thin layer, less than  $0.2 \text{ mg/cm}^2$  to limit the self-absorption of the radiation by the sample material.

### VI.3. Identification of radionuclides mixtures

The in-situ direct measurement of soft beta radiations, as well as low energy and gamma radiations is as difficult to perform as the alpha particle measurements. The high energy surrounding background often prevails. In general, over limited areas, activation or contamination corresponds to a specified mixture of radionuclides. Therefore, the alpha emitters are often associated with beta or gamma emitters.

A radio-chemical or a spectroscopic analysis, carried out on samples in a laboratory, may aid defining the composition of the mixture as a function of the depth of sampling.

The radionuclides included in the mixture and emitting some radiations easily detectable as for instance cobalt 60, are used as indicators to determine quickly the activity of the materials coming from the area where the composition of the contamination was analyzed.

The mixture of radioactive contaminating components is determined for every part of the nuclear power plant. For example, the spectrum of the contaminants surrounding the primary circuit may be different from the one on the walls of the fuel storage pool. The spectrum of an equivalent location, may vary greatly from one reactor to another, because of the different burn up rates of the fuels, cladding ruptures together with the various events that occurred during the reactor's life-time.

The measurements carried out in-situ, consist mostly of beta and gamma measurements. A knowledge of the processes that give rise to contamination in the measured areas confirmed by the sample analysis, generally helps in assessing the total activity from the beta or gamma measurements performed.

#### VI.3.1. Spectrometry

Battery supplied portable analyzers can be used in a shielded cell provided in the working zone. The magnetic tape recording of alpha and gamma spectrum enables a delayed evaluation.

A Ge-Li junction detector is used in the laboratory for precise spectrometry. It has a good resolution, about 1 to 2 keV, according to the energy and to the kind of detector. Its useful volume can reach  $100 \text{ cm}^3$  and its relative efficiency : 30 to 40 %. Unfortunately, this detector is expensive and its use is difficult. The semi-conductors must be kept permanently at a temperature inferior to minus  $50^\circ\text{C}$ . The Ge(Li) detectors are used in liquid nitrogen ( $-196^\circ\text{C}$ ). At such a temperature, the background is very low.

High purity germanium detectors have been improved recently. They are stable at ambient temperature. But the detector as well as its preamplifier must like the Ge(Li) detector, be cooled to liquid nitrogen to reduce background. The cryostat can be autonomous up to 20 days according to its capacity. Its nitrogen consumption is about 1 liter per day.

The spectrometry of X or soft gamma radiations for an energy range of 1 keV to 50 keV use silicon-lithium semi-conductor detectors. Their resolution at 5.9 keV ranges from 0.16 keV to 0.27 keV according to the detector size.

The detection limit in gamma spectrometry is largely dependent on the analysis technique, the external background and the radionuclides that are mixed with the element to identify. Annexe C2 gives the detection threshold for spectrometry measurements.

Sodium iodide scintillators are used in gamma spectrometers ; their capacities are lower but are less expensive. They have a good efficiency but bad resolution (about 10 % of the measured radiation energy, for example 60 keV for Cs 137 and 130 keV for Co 60). Moreover, it is impossible to identify radionuclides with closely separated energy rays.

Background is relatively important. Nevertheless gamma scintillation spectrometers are useful to quickly sort out the materials inside the shielded cell provided on the working site. A thin plastic scintillation detector having a low sensivity to gamma radiations, is used for beta measurements. This scintillation detector is cheap, and is easier to replace, when damaged than the window of a proportional counter.

The proportional counter does not distinguish between beta and gamma radiations. Compensated detectors or anti-coincidence systems, enable only beta radiation to be counted by removing the signals from photons due to natural radioactivity and other gamma sources.

As the beta spectrum is a continuous spectrum, it is difficult to identify pure beta emitters, such as tritium, carbon 14, sulfur 35, strontium 90 and yttrium 90.

In the laboratory, tritium measurements are performed in a physico-chemical extraction, which is followed by a dissolution in a liquid scintillator.

### VI.3.2 Radio-analysis

The radio-analysis of low activity samples often requires samples to be prepared in such a way as to increase their specific activities and to make their measurement easier :

- concentration, crushing, dissolution, evaporation, ...,
- fixation on "carriers" which are chemical materials with the same properties as the radionuclides to be analysed. This process enables elements existing in very small quantities to be extracted,
- physico-chemical separation, in order to isolate the radionuclides sought, especially beta-emitters,
- counting samples with reproducible geometry and small thickness to decrease the self-absorption of alpha and beta-rays. Standardized cupels allow automatic sample changer to be used,
- the quantity of sampled materials must be relatively large to obtain a homogeneous sample from a crushing technique,
- sample weighing, preparation and measurement must be performed with care.

It is important to decrease background due to equipment below natural background. To do so, the shielded cell must be kept perfectly clean.

- the detectors are calibrated daily with standards of various energies. The detection thresholds together with backgrounds are regularly checked to prevent any major error and to detect any contamination of the detection units.

- the samplings which are intended to define and to monitor the usual measurements on the working site, need a radio-analysis and a spectrometry laboratory, with high quality equipment and staffed with skilled technicians and some sampling experts. It may be useful to have this analysis sub-contracted to some specialized laboratories.

#### VI.4. Reference mixtures

Table 3 shows the surface activities of the major contamination components which have a half-life superior to 1 year, for an elapsed time of 5 years, 25 years, 100 years after reactor shutdown. The surface activity of each contamination component is assessed in arbitrary units in relation to the total surface activity (the sum of all components) that has been standardized to 1 becquerel per square centimeter for the 3 'decay periods' considered /1/.

The surface activities were determined from measurements performed in various PWR nuclear plants, in different zones (primary circuit, shield, pipes, effluents, ...). The distribution of activities, due to the different radionuclides, is the result of the average of a great number of measurements. This distribution only gives some indications. As already mentioned, the composition of the mixture of radionuclides varies from one place to another. Nevertheless the average composition, described in the table, does not differ much from the values measured or calculated.

Hence, it seems possible to use one of the components of the mixture, the easiest to measure, as a contamination indicator, in order to simplify the dismantling controls.

The counting rate corresponding to each relative activity, and expressed in pulses per second, for an homogeneously spread source, is shown in the next column. This counting rate is an estimation for alpha sensors (proportional counters) beta sensors (thin-window GEIGER MULLER counters or proportional counters) and gamma sensors (20 cm<sup>3</sup>, gamma scintillation counter (NaI (Tl)) generally used for in-situ measurements.

For a beta sensor, the curves shown in fig. 1 and 2 /5/ illustrate the efficiency variation versus energy for a point radioactive source, and for a homogeneous uniformly spread source. Figure 3 shows the variation in the efficiency of a gamma scintillation detector (NaI) /6/.

The alpha and beta measurements are carried out at a few millimeters from the surface being checked. Gamma measurements are performed at a distance of about 20 cm. The surface is considered as an infinite homogeneous radioactive source.

The nature, energy and percentage of the major radiations emitted by radionuclides, are shown in the second column of table 3.

The third column gives the range of the efficiency of a standard detecting probe (given under  $2\pi$  for a point source) for the radiation considered.

Tables 4, 5, 6 relate to activated materials, for austenitic steel, with a high nickel content (10 %), for steel with a low nickel content (1 %), and for reinforced concrete.

These tables are arranged as follows : mass activity of activating components with reference to a total mass activity standardized to 1 Bq per gram. The activity is homogeneously spread.

The mass activities were calculated taking into account the average values of the neutron flux, the composition of various steels, concretes, iron included in reinforced concrete, and the decay time.

The "dose rate" columns show the relative radiological importance of the various activation products.

Figures 4, 5, 6, 7 show for cooling times of 5, 25 and 100 years, and a standard total activity of 1 Bq/cm<sup>2</sup> or 1 Bq/g, histograms of the counting rates (pulses per second) or dose rates (sievert per hour) corresponding to surface contamination, for activation of a 1 % or 10 % nickel content steel and for the activated reinforced concrete, for the easiest-measured radionuclides. The range of energy corresponding to a significant percentage emission is shown between brackets on the histogram.

In comparison with the individual radionuclides presented respectively in tables 3, 4, 5, 6, the elements shown on the histograms represent the following proportions of the sum of activities and the sum of the counting rates or the dose rates per element of each table, see table 7.

#### VI.5. Soil monitoring

The penetration of the contamination into homogeneous grounds can range from a few centimeters up to one meter. On average, a depth of 50 cm is rarely exceeded. Special attention must be given to the particular features of the ground (trenches, ponds, rivers, differences in heights or soil composition, old sewage pipes, etc.).

An already established method is to use string tied to pegs driven into the ground (in order to prevent blowing away), to divide the field into one-meter squares. Every five meters a post is driven into the ground ; posts ten meters apart are made higher so as to serve as references. Measurements are performed 30 cm above ground. 5 measurements are made every square meter (one at each corner and one at the center). In order to find for "hot-points", the instrument is collimated as indicated in chapter VII.

Existing wheeled ground monitors are supplied with large-window gas flow counters (1,230 cm<sup>2</sup>). They are provided with a commutator for beta-gamma or alpha measurements on even ground (Roads, car-park).

Samples are taken at different depths, by means of an auger for radioanalysis. The sampling distribution takes into account the origin of the contamination and the result of the preliminary measurements.

The monitoring intervals should be closer along the most likely contamination pathways (loading or storage areas, roads where contaminated materials have been transported, pathways or pipe galleries which contained radioactive fluids that may have leaked, etc.).

#### VI.6. Flat surface monitoring by direct measurement

A systematic evaluation of the radioactivity is usually carried out. The user slowly moves measuring instrument over the whole surface, marking on the ground or on the walls, the areas where significant counts occur. The measurement points will be closer in these areas. In fact, the operator makes more measurements than he actually reports on his checking file. Therefore, a significant contamination cannot reasonably be overlooked.

The measurement points registered on the checking file are arranged as points in square elements. The size of these elements is chosen in order to be able to submit any local contamination points to a separate treatment and thus to avoid a considerable increase in the average radioactivity of the materials or of the area considered.

In practice, the size of the monitoring elements is decided from a knowledge of the previous use of the buildings of area, and of any anomalous situations that could have occurred there (thus showing the importance of good records). The measurement grid may be defined in accordance with regulatory bodies.

A scaled map drawn from the results of these measurements clearly shows the areas to be cleaned. Different colours indicate the amount of cleaning to be effected and the materials to be removed. In the most monitored places where several measurements are carried out in every square (the 4 angles and the center), often only the average and/or the highest value is recorded.

#### VI.7. Statistical control

As already stated, the ultimate monitoring or checking cannot be undertaken without taking into account the successive measurements realized during the dismantling, the decontamination and considering the corresponding file. The problem is not only statistical : is the distribution of the checking measures (average, variance) representative of the monitoring measurements ? Is there a risk that an "active spot" may have been overlooked in the choice of measurement points ? As already seen the methods used for gamma-monitoring do not allow the overlooking of any high activity area. Still, areas with an activity only slightly superior to the average might be forgotten. Thus, the surveyor has only to take into account the operators good faith and the reliability of the instruments calibration.

Some statistical methods give criteria for the acceptance of results /7/ but in practice these criteria are based on common sense, on the analysis of the working methods and on the checking of the values obtained in areas of homogeneous activity.

## VII. GENERAL FEATURES OF EXISTING INSTRUMENTS

The measuring devices can be classified in three categories :

- contamination monitors (alpha, beta and beta-gamma sensitive)
- dose-rate meters
- gamma spectrometers

For release procedures, direct, surface measurements employing alpha, beta-sensitive contamination monitors are normally used.

Dose-rate measuring procedures are useful for comprehensive, final inspections in order to prove that decontamination has been sufficient and show the exactitude of the previously made measurements and release procedures.

Random gamma-spectroscopy, carried out under laboratory conditions, mainly serves to confirm the accuracy of contamination measurements and to analyze prevailing nuclide contamination.

The selection of the instruments is based on :

a) The detector and its characteristics which must correspond to the radiations to be measured :

- GEIGER MULLER counter, gas flow counter, scintillation counter associated with a photomultiplier, semi-conductor, ionization chamber, ...
- Effective detection area, window thickness, efficiency, background, detection limit, stability of calibration.

b) The characteristics of the measuring unit :

- Signal processing (amplitude selection according to the radiation energy, automatic deduction of the average background, time constant, measurement duration, ...),
- Information processing (availability of results, display, signalling, remote transmission, cartographic recordings of the places where measurements were carried out, calculation of the mean value and the standard deviation).

c) The adequateness of the material for the in-situ conditions under which it is to be used

- Weight, size, power supply autonomy, detector availability, ease of use, strength, susceptibility to damage.
  - Measuring distance and geometry
- Lighthness is an important factor for instruments carried by an operator several hours per day.

Cables linking the heads to measurement instruments are submitted to large torsional stresses and wrenching forces which may cause faulty contacts. Connectors must be strong.

The measurement devices used on a dismantling site are submitted to very severe conditions (shocks, vibrations, humidity, temperature variations ...).

As far as measurement quality and reliability are concerned, ease of use and strength characteristics are as or even more important than metrological performance.

- d) Monitoring cost price (material, material maintenance and repair, labour cost).

The choice widely depends on individual site conditions and on the user's evaluation.

At the present time two types of instrument are normally used in decommissioning and dismantling practices :

- the land is checked with, a device used for uranium ore prospection,
- the walls and equipment testing is performed with a detector with interchangeable captors : alpha, beta-gamma, large surface, medium surface, small size for places of difficult access and pipes.

For land monitoring instruments, ore prospection portable devices are generally used. They are often scintillation detectors. Here is an example of the main characteristics of these devices :

- gun shaped detection unit, cable linked to an electronic case worn at the waist,
- gamma ( $> 30$  keV) detection by NaI (Tl) crystal and photomultiplier,
- several scale ranges,
- acoustic signal with variable frequency and adjustable switching threshold,
- outputs for earphone, counting scale, recorder,
- the background corresponding to the natural irradiation is about 30 p/s.

The sensitivity of these devices varies with energy. They are contamination detectors rather than measurement devices but may be calibrated for the particular radionuclides measured.

Some instruments are equipped with a discriminator to cover several energy ranges, for example between 50 keV and 2 MeV.

Instruments supplied with a sandwich scintillator leading to a better energy response and a better sensitivity are also currently used.

In order to find the "hot spots" to decontaminate, the scintillation detector is collimated with lead. A non-collimated detector receives a lateral irradiation from the environment which is approximately the same as that due to the one square meter surface located in front of the detector at 30 cm.

For buildings and equipment monitoring, the basic instrument is a self supplied "polyradiometer". Its measurement capacity covers a wide range of pulse rates. The threshold of the acoustic signal can be step by step fixed. The detection heads are removable sensors fixed at the end of a cable. The wide choice of sensors is adapted to match the various situations : G. M. counters, semi-conductors, photomultipliers with scintillators, large-window counters for flat walls, small-sized for cavities and difficult-to-reach areas, alpha, soft beta, beta, X, gamma radiations...



Some instruments are supplied with two sensor switches on the measuring instrument. Each plug is linked to an adjustable independent high voltage power supply. Thus an alpha detector and a beta-gamma detector for example, are immediately available without any change in plugging and adjustments. For all measuring points, the user can record the alpha reading and the beta-gamma reading one after the other without having to make any difficult modifications.

Gas flow counters are fragile and not practical to use because of the cylinders of gas required. However a better calibration stability could be obtained with them. Large window counters (for example 40 x 20 cm) or multi detector assemblies are time-saving for the control of smooth walls, but in this case, the counter windows must be sufficiently resistant. Large scintillation counters connected to a photomultiplier by an optical guide can be useful.

The list of instruments that appear to best fit decommissioning checking has been omitted, for the following reasons :

- it would take too long to include all the very extensive lists of equipment from European manufacturers, and omissions would unfortunately occur,
- nuclear facilities differ ; therefore the choice depends on the characteristics of the particular working site,
- various techniques are rapidly improving ; such list would thus soon become obsolete,
- instruments cannot be seriously and objectively selected because a standardization of the characteristics described in the manufacturer's catalogs does not exist,
- we cannot check and guarantee the characteristics given in catalogs.

The main characteristics are given in table 8.

## VIII. POSSIBLE IMPROVEMENTS

### VIII.1 Use of passive delayed-reading integrating dosimeters

Passive integrating dosimeters may be useful because :

- They are cheap, strong, reliable. They do not need any checking or energy supply, unlike electronic instruments.

- Their use does not require any training nor any particular qualification. As the measurement is delayed, they are not cumbersome on a dismantling site. The introduction and withdrawal of detectors are rapid. The identification of the detector (location, time of exposure, reading), can be automated together with the interpretation of the results. Running costs are low.

- Measurement integration makes it possible to reach high sensitivities. These detectors are small-sized, and they can be put into places difficult to reach.

- These detectors give a contact measurement. The influence of external radiation can be reduced with a shield or it can be balanced by a differential measurement using a reference detector appropriately located with respect to the surface being checked.

#### VIII.1.1. Radio-thermoluminescence

Some previously irradiated crystals emit, when heated, an amount of light proportional to the absorbed dose /8/. The thermoluminescence curve giving the light flux as a function of temperature shows several peaks, the temperatures of which depend on the substance. In order to avoid the loss of dosimetric information or fading, only the higher temperature peaks are used for dosimetry purposes.

The materials used at present are : calcium sulphate activated with dysprosium ( $\text{CaSO}_4$ , Dy) by far the most sensitive, lithium fluoride (Li F), in wide use, and the lithium borate activated with copper ( $\text{Li}_2 \text{B}_4 \text{O}_7$ , Cu).

They are used as powder or as sintered pellets in a waterproof and opaque packaging.

They give a response to any kind of alpha, beta and gamma radiations. For gamma rays between 100 keV and a few MeV, the energy of the incident radiation has hardly any effect on dosimeter response. Below 100 keV, there is an increase in sensitivity which differs according to the dosimeter material. The response of lithium borate is practically unchanged near the maximum thermoluminescence efficiency, which occurs with 40 keV photons, whereas the response of lithium fluoride increases by a factor of 2.8 and that of calcium sulfate by a factor close to 19.

The response to dose can be linear over a large range from the reading threshold up to 100 grays or more, depending on the specific dosimeter material used.

The lower limits for measurable cumulative doses are as follows :

- $\text{CaSO}_4$  : D<sub>y</sub> : 1 E-5 to 5 E-5 Gy  $\pm$  30 %
- $\text{LiF}$  : 1 E-4 to 2.5 E-4 Gy  $\pm$  30 %
- $\text{Li}_2\text{B}_4\text{O}_7$  : Cu : 1 E-4 to 2.5 E-4 Gy  $\pm$  15 %

The stability of the response at normal temperatures is good. The loss of dosimetric information or fading does not exceed 10 % of the result, for a delayed reading of 3 to 6 months after the irradiation of the dosimeter maintained at a temperature inferior or equal to 50°C.

Usually, the dosimeter and reader are calibrated together with a known dose of radiation. The conversion of absorbed dose by the dosimeter in contact with a radioactive surface into a measurement of the activity of the surface is carried out either through a calibration using a standard source of the same composition, or through a theoretical calculation requiring the values of the absorption cross-sections of both the dosimeter and its cover for the different radiations to be measured.

When this kind of dosimeter is used for decommissioning measurements the process must be automated : delivery and withdrawal of the dosimeters, reading including dose or activity results, identification, localization, map drawing.

#### VIII.1.2. Track etch detectors for alpha measurements

The passage of alpha particles through polymers creates irreversible damage along their paths, which can be increased by chemical etching /9/. This development is only possible if the energy loss per unit length exceeds a threshold value specific of the material and etching conditions. With cellulose nitrate, alpha particles with energy between a few hundred keV and about 4 MeV can be detected. These detectors are insensitive to beta and gamma rays, to light, and are little affected by dampness.

The detector, a thin film of a few to hundreds microns thickness, can be placed directly on the surface to be checked or maintained at a certain distance from it. A thin film of a few microns may be inserted between the detector and the surface in order to protect the detector from contamination and to slow down alpha particles.

The detecting materials are manufactured on an industrial scale. Detectors and etching solution are cheap.

The measurement consists of numbering the tracks. It can be carried out by :

- microscope counting, perhaps with an image analyser ;
- electrical discharge : counting sparks through etched holes
- electro-chemical etching leading to larger holes allowing counting by light scanning
- densitometry : light reflexion or diffusion ; possible method for high tracks density.

Depending on the counting method, activity from 3 E-2 Bq/cm<sup>2</sup> up to 1 E+6 Bq/cm<sup>2</sup> can be measured with an exposure duration of 10 minutes.

The same considerations as previously mentioned on radioluminescent dosimeters, relative to the possibility of an automation of the localization and the recording of the exposure duration can also apply to the use of solid alpha detectors.

### VIII.2. Monitoring automation

The characteristics of the detector and previous calibrations (particularly the radionuclide mixture determined from sampling radioanalysis) must be taken into consideration in evaluating measurements.

Each measurement must be identified in relation to its location. It will then be easy to chart contamination. Such operations can be computerized. Consequently, a teletransmission is needed between the portable instrument and computer, but cable transmission must be avoided. Indeed, the measuring instruments must be autonomous if they are to be used in a great number of places and in various monitoring situations.

The mechanization of the motion of measuring instruments over large surfaces, indoors or outdoors, over walls or in fields surrounding the nuclear plant can be studied.

For example, we might consider adapting the already experimental system to the automatic chartering of atmospheric pollution or to field monitoring. This system employs a vehicle which scans the ground, supplied with a mobile laboratory including /10/ :

- suitable detectors,
- data acquisition equipment relative to localize the vehicle and measure the time elapsed,
- a real time data processing and restoring system that would chart the ground radioactivity,
- a small-sized all-track vehicle, operated by remote control and carrying captors, that could operate in areas difficult to reach.

One prospective techniques by helicopter might also be adapted to land monitoring.

The localization of a radium needle (10 mCi) buried at a depth of two meters has been performed with this method with an accuracy of a few meters. Obviously, field scanning would be more precise with a land vehicle, the rolling and the pitching of a helicopter do not favour a sufficient detecting focus.

The response time as well as instrument sensitivity would have to match the moving characteristics of the vehicle.

### VIII.3. Improvement to the detection limit

Improvements on the detector efficiency are desirable.

Activity close to decommissioning standards are very often impossible to measure directly on site. Parts of the plant such as : sheath, metallic pipes, etc. are then dismantled and cut into pieces. Their surface activity is then measured in large shielded cells that protect them from the surrounding radiation.

The alpha and beta radiations from coarsely crushed wastes are measured in the same way. Easily-dismantled cells (transportable from one plant to the other) and their handling and decontaminating supplies as well as measuring outfits, should be manufactured on a large scale. It would undoubtedly improve the metrologic performances and the work would be faster and easier. The transport of the materials to the measuring cell should also be studied.

When materials must remain on the dismantling site, samples are taken. An increased productivity would be gained if decontaminable instrumentation adapted to sampling and to sample preparation was studied.

#### VIII.4. Adaptation of the sensors to various shapes and locations

At present, there are few alpha or beta-gamma sensors, on the market that can be introduced in difficult access zones such as small cracks or slits, corners, gaps between wall and pipes or cables, boreholes or small diameter pipes. A scintillation detector fitted to an optical fiber cable might prove useful.

For falt walls, the effective detecting surface as well as the strength of thin window detectors might be improved. Their change when damaged should also be easier.

Toric-shaped detectors, opening in the same way as a grip-core ammeter, would make the external monitoring of electrical cables small-diameter pipes easier.

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Radio-nuclide	γ Emission		β Emission		α Emission		Half-life	Formation		
	MeV	%	MeV(max)	%	MeV	%	years (1)	(2)		
H 3			0.0186	100			1.23 E+1	AB		
C 14			0.156	100			5.73 E+3	AB - AA		
Si 32			0.213	100			6.5 E+2	AA		
Cl 36			β+ 0.115	2			3 E+5	AB		
K 40	X (Ca)		β- 1.32	89			1.26 E+9	AB		
			EC	11						
Ca 41	X (Sc)		EC	100			1 E+5	AB		
Fe 55	X (Co)		EC	100			2.86 E 0	C - AB - AA		
Co 60			0.314	99			5.27 E 0	C - AB - AA		
			1.173	100					1.48	0.12
Ni 59	X (Cu)		EC	100			7.5 E+4	C - AB - AA		
Ni 63			0.067	100			1.001 E+2	C - AB - AA		
Se 79			0.16	100			6.5 E+4	AA		
Sr 90			0.546	100			2.82 E+1	C		
Y 90			2.27	100			7.31 E-3	C (from Sr 90)		
Zr 93			0.063	96			1.53 E+6	C		
			0.034	4						
Nb 94			0.49	100			2.03 E+4	C		
			0.702	100					0.871	100
Mo 93	X (Tc)		EC	100			3 E+3	AB - AA		
Tc 99			0.292	100			2.14 E+5	C		
Ag 108m	X (Cd)		EC + β <sup>+</sup>	91			1.27 E+2	AB - AA		
									0.08	5
									0.434	89
									0.614	90
Cd 109	X (In)		EC	100			1.27 E 0	AB		
			0.088	3.8						
Cd 113m			0.265	0.1	0.58	99	1.41 E+1	AB		
Sn 121m			0.354	100			5.5 E+1	AA		
Sb 125	X (Te)		0.125	29			2.76 E 0	C		
			0.176	6					0.300	45
			0.427	31					0.444	12
			0.463	10					0.612	14
			0.599	24						
			0.634	11						
Cs 134			0.089	27			2.066 E 0	C - AB		
			0.57	23					0.415	3
			0.605	98					0.658	70
			0.796	99						
			1.038	1						
Cs 135			0.21	100			2.3 E+6	C		
			0.662	85					1.176	6.5
Cs 137	X (Ba)		0.514	93.5			3.02 E+1	C		
			0.662	85					1.176	6.5
Ba 133	X (La)		EC	100			1.06 E+1	AB		
			0.1606	0.75						
			0.22	5						
			0.276	7.1						
			0.303	18.6						
			0.356	62						
0.384	9									

Table 1 : Radionuclides characteristics

Radio-nuclide Tab. 1-4	γ Emission MeV %	β Emission MeV(max) %	α Emission MeV %	Half life years (1)	Formation (2)
La 137	X (Ce)	EC 100		6 E+4	AB
Pm 147		0.224 100		2.62 E 0	C
Sm 151	X <sub>L</sub> (Eu) 0.022 4	0.076 98		9 E+1	C
Eu 152	X (Gd) 0.122 37 0.245 8 0.344 27 0.779 14 0.965 15 1.087 12 1.113 14 1.408 22	0.71 β+ 0.019 1.48 β- 6.75 0.69 β- 13.8 EC 73		1.34 E+1	AB - AA
Eu 154	0.123 38 0.248 7 0.593 6 0.724 21 0.759 5 0.876 12 1.000 31 1.278 37	0.274 29 0.579 38 0.843 17 0.976 4.6 1.198 0.67 1.866 10.8		8.6 E 0	AB - AA
Eu 155	0.087 32 0.105 20	0.152 84 0.252 16		4.96 E 0	AB - AA
Ho 166m	0.081 12 0.184 90 0.280 30 0.412 12 0.532 12 0.711 58 0.810 60 0.830 11	0.065 100		1.2 E+3	AB
Tm 171	γ (% very weak)	0.03 2 0.0965 98		1.92 E 0	AB
Pb 205	X (Bi)	EC 100		1.4 E+7	AA
Pu 238	(% very weak) <0.01		5.46 28 5.5 72	8.77 E+1	C
Pu 239	<0.1		5.11 11 5.14 15 5.16 73	2.4 E+4	C
Pu 240	~ 0		5.12 24 5.17 76	6.55 E+3	C
Pu 241		0.021 100		1.43 E+1	C
Am 241	X <sub>L</sub> (Cm) 0.06 36 0.101 0.04		5.44 13 5.49 85	4.33 E+2	
Cm 244	X <sub>L</sub> (Bk) 0.043 0.02		5.77 23 5.81 77	1.81 E+1	
(1) $N.E+n = N.10^n$ (2) AA = Steel activation AB = Concrete activation C = Contamination					EC = Electronic capture

Table 1 (cont.) : Radionuclide characteristics



mass kg	Counting rate above background (pulses/second)								
	0	10-30	40-60	80-150	200-300	350-500	700-1000	1500-2000	3000
3 - 10	94	17	7	3					
15 - 20	126	53	17	11	6	2	1		
25 - 40	208	62	33	27	5	3	1		
50 - 70	281	144	55	32	20	7	2	2	1
80 - 120	81	73	24	40	24	19	5	1	2
150 - 250	30	74	20	23	15	8			
300 - 500	16	44	12	5	2	4			
600 - 800	1	12	4	1					
1300				1					
1800			1						
<b>Total number of pieces</b>	<b>837</b>	<b>479</b>	<b>173</b>	<b>143</b>	<b>72</b>	<b>43</b>	<b>9</b>	<b>3</b>	<b>3</b>

Table 2 : Numbers of pices according ot their mass and counting rate

Reactor cooling time (years)													
Radio-nuclide	Type of decay	Energy MeV (1)	Emission % (2)	Surface sepsivity (1.s <sup>-1</sup> /Bq.cm <sup>2</sup> ) (3) (4)	5		25		100				
					Relative surface activity Bq/cm <sup>2</sup>	Counting rate i.s <sup>-1</sup>	Relative surface activity Bq/cm <sup>2</sup>	Counting rate i.s <sup>-1</sup>	Relative surface activity Bq/cm <sup>2</sup>	Counting rate i.s <sup>-1</sup>			
Fe 55	EC		100	0	6.7E-2		4.1E-3		3.6E-10				
	X	0.006		0									
Co 60	β	0.310	100	30	6.7E-1	7.7	3.8E-1	4.3	1.4E-4	1.6E-3			
	γ	1.173	100	0.72							0.94	0.53	2E-4
	-	1.332	100	0.69									
Ni 59	EC		100		6.7E-5		5.3E-4		3.6E-3				
	X	0.007											
Ni 63	β	0.067	100	0	1.3E-2		8.9E-2		3.6E-1				
Sr 90	β	0.540	100	15.5	3.3E-2	1.05	1.6E-1	5.1	1.7E-1	5.4			
Y 90	β	2.27	100	16.4									
Zr 93	β	0.063	96	0	6.7E-7		5.3E-6		3.6E-5				
	β	0.034	4	0									
Nb 94	β	0.473	100	15	6.7E-6	1E-4	5.3E-5	8E-4	3.6E-4	5.4E-3			
	γ	0.703	98	0.8							1E-5	8.2E-5	5.6E-4
	-	0.871	100	0.77									
Tc 99	β	0.290	100	30	6.7E-6	7.6E-5	5.3E-5	6E-4	3.6E-4	4.1E-3			
Sb 125	β	0.125	29	4	6.7E-2		3.7E-3		2.1E-10				
	-	0.300	45	11							0.67	3.7E-2	0
	-	0.444	12	14.6									
	-	0.612	14	15.8									
	γ	0.176	6	1.6									
	-	0.427	31	1.04							5.5E-2	3E-3	0
	-	0.463	10	1.01									
	-	0.599	24	0.83									
	-	0.634	11	0.82									
	-	0.660	3	0.8									
Cs 134	β	0.089	27	3	2.7E-2		2E-4		7.1E-15				
	-	0.415	3	14							0.33	2.5E-3	0
	-	0.658	70	16									
	γ	0.57	23	0.83									
	-	0.605	98	0.83							5E-2	3.7E-4	0
	-	0.796	99	0.8									
	-	1.038	1	0.77									
	-	1.168	1.9	0.77									
	-	1.365	3.4	0.77									
Cs 135	β	0.21	100	8	2E-7	1.6E-6	1.6E-6	1.3E-5	1.1E-5	8.8E-5			

Table 3

Surface activities of the major contamination components for different cooling times

	Energy MeV (1)	Decay % (2)	Surface sensitivity $1.s^{-1}/Bq.cm^{-2}$ (3) (4)	5		25		100		
				Bq/cm <sup>2</sup>	1/s	Bq/cm <sup>2</sup>	1/s	Bq/cm <sup>2</sup>	1/s	
Cs 137	β	0.514	94	15.5	6.7E-2	1.04	3.3E-1	5.1	3.9E-1	6.05
	-	1.176	6	16						
	γ	0.662	85	0.8		4.5E-2		0.22		0.26
Pm 147	β	0.224	100	8.5	6.7E-2	0.57	2.6E-3	2.2E-2	3.6E-11	0
Sm 151	β	0.076	100	0	6.7E-5		4.5E-4		1.7E-3	
	γ	0.022	4	0						
Pu 238	α	5.498	71	23	5E-4		3.4E-3		1.25E-2	
	-	5.454	28.7	-		1.1E-2		7.8E-2		0.29
	-	5.359	0.13	-						
Pu 239	α	5.105	11.5	23	6E-5		4.7E-4		3.2E-3	
	-	5.143	15.1	-		1.4E-3		1.1E-2		7.4E-2
	-	5.155	73.3	-						
Pu 240	α	5.014	0.1	23	8E-5		6.3E-4		4.3E-3	
	-	5.115	24.4	-		1.8E-3		1.4E-2		9.9E-2
	-	5.159	75.5	-						
Pu 241	β	0.021	100	0	1.4E-2		4.1E-2		7.5E-3	
Am 241	α	5.443	13	23	1.7E-4		3.5E-3		3E-2	
	-	5.486	86	-		3.9E-3		8.1E-2		0.69
	-	5.513	0.12	-						
	-	5.545	0.25	-						
	γ	0.06	36	0						
	-	0.101	0.04	0						
Cm 244	α	5.764	23	23	2.3E-4		7.9E-4		3E-4	
	-	5.806	77	-		5.3E-3		1.8E-2		6.9E-3
	γ	0.043	0.02	0						
Total	α				1	2.3E-2	1	0.2	1	1.2
	β					11.4		14.6		11.5
	γ					1.1		0.8		0.3

$$N.E-n = N.10^{-n}$$

Contamination is assessed assuming that the sum of the surface activities released by every contamination components is standardized to 1 Bq.cm<sup>-2</sup>

- (1) For β emitters : maximum energy
- (2) Beta-radiations detector : Detector with five Geiger-Muller Counters totalizing an effective area of 75 cm<sup>2</sup>. Surface sensitivity is given in figure 1 for an infinite homogeneous source, set directly in contact with the detector.
- (3) Gamma-radiations detector : Scintillator NaI (Tl) of 8 cm<sup>2</sup> surface ; the counter efficiency for a punctual source is given in figure 2. The efficiency for an homogeneous infinite surface is assessed as twice the efficiency of the surface just in front of the detector window (8 cm<sup>2</sup>).
- (4) Alpha-radiations detector : Scintillator SZn (Ag) with an effective area of 78 cm<sup>2</sup> ; the efficiency for a punctual source is 30 %, the surface sensitivity is then equal to 23 imp/s for 1 Bq/cm<sup>2</sup>.

Table 3 (continued)

Radio-nuclide (3)	Type of decay	Reactor cooling time (years)					
		5		25		100	
		Bq.g <sup>-1</sup> (1)	Gy.h <sup>-1</sup> (2)	Bq.g <sup>-1</sup> (1)	Gy.h <sup>-1</sup> (2)	Bq.g <sup>-1</sup> (1)	Gy.h <sup>-1</sup> (2)
H 3		2.3E-1		6.3E-1		2.E-1	
Cl 36	β	7.5E-5	6.1E-12	5.9E-4	4.8E-11	1.3E-2	1.1E-9
K 40	EC	5.6E-9		4.4E-8		9.6E-7	
	β		7.6E-16		6E-15		1.3E-13
Ca 41	EC	3.5E-3		2.8E-2		6E-1	
Fe 55	EC	6.2E-1		2.4E-2		1E-9	
Co 60	γ	1.9E-2	6.8E-9	1E-2	3.7E-9	1.2E-5	4.4E-12
	β		6.7E-10		3.6E-10		4.3E-13
Ni 59	EC	2.6E-6		2E-5		4.4E-4	
Ni 63	β	2.4E-4	1.9E-12	1.7E-3	1.3E-11	2.3E-2	1.8E-10
Se 79	β	2.2E-8	4.1E-16	1.7E-7	3.1E-15	3.8E-6	7E-14
Zr 93	β	1.1E-8	7.9E-17	8.7E-8	6.2E-16	1.9E-6	1.4E-14
Nb 94	γ	3.8E-6	7.3E-13	3E-5	5.7E-12	6.4E-4	1.2E-10
	β		2.2E-13		1.7E-12		3.6E-11
Mo 93	EC	2.6E-8		2E-7		4.4E-6	
Tc 99	β	3E-9	1E-16	2.4E-8	8.1E-16	5.2E-7	1.8E-14
Ag 108m	γ	3.8E-5	6.9E-12	2.6E-4	4.7E-11	3.7E-3	6.7E-10
Cd 109	γ	1.4E-5	2.3E-15	1.6E-9	0	0	0
Cd 113m	γ	1.1E-4	2.3E-15	3.1E-4	6.4E-15	1.5E-4	3.1E-15
	β		7.3E-12		2E-11		1E-11
Sn 121m	β	1.8E-8	7.4E-16	1.2E-7	4.9E-15	1.3E-6	5.3E-14
Sb 125	γ	8.2E-7	3.8E-14	3.9E-8	1.3E-15	0	0
	β		3E-14		1.4E-15		0
Cs 134	γ	1.2E-3	2.3E-10	1.7E-5	3.2E-12	0	0
	β		6.9E-11		9.8E-13		0
Ba 133	γ	9.9E-5	3.1E-12	1.3E-4	3.9E-12	3.6E-6	1.1E-13
La 137	EC	4E-9		3.1E-8		6.8E-7	
Pm 147	β	7.7E-5	2E-12	3.1E-6	8E-14	0	0
Sm 151	γ	4.7E-4	1.9E-14	3.1E-3	1.3E-13	3.7E-2	1.6E-12
	β		4.1E-12		2.7E-11		3.2E-10
Eu 152	γ	1.1E-1	1.3E-8	2.8E-1	3.6E-8	8E-2	1E-8
	β		2.9E-9		7.4E-9		2.1E-9
Eu 154	γ	5E-3	7.3E-10	1.6E-2	2.6E-9	1.4E-2	2.3E-9
	β		4E-10		1.3E-9		1.1E-9
Eu 155	γ	1.9E-4	2.8E-13	9E-5	2.5E-13	5.6E-8	1.6E-16
	β		3.7E-12		1.7E-12		1.1E-15
Ho 166m	γ	3.5E-6	5.3E-13	2.8E-5	4.3E-12	5.6E-4	8.5E-11
	β		2.6E-14		2.1E-13		4.2E-12
Tm 171	γ	2.8E-5	1.6E-15	1.5E-7	0	0	0
	β		3.1E-13		1.7E-15		0
Total	specific activity	1		1		1	
Total	γ dose		2.1E-8		4.2E-8		1.3E-8
Total	β dose		4.1E-9		9.2E-9		4.9E-9

(1)  $N.E-n = N.10^{-n}$

(2) Relative activity of each radionuclide (total activity normalised to 1 Bq.g<sup>-1</sup>)

(3) Radionuclides characteristics in table 1

Table 4  
Dose rate near activated reinforced concrete

Radio-nuclide (4)	Type of decay	Reactor cooling time (years)					
		5		25		100	
		Relative specific activity (Bq.g <sup>-1</sup> ) (1) (2)	Dose rate (Gy.h <sup>-1</sup> ) (1)	Bq.g <sup>-1</sup> (1) (2)	Gy.h <sup>-1</sup> (1)	Bq.g <sup>-1</sup> (1) (2)	Gy.h <sup>-1</sup> (1)
Fe 55 (3)	EC	4.5E-1		1.8E-2		0	
Co 60	γ β	4.5E-1	1.8E-7 1.6E-8	2.6E-1	1E-7 9.4E-9	2.9E-5	1.1E-11 1E-12
Ni 59 (3)	EC	1E-3		8.4E-3		1.8E-2	
Ni 63 (3)	β	9.6E-2	7.5E-10	6.6E-1	5.1E-9	9.4E-1	7.3E-9
Se 79	β	5E-7	9.3E-15	4E-6	7.4E-14	8.4E-6	1.6E-13
Nb 94	γ β	1.8E-4	3.7E-11 1E-11	1.5E-3	3.1E-10 8.5E-11	3E-3	6E-10 1.7E-10
Mo 93	EC	3.3E-5		2.7E-4		5.4E-4	
Tc 99	β	8.9E-6	3E-13	7.2E-5	2.4E-12	1.5E-4	5.1E-12
Ag 108m	γ	3.9E-5	7.5E-12	2.8E-4	5.4E-11	3.9E-4	7.5E-11
Sn 121m	β	3.9E-7	1.6E-14	2.6E-6	1.1E-13	2.8E-6	1.2E-13
Sb 125	γ β	4.7E-5	2.3E-12 1.7E-12	2.3E-6	7.2E-14 8.3E-14	0	0 0
Sm 151	γ β	2.1E-7	1E-18 1.8E-15	1.5E-6	1E-17 1.3E-14	1.6E-6	1E-17 1.4E-14
Eu 152	γ β	1.2E-3	1.7E-10 3.2E-11	3.1E-2	4.3E-9 8.2E-10	8.5E-4	1.2E-10 2.2E-11
Eu 154	γ β	4E-4	8.1E-11 3.2E-11	1.4E-3	2.3E-10 1.1E-10	1.1E-4	1.8E-11 8.9E-12
Eu 155	γ β	1E-8	1E-17 1.7E-16	8.4E-8	1E-16 1.4E-15	1.6E-7	1E-16 2.7E-15
Total	specific activity	1		1		1	
Total	γ dose		1.8E-7		1E-7		8.2E-10
Total	β dose		1.7E-8		1.5E-8		7.5E-9

- (1) N.E-n = N.10<sup>-n</sup>  
(2) Relative activity of each radionuclide (total activity normalised to 1 Bq.g<sup>-1</sup>)  
(3) Iron content in steel ≈ 90 % - Nickel content in steel : 10 %  
(4) Radionuclides characteristics in table 1

Table 5

Dose rate near activated steel (10 % Ni)

Radio-nuclide (4)	Type of decay	Reactor cooling time (years)					
		5		25		100	
		Bq.g <sup>-1</sup> (1) (2)	Gy.h <sup>-1</sup> (1)	Bq.g <sup>-1</sup> (1) (2)	Gy.h <sup>-1</sup> (1)	Bq.g <sup>-1</sup> (1) (2)	Gy.h <sup>-1</sup> (1)
Fe 55 (3)	EC	5.1E-1		5E-2		0	
Co 60	γ β	4.8E-1	1.9E-7 1.7E-8	6.9E-1	2.7E-7 2.5E-8	2.8E-4	1.1E-10 1E-11
Ni 59 (3)	EC	1.1E-4		2.2E-3		1.7E-2	
Ni 63 (3)	β	1E-2	7.8E-11	1.7E-1	1.3E-9	9.1E-1	7.1E-9
Se 79	β	5.3E-7	9.9E-15	1E-5	1.8E-13	8.2E-5	1.5E-12
Nb 94	γ β	1.9E-4	3.9E-11 1.1E-11	3.8E-3	7.9E-10 2.2E-10	2.9E-2	5.8E-9 1.7E-9
Mo 93	EC	3.5E-5		6.9E-4		5.3E-3	
Tc 99	β	9.5E-6	3.2E-13	1.9E-4	6.4E-12	1.5E-3	5.1E-11
Ag 108m	γ	4.1E-5	7.9E-12	7.3E-4	1.4E-10	3.8E-3	7.3E-10
Sn 121m	β	4.1E-7	1.7E-14	6.7E-6	2.8E-13	2.7E-5	1.1E-12
Sb 125	γ β	5E-5	2.4E-12 1.8E-12	5.9E-6	1.8E-13 2.1E-13	0	0 0
Sm 151	γ β	2.2E-7	1E-18 1.9E-15	3.8E-6	1E-17 3.3E-14	1.6E-5	1E-17 1.4E-13
Eu 152	γ β	1.3E-3	1.8E-10 3.4E-11	8.1E-2	1.1E-8 2.1E-9	8.3E-3	1.2E-9 2.2E-10
Eu 154	γ β	4.3E-4	8.8E-11 3.5E-11	3.6E-3	5.9E-10 2.9E-10	1.1E-3	1.8E-10 8.9E-11
Eu 155	γ β	1.1E-8	1E-17 1.9E-16	2.2E-7	1E-15 3.7E-15	1.6E-6	1E-15 2.7E-14
Total	specific activity	1		1		1	
Total	γ dose	1.9E-7		2.8E-7		8E-9	
Total	β dose	1.7E-8		2.9E-8		9.2E-9	

(1)  $N.E-n = N.10^{-n}$

(2) Relative activity of each radionuclide (total activity normalised to 1 Bq.g<sup>-1</sup>)

(3) Iron content in steel : ≅ 97 % - Nickel content in steel : 1 %

(4) Radionuclide characteristics in table 1

Table 6  
Dose rate near activated steel (1 % Ni)

Cooling time	5 years	25 years	100 years
	Surface contamination		
Number of the major radionuclides included in the mixture	7	6	5
Activity	78 %	91 %	60 %
Counting rate	96 %	99 %	≅ 100 %
	Steel activation (10 % Ni)		
Number of radionuclides	4	6	4
Activity	54 %	95 %	94 %
Dose rate	≅ 97 %	≅ 98 %	≅ 98 %
	Steel activation (1 % Ni)		
Number of radionuclides	4	6	6
Activity	48 %	94 %	97 %
Dose rate	≅ 100 %	≅ 100 %	≅ 100 %
	Reinforced concrete activation		
Number of radionuclides	4	4	4
Activity	37 %	94 %	31 %
Dose rate	≅ 100 %	≅ 88 %	≅ 93 %

Table 7 : Contribution of the major radionuclides to the total activity and to the counting rate

1 Type of detector	2 Useful area (cm <sup>2</sup> )	3 Window thickness (mg.cm <sup>-2</sup> )	4 Energy threshold (MeV)	5 Background (i.min <sup>-1</sup> )	6 Measurement efficiency (4 π) (%)	7 Detectable activity t = 30 s (Bq.cm <sup>-2</sup> )	8 Detectable activity t = 5 s (Bq.cm <sup>-2</sup> )
A Bell type organic Geiger Muller	7	1.8	α 2.5 β 0.030 γ 0.005	25	C 14 (β) 20 %	0.24	(P) > 0.3
B Cylindrical halogen Geiger Muller Ø = 35 mm; L = 190 mm	18	28	β 0.16 γ	50	Sr 90 + Y 90 8 %	0.33	(P) > 0.42
C Air filled counter	300	1	α 2.5	10	U 235 (α) 13 %	(P) > 0.003	(P) > 0.01
	"	"	"	"	Pu 239 (α) 9 %	(P) > 0.005	(P) > 0.015
D Gaz flow counter	246	0.8	α 2.5	1.8	Am 241 (α) (HV = 750 V) 21 %	(P) > 0.0014	
	"	"	β 0.020 γ	180	Tl 204 (β) (HV = 1700 Volts) 31 %	0.012	0.03
E Gaz flow counter	710	0.8	α 2.5	6	Am 241 (α) (HV = 750 Volts) 21 %	(P) > 0.0007	0.018
	"	"	β 0.02 γ	600	Tl 204 (β) (HV = 1700 V) 31 %	0.007	0.018
F Scintillator ZnS (Ag) 70 X 3 mm	29	0.7	α 2.5	1	Am 241 (α) 26 %		
G Fluorescent plastic scintillator 70 X 3 mm	29	0.7	β 0.25	60	Sr 90 + Y 90 (β) 14 %	0.12	(P) > 0.2
H Fluorescent plastic scintillator 40 X 40 mm	12.5	1000	γ 0.1	600	Co 60 (γ) 5 %	2.6	6.4
I NaI (Tl) cristal 32 x 25 mm	8	1000	γ 0.03	600 à 2280 depending on impulse height	Co 60 (γ) 5 %	26 à 8	10 à 20
J NaI (Tl) cristal 32 x 5 mm	7	Al (0.2 mm)	X	600	Fe 55 ; Pu 238 ; Pu 239 (X) 25 %	0.9	2.3

(P) : Background < 101 - Poisson distribution

Table 8 : Characteristics of the detectors



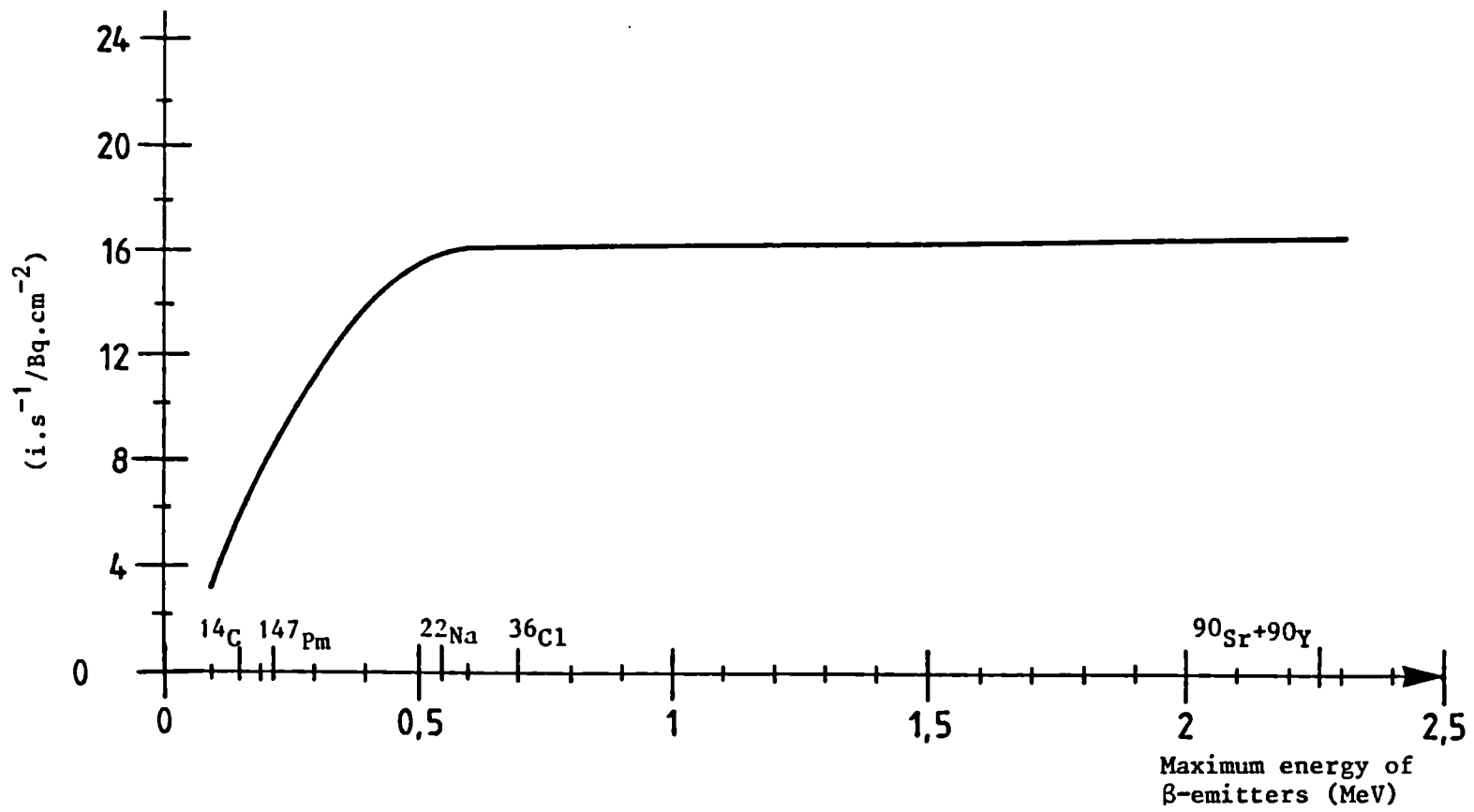


Figure 1 :  $\beta$ -detector surface activity

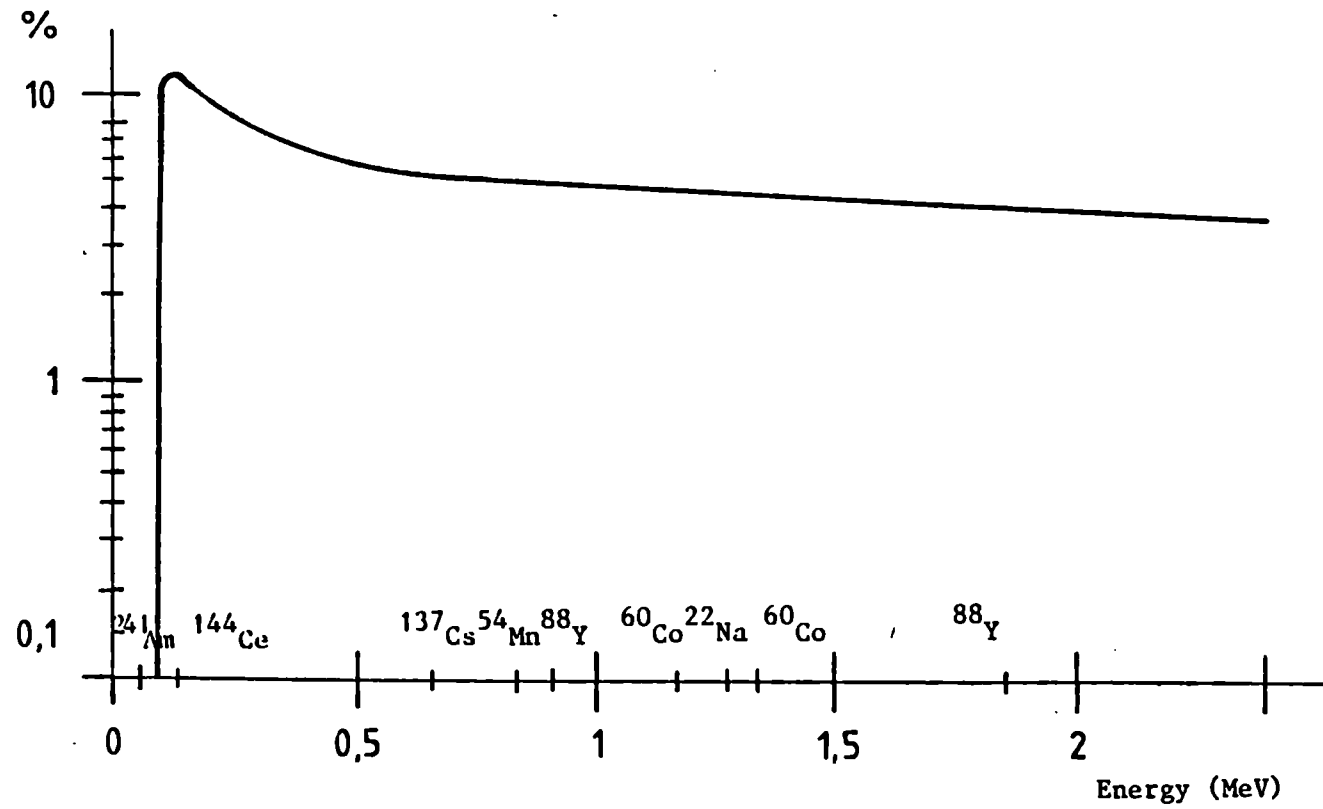


Figure 2 : Efficiency of a  $\gamma$ -detector (NaI)

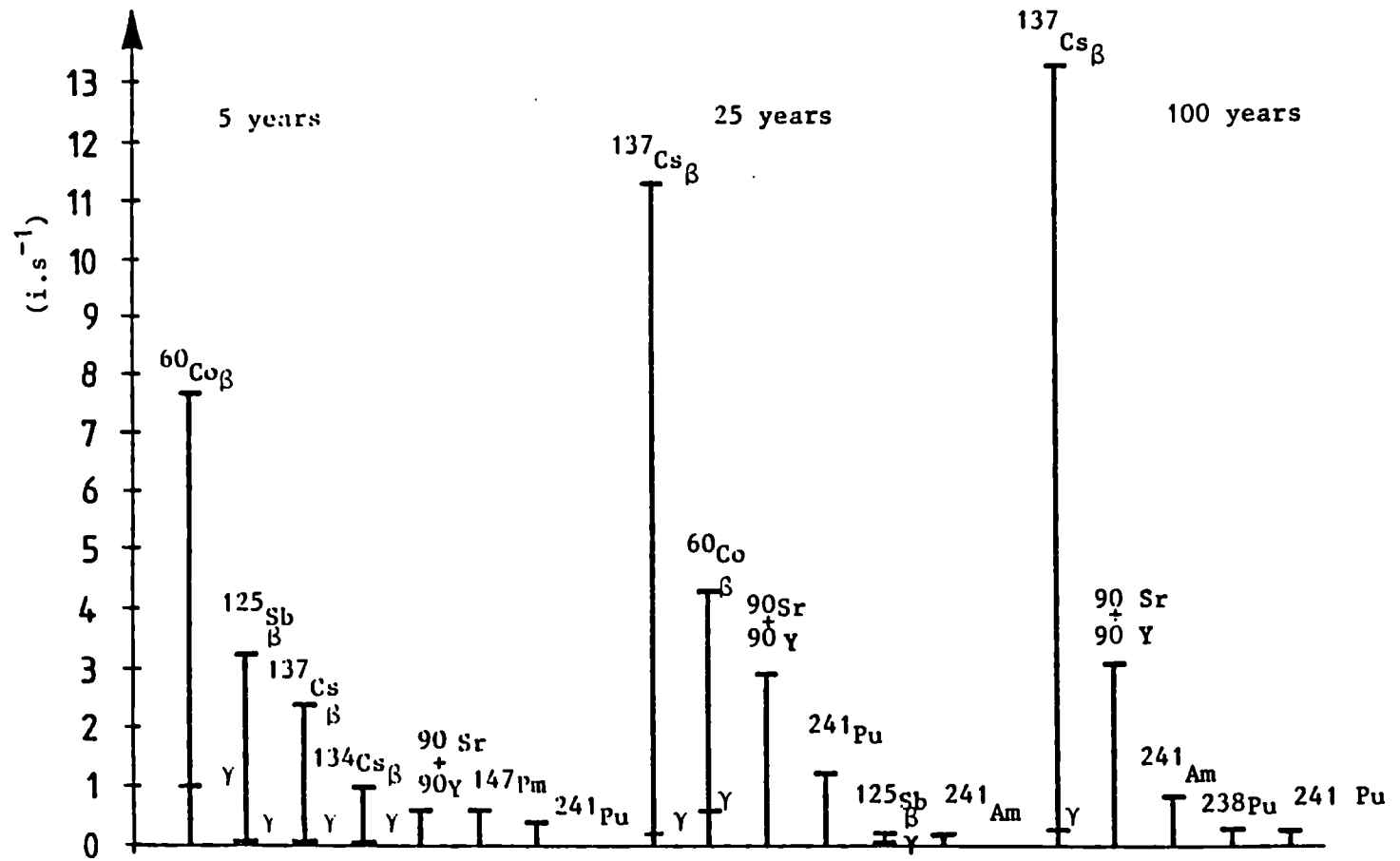


Figure 3 : Surface contamination : Count rates of the major radionuclides  
 The total surface activity is standardized to  $1 \text{ Bq.cm}^{-2}$  for each reactor  
 cooling time

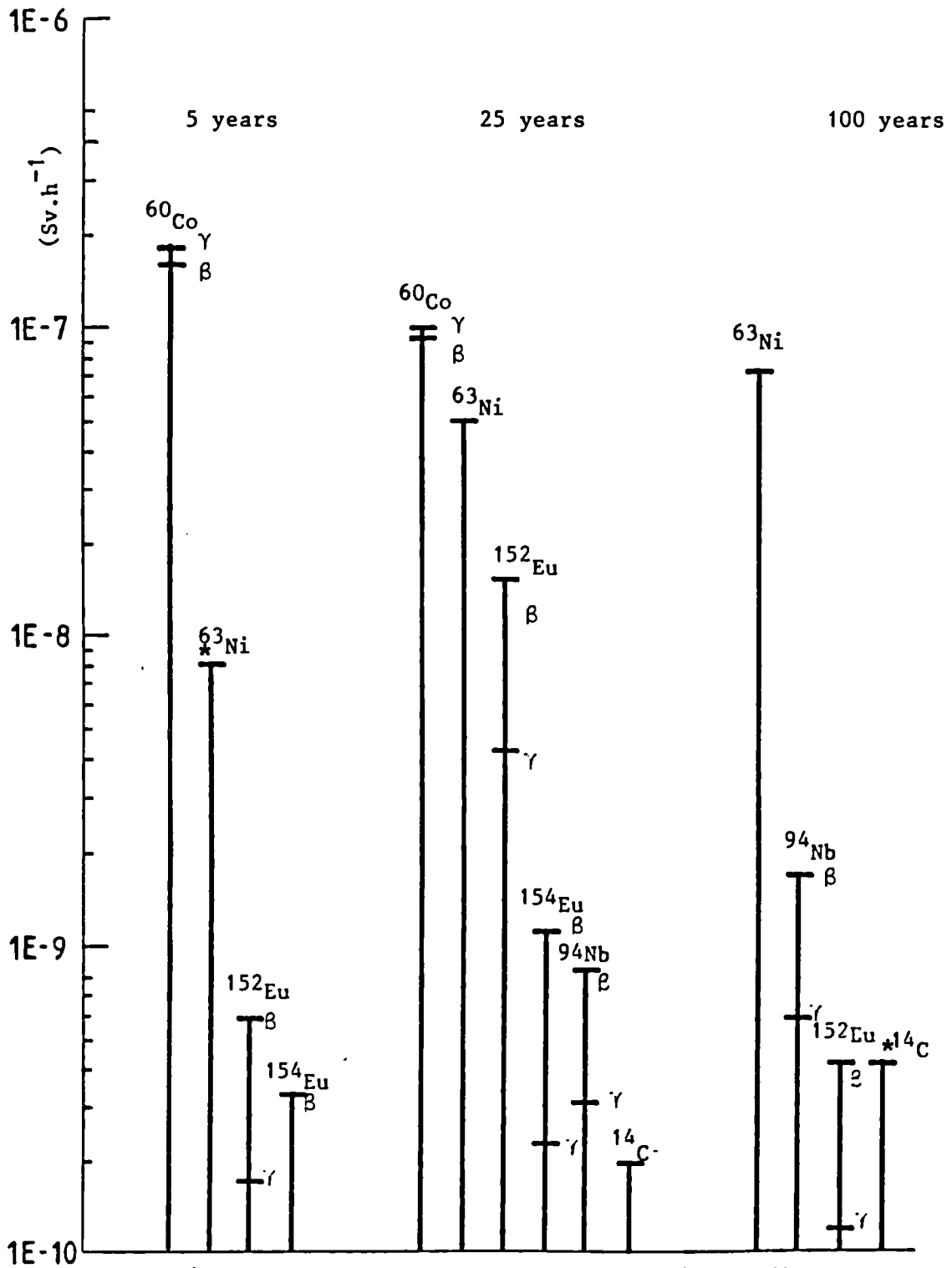


Figure 4 : Dose rates near activated steel (10 % Ni) for different reactor cooling times

\*this dose rate in air is due to very low-energy electrons which give non significant doses in tissue

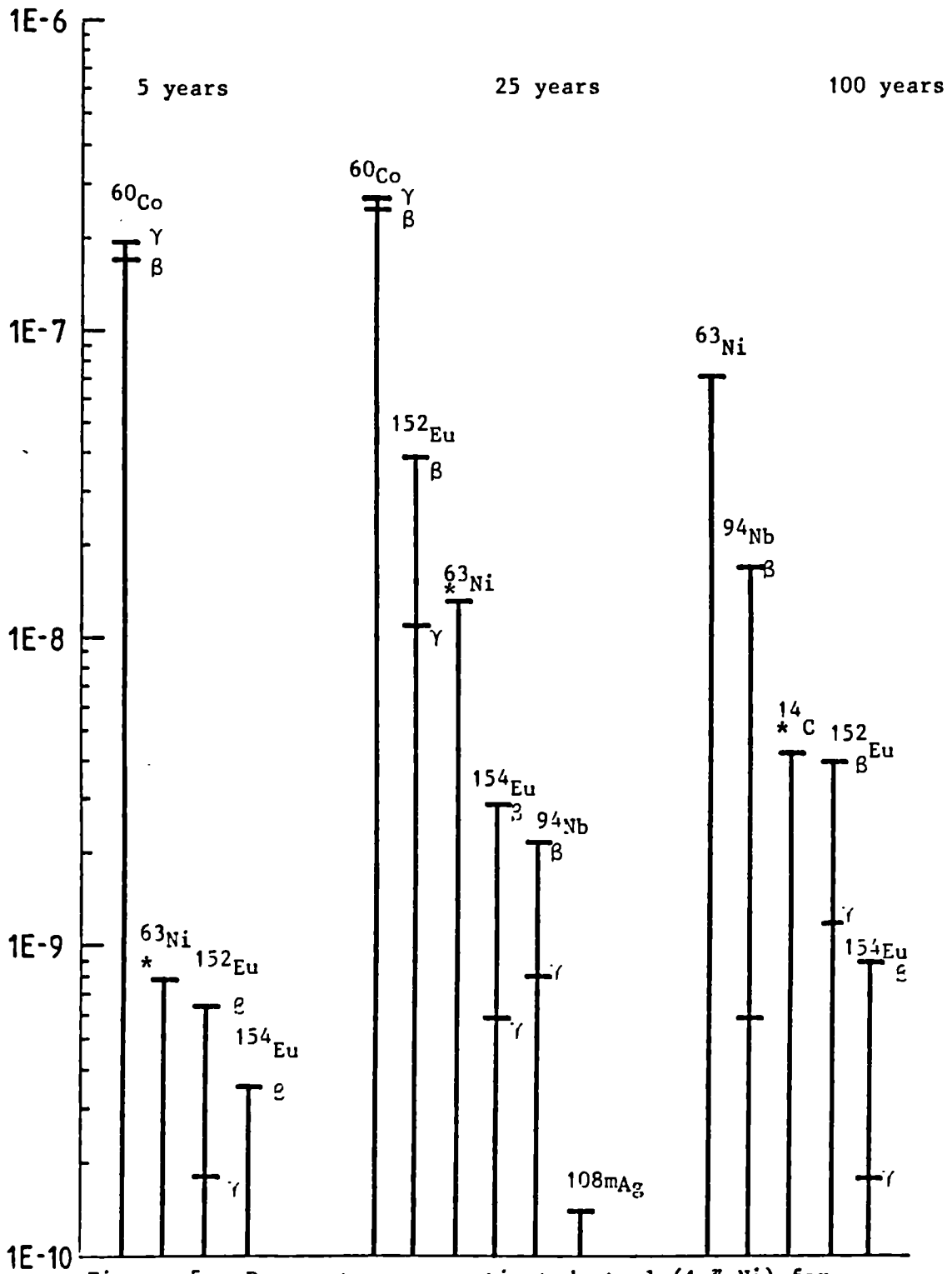


Figure 5 : Dose rates near activated steel (1 % Ni) for different reactor cooling times

\*this dose rate in air is due to very low-energy electrons which give non significant doses in tissue

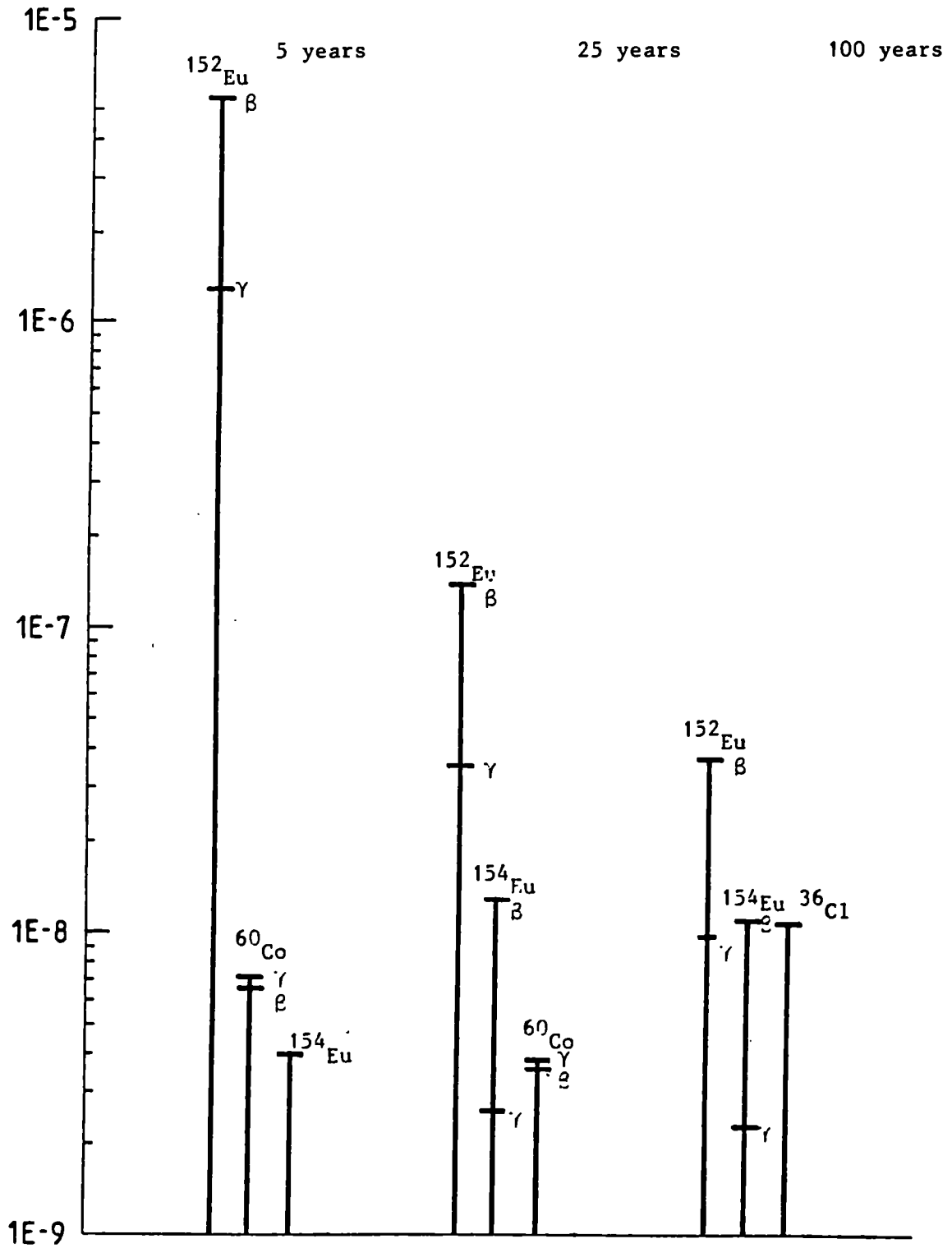


Figure 6 : Dose rates near activated reinforced concrete for the different reactor cooling times.

## APPENDIX A

### DECOMMISSIONING OF N.V. 'OTTO HAHN'

#### A.1 Nuclear System And Control Area

##### A.1.1 Introduction

After 10.33 years of operation, according to 1684 days of full-power operation, the reactor of N.V.'Otto Hahn' was shut down. In May 1979 the fuel elements were dismantled and stored in the vessel's own service pool. In October 1979 the elements were shipped for reprocessing. No fissile matter was left on board the vessel.

After reinstallation of the controlrods, controlrod guides and corner elements, designed as measuring cages, the pressure-vessel was tightly closed. Fuel cladding tubes of corner elements had been removed before.

In the meantime the decommissioning was planned. In spring 1980, the decommissioning concept was presented to the authorities and the licensing procedures began.

After decommissioning authorization had been decreed in December 1980, decommissioning operations immediately commenced. Preparatory operations, restricted by former operation licenses and not interfering with closed nuclear circuits, had already started in June 1980. Thus, from reactor shut-down to decommissioning operation commencement, a decay period of 21 months had passed.

Decommissioning work was finished in June 1982.

Because of the positive results of release measurements, carried out by the vessel's own staff in charge of radiation protection, surveyors (TÜV-Norddeutschland) and authorities (Amt für Arbeitsschutz) and verified by final inspections by an independent surveyor (Physikalisch-Technische Bundesanstalt, Braunschweig), the vessel's release for unrestricted, conventional utilization, inclusive the former nuclear plant area, was declared.

All operation licenses and management regulation related to Atomic Law (AtG) and Radiation Protection Regulations (StrlSchV), were finally suspended by decree in September 1982.

During decommissioning of the 'nuclear area' the elaborated concept, adapted to the decommissioning variant,

- 'Terminated protective storage of the reactor pressure-vessel, including internal structures and shield tank, and of all contaminated components of nuclear auxiliary systems with the option to carry out check-up researches.'

which finally had been chosen from various decommissioning alternatives, flexibly framed the decommissioning procedures by rather establishing protective principles concerning dismantling and storage procedures but permitting detailed from case to case planning to be adapted to growing experience.

Similar preconditions can be postulated with regard to the measuring techniques and procedures implied in the concept.

With respect to the lack of technical know-how on measuring and evaluation of low level activities in the projected extension, the decommissioning license also granted flexibility, but obliged to elaborate acceptable measuring and release procedures and strategies to become compulsory for further proceedings after having been introduced to and approved by the licensing authorities.

#### A.1.2 Plant-Specific Conditions

The nuclear plant of N.V.'Otto Hahn' was characterized by various particular items considerably influencing nature and distribution of activity or kind and quantity of activated or contaminated components and structures and thus the framing of decommissioning procedures as well.

Particulars to be taken into consideration were:

- structure of the biological shield
- construction of the reactor
- extent, arrangement and separation of the nuclear area
- naval design of control area structures

Structure of the biological shield

In its lower part the biological shield consisted of a multi-layer shielding - the neutron shield tank -, alternately formed by water and cast-iron logs; in its upper part, where sufficient neutron shielding was granted by the primary-water level above core, of a solid dome-shaped shield, built of cast-iron rings and plates. Behind primary-shielding neutron and gamma radiation level at full-power performance of the reactor did not exceed 65 mrem/h (n-component) respectively 50 mrem/h (gamma-component) at most exposed locations and permitted visitation of the containment at any time.

Zones of activation were confined to the pressure-vessel, shield tank and their internal components, forming a single structural unit. The quantity of material, the activity of which was predominantly caused by activation or by activation as well as contamination was adequately deficient.

This condition particularly favoured the choice of the decommissioning variant:

'Dismantling and protective enclosure of the reactor pressure-vessel and the shield tank in compact unit'.

The problem to measure and evaluate activity caused by activation in compliance with release requirements consequently did not derive during decommissioning of 'Otto Hahn'. Because of the particular structure of the biological shield activated concrete had not to be disposed off.



### Construction of the Reactor

N.V.'Otto Hahn' was equipped with an 'Advanced Pressurized Light-Water Reactor (FDR)' with integrated steam-generating and pressurizing systems.

Steam generators, primary-water circulation pumps and the pressurizing system were integrated components of the enclosing pressure vessel and according to the decommissioning concept were to be dismantled and protectively stored in compact unit.

Exterior components flushed or contacted by primary-water and consequently high-grade contaminated were confined to the following auxiliary circuits and systems:

- Primary-water Cleaning Circuit
- Blow-down System
- Primary Feeding System
- Pressure and Level Measuring System
- Sealingwater System
- Sampling System

With exception of a minor sampling circuit section installed in the auxiliary plant compartments, the systems were arranged in the containment.

Excepting blowoff pipes of the blowdown system (150 mm), standard diameters of piping of all auxiliary primary systems did not exceed 40 mm. The quantity of contaminated material deriving from dismantling auxiliary primary circuits and systems had to be considered to be relatively deficient. Consequently all exterior primary-system components were provided to be disposed off as radioactive waste.

### Extent, Arrangement and Separation of the Nuclear Area

During the vessel's routine performance the nuclear area (control area) was confined to the following 3 main compartments:

- Containment Compartment
- Aux. Plant Compartment (Aux. Nuclear Installations, Workshops, Laboratories, Shower and Dressing Rooms, Decont-Shop)
- Service-Pool Compartment

In order to carry out decommissioning operations the nuclear area was extended by the 'outer control-area' then additionally comprising hedge no.4 serving as storage compartment, the main-deck in the vicinity of hedge no.4 and the reactor-deck.

Furthermore the wing and doublebottom tanks, enclosing the aforementioned nuclear compartments, and the the ballast double-bottom tank 0801 underneath the vessel's engine room had to be considered subject to decommissioning efforts.

Pursuant to decommissioning concept, the total surface of compartments and remaining installations subject to measurements amounted to approx. 18 000 m<sup>2</sup>. The quantity of the concerned structures came up to 3 000 Mg.

## Naval Design of Control-Area Structures

Contrary to land-site nuclear power plants the nuclear operation compartments and areas of N.V.'Otto Hahn' were designed as naval steel structures. Only the secondary shielding, arranged in the SBR, and the service-pool shielding were constructed as steel enforced concrete structures.

The surface area of concrete structures, subject to release measuring procedures or protective disposal, amounted to approx. 1 000 m<sup>2</sup> and their total quantity to approx. 1 000 Mg. The activity of the concrete structures had been caused by contamination only.

The local effects of operational conditions (pressure, temperature, phase of contaminating media, flow-velocity) in circuit sections or subsections on chemical compound, adhesion and homogeneity of layer thickness of contaminated incrustations were not investigated in detail.

### A.1.3 Radiological Initial-state conditions

The initial-state survey, carried out previous to decommissioning of the nuclear area of N.V.'Otto Hahn' was confined to local dose-rate monitoring in the various control-area compartments and sections and contact dose-rate measurements of particular single components. Measurements should mainly support the elaboration of the 'Man-Rem-Concept' by preliminary assessment of the collective doses probably to be accumulated during various decommissioning proceedings.

The activity inventories caused by activation and contamination were estimated in lump by a study worked out by 'Nuklear-Ingenieur-Service (NIS)' in 1978.

The activity inventory by activation was computed by a program regarding time of operation (10.33 years) and decay (1 year).

In order to assess the lump sum activity inventories caused by contamination the following characteristic grades of contamination were assumed:

Class 1:	E+ 1 $\mu\text{Ci}/\text{cm}^2$ (370 000 Bq/cm <sup>2</sup> ) for components contacted by primary-water
Class 2:	E- 1 $\mu\text{Ci}/\text{cm}^2$ ( 3 700 Bq/cm <sup>2</sup> ) for components contacted by less contaminated media
Class 3:	E- 5 $\mu\text{Ci}/\text{cm}^2$ ( 0.37 Bq/cm <sup>2</sup> ) for components contacted by liquids of shielding-water grade of contamination

For more detailed characterization

Class 4:	E- 5 $\mu\text{Ci}/\text{cm}^2$ ( 0.37 Bq/cm <sup>2</sup> ) for components and structures contaminated by airborne activity
----------	--

was additionally introduced later.

In the aspect of feasibility of decontamination the following groups were specified regarding the origin of activity:

- Group 1: Components, structures and materials predominantly activated
- Group 2: Components, structures and materials activated as well as contaminated
- Group 3: Components, structures and materials exclusively contaminated by contact to active liquids
- Group 4: Components, structures and materials exclusively contaminated by airborne activity.

Contaminated liquids were specified by 2 activity concentration grades:

- Class 1: Activity concentration exceeding release limits (primary-, shield tank- and waste-water)
- Class 2: Activity concentration presumed to be lower than release limits (precipitation water of 'outer control area')

and especially classified to

- Group 5: Liquid materials.

All components and structures classified in group 1 and 2 were part of the reactor pressure vessel or the shield tank and were subject to protective disposal. Feasibility of decontamination could be assumed for components and structures of class 4 and eventually class 3 only.

For the elaboration of measuring and release procedures a characterization of components and materials with regard to nature and grade of activity only is not sufficient.

In order to settle further classification characteristics, the measuring problems, predominantly determined by the regulations of the 'StrlSchV' to become decisive for unrestricted release, had to be defined first.

As authorities announced, that the release of solid materials would predominantly depend on observance of limit values pursuant to § 64 StrlSchV, the measuring problem was to evaluate the specific alpha-, beta-surface contamination averaged over an area of 100 cm<sup>2</sup>.

With aspect to measuring and release procedures the components and structures were additionally classified with regard of component and plant specific characteristics, such as geometry, mobility and local background level.

The following subgroups were determined:

- Group 1: Components and structures with complex geometries
- Group 2: Components and structures with plain geometries, mobility restrained by construction or weight
- Group 3: Mobile Components or structures with plain geometry.

A choice of measuring instruments can only be made after determination of the technical measuring requirements. With respect to radiation sensitivity, requirements depend on component and plant specific (self-absorption with coated surfaces, local gamma-background) and legal conditions (legal reference of release authorization) as well. Requirements with respect to detection efficiency are determined by the release limit values approved by the licensing authorities mainly, but additionally depend on technical conditions such as the maximum measuring area accepted by authorities and the sample area or quantity to be measured.

#### A.1.4 Activity Detection and Monitoring Complying with Decommissioning Concept Requirements

##### A.1.4.1 Measuring Problems According to the Decommissioning Concept

The decommissioning concept of the nuclear area of N.V. "Otto Hahn" provided only a few categories of components and material to be released for unrestricted re-utilization or disposal (see table A 1: Classification of Components and Material). Only these categories were considered subject to contamination measurements and activity evaluation.

Referring to solid matter, the categories 3.2.2, 3.4.2 and 3.4.3 (components and structures of plain geometry, activity caused by contact with low-grade activity media) were concerned. Among the liquids, existent and deriving from decommissioning, unrestricted discharge into surface waters was provided for precipitation water (category 5.2), collected in the 'outer control area' and stored in the vessel's particularly specified storage-tanks, only.

In order to elaborate measuring procedures for solid materials, reference to the regulations and limit values pursuant to § 35 or § 64 StrlSchV for release authorization was presumed.

For approval of unrestricted discharge of precipitation water into the river 'Elbe' the regulations and limit values of § 46 StrlSchV, already valid during the vessel's operation, were assumed to be decisive.

Specific measuring problems with regard to component and material specification can then be defined as follows:

For solid matter (categories 3.2.2, 3.4.2 and 3.4.3):

- evaluation of the specific surface activity ( $\mu\text{Ci}/\text{cm}^2$ ;  $\text{Bq}/\text{cm}^2$ ) averaged over  $100 \text{ cm}^2$
- analysis of the alpha-, beta-compound of contamination emitters

For liquids (category 5.2):

- evaluation of the activity concentration (  $\text{Ci/m}^3$  ;  $\text{Bq/m}^3$  )  
analysis of the nuclide compound

Reference to activity concentration limits pursuant to § 47 and § 4 para.4 s.1 no.2e StrlSchV for release approval was not applied for in the decommissioning concept presented to the authorities.

#### A.1.4.2 Elaboration of Measuring Procedures

All components and structures of the categories 3.2.2; 3.4.2 and 3.4.3 were characterized by their plain geometry and consequently were appropriate to direct, surface covering measuring and monitoring procedures.

After dismantling, components and structures of category 3.4.3 were mobile and could be brought to low-level background areas to be directly measured in compliance with release requirements. Installation of a measuring-site on deck of the vessel remote from interfering sources was provided.

Due to design and construction or to weight, most of the components and structures of category 3.4.2 and the fuel-element storage-pool were immobile.

Regarding background interferences, the decommissioning concept provided to postpone commencement of direct release measurements for components of category 3.4.3 until interfering sources had been removed and the background level sufficiently decreased.

The fuel-element storage-pool was considered to be submitted to release measurements immediately after decontamination. Direct release measurements as well as decontamination surveillance by direct measuring procedures were not applicable for reason of background interference. Indirect measuring by sampling was planned instead.

For sampling a procedure by electrolytic abrasion of the surface, developed by the 'Eidgenössisches Institut für Reaktortechnik (EIR)', was intended to be employed. With this procedure samples, quantified in weight, area and depth of abrasion, are gained and, dissolved by acid agents, can be analyzed chemically as well as radiologically.

Following the decommissioning concept, evaluation of depth of activity intrusion and analysis of the activity compound should serve decontamination surveillance and optimization only and was not stipulated to support release procedures.

For liquids of precipitation water nature and quality homogeneous activity distribution can be presumed. Consequently appliance of an indirect measuring procedure by sampling was provided for evaluation of the activity concentration and for analysis of the prevailing nuclide compound.

A.1.4.3 Chosen Measuring and Sampling Devices - according to the decommissioning concept

Direct Measurement of the Alpha-, Beta-Contamination of Components Category 3.4.2 and 3.4.3

For components and structures of these categories the decommissioning concept proposed the employment of a predominant alpha-, beta-sensitive contamination monitor. The monitor possessed sufficient gamma-sensitivity to detect contamination coated by thin inactive layers (paint, rust):

Manufacturer	:	Herfurth
Type	:	Minikont
Detector		
Type	:	Proportional Counter
Eff. Window Area	:	200 cm <sup>2</sup>
Radiation Sensivity	:	alpha, beta (gamma)
Background Counting	:	12 Cps (approx.)
Monitor		
Indication	:	Cps (analogous)
Delay	:	0 to 20 s (gliding)
Alarm-threshold	:	adjustable

Indirect Measuring of Components Category 3.2.2

- Storage Pool -

Sampling Device

The sampling device of the EIR mainly consists of a rotating sampler carrying 4 small synthetic sponges to be soaked with an electrolyte agent previous to sampling. The device is fixed to position by remote control.

By remote control, sponge 1 with the largest diameter is lowered to the surface until contact. After a certain sampling time, according to the intended depth of abrasion, the sampler is lifted and turned. Next, sponge 2 with a smaller diameter is lowered into position. The procedure continues until all 4 sponges are loaded. Border effects are avoided by gradually decreasing the diameter of the applicated sponges. Thus a step-shaped crater is abraded. Once loaded, the sponges are dissolved by acid agents. The solution can be analyzed by single nuclide detection. The relation between matter transfer per square unit, resp. depth of abrasion and time of DC-supply has to be calibrated by test series in advance. Oxidic surface incrustations are abraded by reversing polarity of the DC-supply. Achievement of abrasion is indicated by alternation of the current-voltage relation.

Measuring Equipment for EIR-Sample Activity Evaluation and Analysis

Primarily the samples obtained by the EIR-procedure should be evaluated and analyzed by gamma-spectroscopy for the purpose of decontamination surveillance and optimization. Therefore it was intended to install a GeLi-detector with multichannel-analyzer and counter remote from nuclear or contaminated zones outside the control-area.

### Measuring Devices for Precipitation-Water Activity Evaluation and Analysis (Material Category 5.2)

Measurements of precipitation-water samples complying with release requirements were routine during the vessel's operation. Sampling procedures, release limits and the measuring devices intended for application had already been approved by the licensing authorities. Sample processing (concentration by evaporation) and evaluation and analysis by GeLi-spectroscopy should be performed in the laboratories of GKSS.

#### A.1.4.4 Measuring Strategies - according to the Decommissioning Concept

##### Measuring Strategy for Components Category 3.4.2 and 3.4.3

Components of the categories 3.4.2 and 3.4.3 should be monitored in a narrow grid. For restricted areas not adapted to monitoring the measured values of adjacent areas should be considered valid if, proven by the results of wiping tests taken before and after decontamination, similarity of contamination resp. decontamination efficiency could be verified.

Because indirect measurements (by sampling) of the fuel-element storagepool should serve decontamination surveillance and optimization primarily, a number of approx. 60 sampling positions, regularly spread over plain and curved tank surfaces, was considered sufficient to obtain ample information on grade of contamination, variety of distribution, prevailing nuclide compound and depth of activity penetration afore and after each step of decontamination performed. Only after proof of sufficient exactitude and applicability this indirect measuring procedure should be employed for release measurements also. A strategy settling sample positions and the requested number of samples was not introduced to the authorities by the decommissioning concept.

Since experience during the vessel's operation proved homogenous activity distribution of the precipitation-water collected in 2 connected ballast tanks, drawing of only one 10 l-sample was presumed sufficient. The quantity of precipitation collected for each discharge amounted to approx. 250 - 350 m<sup>3</sup>.

#### A.1.5 Decommissioning Authorization Decree

After thorough consideration and appraisal of the filed decommissioning concept by experts of TÜV-Norddeutschland decommissioning approval was decreed by the licensing authorities (Amt für Arbeitsschutz -AfA- der Behörde für Jugend, Arbeit und Soziales der Freien und Hansestadt Hamburg) on the 5th of December 1980.

Besides the issues concerning radioprotective measures and dismantling, cutting, packing and conveying procedures, those referring to measuring and release procedures too were essentially accepted. Additional obligations to be observed were mainly accepted from the expert certificate.

The final determination of release limit-values and obligations concerning the defined categories of components, structures and materials was most essential for further proceeding.

A.1.5.1 Release Limit-values and Obligations - pursuant to decommissioning license

Apprehending the problems arising in concern of low-level activity detection and evaluation of components, structures and materials with often complex geometries and acknowledging that N.V.'Otto Hahn' was the very first nuclear power plant in the Federal Republic of Germany to be decommissioned and released for unrestricted conventional utilization, the obligations referring to release and measuring procedures and activity transfer documentation were formulated as follows:

Obligation 13

The activity of all the plant components, structures and materials dissembled during decommissioning has to be evaluated and recorded as precisely as possible.

- Parts and materials, the activity concentration and surface contamination of which unequivocally does not exceed the limit values of § 4 para.4 s.1 no.2e resp. § 64 (app. IX clm.4) StrlSchV, have to be disposed off as inactive waste or scrap without further restriction.
- Parts and materials, the activity concentration or surface contamination of which does exceed the afore mentioned limit-values, have to be disposed off as radio-active waste as far as they are not provided for re-utilization obliged to management restrictions.
- For parts and materials, the activity concentration or surface contamination of which cannot be evaluated unequivocally, proposals, concerning alternative measuring and release procedures and strategies adapted to support release decisions, have to be elaborated and put forward to the authorities.

Obligation 14

A person, responsible for the proper application of measuring procedures, the reliable observance of the limit-values pursuant to § 4 para.4 s.1 no.2e and § 64 StrlSchV, and the correct assignment of parts to be classified according to the categories of obligation 13 and the values of app. IX clms. 2 to 4, has to be nominated.

Obligation 15

Since with N.V. 'Otto Hahn' for the first time decommissioning and dissembling of a nuclear power plant is undertaken in the FRG, great effort has to be made on extensive and accurate documentation of the proceedings.

Beside unforeseen incidents, especially concerning work proceedings, local dose-rates, individual doses, activity evaluating procedures, decision processes, organization measures (surveillance) and the behavior of the staff in charge of decommissioning have to be regarded. After accomplishment of the work licensed by decommissioning authorization a final report has to be elaborated to be issued to the licensing authorities.



#### A.1.5.2 Measuring Problems Pursuant to the Decommissioning License

Obligation 13 requested an evaluation 'as accurate as possible' of the surface contamination as well as the activity concentration of all disassembled components, structures and materials. This had the consequence that measuring problems and expenditures exceeded the extent proposed by the decommissioning concept by far (see table A.1). Components, structures and materials of almost all the defined categories had to be submitted to measuring now.

For exclusively contaminated, plain surface components of the categories 3.2.2, 3.4.2 and 3.4.3 provided for unrestricted release, the obligation to observe concentration limits additionally itself does not alter measuring problems, as the activity concentration can be calculated from the results of surface contamination measurements.

The formulation of obligation 13 para. 3 however, already reflecting the necessity of a radiologically controlled decommissioning of the secondary plant, extended measuring problems by the request to propose measuring procedures and strategies adapted to evaluate the activity concentration of components and structures of complex geometries with sufficient exactitude.

This obligation is additionally emphasized by obligation 15 requesting extensive and accurate documentation of the activity transfer by nature (nuclide compound) and quantity (total activity inventory).

Thus further measuring procedures and strategies exceeding the original concept had to be developed. The adapted measuring devices had to be tested with respect to detection and evaluation requirements.

Because of close contact to the authorities and surveying experts, the measuring procedures and strategies, once developed and approved by authorities, could always be improved by adaption to the growing standard of technical know-how.

#### A.1.5.3 Official Appraisal of the Measuring Procedures Proposed by Concept

The measuring procedures and strategies proposed by the decommissioning concept, concerning components, structures and materials of category 3.2.2, 3.4.2, 3.4.3 and 5.2 were essentially approved by the licensing authorities.

Direct measuring or monitoring of plain surface components and structures had to be performed by a surface covering measuring grid.

Referring to detection sensitivity, the universal application of alpha-, beta-, (gamma)-sensitive contamination monitors for coated and pure metallic surfaces as well was considered not sufficient.

Employment of alpha-, beta-sensitive detectors for pure metallic and beta-, gamma-sensitive detectors for coated surfaces was favoured and requested by the authorities.

The decision, whether the measuring procedures proposed for components category 3.2.2 (FE-storage pool) in context with strategies still to be elaborated would comply with release requirements, was postponed until further knowledge in regard of obligation 13 para 3 had been achieved by the results of the surveillance measurements in the progress of the storage-pool decontamination.

The proposed and already practiced procedure for the release of precipitation-water was approved of without further restriction.

#### A.1.5.4 Additional Measuring Problems Pursuant to Decommissioning License

Additional measuring problems stipulated by obligation 13 of the decommissioning license almost exclusively refer to measuring of components with complex geometries, originally not presumed subject to measuring at all.

In order to evaluate the activity of such components and structures only indirect measuring procedures by sampling were assumed feasible. The sample activity had to be analyzed by gamma-spectroscopy. As these measurements did not have to comply with release criteria, but should serve evaluation and documentation of the activity transfer and improvement of the know-how in the field of low-level activity detection and evaluation of complex geometry components only, the requirements on detection efficiency of the measuring devices to be employed were not stipulated by legal limit values but by the prevailing sample activity.

To avoid unnecessary sampling and measuring expenditure, the formulation 'as accurate as possible' had to be considered with the authorities before the elaboration of sampling strategies.

Considering plant and operation specific conditions, influencing the grade and distribution of the prevailing activity, the following sampling strategies to evaluate the activity inventory of components not to be released were settled in accordance with experts and authorities:

- 5 to 10 samples have to be taken from each circuit or system of very low-grade contamination when only small deviations of activity distribution is proved. (Components and structures of categories 3.2.1 and 3.4.3.)
- 15 to 20 samples have to be taken from each circuit or system of higher-grade contamination with significant deviation of activity distribution. The representative sampling positions have to be determined by local dose-rate monitoring. (Components and structures categories 3.1)
- The activity inventory of a circuit, system or of subsections is calculated from surface area and corresponding average surface contamination given by the sample measuring results.
- The samples are taken by thermal or mechanical cutting methods and are prepared to standardized shape.
- The sample activity is evaluated and analyzed in a calibrated measuring geometry by single nuclide GeLi-detection.

### A.1.6 Applied Measuring Procedures and Devices

Pursuant to the decommissioning license and after considering the consequences stipulated by the obligations 13 to 15 with experts and authorities the following measuring procedures, strategies and devices were actually applied for the various measuring problems.

#### A.1.6.1 Measurements for Compliance with Release Criteria

Measuring of Components and Structures Categories 3.4.2, 3.4.3 and 3.2.2

##### Measuring Procedure

For all plain surface (simple geometry) components and structures direct measurements by a continuous measuring method were applied, either immediately after decontamination at a low-level background measuring site or after background depression by removal of interfering sources.

The fuel-element storage-pool, originally proposed to be released with reference to sample measuring results, was submitted to direct measuring procedures too. Indirect measurements, applied for decontamination surveillance, had proved to be time-consuming and did not comply with release criteria.

##### Measuring Devices

Depending on surface quality and prevailing gamma-background level alpha-, beta- or beta-, gamma-sensitive contamination monitors were employed for final release measurements.

For pure metallic surfaces and/or raised gamma-background the following monitor was used:

Manufacturer	:	Berthold
Type	:	LB 1210 C
Detector		
Mode, Type	:	Proportional Counter, MZ-100
Eff. Window Area	:	100 cm <sup>2</sup>
Counting Gas	:	Liquid Gas (Butane a.o.), refillable
Radiation Sensivity	:	alpha, beta
Background Counting	:	5 Cps (approx.)
Rate (alpha-, beta-, Plateau)		
Calibr. Data	:	f = 1*E-6 μCi/cm <sup>2</sup> , Cps
(Manufact. Declaration)	:	= 0.037 Bq/cm <sup>2</sup> , Cps
Monitor		
Indication	:	Cps (analogous)
Delay	:	2 and 20 s (adjustable)
Alarm Threshold	:	audible (adjustable)

For coated surfaces, low-level gamma-background the following monitor was used:

Manufacturer	:	Berthold
Type	:	LB 1210 B

Detector  
Mode, Type : Proportional Counter, BZ-100XE-P  
Eff. Window Area : 100 cm<sup>2</sup>  
Counting Gas : Xenon, non-refillable

Radiation Sensivity : beta, gamma  
Background Counting : 8 Cps (approx.)  
Rate (beta-Plateau)  
Calibr. Data : f = 2\*E-6 μCi/cm<sup>2</sup>, Cps  
(Manufact. Declaration) = 0.074 Bq/cm<sup>2</sup>, Cps

Monitor  
Indication : Cps (analogous)  
Delay : 2 and 20 s (adjustable)  
Alarm Threshold : audible (adjustable)

By application of the LB 1210 B monitor prevailing gamma-contamination, coated by thin layers of rust or paint, can be detected with restricted but sufficient efficiency. On the other hand, results obtained by LB 1210 B application are influenced by gamma-background irradiation more significantly than by employment of the LB1210 C.

A radiation field-intensity of 0.1 mrem/h causes a counting-rate elevation of 40 Cps by irradiation from the monitor front-side and of 70 Cps by irradiation from the detector window.

For decontamination surveillance the alpha-, beta- (gamma)-sensitive contamination monitor Herfurth 'Minicont' specified afore was employed.

#### Measuring Strategy

Surveillance measurements covered the surface of components and structures completely. Each single measuring position was marked by circumscribing the detector window with an ink-marker. If a single measuring result exceeded a counting rate corresponding to a threshold contamination of 8 E6 μCi/cm<sup>2</sup> (0.296 Bq/cm<sup>2</sup>), the actual rate was entered into the area concerned. These areas were submitted to 2nd-, 3rd- and eventually 4th-step decontamination. If sufficient decontamination apparently could not be achieved, the areas marked by entries were cut out or the whole component was considered radioactive waste.

Release measurements by application of the LB 1210 B and C monitors commenced only, if at all measuring positions the counting rates detected by surveillance measurements did no longer exceed the calibrated threshold values. By release measurements approximately 25 % of the total component and structure surfaces were controlled. Areas with former entries were individually measured by applying a narrow, sometimes surface covering measuring grid.

Release measurements were performed by officially nominated members of the vessel's staff. After verification by further measurements carried out by experts of the 'TÜV' and 'AfA' all components and structures unequivocally not exceeding the calibrated threshold counting rates at each measuring position were provisionally released.

Release as well as surveillance measurements of mobile components were performed at a measuring-site installed on deck of the vessel in a low-level background area. Before the components could be brought there it had to be proved by wiping tests that they were 'free of non-adhesive contamination'.

As proposed by concept, release measurements of immobile components and structures commenced at an advanced stage of decommissioning after removal of dismantlable sources and only after sufficient decontamination of the vicinity had been achieved.

#### Measuring of Precipitation-Water Activity Concentration (Material Category 5.2)

For the precipitation-water the release limit-values measuring procedures and strategies valid during the vessel's operation had been approved by the authorities.

#### Sampling and Measuring Procedure

Preceding each average discharge of 250 to 300 m<sup>3</sup> of precipitation-water, a 10 l-sample had to be prepared by 3 to 5 draws from the 2 ballast storage-tanks. After evaporation to a calibrated standard volume of 1 l, the sample activity was evaluated and analyzed by single nuclide GeLi-detection at the laboratories of GKSS with the following instrument:

Detector	
Manufacturer	: ORTEC
Mode	: GeLi
Type	: Vertical Dipstick
Dimension	: 3"-Diameter
Efficiency	: 20 % rel.
Resolution	: 2 keV at 1.33 MeV
Analyzer	
Manufacturer	: Nuclear Data
Type	: ND 100

#### A.1.6.2 Measurements Complying with Activity Transfer Assessment and Documentation Requirements

Decontamination Surveillance Measurements (Fuel-element storage-pool, component category 3.2.2)

Measurements were performed as provided by the decommissioning concept.

#### Sampling Device and Procedure

For sampling the EIR-sampler was employed. The application of this remote-controlled sampling device was of advantage in areas with elevated dose-rate. The very low sample activity, especially when surface contamination approached release limits with the progress of decontamination, required extremely high detection efficiency.

#### Measuring Procedures, Measuring Device and Measuring Geometry

The 50-ml samples obtained by the EIR-procedure were evaluated and analyzed by a GeLi-multichannel analyzer installed in a low-level background area of N.V. 'Otto Hahn'. At an early stage of decontamination, the measuring time to achieve a sufficient lower limit of detection was 10,000 s and more even for samples taken before 1st-step decontamination. In spite of the extremely high time expenditure the procedure was considered suitable for decontamination surveillance measurements. The release measurements were postponed and should be performed by direct contamination monitoring at the end of decommissioning.

### Measuring Strategy

Considering the high time consumption the number of positions to be sampled was reduced to 10 to 20 per decontamination step. During decontamination surveillance actually approx. 60 samples were taken and analyzed. The results of sample measurements indicated very inhomogeneous distribution of the residual contamination. Vast areas of the pool tank had to be submitted to finishing treatment by locally employed mechanical decontamination procedures.

Considering high time expenditure and insufficient exactitude of the sample measuring procedure, application of direct measurements was provided for the release of the fuel-element finally.

### Activity Evaluation and Analysis of Dissembled Contaminated Components - pursuant to obligation 13 and 15 -

Pursuant to obligations the activity of all dissembled components had to be evaluated and analyzed as 'accurately as possible' with reference to grade of surface contamination, activity concentration, activity inventory and nuclide compound.

### Sampling and Measuring Procedures and Measuring Devices

Because most of the components to be measured had complex geometries (category 3.1; 3.2.1 and 3.4.3), the application of indirect measuring procedures by sampling was feasible only. Destructive sampling was carried out by thermal or mechanical cutting. The samples were prepared to a calibrated standard shape. For sample activity evaluation and analysis the GeLi-multichannel analyzer of GKSS was employed.

### Measuring Strategy

As the measurement pursuant to obligation 13 para. 2 and 3 served activity evaluation of dissembled parts subject to restricted disposal only, the statistical exactitude of the results was of less importance.

Presuming that activity distribution in circuits and systems of unique operation conditions is subject to probability law, the number of samples noted in item A.1.5.4 was considered sufficient. Altogether approx. 150 sample measurements were carried out.

### A.1.7 Calibration of the Measuring Devices and Geometries and Definition of the Lower Limits of Detection

Before the chosen measuring devices and applications could finally be accepted by the authorities it had to be proved, that the lower limits of detection would comply with the requirements.

#### A.1.7.1 Calibration of Contamination Monitors

Stability of calibration data of the contamination monitors LB 1210 B and C employed for release measurements as well as of the 'Minicont' was regularly (daily before start of work and after long pauses) controlled by application of surface standard preparations.

For initial calibration and stability surveillance 2 Co-60 standards were employed.

The following lower limits of detection (LLD), defined for a continuous measuring procedure, were calculated with the evaluated calibration data:

- Berthold LB 1210 C for beta-emitters

the delay set to 2 s : LLD =  $3.35 \text{ E-6 } \mu\text{Ci/cm}^2$  ( $0.124 \text{ Bq/cm}^2$ )

" " " " 20 s : LLD =  $1.06 \text{ E-6 } \mu\text{Ci/cm}^2$  ( $0.039 \text{ Bq/cm}^2$ )

For alpha-emitters the manufacturer grants observance of a LLD of appr.  $5 \text{ E7 } \mu\text{Ci/cm}^2$  ( $0.010 \text{ Bq/cm}^2$ ) (natural background level presumed).

In order to achieve the necessary LLD, the time of delay had to be set to 2 s consequently. In order to obtain sufficient step response, a measuring time of 6 s per position had to be observed.

For components coated with rust or paint the measuring results are debased by absorption of alpha- resp. beta-radiation.

- Berthold 1210 B

the delay set to 2 s : LLD =  $8.50 \text{ E6 } \mu\text{Ci/cm}^2$  ( $0.314 \text{ Bq/cm}^2$ )

" " " " 20 s : LLD =  $2.70 \text{ E6 } \mu\text{Ci/cm}^2$  ( $0.100 \text{ Bq/cm}^2$ )

Applicating the LB 1210 B, the time of delay had to be set to 20 s in order to obtain the requisite LLD. Sufficient response to activity steps could only be achieved by observing a measuring time of 60 s per position. Because, after decontamination and performance of surveillance measurements, homogeneous activity distribution could be presumed, all measurements complying with release criteria were generally performed with a measuring time of 20 s per position.

- Herfurth 'Minicont'

the delay set to 6 s : LLD =  $5.00 \text{ E6 } \mu\text{Ci/cm}^2$  ( $0.185 \text{ Bq/cm}^2$ )

For the surveillance measurements a measuring time of 20 s per position, the delay set to 6 s, was considered sufficient.

#### A.1.7.2 Calibration of the GeLi-Detection Applications at GKSS-Laboratories

In order to carry out the measurements complying with release criteria as well as the measurements stipulated by licensing obligations, the GeLi-multichannel analyzer of GKSS had to be calibrated for the application to samples of the categories 3.1; 3.2.1 to 3.3.1; 3.4.1 to 3.4.3; 4.4.1; 5.1 and 5.2 .

Depending on the standardized sample size (area, quantity or volume) and the corresponding specific activity, evaluated during preliminary investigations for activity assessment or stipulated by release criteria, the sample activity inventory could be precalculated as follows:

- Sampling for Release Measurements

Liquid Samples : drawn volume : 10 l  
 standard geometry : 1 l plastic-bottle  
 concentration : 1 : 10  
 release limit act. : 6 E-7  $\mu\text{Ci}/\text{cm}^3$   
 (Co-60;Cs-137) (0.022 Bq/cm<sup>3</sup>)  
 (purs. app.IV tbl. IV1 a. 3 c1m.6 Str1SchV)

Category	Sample Activity	
	( $\mu\text{Ci}$ )	(Bq)
5.2	6 E-3	222

- Sampling for Activity Assessment (pursuant to obligation 13 and 15)

Solid Matter Samples : stand. sample area : 20 cm<sup>2</sup>

Category	Sample Activity	
	( $\mu\text{Ci}$ )	(Bq)
3.1	2 E-0	74 000
3.2.1 - 3	2 E-2	740
3.3.1	2 E-3	74
4.4.1 - 3	2 E-4	7.4
4.4.1	2 E-4	7.4

Liquid Samples : drawn volume : 1 l

Category	Sample Activity	
	( $\mu\text{Ci}$ )	(Bq)
5.1	1 E-1	3 700

The measuring application for solid matter samples was calibrated with a disc-shaped preparation of 50 mm diameter. The calibration standard contained 9 different nuclides emitting 14 gamma-energies in a spectral range of 81 to 1836 keV, corresponding the energy-range of all nuclides identified by preliminary analysis of the contamination of nuclear components. With the evaluated calibration data the lower limit of detection, defined for digital, single nuclide detection, is:

- Measuring Time 3 000 s : LLD = 2.96 E-5  $\mu\text{Ci}$  (1.10 Bq)

Thus, observing a measuring time of 3 000 s, a surface contamination of 1.50 E6  $\mu\text{Ci}/\text{cm}^2$  (0.055 Bq/cm<sup>2</sup>) could be verified for samples of a standard area of 20 cm<sup>2</sup>. As measurements served activity evaluation of components being subject to restricted disposal only, the detection efficiency was considered sufficient.

In order to calibrate the measuring application for liquid samples a Co-60 solution in a standard 1 l plastic-bottle was employed. The lower limit of detection calculated from the calibration data amounts to:

- Measuring Time 3 000 s : LLD = 2.16 E-5  $\mu\text{Ci}$  (0.80 Bq)



Thus a measuring time of 3 000 s proved to be by far sufficient to evaluate the activity of liquid samples of the prevailing nuclide compound (Co-60 activity  $> 90 \%$ ). Actually, for exact single nuclide analysis and activity evaluation complying with release criteria, a measuring time of 70 000 s was applied to samples of category 5.2. Measurements complying with the requests of the obligations 13 and 15 (nuclide compound analysis and activity assessment of liquids to be treated as radioactive waste; category 5.1) were carried out employing a measuring time of 10 000 s.

A.1.7.3 Calibration of the GeLi-Multichannel Analyzer on Board N.V. 'Otto Hahn'

The measuring application had to be calibrated for samples taken from components of category 3.2.2 by employment of the EIR-sampling procedure. The sample activity to be detected was precalculated as follows:

Component Category : 3.2.2  
 Standard Sample Geometry : 50 ml plastic-bottle

- Sampling for Initial-State Activity Evaluation and Analysis

Sampling Area (cm <sup>2</sup> )	Sample Activity	
	( $\mu$ Ci)	(Bq)
4	max. 4 E-1	14 800
1	" 1 E-1	3 700

- Sampling for Residual Activity Evaluation and Analysis  
 (2.5 E-6  $\mu$ Ci/cm<sup>2</sup> ; 0.09 Bq/cm<sup>2</sup>)

Sampling Area (cm <sup>2</sup> )	Sample Activity	
	( $\mu$ Ci)	(Bq)
4	1 E-5	0.37
1	" 2.5 E-6	0.09

The measuring time required for the exact evaluation of the prevailing initial-state Co-60 activity was 1 s only. In order to evaluate the average residual activity of the samples with sufficient exactitude however, a measuring time of 7 h 15 min would have had to be applied. Here the consequences stipulated by choice of the sampling procedure to be employed are emphasized with reference to the measuring time and sample size (area, quantity) resp. sample activity relation.

Actually the average measuring time, necessary for surveillance of the storagepool decontamination progress and for analysis of the prevailing nuclide compound, amounted to 10 000 s per sample.

### A.1.8 Measuring Time Expenditure for the Decommissioning of the Nuclear Area of N.N. 'Otto Hahn'

#### A.1.8.1 Measurements for Compliance with Release Criteria (Components category 3.4.2 and 3.4.3)

The total surface area of the components concerned amounted to approx. 18 000 m<sup>2</sup>.

##### Measurement of Stationary or Immobile Components (3.4.2)

These measurements could only commence after sufficient background depression had been achieved by removal of sources and by decontamination.

After an initial-state contamination assessment, confined to defined subsections or components of the control area, a first-step decontamination, was carried out, followed by surveillance measurements. For the first time these measurements, employing the Herfurth 'Minicont' contamination monitor, covered the total surface area. 50 single measurements per m<sup>2</sup> had to be performed (window area of the detector 200 cm<sup>2</sup>). The total number of measuring positions for structures and components of this category was approx. 720 000.

After 1st-step decontamination 28 % of the measured area still exceeded the limit value and was marked as mentioned before (A.1.6.1). Mostly the marked measuring positions formed larger, connected areas to be submitted to 2nd-step decontamination by employment of more aggressive decontamination procedures. For example, simple washing was followed by pickling, pickling by grinding.

The area having been subject to 2nd-step decontamination was submitted to surface covering measurements again. Approx. 215 000 single measurements had to be performed.

After 2nd-step decontamination approx. 5.5 % of the total surface of components and structures exceeded the release limit, the marked measuring positions no longer forming larger, connected areas now. Consequently approx. 10 % of the originally contaminated surface had to be submitted to 3rd-step decontamination and subsequent measurements, the requisite number of measuring positions amounting to approx. 72 000.

About 3 % of the originally contaminated surface even had to undergo a 4th decontamination step, now mostly confined to single measuring positions, the progress and result of decontamination being surveyed by immediately applicated measurements. The requisite number of measurements performed for finishing surveillance amounted to approx. 72 000 again.

In total the requisite number of decontamination surveillance measurements employing the Herfurth 'Minicont' amounted to approx. 1 080 000.

With a measuring time of 20 s per position the effective total measuring time was approx. 6 000 h. Adding the time necessary to position the monitor (approx. 10 s per measurement) the actual total time for decontamination surveillance measurement of structures and components of category 3.4.2 amounted to 9 000 h. (see table A2 : Decontamination and Measuring Expenditure)

Surveillance measurements were followed by the actual release measurements employing the Berthold LB 1210 C or B contamination monitors. They were made by members of the vessel's staff and additionally controlled by experts of the surveyor and the authorities.

For assessment of the time necessary for actual release measurements it can be assumed that approx. 25 % of the measuring positions previously marked during surveillance measurements were controlled (180 000). Measurements concentrated on areas, which had been subject to 3rd and 4th step decontamination.

With an average measuring and positioning time of 30 s (20 + 10 s) per position, the total time consumption for the actual release measurements amounted to approx. 1 500 h (1 200 h by the vessel's staff, 300 h by experts). Thus the total measuring time to achieve release of structures and components of category 3.4.2 amounted to 10 500 h.

The quantity of material of category 3.4.2 including steel enforced concrete structures sums up to approx. 2 060 Mg, corresponding an area/mass relation of 70 cm<sup>2</sup> per kg. The quantity related average time consumption then amounts to approx. 5.1 h per 1 000 kg.

For steel and concrete structures the corresponding values are tabulated below:

Material	Quantity (Mg)	Area (m <sup>2</sup> )	Area/Mass (cm <sup>2</sup> /kg)	Specific Time Consumption (h/1 000 kg)
Steel	700	13 400	192	14
Concrete	1 350	950	7	0.5

#### Measurement of Mobile Components (3.4.3)

The total area of the mobile components to be measured amounted to approx. 3 600 m<sup>2</sup>.

In accordance with the decommissioning concept and license mobile components were brought to a measuring site installed at a low level background area directly after they had undergone 1st-step decontamination. First-step decontamination surveillance measurements employing the Herfurth 'Minicont' contamination monitor completely covered the surface of the components. The single measuring positions again were marked as described before. When the release-limit contamination was exceeded at 1 measuring position only, the concerned component was submitted to 2nd-step decontamination followed by surveillance measurements again.

In average the surface area of the components of category 3.4.3 had to be measured twice completely during decontamination survey.

Release measurements by the vessel's staff employing the Berthold contamination monitors again covered the surface of such components completely, while control measurements by the experts confined to appr. 10 % of the positions marked during decontamination survey.

For components of category 3.4.3 the total measuring time including positioning amounted to approx. 4 700 h, about 560 000 single measurements had to be performed.

With an area/mass-relation of  $100 \text{ cm}^2/\text{kg}$ , typical for components of the concerned category, the total quantity of the finally released components amounted to 350 Mg. The specific time consumption was approx. 13.5 h per 1 000 kg.

#### Measurements at the Fuel-Element Storage-Pool (3.2.2)

For reason described before, direct measuring procedures employing contamination monitors were applied for finishing surveillance and release measurements at the fuelelement storagepool too. Measurements were performed as described for categories 3.4.2 and 3.4.3 after sufficient background depression had been achieved in the progress of decommissioning.

According to a contaminated pool surface of  $150 \text{ m}^2$ , completely measured twice in average for finishing surveillance and once for compliance with release criteria, approx. 22 500 single measurement were carried out. Another 100 control measurements were performed by the experts of the surveyor and the authorities, leading to the provisional release of the storage pool. The total time consumption for finishing surveillance and release measurements amounts to approx. 195 h. Related to a material quantity of 12 Mg (area/mass-relation  $125 \text{ cm}^2/\text{kg}$ ) the specific time expenditure computes to approx. 16 h per 1 000 kg.

#### A.1.8.2 Measurements for Activity Evaluation and Analysis of Dismounted

Components (pursuant to the decommissioning license)

Measurement of Components and Structures of the Categories 3.1; 3.2.1; 3.2.2; 3.3.1; 3.4.1 and 4.4.1

To observe the obligations 13 to 15 of the decommissioning license referring to activity evaluation and analysis of materials subject to protective disposal approx. 150 samples were measured and analyzed. Depending on grade of contamination resp. sample activity measuring times from 2 000 to 5 000 s were required in order to achieve sufficient exactitude of nuclide analysis. In a few cases, for instance for the analysis of low contaminated concrete samples (total gamma-activity concentration  $0.037 \text{ Bq/g}$ ) with a sample weight of 10 to 200 g, measuring times up to 70 000 s had to be applied.

As measurements stretched out over longer periods, determined by the progress of decommissioning, time expenditure was considered acceptable.

Application of the described procedure for measurements complying with release criteria (requesting a much larger number of samples to be measured in order to obtain sufficient statistical exactitude and extremely long measuring times for exact residual activity evaluation) was considered uneconomical. Nevertheless important information upon the influence of former operation conditions on the development of contamination prevailing in different systems, circuits and components could be obtained by the results of the measuring series performed.

The measurements resulted in the release of various components (approx. 25 Mg), which were originally provided for protective disposal.

Additionally released components were characterized by either plain geometry (category 3.2.3), permitting application of alpha-, beta-contamination measurements complying with release criteria, or by homogeneous distribution of the residual contamination, being verifiable to a high grade of probability (category 3.3.1 and 4.4.1).

The information obtained during decommissioning of the nuclear area of N.V. 'Otto Hahn', completed by additional measuring and test series, served to elaborate improved decontamination and release procedures for the decommissioning of the secondary plant.

With approx. 150 samples measured (3 000 s per sample, sampling and processing time 10 min per sample), the time expenditure for the evaluation and analysis of the activity of all the dismantled components subject to protective disposal was approx. 150 h.

The total weight of the concerned components was approx. 305 Mg. The specific time expenditure was approx. 0.5 h/1000 kg.

Measurements for Decontamination Surveillance (fuel-element storage-pool, category 3.2.2)

For decontamination surveillance 60 samples were taken (EIR-sampling procedure), processed by dissolution of the sampling sponges and analyzed by single nuclide GeLi-detection, applying a measuring time of average 10 000 s per sample.

Total time consumption, including sampling and processing time (approx. 30 min per sample) amounted to 195 h, according to approx. 16 h/1000 kg.

Measurements for Activity Evaluation and Analysis of System- and Waste-Water (category 5.1)

During decommissioning approx. 430 m<sup>3</sup> of system- and waste-water had to be shipped to waste-water processing. 22 single transports were made to the processing plant. Representative samples were drawn, prepared to a standard 1 l - measuring geometry and analyzed by single nuclide GeLi-detection employing an average measuring time of 10 000 s per sample, the sampling and processing time to be neglected.

The total time consumption amounted to approx. 60 h.

#### A.1.9 Total Activity of Exclusively Contaminated Materials (categories 3.1 to 5.1)

By measurements pursuant to decommissioning license obligations an activity inventory of approx. 3.95 Ci (1.46 E+11 Bq) was evaluated for all dismantled components, structures and waste materials (including the secondary waste produced by decontamination) subject to protective disposal (see table A3 / : Mass and Activity Transfer).

A.1.10 Provisional Release of the Nuclear Areas of N.V. 'Otto Hahn'

In the course of decontamination and after the described release measurements by the vessel's staff and official experts had proved compliance with release criteria, confined subsections of the nuclear area (control area) were successively declared 'free of contamination exceeding release limits'. Admittance to the provisionally released areas was prohibited.

Previous to the provisional release of the containment the aerosol-activity prevailing in the hermetically shut-off compartment was balanced over a period of 3 months. Concentration enrichment by sweating could not be verified.

In June 1982 the decontamination work was finished and the provisional release of all the subsections of the nuclear area was achieved.

## A.2 Secondary Plant

### A.2.1 Introduction

During operation the vessel's secondary circuit had been slightly contaminated by a minor leakage of the steam generators.

By preliminary survey, assessment of total activity inventory of approx. 1 mCi (3.7 E+7 Bq) spread over the whole secondary circuit with an average grade of contamination of approx.  $3 \text{ E-}5 \mu\text{Ci/cm}^2$  ( $1.11 \text{ Bq/cm}^2$ ) was made, the total quantity of secondary circuit components amounting to approx. 325 Mg.

In spite of the activity concentration related to the total quantity of components unequivocally did not exceed limit concentration values pursuant to app. IV, tabl. IV 1, cln. 4 Str1SchV ( $1 \text{ E-}10 \text{ Ci/g}$ ;  $3.7 \text{ Bq/g}$  for Co-60 and Cs-137), licensing authorities, considering the surface contamination exceeding limit values (app. IX, cln. 4) and nature, bond and activity concentration of the prevailing contaminated incrustations ( $1 \text{ E-}10 \mu\text{Ci/g}$ ;  $3.7 \text{ Bq/g}$ ), declared all sections of the secondary circuit to be subject to decontamination previous to unrestricted release or to protective disposal by emphasizing the regulations of § 35 resp. § 64 and particularly § 28 Str1SchV.

### A.2.2 Influence of Plant-specific Conditions and Decontamination Procedures

The vessel's secondary system consisted of a naval steam-turbine plant almost identical to conventional constructions, with HP and LP-propulsion turbine and a reverse-turbine, arranged in the LP-housing, and the necessary auxiliaries, like turbine generators, sea-water evaporators, LP-steam generators and preheaters. 2 oil-fired auxiliary boilers (2-drum water-tube) supplied steam of similar quality (31 bar;  $278^\circ\text{C}$ ).

During operation and of course already before the steam-generator leak occurred, oxidic incrustations of distinct nature (chemical), thickness, distribution and bond had been formed out in different circuit sections and components, depending on the prevailing operation conditions characterized by pressure, temperature and velocity of the flowing medium.

Proved by the results of scrape-sample analysis, carried out during initial-state survey, the intruded activity was almost exclusively bound to the layer material itself. Since, under this premise, complete abrasion of incrustations implied sufficient degradation of contamination, decontamination by pickling could be considered promising.

In order to reduce secondary waste quantities, the layer material, dissolved in the provided pickling agent (mainly HCl-10%) with an precalculated volume of approx.  $60 \text{ m}^3$ , should be precipitated by neutralisation with NaOH and dehydrated by filtration. Filtrate activity concentration aimed at should permit unrestricted release and discharge of the liquid into the river 'Elbe'.

As almost all the single components of the secondary circuit were characterized by complex geometries, application of direct measuring methods could not be considered. Observance of limit values pursuant to § 35 or § 64 StrlSchV could not be proved with equal statistical exactitude as achievable by direct, surface covering measurement of plain surface components.

Therefore the proposition of the release procedure was based on the priority of activity concentration values, which, considering the requests of § 28 StrlSchV, had to be determined more restrictively than settled by § 4 para.4 s.1 no.2e StrlSchV. Strategies to determine the requisite number of samples and significant sampling positions should also grant observance of surface contamination limits with sufficient statistical exactitude but averaged over larger areas.

The provided decontamination procedure enabled to verify the results of release measurements by balancing the activity transfer by the pickling agents or the precipitated sludges with the initial and the residual activity inventory.

Sufficient exactitude of single balancing items and sufficient statistical accuracy of statements about initial-state and residual activity distribution could only be obtained by a large number of samples, the requisite number and sampling positions of which had to be determined regarding the former operation conditions in different circuit sections. Thus classification of plant-sections characterized by operational conditions had to precede the detailed elaboration of measuring strategies.

Acceptability of the sampling and measuring procedures and the devices to be employed had to be considered in the aspect of time consumption regarding the large number of samples to be taken (and evaluated).

### A.2.3 Concept of the Release Procedure

A balance of the activity inventories transferred in the course of secondary circuit decontamination was provided to verify the results of final release measurements. To evaluate the single balancing items diversified sampling and measuring procedures had to be employed.

#### A.2.3.1 Definition of the balancing items

In order to balance the activity transfer by decontamination with the initial-state and residual activity inventory the following items had to be taken into account:

1. Initial-state activity inventory of all secondary component
2. Activity inventory of pickling agents
3. Activity inventory of precipitated sludges
4. Activity inventory of nonsoluble, mechanically removed deposits
5. Activity inventory of agents used during pickling tests
6. Activity inventory of non-decontaminable components
7. Residual activity inventory of components provided for unrestricted release



The balance equation could be considered to indicate the degree of accuracy of the residual activity inventory and residual contamination evaluated by release measurements. By evaluating and balancing the activity transfer with pickling agents and precipitated sludges in parallel further proof of accuracy was given.

#### A.2.4 Activity Inventory Evaluation

##### A.2.4.1 Evaluation of the Initial-state Activity Inventory

Because of complex geometry and elevated background the grade of contamination and the activity inventory of most components and sections of the secondary circuit could only be evaluated by application of indirect sample measuring procedures.

Thus the measuring problem was defined as: 'evaluation of the surface contamination of representative samples'. The activity inventory then had to be calculated with the mean value of samples assigned to a defined component or circuit section and its contaminated surface area. Observing the requirements of § 4 para.4 s.1 no.2e StrlSchV, the nuclide compound of contamination had to be analyzed additionally.

In order to determine representative samples, the influence of operation conditions on the formation of incrustations and activity distribution had to be investigated in more detail.

#### Classification of the secondary circuit

Classification of secondary circuit components and sections depended on pressure stage and phase of the flowing medium. Investigation of contamination and incrustation formation had the following results:

- In sub-sections of similar operating conditions uniform incrustations of almost homogeneous thickness had formed out.
- The activity concentration of layer material in sub-sections with similar operating conditions was almost identical and, varied within only one order of magnitude. Exceptional concentration values approx. tenfold higher than average were found in sections of phase transition from water to steam and in zones with varying water levels.
- The grade of contamination and its homogeneity in different circuit sections was almost exclusively determined by the thickness of incrustations.
- In sections constantly flushed by dry steam, condensate or feed-water, strong adhesive incrustations of almost homogeneous thickness and thus homogeneous grade of contamination had formed out.
- In components or sub-sections where phase transition (steam to water) combined with higher corrosivity of the medium (oxygen-enrichment) commenced, less adhesive layers of strongly varying thickness had developed.



- In sub-sections or components flushed with low flow velocity of the medium and in their lowest areas (wetwells, drainages) sediments of various thickness had deposited. The deposits could be removed simply by flushing or brushing. The activity concentration and contamination distribution of stronger adhesive incrustations underneath the deposits corresponded with the values of sections with similar operating conditions (pressure, temperature and phase).

The number of samples was determined with regard to the the results of the preliminary investigations. In order to evaluate the initial-state activity inventory 47 representative circuit sections were determined. Depending on activity distribution 3 to 5 samples had to be taken from each the sections.

#### Sampling for initial-state activity evaluation

The decontamination procedure provided by the decommissioning concept for the secondary system stipulated a non-destructive sampling procedure. Thus, the circuit sections and components, determined as representative sampling positions, were opened. At various spots and area of 100 cm<sup>2</sup> was marked. Then the layer material in the area was abraded down to the metallic base material by scraping. Approx. 150 samples with an average quantity of 1.4 g were taken by this procedure.

At several sampling spots (e.g. valve-cones, steam strainers or similar detachable components), the destruction of which did not hinder decontamination operations or the geometry of which permitted direct measurement by GeLi- or NaJ-detection, the efficiency of the sampling procedure was determined by evaluation of the contamination prevailing previous to and after abrasion of the incrustations.

The average efficiency of the sampling procedure by scraping was 85 %.

#### Time consumption for sample activity evaluation

For sample activity evaluation and analysis gamma-detection had to be applicated. Adapted detector types had to be chosen with reference to the opposing requirements on energy resolution (requisite exactitude of single nuclide analysis) and on detection efficiency (measuring time minimization).

The calibration data of a certain measuring geometry known and the sample activity precalculated, the required measuring time can be determined. For initial-state activity evaluation an average sample activity of 2.5 E-3  $\mu$ Ci (92 Bq) could be presumed. Corresponding to the contamination release limit, the total activity of samples taken by cutting would amount to only 2 E-4  $\mu$ Ci (7.4 Bq), after decontamination.

If an acceptable error of 0.13 % has to be observed for allotment of samples to inactive as well as active material categories (error alpha = error beta), the requisite LLD of the employed measuring application has to be at least 6 \* SD (SD = standard deviation of background counting rate) lower than the calculated sample limit activity.

The residual activity of samples was determined by Co-60. The minimum measuring time required for activity evaluation by application of the GKSS Geli-detector amounted to approx. 4.4 min.

For analysis and evaluation of a nuclide in the energy range of Cs-137, potentially release-decisive with a component of 10 % of the total sample activity, a measuring time of 515 min is required in order to achieve sufficient exactitude.

Because of the large number of measurements, to be performed for activity transfer balancing and residual activity evaluation, the application of single nuclide Geli-detection for all the samples had to be considered economically not feasible.

Therefore most of the samples were measured by integral NaJ-detection within a limited energy range. Thus, the total activity of all significant nuclides, the quantified compound of which previously had been analyzed by random single nuclide Geli-detection, could be evaluated sufficiently accurate with minimum time consumption.

In order to achieve unrestricted release of a circuit section or component, the contamination limit of all concerned samples had been set to  $8 \text{ E-6 } \mu\text{Ci/cm}^2$  ( $0.3 \text{ Bq/cm}^2$ ) by licensing authorities. Even then, with a maximum sample activity of  $1.6 \text{ E-4 } \mu\text{Ci}$  ( $5.9 \text{ Bq}$ ) and observing an error of 1.3 %, a measuring time of only 1 min was required.

For exact nuclide analysis, random samples were taken out of every measuring series to be measured by single nuclide GeLi-detection in the laboratories of GKSS.

#### Description of the combined measuring procedure

The total activity of the samples was evaluated by integral NaJ-detection on board of the vessel. The total counting rate within an energy range of 100 to 1,260 keV (2nd Co-60 Peak faded out) was determined. Within the energy range all relevant nuclides, identified by single nuclide Geli-detection (Co-60, Cs-137, Cs-134, Mn-54), were detectable. The nuclide compound and its uniformity was visually controlled at the monitor of the multi-channel analyzer. From every measuring series (samples from 1 sub-section or component) random samples were chosen to be exactly analyzed by single nuclide Geli-detection in the laboratories of the GKSS.

Because of the inclusion of Compton-scattering generated background counts, the total sample activity evaluated by NaJ-detection was slightly higher than the total single nuclide activity of all random samples measured by single nuclide Geli-detection and thus on the 'safe side' with regard to release limit observance.

#### A.2.5 Calibration of the Measuring Applications

##### A.2.5.1 Calibration of the Geli-detector of GKSS

The Geli-detector was calibrated for samples of the following specification:

Kind of sample	Measuring geometry	Concentr.	Precalculated initial state	Sample activity residual
<b>Solid matter:</b>				
scraped	sample scale (50 mm Ø) ca. 1,4 g		2,5 E-3 µCi (92 Bq)	
cut-out	disc- shape ca. 20 cm <sup>2</sup>	-	8,0 E-4 µCi (30 Bq)	1,6 E-4 µCi (6 Bq)
<b>Liquids:</b>				
pickling agent	plastic bottle standardized 1 l		1,0 E-1 µCi (3 700 Bq)	
filtrate	plastic bottle standardized 1 l	1 : 10	1,3 E-4 µCi (ca. 5 Bq)	

#### Calibration for measurement of solid matter samples

To calibrate the measurement of scraped or cutout solid samples a discshaped standard preparation with a diameter of 50 mm was employed. The preparation consisted of 9 different nuclides, verified to prevail in the systems of the nuclear plant, emitting 14 various gamma-energies in a range from 81 to 1,836 keV. For calibration the standard was arranged directly on top of the detector.

Applicating a measuring time of 3,000 s, for Co-60 a lower limit of detection (LLD, safety-factor  $k = 3$ ) of  $LLD = 2.96 \text{ E-}5 \text{ µCi}$  (1.1 Bq) was determined.

Provided that the nuclide compound (Co-60 > 85%, Cs-137 < 15%, Cs-134 < 5%, Mn-54 < 1%) did not decisively change, for these samples a measuring time of 3 000 s was sufficient even for residual activity evaluation.

#### Calibration for measurement of liquid samples

To calibrate the measurement of pickling agent and filtrate samples a liquid preparation in standard 1-l plastic bottle was employed. Applicating a measuring time of 3 000 s a lower detection limit ( $k = 3$ ) of  $LLD = 2.70 \text{ E-}5 \text{ µCi}$  (1.0 Bq) was evaluated for Co-60.

With the prevailing nuclide compound of activity a measuring time of 3 000 s was sufficient to evaluate the activity of samples drawn from the pickling agent. Samples drawn from filtrate had to be concentrated by a factor of 10 before measurement.

#### A.2.5.2 Calibration of the NaJ-detector of N.V "Otto Hahn"

The NaJ-detector on board of the vessel had to be calibrated for the following kinds of samples:

Kind of sample	Measuring geometry		precalculated Concentr.	Sample activity initial-state residual
<b>Solid matter:</b>				
scraped	sample scale (50 mm Ø) approx. 1,4 g	-		see A.2.5.1
cut-out	disc- shape approx. 20 cm <sup>2</sup>	-		see A.2.5.1
<b>Liquids:</b>				
pickling agent	plastic bottle standardized 50 ml	-	5,0 E-3 µCi (185 Bq)	
sediment or sludge	sample scale (50 mm Ø) ca. 10 g	-	1,0 E-3 µCi (37 Bq)	
sediment or sludge (dispersed)	plastic bottle stand. 50 ml approx. 10 g	-	1,0 E-3 µCi (37 Bq)	

#### Calibration for measurement of solid matter samples

To calibrate the measurement of scraped and cut-out solid samples, a certain quantity of incrustation material, taken from a contaminated circuit of the nuclear plant, coated by cast resin and prepared to disc-shape (diameter 50 mm) was assembled directly on top the crystal. Previously the nuclide compound Co-60: 47.9 %; Cs-137: 38.6 %; Cs-137: 10.7 %) and the activity of the standard preparation had been analyzed and evaluated by single nuclide GeLi-detection in the laboratories of GKSS.

Applicating a measuring time of 1 min, for gamma-emitters of the calibrated compound and within the confined energy range the lower limit of detection was LLD = 6.90 E-5 µCi (2.55 Bq).

Regarding the actual nuclide compound even for residual sample activity evaluation a measuring time of 1 min was sufficient.

#### Calibration for measurement of pickling samples

To calibrate the measurement of pickling agent samples by NaJ-detection, liquid preparations in standard 50 ml plastic bottles of the following nuclide compound were employed: Co-60: 60.5 %; Cs-137: 39.5 % resp. Co-60: 64.4 %; Cs-137: 35.6 %.

With a measuring time of 1 min, the lower limit of detection of the measuring application was LLD = 2.23 E-4 µCi (8.25 Bq).

Thus, for the evaluation of the activity of pickling agent samples, a measuring time of 1 min. was sufficient.

## Calibration for measurement of sediment and sludge samples

The activity of sediment and sludge samples could be evaluated in both the described measuring applications. In both cases a measuring time of 1 min. was sufficient.

### A.2.6 Activity Inventory Evaluation

The activity balance based on the results of the following measurements is recorded in table A4.

#### A.2.6.1 Initial-state activity

For initial-state activity evaluation approx. 200 samples from 47 defined sampling positions were taken by scraping or cutting.

##### Activity evaluation by integral NaJ-detection

All samples gained by scraping or cutting were measured on board the vessel by integral NaJ-detection in an energy range from 81 to 1280 keV first. The surface contamination of each the 47 defined components, sections or sub-sections was averaged from the results of the number of sample measurements (3 to 5) previously determined. Thereby the single results of a sampling series showed minor deviation only.

By visual surveillance of the nuclide compound, employing the monitor of the multi-channel analyzer, for more than 90 % of the samples exclusively Co-60 activity could be identified. The remaining measurements indicated a minor Cs-137 component.

For all measurements a measuring time of 1 min/sample was sufficient. The requisite sampling and preparation time for cut-out samples amounted to approx. 10 min/sample. For sampling and preparation of samples taken by scraping approx. 30 min/sample were required.

##### Activity analysis and evaluation by single nuclide GeLi-detection

For each of the 47 defined sampling positions one sample was chosen at random from the series measured by application of integral NaJ-detection to be analyzed by single nuclide GeLi-detection at GKSS. All the random GeLi-measurements verified the results obtained by NaJ-detection. Even by long-term analysis ( $>10\ 000$  s), Co-60 activity could be identified only, with most of the samples. For 6 samples a nuclide compound of Co-60  $>80$  %; Cs-137  $<15$  %; Cs-137  $<5$  % was analyzed.

For sufficiently exact activity evaluation a measuring time of 3 000 s had to be applied. Sampling and preparation time have already been mentioned.

##### Initial-state activity inventory

The average sample contamination, was sectionally assigned to the secondary circuit sections and main components. With the surface area of the sections and components and the average surface contamination concerned, the sectional activity inventory was computed. The total activity inventory of the secondary circuit amounted to 778  $\mu\text{Ci}$  ( $28.79\ \text{E}+6\ \text{Bq}$ ).

#### A.2.6.2 Activity Transfer by Acid Agents

For decontamination surveillance and evaluation of the activity transfer achieved by the two pickling steps, 120 samples were drawn from the circulating acid agent in terminated intervals.

##### Activity evaluation by integral NaJ-detection of pickling agent samples

All samples drawn from the pickling agent were measured on board the vessel by integral NaJ-detection. In order to evaluate the activity transfer (activity inventory of the acid agent applied) achieved with each pickling step, the measuring results of 5 samples, drawn at the end of the step and showing minor deviation as activity saturation had occurred, were averaged. By visual surveillance of the nuclide compound Co-60 activity could be identified only. Sampling and preparation time to be neglected, for all measurements a measuring time of 1 min was sufficient.

##### Activity analysis and evaluation by single nuclide GeLi-detection

In order to control the exactitude of measurements achieved by NaJ-detection application and for exact analysis of the nuclide compound 5 samples were drawn to be analyzed by single nuclide GeLi-detection at GKSS, one sample to be taken at the end of each pickling step.

The results of the control measurements proved sufficient accuracy of integral NaJ-detection. Matching the results of initial-state analysis of solid matter samples, a Co-60 activity component more than 99 % was analyzed for the pickling agent. Sampling and preparation time to be neglected, the requisite measuring time amounted to 3 000 s/sample.

##### Activity inventory of acid agents

For the first pickling step (flushing all circuit-sections and components at least once for a minimum time of 1 h) the volume of the acid agent applicated amounted to 16.3 m<sup>3</sup>.

After activity saturation was attained at the end of the first pickling step an average specific activity of the agent of 1.40 E-5  $\mu\text{Ci}/\text{cm}^3$  (0.52 Bq/cm<sup>3</sup>) was evaluated.

For the 2nd pickling step, during which the voluminous components were decontaminated by repeated flooding and draining or by swelling (auxiliary turbines) 43.2 m<sup>3</sup> of acid agent had to be applied. As the grade of contamination had been depressed by first step pickling, saturation of the 2nd agent was already attained at 4.90 E-6  $\mu\text{Ci}/\text{cm}^3$  (0.18 Bq/cm<sup>3</sup>) now. The large volume of 2nd-step acid agent could only be processed in 3 charges. The residual quantities, buffered in the main condensor during 1st- and 2nd-charge processing, were employed for finishing decontamination of components, the residual contamination of which still exceeded release limits.

After finishing decontamination the activity concentration of the residual agent (3rd charge) amounted to 7.03 E-6  $\mu\text{Ci}/\text{cm}^3$  (0.26 Bq/cm<sup>3</sup>).

The activity inventories of the single agent volumes and the total activity transfer by pickling are given in table A4.

#### A.2.6.3 Activity Transfer by Precipitated Sludges and Filtrate

In order to evaluate the activity inventory of the sludges precipitated by acid agent processing, a certain quantity of filter cake sludge was sampled from approximate 200 filter tappings and blended to 30 representative measuring samples. From each of the blends a quantity of approx. 10 g was exactly weighted and prepared to be measured in the calibrated geometries. The filter effluent was drained into the former destillate tanks. By 4 tappings samples were drawn from the 2 tanks to be blended to 2 representative 1 l filtrate samples.

##### Activity evaluation of sludge samples by integral NaJ-detection

By visual surveillance of the nuclide compound employing the analyzer monitor, Co-60 activity could be identified only. To evaluate sludge sample activity a measuring time of 1 min/sample was sufficient. Sampling, weighting and processing required an average time of approx. 10 min/sample.

##### Activity analysis and evaluation of sludge samples by single nuclide GeLi-detection

The analysis of the sludge activity by single nuclide GeLi-detection could be abandoned, because the nuclide compound of the acid agent had been analyzed before.

##### Activity evaluation of filtrate samples by integral NaJ-detection

Because of the low residual activity, the limited number of samples and because the measurements should result in immediate release, filtrate samples were not measured by integral NaJ-detection.

##### Activity analysis and evaluation of filtrate samples by single nuclide GeLi-detection

The filtrate samples were analyzed by single nuclide GeLi-detection applying a prolonged measuring time of 70 000 s. Even then only Co-60 activity could be identified. The time needed to draw and prepare the filtrate samples can be neglected.

##### Activity inventory of precipitated sludges and filtrate

The quantity of sludge, obtained by processing the 4 charges of pickling agent, amounted to total 5 390 kg. By filtration of the first charge 1 615 kg of filter cake with an average activity concentration of  $1.29 \text{ E-}4 \text{ } \mu\text{Ci/g}$  ( $4.77 \text{ Bq/g}$ ) was gained. With the 3 processing charges of 2nd-step pickling agent the quantity of sludge with average activity concentration of  $7.64 \text{ E-}5 \text{ } \mu\text{Ci/g}$  ( $2.83 \text{ Bq/g}$ ) amounted to 3 775 kg.

The quantity of filtrate, drained into the destillat tanks by 1-step pickling agent processing and filtration with an activity concentration of  $5.80 \text{ E-}7 \text{ } \mu\text{Ci/cm}^3$  ( $0.021 \text{ Bq/cm}^3$ ) amounted to  $18.0 \text{ m}^3$ . With 2nd-step pickling agent processing and filtration the quantity of filtrate with an average activity concentration of  $1.35 \text{ E-}8 \text{ } \mu\text{Ci/cm}^3$  ( $0.0005 \text{ Bq/cm}^3$ ) amounted to  $46.0 \text{ m}^3$ .



The evaluated activity concentrations and inventories of filtrates and sludges are given in table A4.

#### A.2.6.4 Activity Transfer by Mechanically Removed, Non-soluble Deposits

Non-soluble constituents of the original layer material had partly deposited in the progress of pickling, especially during standstill of pickling agent circulation in areas of low flow velocity. The sediments as well as residual precipitation sludges deposited in the deaerator were removed from the sections and components concerned (boiler drums, main condenser, deaerator, condensate tank) by mechanical finishing decontamination.

For each of the finished components a representative sample was prepared from the particular quantity of removed sludges or deposits by blending several draws.

##### Activity evaluation by integral NaJ-detection

The activity concentration of all the non-soluble deposit samples and of residual sludge was significantly higher than the initial-state activity concentration of the layer material itself.

The results indicate distinct solubility of the inactive layer constituents, mainly iron oxides, and the active Co-60 (Cs-137) compounds. The increased activity concentration of residual sludges is explained by deposition of nonsoluble matter in the bottom area of the deaerator in the progress of pickling. By visual surveillance of the nuclide compound only Co-60 could be identified. For sample activity evaluation a measuring time of 1 min/sample was sufficient. Sampling and sample processing took approx. 10 min/sample.

##### Activity analysis and evaluation by single nuclide Geli-detection

Only samples drawn from boiler and main condenser deposits were analyzed by single nuclide Geli-detection. The activity concentration evaluated matched the results gained by integral measurements. The Co-60 activity constituent was analyzed to be more than 99 %. For the prevailing sample activity and its compound a measuring time of 3 000 s/sample was sufficient.

##### Activity of the deposits and residual precipitation sludges

The quantity, activity concentration and the activity inventory of non-soluble sediments and residual precipitation sludges are given in table A4.

#### A.2.6.5 Activity Transfer by Pickling Tests

Previous to actual decontamination, pickling tests applying various acid agents were performed at the subsections LP-preheater (with sealing-steam condenser) and HP-preheater. The activity loaded agents were stored in 5 drums with a capacity of 400 l each to be processed in particular. To evaluate the activity transfer by test agents a 50 ml sample was drawn from each drum.

#### Activity evaluation by integral NaJ-detection

All the test agent samples were evaluated by integral NaJ-detection only. By visual surveillance of the nuclide compound exclusively Co-60 activity could be identified.

For sample activity evaluation a measuring time of 1 min/sample was sufficient, sampling and processing time to be neglected.

#### Activity inventory of the pickling test agents

The quantities of agents applied during pickling tests and the evaluated activity concentration and inventories are given in table A4.

#### A.2.6.6 Activity Inventory of Non-decontaminable Components

The finishing decontamination procedures were surveyed by application of direct contamination monitoring employing the contamination monitors Berthold LB 1210 B and C for all the circuit sections and components opened and accessible for inspection now. Surveillance measurements, feasible after sufficient background depression had been achieved, served to detect areas, subsections or components, above release limits. Finishing decontamination was methodically applied to all the sections and components still contaminated and mostly resulted in the unrestricted release of the concerned sections and components. In only a few cases extreme thick incrustations could not be completely removed or nature and distribution of residual contamination as well as component geometry did not permit sufficient proof of compliance with release limits. These sections or components were cut-out or dismantled to be provided for protective disposal.

In order to evaluate the activity inventory of the non-decontaminable components 35 samples were cut-out.

#### Contamination monitoring for decontamination surveillance

For finishing decontamination surveillance and 'hot spot' detection approx. 4 000 single measurements were carried out by application of the contamination monitors LB 1210 B or C. The requisite measuring time amounted to 20 s/single measurement.

#### Activity evaluation by integral NaJ-detection

For sample activity evaluation by integral NaJ-detection a measuring time of 1 min/sample was sufficient in spite of the activity depression achieved by decontamination. By visual surveillance of the nuclide compound Co-60 activity could be identified only.

#### Activity analysis and evaluation by single nuclide Geli-detection

Samples cut from non-decontaminable components and sections to be protectively disposed off were not analyzed and evaluated by single nuclide Geli-detection.

#### Activity inventory of non-decontaminable components

The average specific activity and the activity inventory of components to be protectively disposed off is given in table A4.

The quantity of non-decontaminable components to be protectively disposed off amounted to 4 060 kg, i.e. approx. 1.25 % of the total quantity of secondary plant structures. Relating the evaluated residual activity inventory to the quantity of material to be protectively disposed off, the resulting activity concentration was  $7.29 \text{ E-13 } \mu\text{Ci/g}$  ( $0.027 \text{ Bq/g}$ ) or approx. 1/137 of the limit value pursuant to app. IV, table IV1, column 6 Str1SchV.

#### A.2.6.7 Residual activity inventory of components and sections provided for unrestricted release

The evaluation of residual contamination and the residual activity inventory was based on the measurement of a number of samples to be taken from the 47 sampling positions already determined for initial-state activity evaluation.

Considering the results of contamination measurements obtained during finishing decontamination surveillance, for 5 particular sampling positions the number of samples had to be increased.

The particular sampling positions were limited to components with contaminated areas concentrated to small volumes (e.g. heat exchangers, turbine-blading). Because counting rates do not relate to defined areas, without additional reference, application of direct contamination measurements or monitoring is unfit to quantify the detected residual surface contamination of such components.

Regarding secondary plant components, elevated counting rates were detected at the HP- and LP-preheater, both the auxiliary boilers and the auxiliary turbines. Because extremely low residual contamination had been achieved by decontamination, the counting rates detected at other, similar components, e.g. the main and auxiliary condensers, fell below the calibrated threshold in spite of unfavourable geometry.

In order to determine the number of samples and to locate significant sampling spots, direct contamination measurements employing the contamination monitor LB 1210 B were carried out.

160 samples were cut from the 47 defined sampling positions. The number of samples to be taken from problematic positions was increased to 6 or 8. Samples cut from areas of very low level residual contamination (approx. 25 %) were prepared to an area of  $40 \text{ cm}^2$ .

#### Contamination measurements for sampling localization and sample number determination

For sampling localization and sample number determination as well as 'hot-spot' detection approx. 8 000 single measurements by application of the contamination monitor LB 1210 B were performed, geometry of the components permitting. The measuring time amounted to 20 sec/single measurement.

#### Activity evaluation by integral NaJ-detection

All the samples cut from the defined sampling positions were evaluated by integral NaJ-detection on board the vessel.

For exact activity evaluation of approx. 60 % of the samples (20 and 40 cm<sup>2</sup>) a measuring time of 1 min/sample had to be applied. Due to the very low level residual contamination prevailing after decontamination, for approx. 40 % of the samples (20 cm<sup>2</sup> only), the measuring time had to be prolonged to 10 min/sample in order to obtain accurate values. Sampling and sample preparation again required about 10 min/sample.

By visual surveillance of the nuclide compound exclusively Co-60 activity was identified at all the samples.

#### Activity analysis and evaluation by single nuclide Geli-detection

Since the prevalent Co-60 activity constituent of nuclide compound and sufficient exactitude of activity evaluation by integral NaJ-detection had been proved by preceding measurements, only 10 samples were chosen at random from the measuring series for residual activity evaluation by single nuclide Geli-detection.

Because of the low-level activity of samples the measuring time to be applied amounted to 10 000 s/sample. The sample activity evaluated by integral NaJ-detection was confirmed by Geli-measurements. The Co-60 activity constituent was analyzed to be more than 99.9 %.

#### Residual activity inventory of components provided for unrestricted release

Complying with release obligations, the surface contamination of none the evaluated samples exceeded the release limit of  $8 \text{ E-}6 \mu\text{Ci}/\text{cm}^2$  ( $0.3 \text{ Bq}/\text{cm}^2$ ). The residual activity inventory was calculated as described above. For computation of the sectional activity inventories, the maximum value evaluated for each of the 47 sampling positions was taken.

The total activity inventory of components provided for unrestricted release amounted to:  $40.7 \mu\text{Ci}$  ( $1.51 \text{ E} + 6 \text{ Bq}$ ).

#### A.2.7 Interpretation of the Measuring Results and the Activity Balance

The surface contamination of none of the samples, measured for residual activity evaluation, exceeded the defined release limit and no 'hot-spots' were detectable by direct contamination measurements. Thus important prerequisites for the unrestricted release of the secondary circuit components, remaining after the non-decontaminable parts had been dismantled, had been complied with.

The results of activity evaluation measurements and non-quantified contamination monitoring were essentially supported by the balance of activity transfer.

Balancing the activity transfer by acid agent and protectively disposed off components with the difference between initial-state and residual activity inventory of components provided for unrestricted release, the balancing deficiency was  $23 \mu\text{Ci}$  ( $0.85 \text{ E}+6 \text{ Bq}$ ), corresponding an error of approx. 3 % related to the evaluated initial-state inventory.

By parallel balance of the activity transfer by precipitated sludges and filtrate a balancing surplus of  $13.5 \mu\text{Ci}$  ( $0.5 \text{ E}+6 \text{ Bq}$ ) corresponding to an error of approx. + 1.7 % was calculated.

Related to the total area exposed to contamination the average residual surface contamination of components provided for unrestricted release amounted to  $1.91 \text{ E-}6 \mu\text{Ci/cm}^2$  ( $0.07 \text{ Bq/cm}^2$ ).

The actual residual activity inventory ficticiously including the balance deficiency - allowing for an error of approx.- 57 % related to the evaluated residual activity inventory - the average surface contamination results in to  $2.99 \text{ E-}6 \mu\text{C/cm}^2$  ( $0.11 \text{ Bq/cm}^2$ ).

Relating the evaluated residual activity inventory to the total mass of components (approx. 320 Mg) provided for unrestricted release, the average activity concentration is  $1.28 \text{ E-}13 \text{ Ci/g}$  ( $0.0047 \text{ Bq/g}$ ) or, a Co-60 activity constituent of almost 100 % proven, to a 780th part of the no-restriction limit pursuant to app. IV tbl. IV1 cln. 4 Str1SchV. Calculating activity concentration with the balancing deficiency included, the corresponding values is to  $2.0 \text{ E-}13 \text{ Ci/g}$  ( $0.0074 \text{ Bq/g}$ ) or 1/500 of the limit.

The results of measurements for residual activity evaluation combined with the balance of activity transfer were considered sufficient proof of compliance with the release limits and insignificance of residual activity by experts and licensing authorities.

Because of the significant difference between actually evaluated activity and the determined limit values, surface contamination locally exceeding release limits could be considered improbable for most of the defined circuit sections. Provisional release was decreed for these sections after control measurements had been performed by experts of the surveyor and the licensing authorities employing the Berthold LB 1210 B contamination monitor.

Final release of all the secondary plant components was determined to depend on the results of additional inspections to be carried out by an independent surveyor nominated by the authorities. Especially the problem of interpreting the results of contamination measurements at heat exchangers and auxiliary turbines had to be settled previous to final release.

#### A.2.8 Analysis of Measuring Time Expenditure

Total time expenditure amounted to about 365 h or approx. 1.1 h/ 1 000 kg.

A measuring expenditure of this amount finally resulting in the release of almost all the components must be regarded as favourable. Minimization of measuring time expenditure to this extent could be achieved by combined application of integral NaJ and single nuclide Geli-detection only, additionally providing that indirect measurements could be approved of by authorities and compliance with activity concentration limits would be decisive for final release predominantly.

Evaluating all the various kinds of samples by single nuclide Geli-detection only, the requisite measuring time would have amounted to approx. 740 h or 2.3 h/1 000 kg, without improving the accuracy of activity evaluation and nuclide compound analysis. Resulting in the release of the concerned components, a measuring time expenditure to this extent might still be considered acceptable.

### A.3 Final Inspections by Independent Experts

#### A.3.1 Introduction

In mid June 1982, after control measurements had been performed by the experts of licensing authorities and surveyor, provisional release was achieved for all areas of the nuclear and secondary plant. From June 14th to 16th 1982 final inspections were carried out by experts of the 'PTB, Braunschweig', officially appointed independent surveyors by the licensing authorities.

Final inspection measurements commencing, all active materials and sources influencing the background level had previously been dismantled, packed and conveyed to protective disposal facilities established at GKSS site.

Final inspections had to cover the following areas of the vessel.

Area	Compartment or component	Sub-section
Inner control area :		
containment		
containment-compmnt.	containment, (outer surface) second. shielding	
auxiliaries-compmnt.	main-deck	laboratories, shops, washing a. dressing- rooms
	tween-decks (3)	entrance air lock
	lower-deck	sampling cab.shield. waste-wat.tk.shldg.
service-compartment.	fuel storage pool pool shielding	
Outer control area :		
open decks	reaktor-deck main-deck	inside border fence " " "
hedges	hedge IV hedge III	empty decontaminated com- ponents - 375 Mg
tanks (nucl. area)	wing-tanks double-bottom tanks bulkhead tanks	
Observation area :		
secondary plant	engine room boiler room tanks	machinery aux. boilers double-bottom 0801
superstructure	crews' quarters open decks	

### A.3.2 Measuring Procedures and Applications

For final inspection measurements by independent experts, application of alternative measuring procedures was stipulated by the licensing authorities. Pursuantly PTB-Braunschweig decided to base the inspection on results obtained by dose-rate monitoring mainly.

In order to perform final inspection measurements a high-sensitive doserate meter, developed by PTB, was used:

Manufacturer	:	Münchener Apparatebau Kimmel
Type	:	MAB 604/J
Detektor	:	
Mode	:	plastic-szintillator with silver-activated ZnS-foil coating
Radiation sensivity	:	Gamma, Röntgen
Energy range	:	20 keV - 1,5 MeV
lowest indicated value	:	0,1 $\mu$ R/h
LLD (background 8 uR/h)	:	1 $\mu$ R/h
Indicator	:	
Indication	:	$\mu$ R/h           analogous
Alarm threshold	:	adjustable   visual/acoustical

### A.3.3 Strategy of Final Inspection Measurements

As a first indication of sufficient decontamination or of insignificance of residual contamination, the grade of conformance between natural background level and dose-rates, actually evaluated by measurements generally prospecting all the compartments belonging to the former nuclear area or having been contaminated otherwise, should be taken into consideration.

Measuring the surface contamination by application of the LB 1210 B or C contamination monitors at random and especially if, approaching certain components or areas, significant doserate elevation had been detected, the relation between both the measuring procedures should be determined by comparison.

At special events employment of a portable NaJ-multi-channel analyzer was provided in order to analyze the nuclide compound of eventually detected residual sources.

Furthermore the accuracy of results previously obtained by release measurements should be proved by measuring a certain number of retained samples.

### A.3.4 Detection Capability of PTB-Dose-Rate Meter Application

In order to assess the detection capability of the dose-rate measuring procedure the dose-rates on the center-axis of disc-shaped sources, the active surface of which ranged from 100 cm<sup>2</sup> to 100 000 cm<sup>2</sup>, was computed for measuring distances up to 50 cm by model calculation, assuming a Co-60 contamination of 1 E-5  $\mu$ Ci/cm<sup>2</sup> (0.37 Bq per cm<sup>2</sup>).

Approaching a contaminated surface with an area of 1 000 cm<sup>2</sup> to a measuring distance of 10 cm for instance, the dose-rate would rise by 0.7 μR/h (7 nSv/h) or, related to a background level of 5 μR/h (50 nSv per hour), by 14 %.

Areas of this extension, the contamination of which exceeding the defined limit value (8 E-6 μCi/cm<sup>2</sup>; 0.3 Bq/cm<sup>2</sup>) by a factor of 1.25 (1 E-5 μCi/cm<sup>2</sup>; 0.37 Bq/cm<sup>2</sup>) were detectable by application of the PTB dose-rate meter, prevailing contamination by Co-60 assumed.

In order to evaluate the dose-rate due to natural background radiation in the various areas to be measured during final inspection, background monitoring was performed in the vicinity of the vessels' mooring area as well as in non-nuclear sections on board.

The measurements showed that the cosmic (approx. 3.6 μR/h; 36 nSv/h) and the terrestrial (1.4 μR/h; 14 nSv/h) component of natural background radiation averaged in the vessels' mooring vicinity were differently influenced by steel hull structures in sections above water level and by water in submerged compartments.

In water-side, closed decks above water level a dose-rate depression down to total 3.6 μR/h (36 nSv/h) was registered (shielding of the terrestrial component). Beneath water-level and particularly in double-bottom areas the natural background declined to 3.0 μCi/cm<sup>2</sup> (30 nSv/h) (shielding of both components).

Residual contamination, eventually exceeding release limits, could be detected particularly well in such very low background areas.

#### A.3.5 Final Measurements Performance, Results and Interpretation

For the final inspection the plant sections were scanned grid-wise, keeping a measuring distance of approx. 10 cm from walls, floors and component surfaces.

Values of 5.0 μR/h (50 nSv/h) were detected mainly. The results evaluated at approx. 150 measuring positions were recorded. The recorded positions were characterized by a slight but significant elevation of the average local dose-rate level and were submitted to direct contamination measurements by LB 1210 B or C application again.

A representative selection of dose-rate and corresponding contamination values is tabulated below:

Section	Sub-section Component	Dose- rate μR/h(nSv/h)	Surface contamination μCi/cm <sup>2</sup> (Bq/cm <sup>2</sup> )
Inner control area :			
containment	bottom plate	5,0 ( 50)	8 E-6 (0,29)
containment-cmptmnt.	floor	6,0 ( 60)	8 E-6 (0,29)
	concrete shield	12,0 (120)	1 E-5 (0,37)
service-compartment	wall	5,0 ( 50)	8 E-6 (0,29)
	pool, inside	6,0 ( 60)	8 E-6 (0,29)
	concrete shield	10,0 (100)	1 E-5 (0,37)
auxiliary-cmptmnt.	lower deck	6,0 ( 60)	8 E-6 (0,29)



Outer control area :

main deck	floor	5,0 ( 50)	8 E-6 (0,29)
hedge IV	floor	5,0 ( 50)	8 E-6 (0,29)
hedge III	decontaminated components	5,0 ( 50)	8 E-6 (0,29)

Observation area :

engine room	main-condenser	7,0 ( 70)	1 E-6 (0,04)
	main-turbine	14,0 (140)	6 E-6 (0,22)
	HP-preheater	17,0 (170)	6 E-6 (0,22)
boiler room	main-boiler	5,0 ( 50)	3 E-6 (0,12)
double-bottom	tank 0801	3,0 ( 30)	8 E-6 (0,29)
superstructure	open decks	6,0 ( 60)	not contam.

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Significant background increase was only detected approaching concrete structures and particular secondary circuit components.

For concrete the increase is due to natural radionuclides (mainly K-40), present in its mineral components. Applying the portable NaJ-multi-channel analyzer, to the concrete surfaces, the gamma-energy spectrum of K-40 was significant. Single nuclide GeLi-analysis of concrete samples verified, that the residual surface contamination of  $1 \text{ E-5 } \mu\text{Ci/cm}^2$  ( $0.37 \text{ Bq/cm}^2$ ), generally prevailing at concrete structures, was determined by Co-60 to minor extent only.

The relation between residual contamination, prevailing on formerly contaminated surfaces of heat-exchangers and auxiliary turbines, and dose-rates, detected by applying the dose-rate meter to cut-open sections of the concerned components, was determined by model calculation.

The accuracy of residual contamination evaluated by release measurements was checked by measuring retained samples in the laboratories of PTB as follows:

- The surface layers of the retained samples were chemically abraded, the abraded and dissolved material to be transformed to solid matter again.
- The rate of chemical contamination abrasion was determined by measuring the retained samples previous to and after treatment employing single nuclide GeLi-detection.
- An exactly weighted quantity of the solid matter was analyzed by single nuclide GeLi-detection employing a bore-hole crystal.

The results sufficiently conformed the results of previous measurements carried out for initial-state as well as residual contamination evaluation, confirming applicability of the measuring procedures.

To compute the adequate source activity of a calculation model corresponding the HP-preheater for instance, the surface contamination evaluated by release measurements could be referred to.

Assuming a maximum residual contamination of  $1 \text{ E-5 } \mu\text{Ci/cm}^2$  ( $0.37 \text{ Bq/cm}^2$ ), with the total surface area of preheater interiors amounting to  $100 \text{ m}^2$ , the activity inventory computes to  $10 \mu\text{Ci}$  ( $0.37 \text{ E+6 Bq}$ ). Related to the total mass of interiors the fictitious activity concentration amounts to  $7.9 \text{ E-12 Ci/g}$  ( $0.29 \text{ Bq/g}$ ).

For model calculation of dose-rates, probably detectable in various distances from the preheater housing, the preheater interiors were considered to be a cylindrical volume-source, the homogeneous activity concentration of which amounted to  $7.9 \text{ E-12 Ci/g}$  ( $0.29 \text{ Bq/g}$ ).

Dose-rates actually detected directly at the cut-open housing of the preheater and at defined measuring positions of various distances from the preheater surface significantly fell short of the calculated values. The detected increase of the dose-rate was confirmed to be caused by residual contamination not exceeding the limit values.

The vessel being free from materials relevant in the aspects of radiation protection and the reliability of the preceding release measurements proved by final inspection the final report of PTB closed with the statement:

"The vessel has sufficiently been contaminated. Further measures are not required."

#### A.3.6 Qualification of Final Inspection Procedures and Expenditures

Regarding release measurements dose-rate monitoring does not render quantified, area- or mass-related values and cannot replace the measuring procedures described above.

In case of low background level dose-rate monitoring is suitable to locate 'hot spots' and areas with residual contamination.

Furthermore, the detected dose-rate values by the licensing authorities may be considered as a proof of insignificant residual contamination.

Employing high-sensitive meters, dose-rate monitoring can be of advantage to determine the number and position of samples to be taken for indirect measuring procedures and may considerably minimize time expenditure.

PTB succeeded to perform the final dose-rate measurements within a period of 3 days only. Time expenditure therefore can be considered to be extremely low.

Classification of Components and Materials

Group No. pursuant	Class No. to NIS	Class No. addition.	Classification Criteria			Contaminating Medium	Systems or Components	A* B* C*			Categories re-released	Measuring and Sampling Procedures provided (Concept) performed	
			Origin of Activity	Grade of Contamination	Component Characterist.								
1	1	-	Solids predominant, activated	370000 Bq/cm <sup>2</sup> 0,37 "	-	primary water shield-tank water	pressure-vessel + interiors (core zone) shield-tank + int.						
2	1	-	activated as well as contaminated	370000 Bq/cm <sup>2</sup>	-	primary water	pressure-vessel + int. (remain. parts) e.g. steam generat.						
3	1	-	exclusively contaminated	370000 Bq/cm <sup>2</sup>	-	primary water and concentrates	prim. nuclear syst: e.g. cleaning circuit, seal.-water	*	*	*			single-nuclide GeLi-detection standard-shape solid samples
3	2	1	"	3700 Bq/cm <sup>2</sup>	complex geometry	aux. nuclear system water of lower grade contamination	aux. nucl. syst.: e.g. active water syst. waste-water syst.	*	*	*			
3	2	2	"	3700 Bq/cm <sup>2</sup>	simple geometry immobile		fuel-element storage-pool	*	*	*	*	single-nuclide GeLi-detection EIR-electrolytic sampling	alpha-beta-contamination monitoring (flight grid)
3	2	2	"	3700 Bq/cm <sup>2</sup>	"		exhaust-gas system (gas-tanks)	*	*	*	*		
3	3	1	exclusively contaminated	0,37 Bq/cm <sup>2</sup>	complex geometry	aux. system of water of very low gr. cont.	aux. systems: e.g. shield-tank, emerg. interm. cool. syst.	*	*				single-nuclide GeLi-detection + random cont. monitoring
3	4	1	"	0,37 Bq/cm <sup>2</sup>	complex geometry	predominantly contaminated by airborne activity	aux. systems: e.g. deionate, compr.-air systems	*	*	*			standard-shape solid samples
3	4	2	"	0,37 Bq/cm <sup>2</sup>	simple geometry immobile		naval structures: compart., decks, holds and tanks	*	*	*	*	alpha-beta-contaminat. monitoring (flight-wise) and rand. GeLi-detection	
3	4	3	"	0,37 Bq/cm <sup>2</sup>	simple geometry mobile		shielding structures and auxiliary devices	*	*	*	*		
4	4	1	"	(0,37 Bq/cm <sup>2</sup> )	complex geometry	exclusively contaminated by airborne activity	insulating material, cables, cable-shelves, smallparts	*	*	*	*		single-nucl. GeLi-detect. + alpha-beta-contamination monitoring
5	1	-	liquids	> release limit (846 Str. Sch.V.)	homogeneous distribution		nuclear system water (liquids)	*	*	*			single-nucl. GeLi-detection 1 l - sample
5	2	-	"	< release limit	homogeneous distribution		precipitation water	*	*	*	*		single-nucl. GeLi-det. 10 l-sample conc. to 1 l

Table A 1

\* ) A-provided (concept); B-required (decom. license); C-meeting release requirements

Decontamination and Measuring Expenditure (for Components of Cat. 3.4.2)

Area Compartment	Total Surface Area m <sup>2</sup>	Initial-state Monitoring		1st Step Decont.		Surveillance Measurements		2nd Step Decont.		Surveillance Measurements		3rd Step Decont.		Surveillance Measurements		Further Steps Decont. 2)	Surveillance Measurements		Summation	
		1)	2)	1)	2)	1)	2)	1)	2)	1)	2)	1)	2)	1)	2)		2)	3)		
Inner Contr. Area Containment	900	480	440	Washing	440	120	Cauteri- zing	120	23	Grinding	23	-	-	-	900	1063	1063			
		40200		900	22000		120	6000		23	1150									53150
Containment- Compartment	1200	895	410	Washing/ Brushing	405	170	Rust Re- moval	170	70	Pickling	70	-	-	-	1200	1250	1540			
		17750		410	18000		770	8500		70	3500									47750
Auxiliary- Compartment	2938	1460	1896	Vacuum Cleaning /Washing	1780	787	Cauteri- zing/ Pickling	787	180	Grinding Milling	182	95,5	Grinding Milling/ Pickling	224	2938	4138	4433			
		68830		2896	89000		787	39325		180	9100		320	11200						217455
Service- Compartment	993	30	172	Washing	440	70	Grinding Milling	70	13	Grinding Milling	13	-	-	-	933	1016	553			
		1500		983	22000		70	6500		13	650									30650
Outer Contr. Area Main and Reactor Deck	1310	1310	46	Paint/ Rust Rem.	46	8	-	8	-	-	-	-	-	-	1310	54	1364			
		65500		46	2300		8	400												68200
Hedge No. 4	480	170	218	Washing	480	142	Rust Re- moval	202	102	Grinding Milling	102	100	Grinding Milling	100	480	874	1054			
		8500		480	24000		192	1100		102	5100		100	5000						43700
Observation-Area Double Bott. Tank 0801 (Eng. Room)	1200	-	1200	Rust Re- moval	-	1200	Rust Re- moval	1200	320	Rust Re- moval	350	4	Milling/ Pickling	4	1200	2724	1554			
		-		1200	-		1200	60000		320	17500		4	200						77700
Wing and Double Bottom Tanks (Nuclear Area)	5660	580	1605	Rust Re- moval	2640	815	Grinding Milling	835	85	Grinding Milling	105	41	Milling	41	5660	3221	4201			
		29000		2280	132000		815	31750		85	5250		41	2050						200050
T o t a l	14681	4925	5987	9165	6231	3312	3962	3392	793	793	845	241	465	369	14621	14385	15762			
		215080			309300			153575			42250			18450			738655			

- 1) left upper number: measured area, m<sup>2</sup>  
left lower number: measuring positions  
right upper number: area exceeding release limit, m<sup>2</sup>  
2) upper entry: decontamination procedure  
lower number: area treated, m<sup>2</sup>  
3) upper number: total measured area, m<sup>2</sup>  
lower number: total measuring positions

Mass and Activity Transfer

Year Month	Safe-stored Components Quantity		Activity		Released Components Cat.3.2.3;3.4.3;4.4.1		Total Transfer (Solids)		WaterTransf.Cat.5.1 System <sup>1</sup> Waste <sup>2</sup> -Water		Aerosol-Emission Exhaust Air	
	Contain. kg	Drums kg	Contain. Bq	Drums Bq	Quantity kg	Activity Bq	Quantity kg	Activity Bq	Quantity m <sup>3</sup>	Activity Bq	Quantity m <sup>3</sup>	Activity Bq
1980 June	8800	-	5,18 E+5	-	-	-	8800	5,18 E+5	65,6 16,4	1,22 E+9	N.A.	N.A.
July	8100	4160	2,78 E+9	3,66 E+9	40200	1,81 E+5	52460	6,44 E+9	-	-	"	"
August	-	1820	-	4,81 E+8	68070	1,92 E+5	69890	4,81 E+8	-	-	"	"
Sept.	18400	3400	5,55 E+9	3,40 E+9	29543	1,74 E+5	51343	8,88 E+9	22,0	2,18 E+9	"	"
October	-	3100	-	1,41 E+9	5637	1,52 E+5	8737	1,41 E+9	-	-	"	"
November	6899	4675	1,26 E+8	1,78 E+9	8053	3,18 E+5	19627	1,89 E+9	-	-	8,2 E+6	2,59 E+3
December	11592	3630	1,70 E+9	5,55 E+7	500	1,30 E+3	15722	1,76 E+9	21,3	2,66 E+8	1,0 E+7	1,11 E+4
1981 January	19552	6050	1,67 E+10	5,92 E+9	11530	2,59 E+6	37132	2,26 E+10	19,0	1,04 E+8	8,1 E+6	5,18 E+3
February	20909	-	3,22 E+8	-	25866	1,55 E+5	46775	3,22 E+8	-	-	7,4 E+6	2,40 E-7
March	26973	3240	3,52 E+10	9,99 E+7	13330	1,81 E+5	43543	3,52 E+10	22,0 21,5	3,18 E+8 5,55 E+8	8,6 E+6	1,63 E+4
April	8600	4270	2,33 E+6	3,44 E+8	17504	6,66 E+5	30374	3,44 E+8 <sup>3</sup>	22,0 22,0	4,81 E+8 2,11 E+8	8,7 E+6	6,29 E+3
May	6300	2840	8,14 E+8	2,78 E+9	7004	1,63 E+5	16144	3,59 E+9	43,0	1,3 E+8	7,8 E+6	5,18 E+3
June	15224	-	2,07 E+9	-	2233	1,37 E+5	17457	2,07 E+9	20,8 22,2	7,77 E+8	8,6 E+6	5,92 E+3
July	-	3180	-	6,29 E+8	17061	6,66 E+5	20241	6,29 E+8	22,0	2,48 E+7	1,3 E+7	2,55 E+3
August	3276	920	2,11 E+7	6,66 E+8	7195	4,81E+5	11391	6,88 E+8	22,0	4,07 E+8	4,9 E+6	1,67 E+2
Sept.	6600	-	1,89 E+8	-	4334	2,00E+5	10934	1,89 E+8	21,2	7,40 E+7	8,6 E+6	1,04 E+3

Table A 3

<sup>1</sup>) upper number  
<sup>2</sup>) lower number

Table A 3 (cont.)

Year Month	Safe-stored Components Quantity Activity				Released Components Cat.3.2.3;3.4.3;4.4.1		Total Transfer (Solids)		WaterTransf.Cat.5.1 System <sup>1</sup> )Waste <sup>2</sup> )-Water		Aerosol-Emission Exhaust Air	
	Contain. kg	Drums kg	Contain. Bq	Drums Bq	Quantity kg	Activity Bq	Quantity kg	Activity Bq	Quantity m <sup>3</sup>	Activity Bq	Quantity m <sup>3</sup>	Activity Bq
October	-	4380	-	9,25 E+8	5154	5,55 E+5	9534	9,25 E+8	22,0	2,22 E+7	1,0 E+7	1,04 E+3
November	8500	5050	1,15 E+7	2,70 E+9	42214	2,37 E+5	55764	2,71 E+9	20,0	2,37 E+7	5,8 E+6	7,77 E+2
December	-	-	-	-	-	-	-	-	14,0	6,66 E+7	8,4 E+6	2,96 E+3
1982 January	8176	11810	4,44 E+6	4,44 E+6	-	-	19986	4,44 E+10	12,5	6,59 E+7	8,0 E+6	1,22 E+3
February	1424	5310	3,33 E+6	1,67 E+9	17650	5,55 E+5	24384	1,67 E+9	-	-	8,6 E+6	3,26 E+3
March	-	-	-	-	8495	2,11 E+5	8495	2,11 E+5	15,0	1,30 E+6	8,06E+6	6,66 E+2
April	-	24470	-	2,48 E+9	1560	1,37 E+5	26030	2,48 E+9	15,0	9,25 E+5	5,47E+6	1,52 E+3
May/June	-	33630	-	5,55 E+8	41025	3,63 E+5	74655	5,55 E+8	2,0	8,88 E+5	-	-
Total	179325	125935	6,59 E+10	7,36 E+10	374158	8,14 E+6	679418	1,40 E+11	151,4 332,1	6,66 E+9	1,4 E+8	7,77 E+4

<sup>1</sup>) upper number

<sup>2</sup>) lower number

Balance of the Activity Transfer  
Initial-State and Final-State Activity-Inventory

Contaminated Component or Material	Reference Area or Quantity	Specific Activity	Activity Inventory
<u>1. Initial State</u>			
Component	cm <sup>2</sup>	Bq/cm <sup>2</sup>	Bq
Piping	2,25 E+6	1,02	2,29 E+6
Main Turbine	7,00 E+5	0,78	0,55 E+6
Main Condenser	1,30 E+7	1,61	20,94 E+6
Aux. Condenser	1,10 E+6	1,11	1,22 E+6
Aux. Boilers	3,83 E+6	0,74	2,83 E+6
Deaerator	4,80 E+5	1,81	0,87 E+6
Condensate Tank	1,00 E+5	0,88	0,09 E+6
<b>Total</b>	<b>2,15 E+7</b>	<b>1,34</b>	<b>28,79 E+6</b>
<u>2. Final State</u>			
Component	cm <sup>2</sup>	Bq/cm <sup>2</sup>	Bq
Piping	2,05 E+6	0,12	2,24 E+6
Main Turbine	7,00 E+5	0,22	0,16 E+6
Main Condenser	1,30 E+7	0,037	0,48 E+6
Aux. Condenser	1,10 E+6	0,044	0,05 E+6
Aux. Boilers	3,83 E+6	0,12	0,47 E+6
Deaerator	4,80 E+5	0,185	0,09 E+6
Condensate Tank	1,00 E+5	0,22	0,02 E+6
<b>Total</b>	<b>2,13 E+7</b>	<b>0,071</b>	<b>1,51 E+6</b>
<b>Activity Transfer</b>			<b>27,28 E+6</b>

Table A 4

Balance of the Activity Transfer  
Activity Inventories of the Removed Components and the Decontamination-Agent  
respectively the Secondary-Waste

Contaminated Component or Matter	Reference Area or Quantity	Specific Activity	Activity Inventory	Balancing Items	
				A (Bq)	B
<u>2. Pickling Acid Agent</u>	m <sup>3</sup>	Bq/cm <sup>3</sup>	Bq		
Charge 1	16,3	0,52	8,48 E+6		
" 2.1	15,0	0,18	2,72 E+6		
" 2.2	13,2	0,18	2,39 E+6		
" 2.3	15,0	0,26	3,90 E+6		
Total	59,5	0,29	17,46 E+6	17,46 E+6	-
<u>3. Waste-Processing Precip. Sludge</u>	kg	Bq/q	Bq		
Charge 1	1615	4,77	7,70 E+6		
" 2.1 - 3	3775	2,83	10,66 E+6		
Total	5390	3,42	18,36 E+6	-	18,36 E+6
<u>Filtrate</u>	m <sup>3</sup>	Bq/cm <sup>3</sup>	Bq		
Charge 1	18,0	0,02	0,38 E+6		
" 2.1 - 3	46,0	0,0005	0,02 E+6		
Total	64,0	0,0064	0,41 E+6	-	10,41 E+6
<u>4. Deposits (non soluble or residual.)</u>					
Dep. removed from:	kg	Bq/q	Bq	Bq	Bq
Boiler Drums	3,1	144,30	0,45 E+6		
Condensate-Tank	40,0	11,66	0,47 E+6		
Main Condenser	5,0	138,38	0,69 E+6		
Deaerator	80,0	70,67	5,65 E+6		
Total	128,1	56,61	7,26 E+6	7,26 E+6	7,26 E+6
<u>5. Pickling-Tests Agents used f. Test-Section</u>	m <sup>3</sup>	Bq/cm <sup>3</sup>	Bq	Bq	Bq
LP-Preheaters	1,28	0,46	0,59 E+6		
HP-Preheaters	0,36	2,78	1,00 E+6		
Total	1,64	0,97	1,59 E+6	1,59 E+6	1,59 E+6
<u>6. Non-decontaminable Component (safe-stored) Components</u>	cm <sup>2</sup>	Bq/cm <sup>2</sup>		Bq	Bq
Test- and Seal-Steam Condenser, Receiver and others	2,00 E+5	0,74	0,15 E+6		
Total	2,00 E+5	0,74	0,15 E+6	0,15 E+6	0,15 E+6
Grand Total				26,46 E+6	27,22 E+6
Surplus					0,49 E+6
Deficiency				0,82 E+6	

Table A 4 (cont.)



## APPENDIX B

### DECOMMISSIONING OF THE POWER HOUSE A OF THE NUCLEAR POWER PLANT GUNDREMMINGEN

#### B.1 Introduction

End of 1976 after about 10 years of operation the nuclear power plant Gundremmingen was shut down due to operational difficulties. Economic investigations led to the conclusion to put the plant out of service finally.

After various alternative decommissioning concepts had been studied with respect to radioprotection and economy, the reactor and most of the components in the reactor area were provided for "protective enclosure" for an undefined period. The powerhouse, containing the components of the turbine circuit, should be submitted to immediate decommissioning by decontamination of all the structural parts concerned.

In order to obtain decommissioning approval preliminary tests, limited to 100 Mg of material to be dismantled and decontaminated, had to demonstrate the acceptability of the decommissioning concept. Furthermore decontamination procedures had to be tested in order to chose the most efficient methods selected for application.

During the tests 110 Mg of various contaminated components, representing the overall decommissioning requirements with regard to specification and geometry of parts and of kind and grade of contamination, were dismantled from circuit sections with typical operating conditions. The components were predominantly decontaminated by electropolishing, which had proved to be the most efficient procedure.

Meanwhile approval of release has been obtained for approx. 106 Mg of dismantled and decontaminated components.

The authorities held out prospect to approve complete decommissioning of the power-house. Sectional decommissioning, continuing the test operations, was authorized immediately.

#### B.2 Influence of Plant-specific Conditions and Decommissioning Procedures

##### B.2.1 General Description of the Power Plant

Plant A of the KKW Gundremmingen is equipped with a 2 loop boiling water reactor. Primary steam was directly supplied to the HP-turbine. The "secondary steam", generated by circulating primary water through 3 steam generators installed in parallel, entered the HP-turbine at the 2nd pressure stage and served to govern power output at fullpower performance mainly. Furthermore all auxiliary systems were supplied by steam of this quality.

The plant is equipped with an ion-exchange cleaning system, designed to continually process the total throughput of condensate of approx. 1 400 t/h. The principle of total condensate purification stipulated to resign from regenerative heat-recycling by direct-contact- or HP-preheating. The heat exchanging surface of the LP-preheaters had to be rated adequately.

### B.2.2 Operating Conditions and Initial-state Contamination Distribution

Before the '100 Mg-decontamination test' and for licensing purposes an assessment of the quantities, contaminated surface areas and activity inventories of components was made.

In order to carry out the survey, the various plant sub-sections were classified with regard to nature, phase, pressure and temperature of the contaminating medium, according to operating conditions. Furthermore, similar components and structures were classified with reference to kind, geometry, surface structure and typical surface/mass-ratios.

Assigning the results of measurements of a large number of samples to the defined categories, the profile of contamination typical for plant sections and various components was established.

The sectional activity inventories were calculated and summed up to the total activity inventory of power-house components. The relatively low contamination (peak values of approx. 1 800 Bq/cm<sup>2</sup> for components of group A (valves, fittings) in section class DP 1 (contaminated by primary steam 70 bar)) is explained by effective condensate purification and the time of decay (6.5 years from reactor shut-down to initial-state survey commencement).

### B.2.3 Initial-state Nuclide Compound

The nuclide compound of contamination was analyzed by single nuclide GeLi-detection and by radio-chemical analysis for all categories of sections and components.

With exception of a subsection of section class DP 1 the Co-60 constituent of the total gamma-activity amounts to 90 % in average. In narrow crevices (press fit clearances, cracks a.o.) of e.g. valves and machinery parts a Cs-137 constituent of approx. 30 % was evaluated. In general the Cs-137 constituent amounts to < 10 % of total gamma-activity.

For the above mentioned particular sub-section of section class DP 1 a Ni-63 constituent of total activity of approx. 50 % was evaluated by radio-chemical analysis. The total low energy beta- and gamma-constituent (Ni-63, Fe-55, Ni-59 a.o.) of all the other samples evaluated by radio-chemical analysis amounts to < 10 % of total activity.

### B.2.4 Decontamination Procedure and Residual Activity

All components provided for unrestricted release in the course of the '100 Mg-decontamination test' and continuing decommissioning operations, which had been contaminated by primary or secondary steam, condensate or feedwater were decontaminated by electro-polishing.

With exception of components with rough surfaces or imperfections of fabrication (cracks, laps, slivers, cold shots and seams), the residual contamination was homogeneous after abrasion of the incrustations and a thin layer (approx. 0.005 mm) of base material.

According to the decommissioning concept, components contaminated by airborne activity were decontaminated by application of conventional decontamination methods like washing, brushing, pickling and acid treatment. Residual contamination distribution after decontamination was homogeneous for these components also.

### B.3 Concept of Release Procedures

#### B.3.1 Limit Values

Limit values for the unrestricted release or reutilization of components and materials dismantled and decontaminated in the course of the '100 Mg-decontamination test' were determined by the licensing authorities with reference to § 4 and § 35 StrlSchV.

Because of the non-detectable low energy beta- and gamma-constituents of the prevailing nuclide compound, the surface contamination limit (according to app.IX clmn. 4 StrlSchV at 0.37 Bq/cm<sup>2</sup>) was lowered to 0.22 Bq/cm<sup>2</sup>.

Activity concentration limits according to § 4 were defined by the expression:

$$A_m = F_m \left( 0.9 * 10^{-4} - \frac{A_n}{F_n} \right)$$

$A_m$  = limit value of activity concentration detectable by gamma-spectrometry

$F_m$  = limit value according to app.IV tbl.IVlclmn.4 StrlSchV for detected radio-nuclides

$A_n$  = lower limit of detection (LLD) for not identified nuclides, but practically present in nuclear systems

$F_n$  = limit value according to app.IV tbl. IVlclmn.4 StrlSchV of not identified nuclides, but practically present in nuclear systems and

This method enables a lumpsum regard of the radiological relevancy without consideration of the actual constituent of nuclides not detectable by gamma-spectrometry but eventually present. Because the LLD for non-detectable nuclides is not clearly defined, this method may restrict limit values unnecessarily strong.

Assuming Co-60 to be the only nuclide present, the resulting limit of activity concentration is 3.33 Bq/g.

Observance of activity concentration limits can be demonstrated by relating the results of the surface-covering contamination measurements to the mass of the components concerned.

Because of the area/mass-ratio of the processed components, the area related contamination limit turned out to be more restrictive, during the 100 Mg-decontamination test' (average activity concentration of the released components 0.065 Bq/g).

Since the average area/mass-ratio of powerhouse components was approx. 240 cm<sup>2</sup>/kg, observance of concentration limits could be assumed for most of the components, which were below the surface contamination limits.

### B.3.2 Release Procedures

The following release procedure was proposal to the authorities:

- The alpha-, beta-surface contamination of components to be released is evaluated by gridwise application of a direct, continuous measuring procedure covering the total surface area.
- In order to perform the release measurements, a measuring site will be installed in a low-level background area outside the control area.
- After positive results of decontamination surveillance measurements (carried out in a low-level background area of the powerhouse), the components, cut to decontaminable pieces, are brought to the release measuring site by a platform wagon.
- A release-decisive threshold counting rate will be determined, taking into account the calibration data for the measuring device and the geometry.
- If the threshold counting rate is at no measuring position (window area of detector) exceeded, the component is provisionally released and stored outside the control area until final release.
- The detected maximum concentration value or the lower limit of detection of the concerned measuring application is recorded.
- The activity inventory is calculated by relating the recorded residual contamination value to the total surface of the single components. Relating the calculated inventory to the mass of single components, the activity concentration is evaluated.
- If the calibrated threshold counting rate is exceeded at one position, the component is submitted to decontamination again. The area exceeding limit values is marked.
- Additionally, random samples are analyzed and evaluated by single nuclide GeLi-detection.
- Samples are cut out from an area of components showing maximum residual contamination values and prepared to calibrated size and geometry.
- Furthermore wiping and scraping samples will be taken and analyzed by single nuclide GeLi-detection.

B.4 Measuring Procedures and Measuring Devices

Continuous measurements with alpha-beta sensitive contamination monitors should be made. The lower limit of detection for continuous measuring procedures is defined as follows:

$$LLD = f * k * \sqrt{\frac{nu}{2 * \tau}}$$

f = calibration factor (of the measuring device applied to a certain measuring geometry)  
 k = safety factor (3)  
 nu = background counting rate  
 τ = delay

Alpha-beta sensitive contamination monitors of the following specification were used:

Monitor:	manufacturer:	Herfurth
	type:	H 13.59 B
Detector:	manufacturer:	Herfurth
	type:	HGZ I-T
	radiation sensitivity:	alpha, alpha + beta
	counting gas:	liquid gas (butane, propane)
	measuring area:	228 cm <sup>2</sup>
	eff. window area:	160 cm <sup>2</sup>

For release measurements to be performed during the '100 Mg-decontamination test', the measuring devices were calibrated for measuring distances of 0 and 2 cm from the surfaces of components using a plain Co-60 standard preparation with an active surface area of 100 cm<sup>2</sup>.

The average background counting rate at the provided measuring site was 5.5 cps. The lower limits of detection, computed with the calibration data evaluated for both measuring distances, are tabulated below (delay set to 10 s).

Measrng.dist. cm	Calib.factor Bq*s/c*cm <sup>2</sup>	LLD (k=3)	
		μCi/cm <sup>2</sup>	Bq/cm <sup>2</sup>
0	6.76 E-2	2.86 E-6	0,106
2	1.33 E-1	5.64 E-6	0,209

The measuring time to achieve sufficient step response was 30 s per position.

The lower limits of detection for a distance of 2 cm just met the requirements of the release procedures.

Measuring at contact, the counting rate calibrated to resemble limit contamination (0.22 Bq/cm<sup>2</sup>) exceeded the LLD-counting rate by 1.5 cps (approx. 3 times the standard background deviation). Measuring at a distance of 2 cm however both the counting rates were almost similar, allowing for accordingly high magnitudes of error.

Because calibration by 100 cm<sup>2</sup> standard preparation does not conform to actual contamination distribution, many components will have to be submitted to another step of decontamination unnecessarily.

In order to reduce measuring as well as decontamination expenditure, the measuring procedures provided for application had to be improved with respects to:

1. Minimization of the necessary single measuring time
2. Minimization of the probable error in assigning detected activity to contamination exceeding limit values

The request to relate the detected activity to an area of 100 cm<sup>2</sup> had to be refrained from. Consequently, monitors had to be realistically calibrated by application of standard preparations adequate to the window area of detectors.

Because a homogeneous distribution of residual contamination could be shown, authorities and experts agreed to relate the detected activity to an area of 228 cm<sup>2</sup> (detector window area).

To reduce the measuring time of 30 s per position, required for application of continuous measuring procedures in order to achieve sufficient step-response even after improvement of calibration data, and to restrict probable errors in assigning counting rates to background as well as contamination exceeding release limits, the previously proposed release procedure was amended as follows:

- The delay set to 10 s, continuous measurements are carried out with a measuring time of 10 s per position.
- In parallel with continuous measurements, signals will be evaluated by a digital, integral measuring procedure employing an additional, programmable data unit supplied by a special output of the contamination monitors. Applying a measuring time of 10 s per position for digital, integral measurements also, the net counting rate is registered.
- Background counting rates are subtracted by routine.
- The digital, integral measuring period is started by a controller mounted to the handle of the detector. An acoustic signal indicates the end of the measuring period.
- Counting rates, analog displayed by the contamination monitor employing the continuous measuring method, are immediately interpreted considering the actual measuring distance. Adequate threshold counting rates, related to an acceptable magnitude of errors, will be determined. If the threshold counting rates is exceeded, the position is marked. The component is submitted to decontamination.
- The threshold alarms of the data unit, employing the digital, integral measuring mode, is set to the determined threshold counting rate also. If the threshold counting rates is exceeded by immediate response to a sudden increase of contamination, the tripped threshold alarm indicates to prolong continuous measuring time to 30 s.

Combining continuous and digital, integral measuring mode the inherent disadvantages of both measuring procedures (slow step response for continuous measurements, higher LLD for digital, integral measurements) could be compensated.

The lower limits of detection, calculated for both procedures from calibration data evaluated for both extreme measuring distances ( 0 and 2.5 cm) are tabulated below:

Measuring procedure	Distance cm	$\mu\text{Ci}/\text{cm}^2$	LLD (k=3) Bq/cm <sup>2</sup>
continuous	0	1,20 E-6	0,044
	2,5	1,87 E-6	0,069
digital, integral	0	1,69 E-6	0,063
	2,5	2,64 E-6	0,098

The lower limits of detection (with a safety factor of 3), calculated with realistic calibration data for a measuring time of 10 s per position (difference to limit contamination values at least 2 times standard gross rate deviation), permitted to determine threshold counting rates granting observance of an error of 2.4 % for the assignment of components, the actual activity of which exceeds limit values, to the materials provided for unrestricted release (error b) or an error of 0.13 % for the assignment of components, the actual contamination of which does not exceed limit values to the material to be submitted to finishing decontamination (error a).

As the surface of a considerable portion of the components to be measured by the described procedure was one-dimensionally curved (pipes, sheet coverings), the error-related threshold counting rates were computed with reference to the mean surface - detector distances calculated for various radii of curvature.

For plane surface components and pipes with a minimum diameter of 250 mm the threshold counting rates tabulated below were determined and approved by authorities and experts.

mean Distance (cm)	Component geometry	Threshold-counting rate (Imp/s)
0	plane surface	5,5
0,5	pipe diam. 800	5,5
1,0	" " 400	5,0
1,5	" " 300	4,5
2,0	" " 250	3,5

With delay time for continuous measurements of 10 s and a measuring time of 10 s (giving sufficient step response as well as detection efficiency) the necessary measuring time could be reduced to 1/3 of the originally proposed release procedure. Further reduction of measuring time and retention of error b = 2.4 % would increase the portion of components unnecessarily submitted to further decontamination.

If these contamination monitors are used for measurement of pipes and valves less than 250 mm in diameter the maximum mean measuring distance is exceeded. Because the irradiating area of complexly structured components does not resemble the detector window area, the residual contamination of such components cannot be evaluated with the accuracy necessary.

Alpha-, beta- or beta-, gamma-sensitive detectors, adapted to particular geometries (e.g. narrowly dimensioned plane detectors, tubular or corner detectors and detector assemblies permitting the measurement of the outer surface contamination of tubes) have not been calibrated in order to determine requisite measuring times and threshold counting rates related to admissible errors yet.

Economy of area-related contamination measurements, apart from the necessary measuring time per position, depends on the area/mass-ratio of the components to be released.

Measuring the tubing of heat-exchangers, thin plates and sheets (e.g. insulating casing), the described measuring procedure is not feasible. Approx. 30 % of the components assessed by initial-state survey belong to such categories.

Furthermore the described measuring procedure is unqualified to evaluate the residual activity or contamination of materials the surface area of which cannot be defined (e.g. insulating material, brick-work and concrete rubble).

Of course the application of alpha-, beta-contamination measuring procedures is also unqualified to evaluate the activity concentration and inventory of activated components or materials.

#### B.5 Future Measuring and Release Procedures

With the progress of decommissioning the amount of components and materials for technical or economical reason not suited for measurements with the introduced procedure will increase. Considering acceptability from the aspects of radio protection as well as economy, new measuring procedures have to be developed.

As a prerequisite, the request to relate contamination limit values to an area of 100 cm<sup>2</sup> will have to be refrained from, the activity concentration limits consequently becoming more significant.

Restricting concentration limits to a 1/10 or even 1/100 of the values tabulated in app. IX StrlSchV with regard to the radiological hazards to the environment, the quantity of releasable components will hardly be reduced because of the prevailing area/mass-relation.

Acceptability of future release procedures may be improved by the obligation to process quantified charges of the materials provided for release by smelting or grinding.



## APPENDIX C

### DETECTION THRESHOLDS

#### C.1. Detection threshold for pulse counting

The possibilities to detect and to measure low activities can be assessed from the general characteristics of the various radiation detectors /C1, C2, C3, C4, C5/.

A radiation detector gives a mixture of impulses due to the background and to the detection of the radiation emitted by the surface checked by the detector. In fact, this background includes the environmental background effect surrounding the surface tested. The detector shielding is not sufficient to remove the total surrounding radiation.

The temporal distribution of impulses is randomly fluctuating. If the measurement is carried over a long time or if a great number of short-timed measurements are carried out, a mean value can be obtained together with a low statistical error.

In practice, taking into account the quantity of measurements to be undertaken, every test point is submitted to a single and short-time measurement. Therefore, we would consider, in relation with the average background :

- the probability to consider the number of impulses as a value belonging to the distribution of both the signal and the background ( $S + B$ ), whereas in reality, it belongs to the background distribution ( $B$ ). This probability is commonly called type-one error,

- the probability to reject the number of impulses as belonging to the background distribution ( $B$ ), whereas in reality, it would belong to the distribution of both the signal and the background ( $S + B$ ). This probability is called type-two error,

- the maximum error made on the signals from the radioactivity detection (after deduction of the average background  $B$  from the indication  $S + B$  given by the instrument).

The detection threshold of the instrument corresponds to the lowest counting (or lowest activity) that can be distinguished from the background for a given probability /C6/.

We only pay attention to very low level measurements and we can neglect the counting loss due to the detector 'dead-time' and to the electronic circuit resolving time.

Statistically, the detection threshold  $S_D$  can be defined as the measurement value for which the relative uncertainty equals  $\pm 100\%$  at 95% probability level.

Impulses relative to the signals and to the background B are considered as random variables. If the distribution of impulses is assumed to follow a normal law (or Gaussian law) which is roughly what happens when the number of impulses is larger than ten, the detection threshold  $S_D$  is :

$$S_D = 2 \sigma_{SD} \quad (1)$$

The signal S is evaluated from two countings :

- $N_1$ , on the background B during a time  $t_1$
- $N_2$ , on the signal and on the background during a time  $t_2$

The variables considered are :

- $N_1$  with a mean value =  $Bt_1$  and a variance  $\sigma_{N_1}^2 = B \cdot t_1$
- $N_2$  with a mean value =  $(B + S)t_2$  and a variance  $\sigma_{N_2}^2 = (B + S)t_2$

In order to assess S, the effective counting rate, the following variable is used :

$$\hat{S} = \frac{N_2}{t_2} - \frac{N_1}{t_1}$$

As the two measurements are independant, the variance of S is :

$$\sigma_{\hat{S}}^2 = \frac{1}{t_2^2} \cdot \sigma_{N_2}^2 + \frac{1}{t_1^2} \sigma_{N_1}^2$$

$$\sigma_{\hat{S}}^2 = \frac{B + S}{t_2} + \frac{B}{t_1}$$

When times  $t_1$  and  $t_2$  are chosen equal :

$$\sigma_{\hat{S}}^2 = \frac{2B + S}{t}$$

and :

$$\sigma_{\hat{S}} = \frac{2B + S}{t}^{1/2}$$

the detection threshold  $S_D$  (relation (1)) is written down as follows :

$$S_D = 2 \sigma_{SD} = 2 \frac{2B + S_D}{t}^{1/2} \quad (2)$$

Therefore, the value  $S_D$  is the solution to the quadratic equation :

$$S_D^2 = 4 \frac{S_D}{t} + 8 \frac{B}{t}$$

the solution to this equation is :

$$S_D = \frac{2}{t} \sqrt{1 + (1 + 2 Bt)} \quad (3)$$

In theory, we need to know B.t, but in practice it is estimated as  $N_1$ ,

$B = \frac{N_1}{t}$ , consequently the detection threshold is :

$$S_D = \frac{2}{t} \sqrt{1 + (1 + 2 N_1)}^{1/2}$$

If  $2B.t$  is great compared to 1, a simpler formula is the following :

$$S_D = 2 \sqrt{\frac{2B}{t}} \approx 2,8 \sqrt{\frac{B}{t}} \quad (4)$$

or, related to the counting rate :

$$S_D = \frac{2,8}{t} (N_1)^{1/2} \quad (5)$$

The greater is the time of measurement, the smaller is the detection rate expressed in counting rate :

$$S_D = 2,8 N_1^{1/2} = 2,8 \sigma_{N_1} \quad (6)$$

If R is the detector efficiency, the activity threshold defined by the formula (4) is written down as follows :

$$S_A (\text{Bq}) = \frac{S_D}{R} = \frac{2,8}{R} \sqrt{\frac{B}{t}} \quad (7)$$

### C.2. Counting ratemeter detection threshold

The average voltage existing at the terminals of a RC integrator circuit (or the average intensity going through the resistance) can be considered as the average number of impulses  $n$  after a time  $t > 2 RC$  (the correct value is obtained with an error of 1 % after a time = 5 RC).

The usual instruments have time constants ranging from three to twelve seconds according to their sensitivity. By analogy with countings, we can consider that :

$$\sigma = (2 R.C. \bar{n})^{1/2} \quad (8)$$

The relation (5)  $S_D = \frac{2,8}{t} (N_1)^{1/2}$  can be written down as follows :

$$S_D = \frac{2,8}{2R.C} \sqrt{2R.C. \bar{n}_B} = 2,8 \frac{\sqrt{\bar{n}_B}}{2R.C} \quad (9)$$

where  $\bar{n}_B$  represents the average background counting during the time  $2R.C.$

C.3. Detection threshold for a detector producing an average current proportional to the dose rate

This is the case of the ionization chamber :

a) The average current is measured throughout a circuit having a constant time R.C. :  
the formula (8) must be used.

b) The measurement is carried out by sampling with a digital voltmeter :

- n readings of  $x_i$  are made (a minimum of ten)
- the average  $\bar{x}$  of these n readings is calculated
- the estimation of the variance is calculated as follows :

$$\sigma^2 = \frac{\sum (x_i - \bar{x})^2}{n(n-1)}$$

- the detector threshold is calculated according to the formula (6) :  
 $S_D = 2.8 \sigma$ .

C.4. Detection threshold for a low background detector (lower than 10 impulses during the measurement time)

This is particularly the case for alpha-rays detectors. It is impossible to use the previous relations because the distribution of the impulses can't be associated to a Gaussian law. It should be better to use the Poisson law /C7/.

a) Poisson's distribution

The distribution fonction is :

$$p \{x\} = e^{-\xi} \cdot \frac{\xi^x}{x!} ; \quad x = 0, 1, \dots, n$$

It expresses the probability for this event to happen x times on n observations, the mean of the distribution being  $\xi$ .

If we look in an integrated Poisson's law table, for the value of x corresponding to a type-one error of 0.05 for the values  $\xi = 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 12, 15$ , we can draw the simplified table C1 (round values).

Therefore, the threshold column gives the lowest impulses number which allows to consider that impulses coming from the detector are produced by radioactivity detection, besides the background. The background is then supposed to be known.

b) Practical threshold

In practice, the detection threshold will hardly depend on the background, for a short-time measurement carried out with a low background detector protected from the surrounding radiation or insensitive to the beta-gamma background (which is the case for alpha radiation detectors). If there would be no more limitations particular to the measuring instrument, the detection threshold, when there is no background, would be very low.

The practical threshold will be determined by the other characteristics of the measuring instrument. For example, it could be subjectively set to a tenth of the maximum range of the most sensitive scale. Moreover a counting rate based on a few number of impulses is almost meaningless. If the background can be neglected during the counting time, the useful threshold could be admitted to represent the activity necessary to count an average number of 10 impulses.

### C.5. Detection threshold for various detectors or monitoring instruments

It's not easy to classify the various instruments in regards with their specific use because of the lack of standardization of the characteristics given by the manufacturer. That is why we did not pay attention to the features mentioned in the manufacturers' catalogs. Therefore, as shown in Table C2, we only took as reference the few detectors that are most used in surface activity checking /C1/.

According to the purpose of the previous paragraphs and for measurement times of 30s and 5s we calculated the activity detection threshold.

Considering a background of  $N_B$  impulses during the time  $t$ , the detection threshold corresponding to this measurement duration  $t$  is expressed in number of impulses :

$$S_D = 2.8.(N_B)^{1/2}$$

The total number of impulses given by the detector is  $B + S_D$ . The detection threshold expressed in counting rate or in number of impulses per second (i/s) is :

$$S'_D = \frac{S_D}{t}$$

The background is given for a surrounding activity of 10  $\mu$ rad/h.

The efficiency  $R$  is the ratio of the number of particles detected by the number of desintegrations during the same time (column 3).

$$S_A \text{ (Bq)} = \frac{S'_D \text{ (i.s}^{-1}\text{)}}{R} \quad (10)$$

Dividing by the effective area, we obtain :

$$S'_A \text{ (Bq.cm}^{-2}\text{)} = \frac{S_A \text{ (Bq)}}{S \text{ (cm}^2\text{)}} \quad (\text{columns 4 and 5})$$

The effective area takes into account the transmission factor of the sensor protection grid.

C.6. Comments on tables and curves briefing the characteristics of detectors usually used in low radioactive portable monitoring instruments

a) As an example, table C2 shows an investigation on the values detection capacity of the following counters : a small surface GEIGER MULLER counter commonly called 'BELL type GEIGER MULLER counter (line A), an average-sized tubular GEIGER MULLER counter (line B), some wide window proportional air or gas flow counters (lines C, D, E), as well as different types of scintillation counters (lines F to S).

The background of a proportional counter can vary widely according to the level above which the impulses are counted (according to the radiation energy to be detected). Such variation was noticed for the scintillation detector on the line I. The impulses produced in the detector by low energy radiations have a low magnitude. Therefore, there is a substantial influence of background and a high detection threshold.

The longer the time of measurement, the lower the activity measured. Depending on the time counting, the most sensitive detectors listed in table C2, would detect the activities whose importance is shown in table C3.

The values shown in table C2 in the detection activity threshold columns (4 and 5) associated with the letter "P", are related to an estimation according to the Poisson's distribution. This distribution assumes the possibility to detect a low radioactivity from the background (when the background is lower than 10 impulses during the time of measurement).

b) The error risk coming from the definition of the detection threshold, must be taken into account to get a proper interpretation of the concepts previously expressed and of the values written down in the tables. The hatched areas are indicating the error risks.

Fig. C1 represents the position of the signal calculation (in regards with the value 0 of the measurements) for a normal distribution with a standard deviation  $\sigma_S$  when the mean value S equals the detection threshold  $S'_D$  defined by the relation (1).

$$S_D = 2 \sigma_{SD} \quad (1)$$

Fig. C2 represents the distribution of the results when the mean value S is twice the value  $S_D$  of the detection threshold.

It appears that if the calculated value  $N_2 - N_1$  equals the detection threshold  $S_D$ , the confidence limits are 0 and  $2S_D$ .

If  $N_2 - N_1 < S_D$ , the signal S is not detected, or the hypothesis that it is only the background is admissible.

The value  $\alpha = 2.5 \%$ , figured out by the hatched area on the right hand-side of the background distribution curve, is a probability called type-one error. Any value  $N_2$  included in this area can be considered as making part of the distribution signal plus background whereas in reality, it would belong to the background distribution only.

The value  $\beta = 2.5 \%$ , figured out by a hatched area and located on the left hand-side of the distribution curve of the signal plus background is a probability called type-two error. Any value  $N_2$ , included in this area, implies to associate the value  $S$ , shown by the instrument, to the distribution  $B$ , whereas in reality it would make part of distribution  $S + B$  (fig. C3).

c) If a better evaluation of a radioactivity measurement is required (instead of only detecting radioactivity), we should double the activity corresponding to the detection threshold calculated according to the relation (6).

d) The previous results have been generalized :

- Table C4 together with the charts of fig. C4 and C5 show in counting rate (impulses per second) for measurement times ranging from 5 to 60 seconds, what would be the detection threshold of monitoring instruments having a background varying from 0.17 to 33 impulses per second.

The minimal time of measurement to detect a given activity can be determined, according to the detector efficiency, from the chart of fig. C6.

#### C.7. Detection threshold for spectrometry measurements

The detection limit through gamma spectrometry is widely dependant on the analysis technique, the external background, the radionuclides mixed with the one to identify.

In the detector or in its surrounding, every radionuclide produces a background (Compton diffusion and bremsstrahlung) that may hide the low intensity peaks. The detection limit is improved when the background ( $B$ ) is decreasing and when the detector efficiency  $e(E)$  increases. This efficiency depends on the radiation energy  $E$  and on the effective volume of the detector /C8/. The lowest gamma flux that can be detected, for a given time of measurement and a resolution  $r$  of the spectrometer, is as follows :

$$\frac{(r.B)^{1/2}}{e(E)} \quad (1)$$

The average value of the background is proportional to the duration of measurement. The minimum gamma flux ( $G_m$ ) that can be measured is decreasing according to the reverse square root of the measurement time.

A total absorption peak, the shape of which is approximately described by a gaussian curve, could be detected if its surface exceeds the error corresponding to the background surface, supposed constant inside the interval defined by the adopted confidence level.

For a confidence level of 0.95, the minimum detectable surface  $P_m$  of the peak is demonstrated to be :

$$P_m \approx 2,4 \quad (1 + (1 + 2,4 r.B))^{1/2} \quad (2)$$

The minimum detectable gamma flux  $G_m$  which is significant at a probability of 0.95 is :

$$G_m = \frac{P_m}{t \cdot e(E)} \quad (3)$$

The peak resolution being :

$$R = 2.4 \sigma \quad (4)$$

$t$  is the counting time

$B$  is the average value of the background under the peak (external radiation, Compton diffusion, bremsstrahlung, detector thermal noise) ...

$e(E)$  is the absolute total absorption efficiency for the energy  $E$ .

For an energy line  $E$  of absolute intensity  $I(E)$ , with a peak surface  $P(E)$  with an active counting time  $t$ (seconds), corrected in relation with the 'dead-time' and with a radioactive source  $A$  (expressed in Becquerels), the efficiency  $e(E)$  is calculated as follows :

$$e(E) = \frac{P(E)}{t \cdot I(E) \cdot A} \quad (5)$$

The measurable minimum volumic activity  $A_m$ , for a sample of volume  $V$ , is :

$$A_m = \frac{G_m}{I(E) \cdot V} \quad (6)$$

For a radionuclide emitting several gamma-rays, the absolute intensity is calculated for each ray, and the smallest value is used in the formula (6).

Owing to the various precautions, activities about  $1 \text{ E-4 Bq/cm}^3$  can be measured.

Table C5 shows backgrounds and detection limits that are measured on gamma spectrometers. The corresponding minimum gamma fluxes are written in table C6.



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Background $\xi (= B)$ impulses	Threshold (for $P_x = 0.95$ ) impulses
1	3
2	4
3	6
4	7
5	9
6	10
7	11
8	13
9	14
10	15
12	18
15	21

Table C1 : Detection limits (Poisson distribution)

1 Type of detector	2 Background (i.min <sup>-1</sup> )	3 Measurement efficiency (4 π) (%)	4 Detectable activity t = 30 <sup>s</sup> (Bq.cm <sup>-2</sup> )	5 Detectable activity t = 5 <sup>s</sup> (Bq.cm <sup>-2</sup> )
A Bell type organic Geiger Muller	25	C 14 (β) 20%	0.24	(P) > 0.3
B Cylindrical halogen Geiger Muller ∅ = 35 mm; L = 190 mm	50	Sr 90 + Y 90 8 %	0.33	(P) > 0.42
C Air filled counter	10	U 235 (α) 13 %	(P) > 0.003	(P) > 0.01
	"	Pu 239 (α) 9 %	(P) > 0.005	(P) > 0.015
D Gaz flow counter	1.8	Am 241 (α) (HV = 750 V) 21 %	(P) > 0.0014	
	180	Tl 204 (β) (HV = 1700 Volts) 31 %	0.012	0.03
E Gaz flow counter	6	Am 241 (α) (HV = 750 Volts) 21 %	(P) > 0.0007	
	600	Tl 204 (β) (HV = 1700 V) 31 %	0.007	0.018
F Scintillator ZnS (Ag) 70 X 3	1	Am 241 (α) 26 %		
G Fluorescent plastic scintillator 70 X 3 mm	60	Sr 90 + Y 90 (β) 14 %	0.12	(P) > 0.2
H Fluorescent plastic scintillator 40 X 40 mm	600	Co 60 (γ) 5 %	2.6	6.4
I NaI (Tl) cristal 32 x 25 mm	600 à 2280 depending on impulse height	Co 60 (γ) 5 %	2.6 à 8	10 à 20
J NaI (Tl) cristal 32 x 5 mm	600	Fe 55 ; Pu 238 ; Pu 239 (X) 25 %	0.9	2.3

(P) : Background < 10i - Poisson distribution

Table C2 : Characteristics of the detectors

Detector volume cm <sup>3</sup>	21		50		114	
Relative efficiency at 1.3 MeV	4 %		10 %		25 %	
Resolution at 1.3 MeV	2.4 keV		2.2 keV		2.7 keV	
Peak efficiency at 1 MeV	0.1 %		0.4 %		0.87 %	
Energy keV	Background $1.s^{-1}.keV^{-1}$ $\times 10^{-3}$	Detection limit $1.s^{-1}$ $\times 10^{-3}$	Background	Detection limit	Background	Detection limit
100	9.1	1.3	6.1	0.7	3.8	0.66
300	3.7	0.85	2.4	0.4	1.8	0.48
500	2.4	0.7	1	0.26	0.86	0.34
1000	1.9	0.47	0.36	0.16	0.6	0.28
1500	0.27	0.24	0.1	0.08	0.17	0.15
2000	0.16	0.19	0.07	0.07	0.1	0.12

Table C5

Characteristics of some  $\gamma$  - spectrometers (Ge-Li)  
Counting time : 15 hours

Type of emission	t = 30 s	t = 5 s
X	0.9	2.3
beta, gamma	0.007	0.018
alpha	0.002	0.014

Table C3

Detectable surface activities for the most sensitive detectors  
Bq/cm<sup>2</sup>

Counting time (s)	Background (imp/s)								
	0.17	0.42	0.83	1.7	3.3	8.3	17	33	
60	0.15	0.23	0.33	0.47	0.66	1	1.5	2	
30	$N_B < 10i$	0.33	0.47	0.66	1	1.5	2	2.6	
10	$N_B < 10i$	$N_B < 10i$		1.14	1.6	2.6	3.6	5.1	
5			$N_B < 10i$		2.3	3.6	5.1	7.2	

Table C4

Detection limit depending on background and counting time (imp/s)

Energy (keV)	Detector volume (cm <sup>3</sup> )		
	21	50	114
100	6.4	3	1.3
300	12	5	2.3
500	19	5	2.5
1000	23	6	3.8
1500	24	6	3
2000	26	5	3

Table C6

Minimum detectable  $\gamma$  flux by spectrometry Ge-Li ( $\gamma \cdot s^{-1} \times 10^{-2}$ )

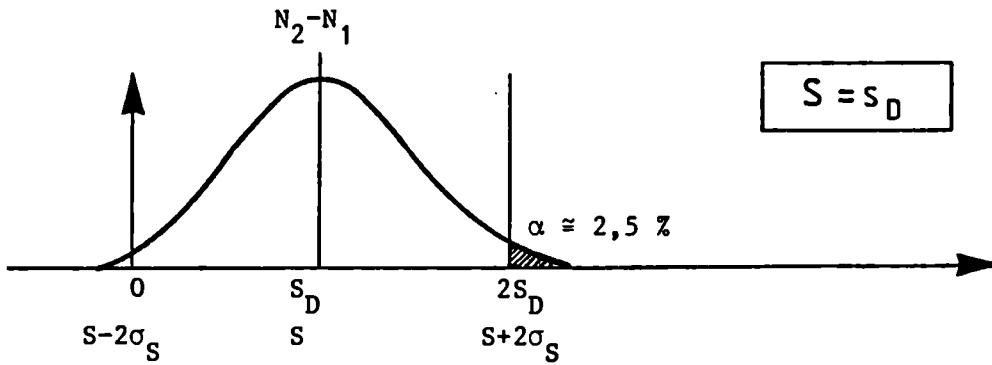


Figure C1 (The mean  $S$  is equal to the threshold)

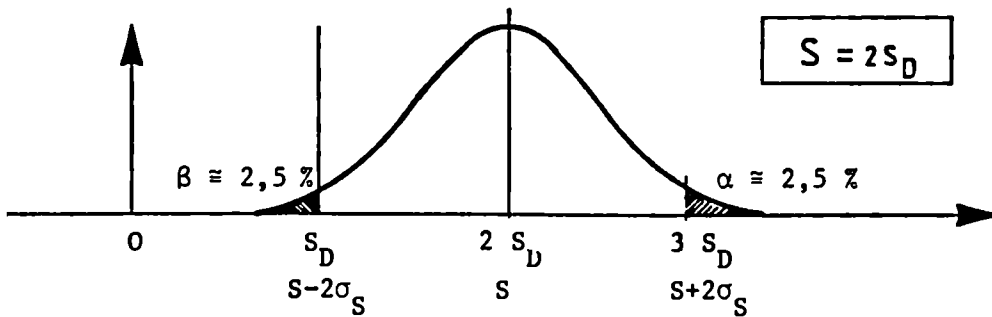


Figure C2 : (The mean  $S$  is twice the threshold)

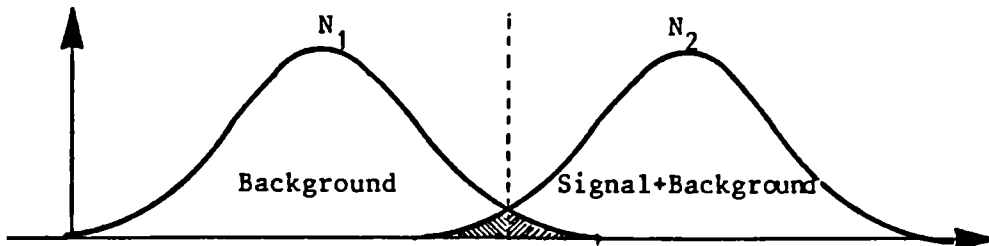


Figure C3

Figures C1, C2, C3 : Distribution of the signal and background  
 Position of the signal measure results function of the threshold  
 and the choosed type-one and type-two errors.

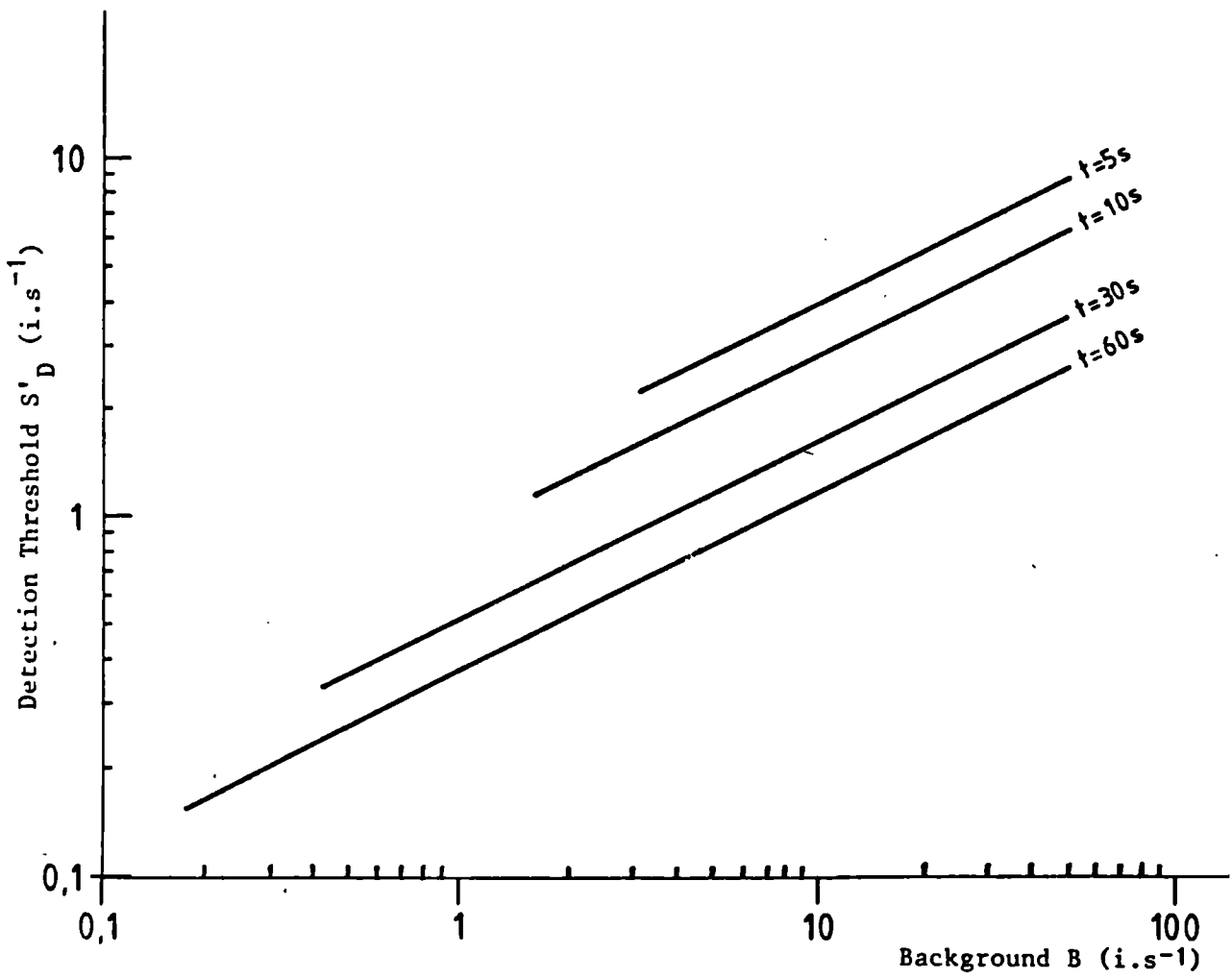


Figure C4 : Relations between counting time, threshold and background



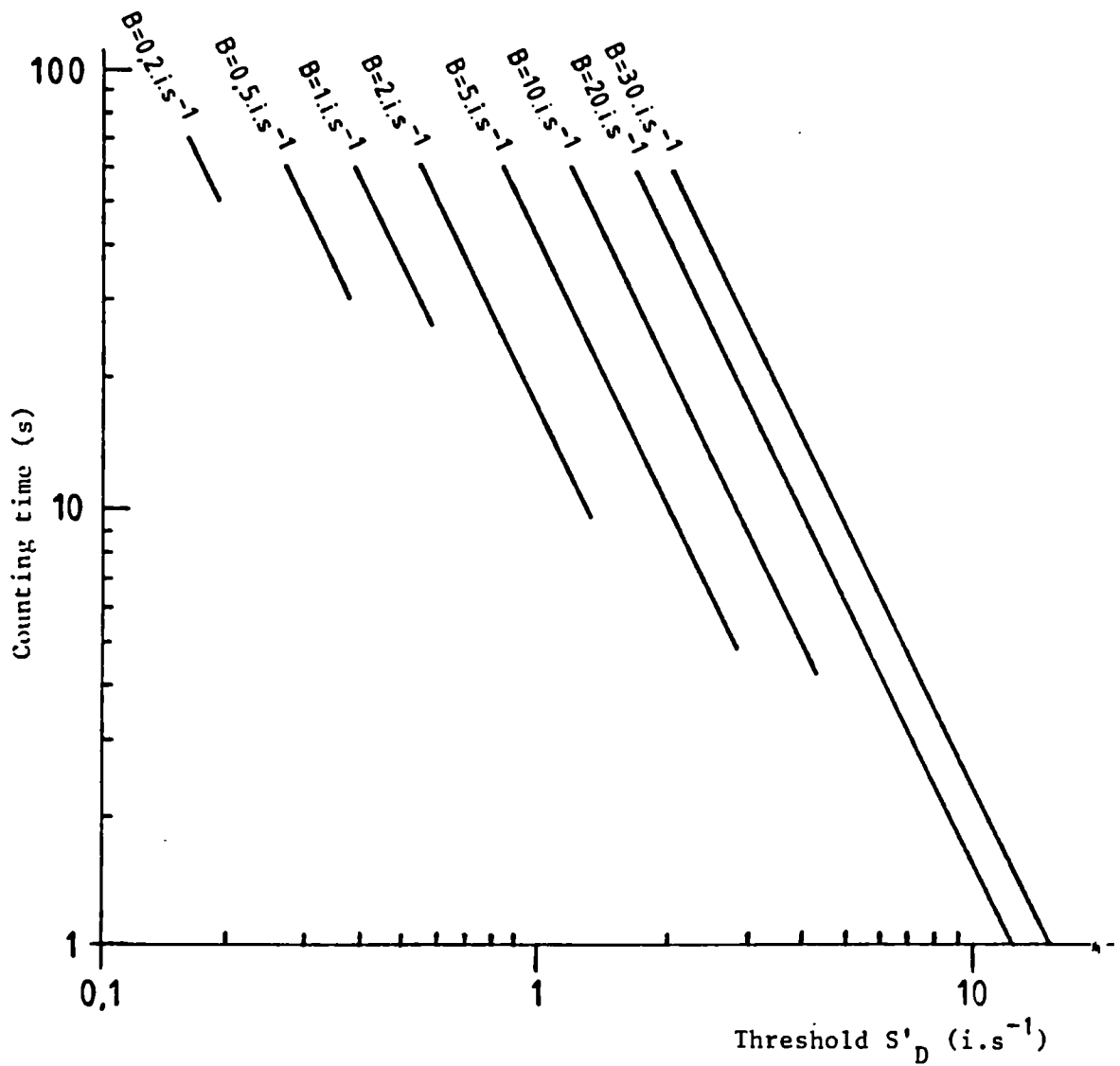


Figure C5: Relations between background, counting time and threshold

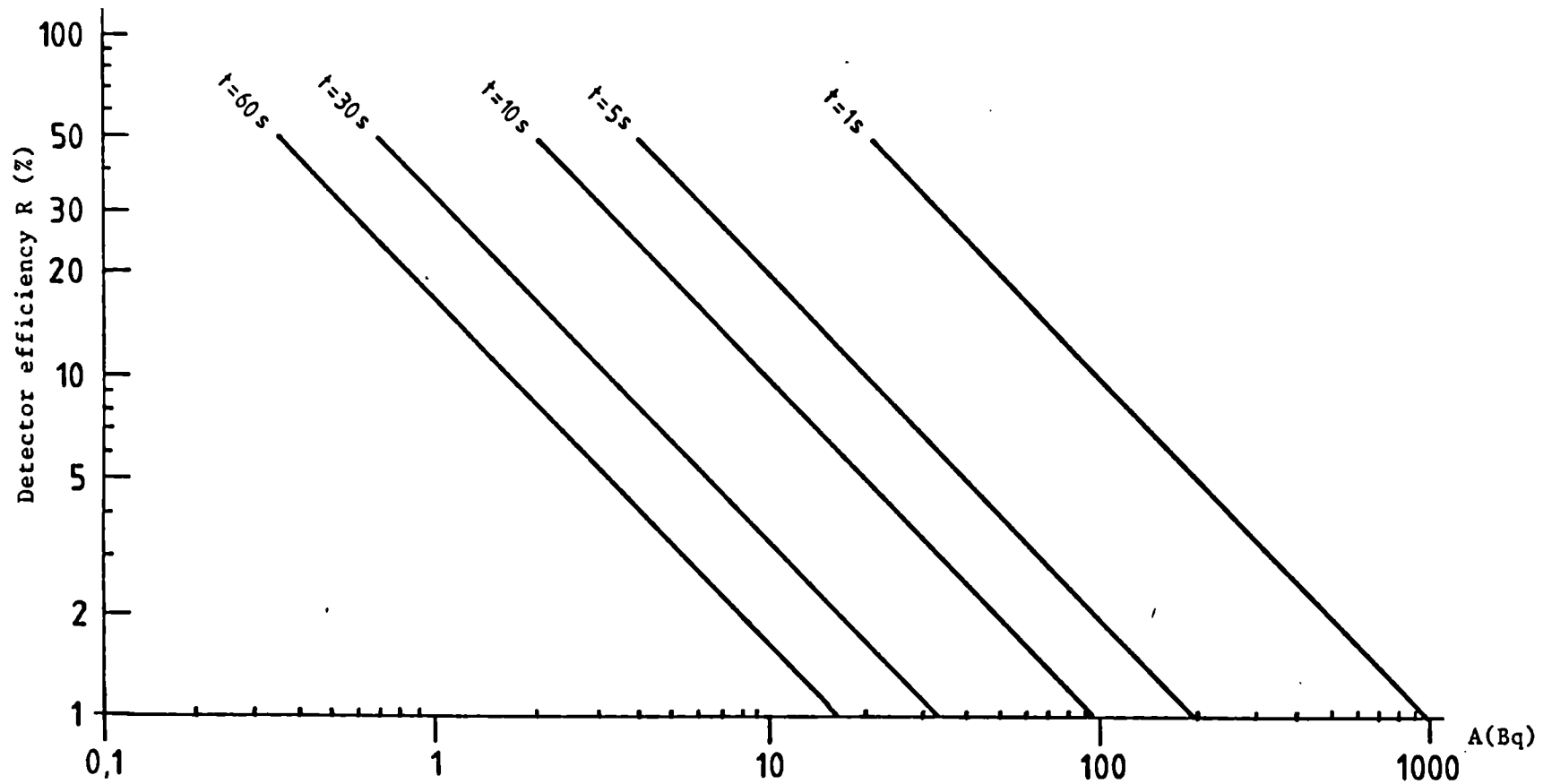


Figure C6: Minimal counting time to detect and given activity depending on the detector efficiency





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**EUR 10643 — State-of-the-art review on technology for measuring and controlling very low level radioactivity in relation to the decommissioning of nuclear power plants**

*M. Hulot, A.M. Chapuis, H. Garbay, W. Jager, J. Kröger*

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Information in relation to the measurement devices for very low radioactivity, applied in the decommissioning of electronuclear power plants, is scarce and scattered. However, some realistic information can be obtained from the specialists working on nuclear plant dismantling sites.

In practice, the *in situ* measurements deal with two or three radionuclides only. The other containments must be determined on some samples in laboratories by means of radioanalysis and spectometry technologies. These technologies are briefly described.

One of the problems concerns the detection threshold of the measuring instruments. This threshold varies in relation with the detector efficiency, the background and the measurement duration. The application of this concept of threshold to various types of instrument is studied. Tables and graphs are given that sum up the various possibilities of detection for the usual detectors according to their effective area.

The possibility of reducing the number of monitoring points using statistical methods is presented.



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