Radioactivity, Radiation Pr otection and Monitoring During Dismantling of Light-Water Reactors

J. B. Zech, L. Hummel

TÜV Süd Group / D-80686 Munich, Westendstr. 199

Abstract:

Based on the radioactivity inventory in the systems and components of light-water reactors observed during operation, the impact of actions during plant emptying after the conclusion of power operation and possible subsequent long-term safe enclosure concerning the composition of the nuclide inventory of the plant to be dismantled will be described. Derived from this will be the effects on radioactivity monitoring in the plant, the monitoring of emissions in the exhaust air and wastewater from the plant, physical radiation protection monitoring, and the measured characterization of the residual materials resulting from the dismantling. An area of concentration will be the monitoring of the material flows resulting from the dismantling, which are to enjoy restricted or unrestricted release (“clearance”) or disposal in areas outside of nuclear facilities; however, the impact of long-term interim storage will also be addressed in the discussion.

The talk should provide an overview of the interrelationships between source terms, decontamination strategies, decay times and the radioactivity monitoring requirements of the various dismantling concepts for commercial light-water reactors.

1 PRINCIPAL PROCEEDING

Shutdown and dismantling are phases in the life cycle of a nuclear power plant, which must be taken into consideration from the outset in the design and development stage. Permanent use of nuclear power to meet our energy needs can only be justified if plants can be safely and properly dismantled and the resulting waste material disposed of. To this extent, the dismantling of a nuclear power station is not first and foremost a step towards bidding a final farewell to this technology, but can equally well be the starting point for building and commissioning a new generation of power plants, whether nuclear or conventional.

This transition from a nuclear power plant in full operation to a location with unrestricted availability for a new plant ("from the last megawatt to a green field") necessitates going through a series of operating states which are described in more detail in the following section. In all these phases, radioactivity in the power plant (buildings, systems and waste) represents a boundary condition of crucial importance when planning and carrying out such work.

The general process can be divided into the following phases:
–Full operation
–Post-operational phase
–Residual operation
–Safe enclosure
–Dismantling

These phases follow each other for varying durations of time and may partially overlap. The phase of safe enclosure may not be necessary and it may be possible to dismantle the plant immediately. These phases can be characterised as follows with regard to the question of radioactivity and radiation protection:

In full operation, the radioactive inventory of the plant is heavily dominated by the inventory of the reactor core, which, in turn, is dominated by short-life fission products. The inventory of long-life radionuclides is determined by the number of spent fuel elements in the storage facilities and the amount of treated and untreated waste in the plant's waste storage areas. After the plant has been switched off and the fuel elements removed, this inventory dominates together with the activated systems and components in the reactor area.

The post-operational phase basically represents the continued running of the plant but can be compared to a lengthy downtime period. The main activities here are the removal of spent reactor fuel, the decontamination of parts of the plant and the treatment of operational waste. The system is prepared for the subsequent operational phases, but remains unchanged in terms of its essential technical facilities. At the end of this post-operational phase comes a scientific analysis of the radioactive inventory in different parts of the plant by level of activity, proportion of radionuclides and types of material. This analysis forms the basis for planning the actual dismantling phase and for deciding how the resulting materials are to be disposed of.

The post-operational phase may be followed by a period of safe enclosure of varying duration in which no work is carried out in the plant besides essential maintenance and monitoring activity. The purpose of this period is to allow the radioactive inventory to diminish further through decay, thereby making the later dismantling process easier. However, work can also start without further delay on the residual operation phase or on dismantling the plant.

Residual operation comprises setting up and operating the facilities required for the dismantling phase as well as setting up and operating facilities for treating and storing the residual material and waste resulting from the dismantling process. Major alterations are carried out on the plant in order to gain the space needed for dismantling it. In many cases, new buildings have to be erected in this phase for collecting, treating and storing waste in order to guarantee the free flow of material in the later dismantling stage.

Dismantling itself comprises the step-by-step removal of system parts inside the plant, in some cases directly and in some by means of remote handling devices depending on the level of activity and dose rates. Greater attention must be paid to the protection of operating personnel in this phase as a result of opening systems and dismantling radioactive parts of systems and buildings. The dismantling sequence is based on the one hand on technical aspects (availability of facilities in later dismantling steps) and constructional circumstances, and on the other on the requirements of containing activity with regard to the protection of operating personnel and the environment. In particular, the restrictions imposed by atomic power legislation make it essential to ensure that the release of radioactive material in exhaust air and waste water is monitored and a record kept of the type and quantity measured in all phases of the dismantling process until final clearance of the buildings and land.

2 ACTIVITY IN LIGHT-WATER REACTORS AFTER SHUTDOWN

A modern nuclear power plant with an electrical power output above 1,000 MW contains a huge amount of radioactive nuclides. During operation the total inventory is above 10^{20} Bq which is composed of some hundreds of different, partly short-lived isotopes. After the final shutdown of this plant the radioactive inventory decreases by several orders of magnitude. Without any human intervention this decrease is caused by radioactive decay. Nevertheless, the main contribution is from de-fuelling the plant.

In this stage, typical inventories are on the order of 10^{15} (e.g. NPP Mülheim-Kärlich) to 10^{17} Bq (e.g. NPP Stade). More than 99% of this activity is located in the core, its internals and the reactor cavern. After dismantling and removing these systems and the reactor coolant system, the inventory is reduced by another three to four orders of magnitude.

The main radiological boundary conditions for dismantling a nuclear power plant are established by its design, in particular the choice of materials, and operation of the plant:

- The quantity and composition of the radioactive materials generated in running the plant are determined by the choice of materials used for the reactor elements and the reactor circuit components. Characteristic radionuclides include Co-60, Fe-55, Ni-63, Sb-125. Co-60 is used as a "key nuclide" for monitoring operation of the power plant because it is well quantifiable with gamma measurements.
- Additional fission products occur in the reactor cooling circuit if the fuel rod cladding tubes are damaged. Typical fission products are I-131, Sr-90 and Cs-137. I-131 is of no significance during dismantling due to its short half-life (8.02 d). Cs-137 is generally used as a "key nuclide" for the proportion of fission products.
- If there is considerable damage to the fuel rod cladding tubes, reactor fuel can also get into the reactor circuit, which leads on the one hand to contamination by alpha emissions from the reactor fuel. Typical of such emissions are the U, Pu and Am nuclides. Am-241 is normally used as the "key nuclide" for establishing the proportion of these nuclides in waste. On the other hand, these fission materials produce further fission products in the reactor's neutron field on a continuous basis which can lead to local contamination with a very high level of activity ("hot spots").

These processes are mitigated by the use of cleaning systems when the plant is in operation, by the way in which the reactor is started up and shut down and by the decontamination of the circuits carried out during plant inspections. The overall result is that a typical radionuclide mixture ("finger print") is formed in separate parts of a nuclear power plant, which is characteristic of the composition of the activity produced by waste occurring in this area. This applies to the operating phase of a plant as well as to each phase of the dismantling process. The composition of this radionuclide mixture changes continually as a result of radioactive decay.

2.1 Variability of the “finger print”

The actual composition of radioactive nuclides in a certain system or area of the facility is called nuclide vector or finger print. The variation of this finger print may be enormous. It depends on the history of the facility (damaged fuel rods, fault conditions), changes though the different systems (determined by physical and chemical processes, different mobilibility through barriers) and may also be modified after decontamination (e.g. worse decontamination factor in pipes for alpha emitters, different migration of radionuclides in concrete).

These effects are considered for the exposure of the personnel through possible effluents in the air or water. If a calculation is performed on impact to the public, the diversion of different elements or compounds in the ecosystem has to be considered.

In any case the finger print changes with time. According to the main topic of this presentation, the discussion of different options for the decommissioning of a nuclear power plant, radioactive decay is the main parameter in the following discussion.

2.2 Characteristic Nuclides and Their Essential Properties

It is beyond the scope of this presentation to discuss all relevant radionuclides and their radiologic relevance in detail. Nevertheless Table 1 gives a summarizing overview. For this table 16 radionuclides have been selected which are of importance according to their contribution to the finger print *or* their half-life *or* their radiological relevance.

1	2	3	4	5	6	7	8	9	10	11	12
					External	External	Contamination	Incorporation	Ingestion	Inhalation	
Nuclide	**Half life** *[y]*	**Assignment**	**Exemption level** *[kBq]*	**Surface contamination** *[Bq/cm²]*	**point:30cm; skin dose,beta** *[µSv/h/MBq]*	**point:30cm deep,gamma** *[µSv/h/MBq]*	**50µl-droplet; skin dose** *[µSv/h/kBq]*	**Highest dose organ**	**20mSv ALI** *[kBq]*	**20mSv ALI** *[kBq]*	**Monitoring**
H-3	12.3	fission, act?	1000000	100	0	0	0	whole body	480000	1100000	- / LSC / -
C-14	5730	fission, act?	10000	100	0	0	2.7	whole body	34000	34000	- / **+** / -
Mn-54	0.85	activiation	1000	1	0	1.4	15	lungs	28000	13000	- / - / **++**
Fe-55	2.7	activiation	1000	100	0	0	?	spleen	61000	22000	- / LSC / -
Co-60	**5.3**	**activiation**	100	**1**	**13**	**3.9**	**220**	**lungs**	**5900**	**6900**	- / **++** / **++**
Ni-63	100	activiation	100000	100	0	0	0	lower large intestine	130000	38000	- / **+** / -
Sr-90/Y-90	29.1	fission	10	1	200	0	2000	lungs	710	130	- / **++** / -
I-131	0.022	fission	1000	10	86	0.7	570	thyroid	910	1800	- / **++** / **++**
Cs-134	2.1	activation?	10	1	70	2.7	520	soft tissues	1100	2100	- / **++** / **++**
Cs-137/Ba-137m	30.2	**fission**	10	1	210	1.1	710	soft tissues	1500	3000	- / **++** / **++**
U-235	7.00E+08	fuel	10	1	0	0.33	9	lungs	430	2.6	**++** / - / **++**
U-238	4.50E+09	fuel	10	1	0	~ 0	1.4	lungs	450	2.7	**++** / - / -
Pu-238	88	fuel	10	0.1	0	~ 0	0	bone surface	87	0.47	**++** / - / -
Pu-239	24000	fuel	10	0.1	0	~ 0	0.9	bone surface	80	0.43	**++** / - / -
Am-241	433	act. fuel	10	0.1	0	0.15	6.1	bone surface	100	0.51	**++** / - / **+**
Cm-244	18.1	act. fuel	10	0.1	0	~ 0	1.7	bone surface	170	0.8	**++** / - / -

Table 1: Characteristic Nuclides and Their Essential Properties

After the fundamental data in columns 1 to 3 of Table 1, the exemption levels and the limits for surface contamination according to the German radiation protection ordinance [1] are given. Columns 6 to 11 provide relevant information [2] for radiation protection purposes (factors for external irradiation or contamination and annual limits of intake (ALI) for incorporation). The last column indicates the general measurability of these nuclides by detecting $\alpha/\beta/\gamma$-radiation. The signs are self-explanatory; in cases, where only very weak beta radiation is emitted, liquid scintillation counting (LSC) as measuring device is recommended where possible.

The shaded displayed radionuclides H-3, C 14 and I-131 are normally not of importance during decommissioning, but may be of importance during the normal operation of a NPP. Their properties are indicated for comparison purposes.

For the major nuclide groups (activation products, fission products and nuclear fuel) key nuclides are defined. These nuclides should be “easy-to-measure” and should have a comparatively high fraction to the total radiological impact of this group. Normally the nuclides Co-60 (activiation), Cs-137 (fission) and Am-241 (alpha emitter) are used as key nuclides.

2.3 Model Finger Print, Representativity, Radioactive Decay

The following short list shows the variability of the finger print for six selected radionuclides. The left values (P) are the weighted average mean from the analysis of the water of the storage pond for fuel elements of eight NPP´s in Germany in one year (without H-3). The right values (W) are the finger print of nuclear waste of one NPP in Germany, which will be used as the model finger print in the following.

	P	W
Fe-55	**0.7%**	**19.0%**
Co-60	**37.4%**	**55.5%**
Ni-63	**24.4%**	**2.1%**
Sb-125	**23.1%**	**0.8%**
Cs-134	**2.2%**	**2.2%**
Cs-137	**12.2%**	**6.5%**

The very small fraction of alpha emitters is not indicated here. It will be shown later, that the amount of alpha activity has a major influence upon the monitoring and assessment of radioactivity in a NPP under decommissioning. For the moment, the effect of radioactive decay will be discussed.

Table 2 shows more detailed the nuclide mixture (W) for contaminated waste from the operation of a light-water reactor plant (only the main radionuclides). A comparable nuclide mixture will also emerge when the operation of a nuclear power plant is terminated and it is subsequently changed by the process of radioactive decay. Beneath the relative amount of each nuclide, the ratios of the three key nuclides have been calculated. Especially the high value for the ratio of Co-60 to Am-241 is a indicator for a facility with no special problems with alpha emitters.

		Decay time (years)			
Nuclide	**Start**	**3**	**10**	**40**	**100**
Fe-55	**22.01%**	15.97%	6.41%	0.02%	0.00%
Co-60	**64.29%**	66.97%	62.38%	6.34%	0.01%
Ni-63	**2.39%**	3.61%	8.05%	34.33%	60.35%
Sr-90/Y-90	**0.27%**	0.39%	0.77%	1.97%	1.24%
Sb-125	**0.95%**	0.69%	0.28%	0.00%	0.00%
Cs-134	**2.50%**	1.41%	0.31%	0.00%	0.00%
Cs-137/Ba-137m	**7.58%**	10.93%	21.75%	57.25%	38.31%
Pu-α / Cm	**0.0009%**	0.0014%	0.0032%	0.0169%	0.0448%
Pu-241	**0.0212%**	0.0283%	0.0473%	0.0582%	0.0085%
Am-241	**0.0002%**	0.0005%	0.0018%	0.0150%	0.0408%
Total	**100%**	**100%**	**100%**	**100%**	**100%**
Co/Cs	***8.49***	***6.13***	***2.87***	***0.11***	***0.00***
Co/Am	***283163***	***135120***	***35167***	***423***	***0***
Cs/Am	***33367***	***22047***	***12262***	***3817***	***939***

Table 2: Finger print of the nuclear waste of a German NPP in operation and its change by radioactive decay ***("model finger print")****.*

This finger print for a defined start time was corrected by radioactive decay. The intervals 3, 10, 40 and 100 years were chosen according to the fact that typically

- after three years, first decontamination work is likely to be done in the primary coolant system (end of post-operational phase),
- after ten years, decommissioning of a NPP can be in a late phase in the case of "immediate" dismantling (end of dismantling process or end of phase of safe containment),
- the time scale in the german discussion about intermediate storage facilities is 40 years and (example assumption of maximum interim storage time for light and medium radioactive waste)
- for comparison purposes concerning a long safe enclosure, values for 100 years are given (example assumption for the maximum decay storage time).

It can be seen clearly that over the decades the finger print completely changes. The influence on effective doses on personnel will be discussed in section 3.4. This finger print change and the way it can be measured with state-of-the-art instrumentation will be assessed in chapter 4. In particular, the table shows that the proportion of Co-60, which is important for measuring the level of activity when monitoring the plant, declines significantly after 10 years, and the proportion of alpha emissions increases significantly.

2.4 Radiological Relevance of the Model Finger Print as a Function of Time

Table 2 clearly indicates the activity-based dominance of Co-60 in the first ten years. After this period, the situation changes significantly. The next diagrams show in not normalised y-axes the change of the total activity, the contribution to the deep dose [H*(10)], the skin dose in the case of local contamination [H´(0.07)] and the effective dose by inhalation in the case of air borne activity.

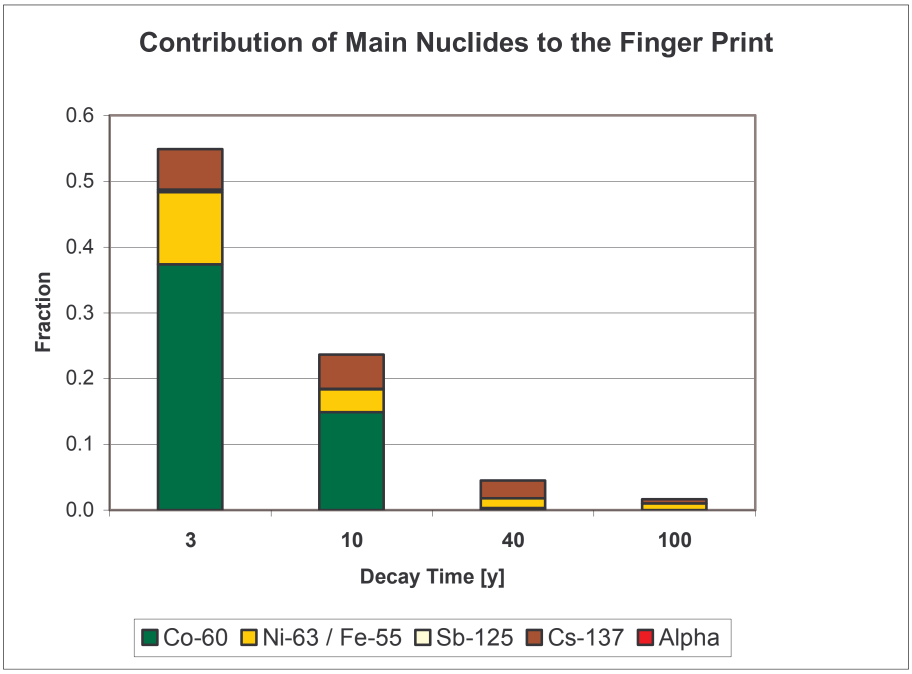


Figure 1: Reduction of the activity displayed by the decay of important radionuclides for the model finger print

The effect of the decay time on major radiological variables is shown in Figs. 1 to 4. Figure 1 shows the proportion of Co-60 in the total nuclide mixture. This declines considerably while the proportion of other nuclides increases. This means that the usual measurement technology used in nuclear power plants, which is based essentially on measuring Co-60 as a calibration nuclide and on calculating limit values as a result of the nuclide mixture, is no longer sufficient after a certain length of decay time. Other measuring techniques must be used.

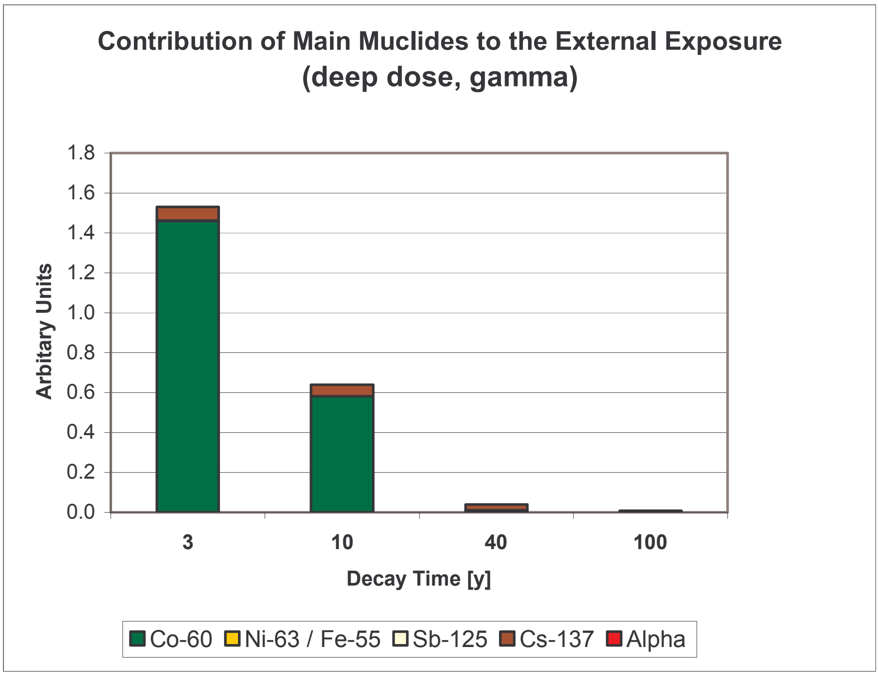


Figure 2: Reduction of the dose rate by the decay of important radionuclides for the model finger print

Figure 2 shows the main benefit of safe enclosure from the radiological point of view, if there is no contact with radioactive material. According to the half-life of Co-60, the dose rate in the facility decreases sharpely in units of decades. On the other hand, if Co-60 vanishes, the easy-to-detect indicator for internal contamination also gets lost.

In the case of open contamination the situation is gradually different (fig. 3). Additionally to the six-times larger half-life, the average beta energy of Cs-137 is higher than that of Co-60. As a result, Cs-137 contributes to the skin dose in any time and dominates this pathway for exposure after large decay times.

Without alpha emitters, the situation would be similar for any kind of incorporation. Unfortunately, certain alpha emitters have very high dose factors for inhalation (see small values for the ALI in Table 1) with the effect that even low fractions of activity alpha emitters have a major impact in terms of effective dose in the case of inhalation. The situation becomes even worse if the ratio of beta activity to alpha activity varies inside the plant, which is not an unusual situation. Like in older facilities, which often exhibit more malfunctions during operation, alpha emitters may dominate radiation protection during decommissioning.

At the same time, the contribution made by Co-60 to external radiation (Fig. 2) declines while the contribution from alpha emissions to exposure to radiation through inhalation increases considerably (Fig. 3). This means, for example, that monitoring room air for gamma emissions (Co-60 measurement) can no longer guarantee the protection of operating personnel. This also applies to the monitoring of emissions in exhaust air. Figure 3 clearly shows that even in this example with a very low proportion of alpha emissions in the original nuclide mixture, alpha emissions can become after a certain period of time the major determinants for monitoring activity required in the plant.

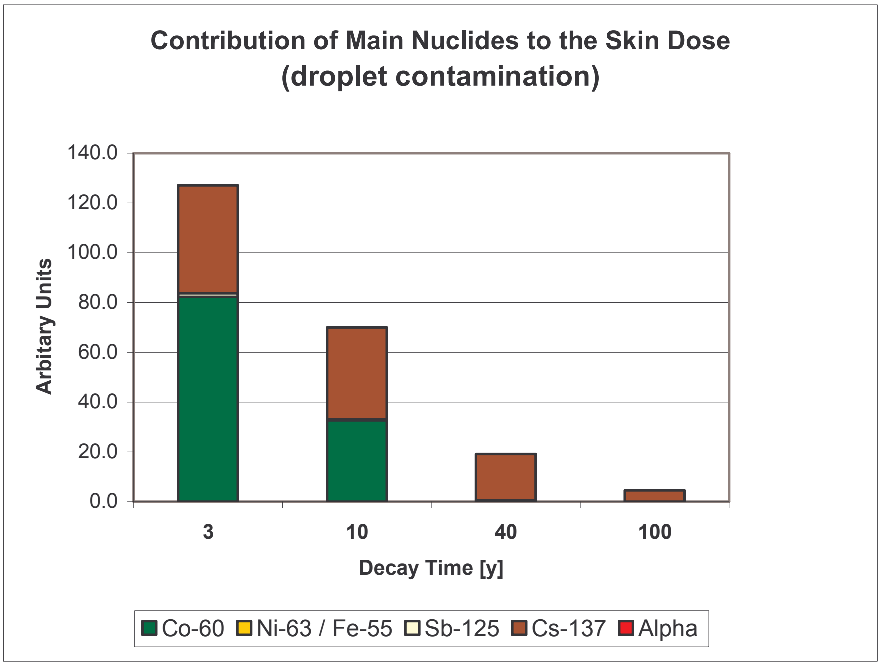


Figure 3: Reduction of the skin dose caused by contamination through by decay of important radionuclides for the model finger print

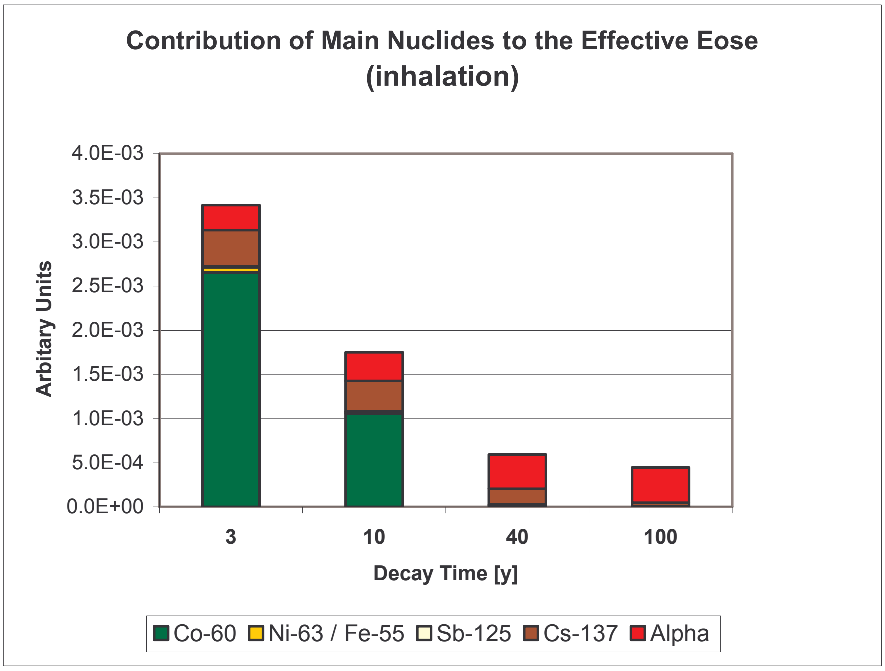


Figure 4: Reduction of the effective dose by the decay of important radionuclides for the model finger print

Figure 4 shows the effect of decay time on the disposal of radioactive waste and residual material from the plant for the nuclide mixture used in the example:
It shows the total activity over the decay time (0 to 40 years), the ratio of specific activity to specific activity for the unrestricted clearance of building rubble and for the restricted clearance of residual metallic material for re-cycling (in accordance with [1]). These values are standarized in each case to ensure that factor 1 is reached after 40 years. It can be seen that in the case shown, total activity declines by a factor of approximately 22, while the activity figures determining unrestricted clearance do so by a factor of 74 and those determining the recycling of residual metallic materials do so by a factor of only approximately 12.

The following radiation protection requirements must be met during the dismantling phase of a nuclear power plant:

- protection from contamination being spread within the plant and out of it
- protection of operating personnel from external radiation and incorporation
- monitoring and calculation of emissions of radioactive materials in exhaust air and waste water
- characterisation and minimisation of radioactive waste
- clearance of residual materials for recycling as well as of buildings for conventional re-use or destruction
- clearance of land

The following radiation measurement techniques are predominantly used in meeting these requirements:
- dose rate and personal dose measurement devices
- monitoring devices for air-borne activity
- personal contamination measurements and incorporation measurements
- contamination measurements on objects and surfaces.

The suitability of the proven devices used in the operating phase of nuclear power plants for their dismantling with or without a phase of safe containment is to be discussed in the following section:

3 MONITORING AND ASSESSMENT OF RADIOACTIVITY

In Germany, the regulatory framework for the surveillance of radioactivity inside a NPP in operation and monitoring its effluents is defined by the standards of the KTA [3]. The effort for surveillance and monitoring is likely to be reduced during decommissioning. The German “Leitfaden” [4] provides general information on the necessity of maintaining instrumentation for the operation of the facility during the phase of decommissioning. This chapter contains some examples for basic techniques and their suitability for measuring the radioactivity decribed by the model vector defined above and its modification according to radioactive decay.

3.1 Measuring Devices

The most impostant tasks for the health and safety of staff inside the plant is the
- measurement of the dose rate,
- monitoring of air-borne radioactivity (filters, semi-conductors): inhalation,
- contamination (portable monitors, smears): skin (personal precautions), incorporation,
- monitoring leaving the controlled area: (skin, lung).

Direct measurement of dose rate and contamination will not be discussed here in more detail. The following two sub-chapters deal with the indirect measurement of air-borne radioactivity and contamination.

3.1.1 Indirect measurement of Air-borne Radioactivity and Contamination

Basing on a mobile measuring device originally designed for the analysis of filter samples the suitability of these devices will be investigated:

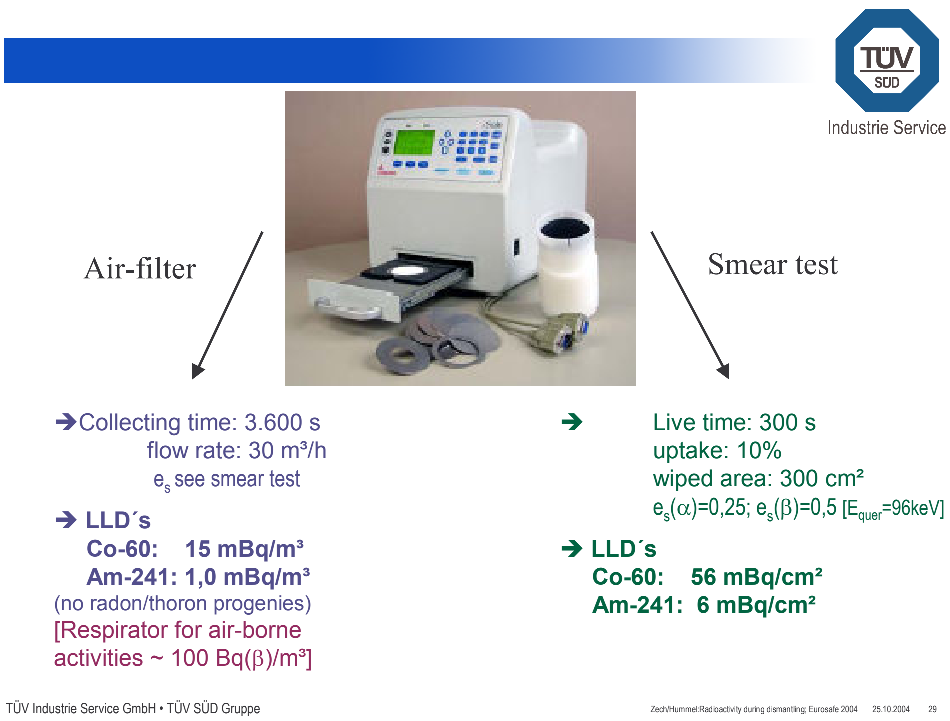
Properties:
Silicon detector (gasless)
Scintillation guard detector
Radon/Thoron interference calculation

Specifications:
Background (α/β): 0.05 / 9.0 cpm
Efficiency (Co-60/Am-241): 10.1% / 33.8%
typ. values for 20cm² 0.3mm PIPS

These specifications lead to the following “theoretical” detection limits. Especially for alpha emitters realisitic values are clearly higher (presence of radon/thoron progenies and its interference correction).

Detection limits (LLD)
(according to DIN 25482):
Time: 3,600 s Measurement
50,000 s background
5% stat. uncertainty

Co-60: 226mBq
Am-241: 7.3mBq



The results for the LLD´s of key nuclides for the two measuring tasks are given in the slide above. In the next step, these will be compared with the timely variation of the model finger print and the limit of the surface contamination according to the German RPO.

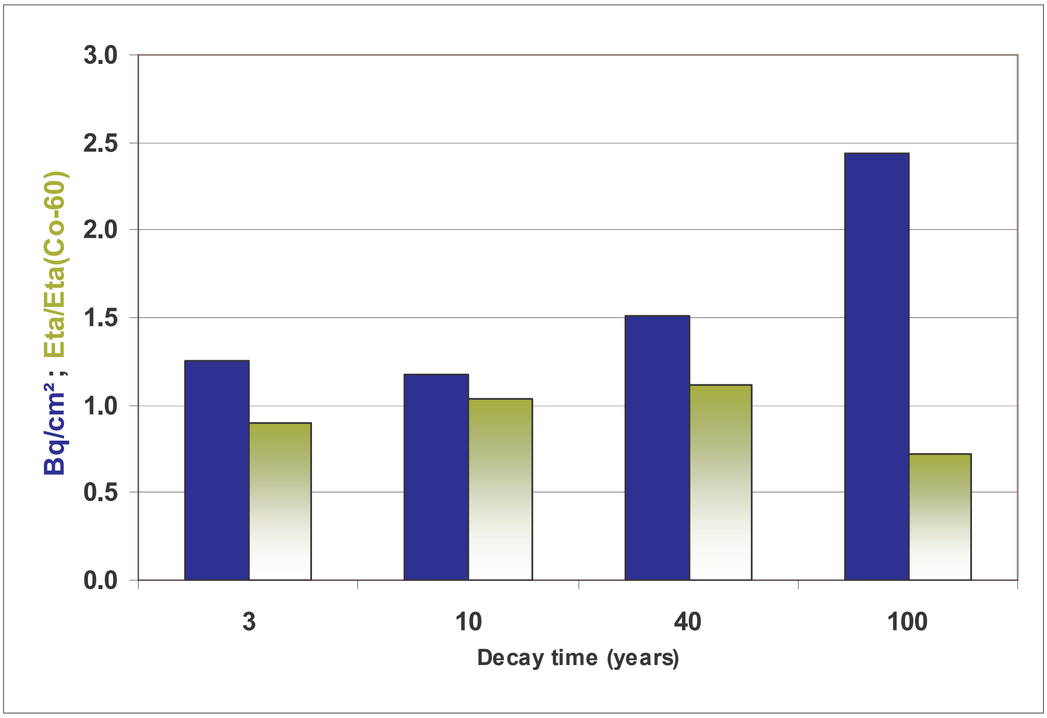


Figure 5: Limit of the surface contamination for the model vector acc. to the German RPO (left bars)and the ratio between the efficiencies for the model vector and the calibration nuclide as a function of time after shutdown (right bars)

The diagram shows the increase of the limit of the surface contamination mainly due to the remaining Ni-63 in the finger print, which has low radiological relevance. For the same reason the efficiency of the measuring device decreases compared with the efficiency for the nuclide used for calibration (Co-60).

An important note regarding the trend of the right bars in the diagram above is that in [5] we showed that the use of Co-60 as the calibration nuclide for beta emitters and the use of detectors with somewhat thicker dead layers (or foils) lead to the best results in terms of the stability against variations of the finger print.

Alpha emitters, which contribute to the finger print as in the model vector, cannot be detected with these routine measurements (neither as air samples, nor as smear test) before the limits for the beta emitters are reached.

3.1.2 Monitoring Leaving the Controlled Area

In the last years more and more so-called “two-step”-monitors were installed to measure contamination of workers before they leave the resticted area. The name “two-step” results from the fact that the front and the back of the body of a person is measured seperately. As a result, these monitors show an improved sensitivity profile for contamination of the body. In addition, monitors were sold with large plastic scintillation counters behind the proportional counters in the thorax region. Obviously, the scintillation counters show a much better sensitivity for inhalated gamma emitters in the lung than proportional counters for contamination do.

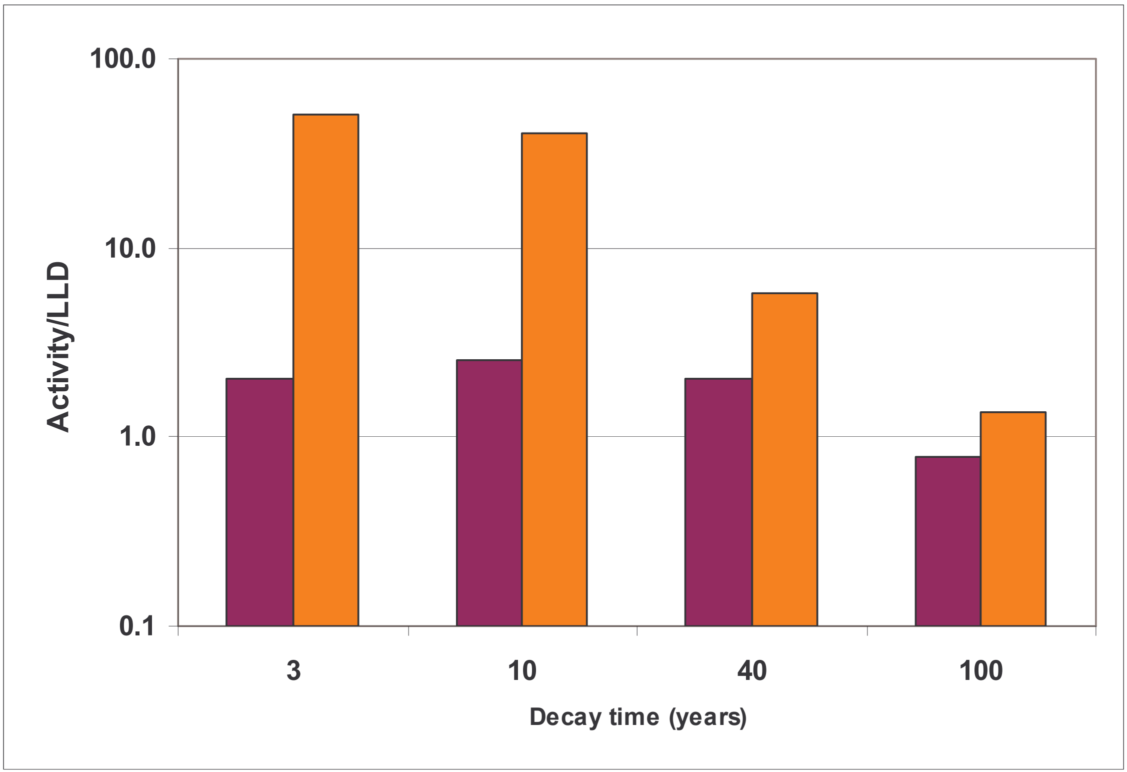


Figure 6: Ratios between activity that has to be detected and the LLD of the monitor for Contamination of the body (left bars) and Inhalation (right bars; on the condition, that scintillation detectors are mounted behind proportional counters in the thorax region) calcuated for the model finger print basing on the nuclides Co-60 and Cs 137

In this diagram, values larger than 1 indicate that the measuring device is appropriate. Both tasks of surveillance can be performed without restrictions. For contamination, the properties of the body detectors were chosen. Experience shows that the detectors for hands and feet normally lead to better results and the efficiency for the head is worse. To calculate the ratio for inhalation a daily dose limit of 0.5 mSv_{eff} was assumed.

3.2 Radiological Relevance and Measurability

It can be summarized that in a NPP under decommissioning with a history without larger damages of the fuel rods, it is sufficient in most cases to monitor beta/gamma activity. In most cases the measurability of a certain radionuclide correlates well to its radiological relevance. This is not surprising considering the fact that the higher the energy of nuclear disintegration, the higher the biological effect *and* the signal from the emitted beta/gamma radiation are.

The situation changes with the presence of alpha activity. In Chapter 3, it was shown that the minute amount of alpha activity leading to significant incorporation doses if barriers are broken during dismantling and air-borne alpha activity may become relevant in the workplace. When systems are opened, the use of respirators may become necessary, if the easily measured beta/gamma emitters have been decayed after a long period of safe enclosure of the plant.

Therefore in actual decommissioning projects – with higher amounts of alpha activity – it cannot be recommended in most case to introduce a long period of safe enclosure. In future decommissioning projects, alpha activity may have a reduced impact on the decision regarding when a facility should be disassembled and what personal protective equipment is needed for the staff. The option of safe enclosure has been discussed in the past without due consideration of aspects such as minimisation of the risk of the total facility by decontamination of the systems and conditioning waste.

4 OPTIMISED PROCEEDING

Taking into consideration all the boundary conditions discussed, the following optimised procedure emerges from the perspective of radiation protection and the disposal of residual materials for the future dismantling of a nuclear power station:

The final termination of power production is followed by a post-operational phase lasting approximately 3 years in which spent fuel elements are removed, waste created by operation of the plant treated, and the circuits and systems decontaminated. Besides the reduction of the radioactive inventory in the plant itself, the radioactive materials are immobilised and the potential for emissions in the event of accidents efficiently reduced. Operation of the plant during this phase differs little from that during the inspection of an active nuclear power plant. Extensive experience has been gathered, therefore, from the preceding years of operation for planning and carrying out the activities, and appropriately trained personnel are available.

At the end of the post-operational phase, there follows an intensive analysis of the contamination and activation of individual parts of the plant by means of measurements and reports which serve as a basis for planning the dismantling of the plant, in particular, for handling and disposing of the resulting materials. It is of major importance to quantify all radionuclides occurring in the plant through chemical and physical analyses, whereby a distinction is made between activated and contaminated materials. Special attention must be paid to the proportion of alpha emissions in contaminated materials. The experience of the operating team and documentation from operating the plant represent an important basis for carrying out these measurements.

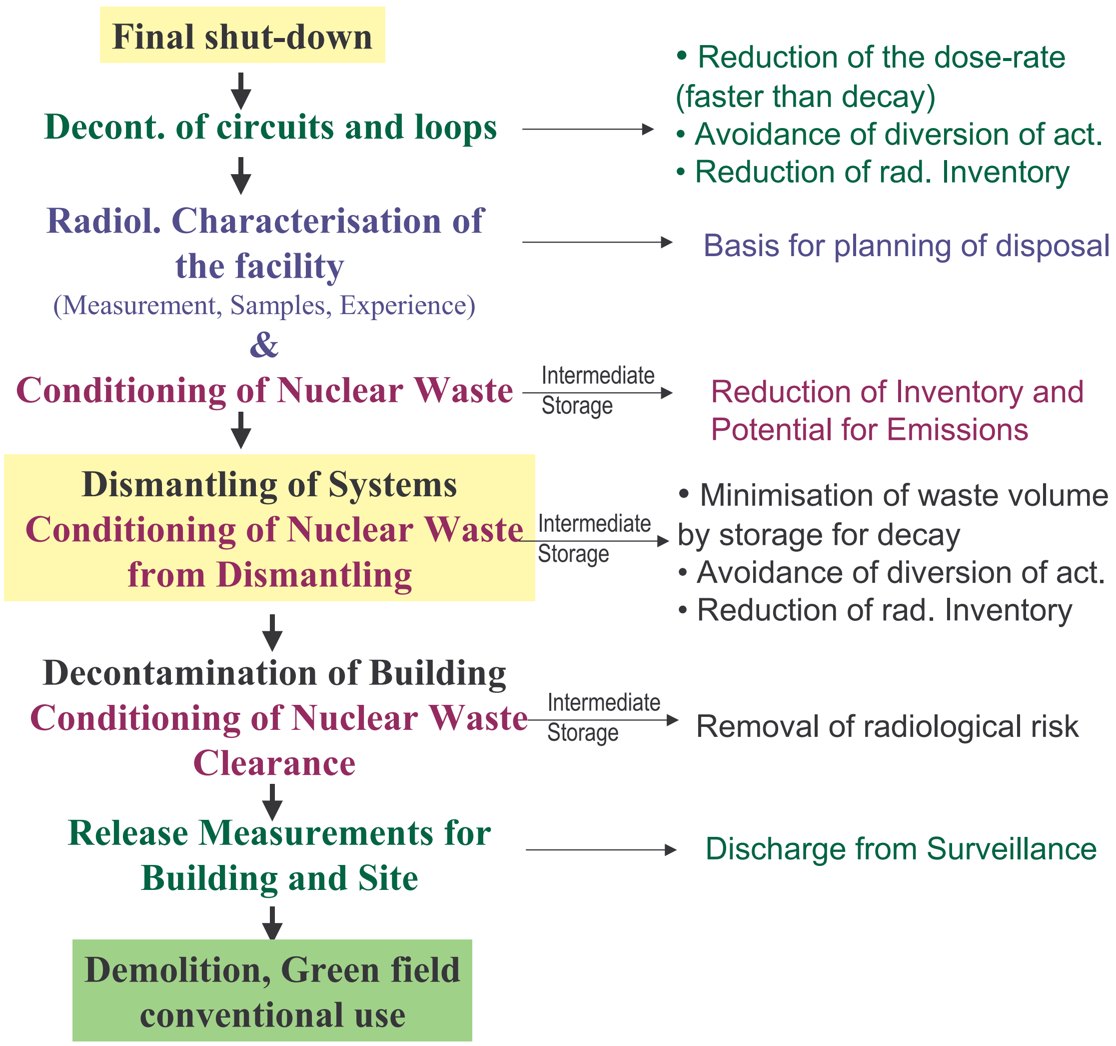
In order to be able to use the necessary experience in the whole dismantling process, it is appropriate to start dismantling the plant immediately after the post-operational phase and after approval has been issued with no phase of safe containment.

The actual shutdown begins with dismantling the decontaminated primary systems. By far the largest proportion of primary nuclear waste is created in removing the reactor pressure vessel and its core assemblies as well as the primary circuit and the nuclear steam generation system. With the removal of these systems and the offloading of the nuclear fuel, the radioactive inventory of the plant is again reduced by several magnitudes. In the course of these dismantling measures, the activated structures (with the exception, for example, of biological shield) are almost entirely removed. The risk of spreading or carrying over radioactivity is also reduced by the considerable reduction in overall activity from contamination.

With the subsequent decontamination of the building structures and the removal of any remaining auxiliary and peripheral systems, the radiological risk, which originally emanated from the plant, is reduced to more or less zero. It is characteristic of this phase that the activity flows from the plant through the removal of radioactive waste decline successively while the mass flows increase. One task which is very different to those occurring in the operational and post-operational phases of the plant, is the clearance of large quantities of material with, in some cases, very low levels of activity. Before the building structures are torn down, it is typically necessary to assess some 10,000 Mg of ferrite metals and 1,000 Mg of non-ferrite metals.

Finally, depending on the type of plant, considerably more than 100,000 Mg of concrete has to be measured as a standing structure, and if evidence can be provided that the clearance criteria have been met, approval can be issued for demolishing the structure. At the same time as the decontaminated building structures are radiologically assessed, the site of the plant must also be subjected to a suitable clearance procedure.

With the declaration that neither the demolishment of the building structures nor the further use of the earlier location pose radiological risks to individuals in the population, the owner of the approval is released from his obligations under nuclear power legislation and the site of the plant can be made available for future use.



5 CONCLUSIONS

Experience with in some cases remotely controlled dismantling techniques and various decontamination procedures in dismantling projects which have been largely or completely finished, has shown that nuclear power plants can be dismantled without any long periods of safe containment as long as high safety standards are maintained and attention paid to the radiological protection of the dismantling personnel.

Besides practical aspects such as:

- Documentation of Surveillance and History of Operation are available under unreduced circumstances
- Experienced personnel is available
- The risk factor can be decreased faster by dismantling and conditioning
- The leading nuclides for monitoring can be used as during the operational phase
- The existing instrumentation can be used to a wide extent, no new tasks in principle
- Reduction of nuclear waste can be achieved by decay during intermediate storage as by safe enclosure,

direct dismantling of a nuclear power plant also has positive effects

- economically (use of the location and its infrastructure),
- socially (part of the workforce employed for a longer period of time) and
- socio-politically (demonstration of the practicability of safely removing nuclear systems).

In Germany, two prototype reactors have been successfully dismantled to "green field" status with the participation of TÜV Süd Group as the expert organization, namely, the nuclear power plants in Niederaichbach (KKN, 1994) and Großwelzheim (HDR, 1998). In many cases, there are arguments relating to radiation protection and the corresponding measurement monitoring which speak for direct dismantling, in particular for older plants which are currently in the shutdown phase.

References:

[1] Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (StrlSchV) vom 20.07.2001, zuletzt geändert am 18.06.2002

[2] Radionuclide and Radiation Protection Handbook 1998
Rad.Prot.Dos., ISBN 1 870965 51 5

[3] Sicherheitstechnische Regeln des Kerntechnischen Ausschuß

[4] Leitfaden zur Stilllegung von Anlagen nach § 7 Atomgesetz vom 14.06.1996

[5] Martin Filß, Lothar Hummel, Johann B. Zech
Kontaminationsgrenzwerte beim Transport radioaktiver Stoffe:
Das radiologische Konzept und seine messtechnische Umsetzung bei Kernkraftwerken
Symposium: „40 Jahre Strahlenschutz beim TÜV Süddeutschland vom 28. bis 30. Juni 2000 in München"