STATUS OF WASTE MANAGEMENT PRACTICES IN THE

FEDERAL REPUBLIC OF GERMANY

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ABSTRACT

Nuclear power stations in the FRG are owned and operated by the private sector. Consequently, the electricity utilities are obliged to take care of the back-end of the nuclear fuel cycle including waste conditioning. The ultimate storage of wastes, however, stays within the responsibility of the federal government. The German Atomic Energy Act appoints the Federal Agency, Physikalisch-Technische Bundesanstalt (PTB) in Braunschweig to undertake the waste disposal on behalf of the federal government. The power plant operators are liable to utilize the radioactive residues, e.g. the spent fuel elements, in such a way that excludes any adverse effects to mankind and the environment. Considerable progress has been made over the past decade within the FRG to develop a national policy and strategy for the management of nuclear wastes. Reference is made to the development work already undertaken and the experience gained to ensure that appropriate treatment, conditioning techniques and facilities are available. Further R&D will have to be focused on subjects of improved and advanced procedures with the prime goal of minimizing waste volumes and reducing workers radiation exposures as well as the development of control measures which ensure observance of the specifications of the waste to be disposed of.

HISTORICAL OVERVIEW

The concept of nuclear waste management in the FRG is based upon the principle that the disposal of radwaste arising from nuclear power plants is an indispensable prerequisite for the peaceful utilization of nuclear energy. The german government and the federal authorities have laid down special regulations mentioning that nuclear power stations will be licensed only if sufficient evidence of secure and ultimate waste disposal practices are brought out.

According to the German Atomic Energy Act[1], the nuclear fuel cycle must be an integrated closed cycle and include reprocessing of the spent fuel elements and recycling the recovered fissile isotopes. Only in exceptional cases, e. g. HTGR's or other prototype reactors, where reprocessing and recycling have not proven to be technically and economically feasible, the spent fuel elements are to be disposed of as radioactive wastes in an orderly manner.

All sectors of the back-end of the nuclear fuel cycle have to be taken care of by the private sector. However, the federal government is responsible for the disposal of radioactive waste. All costs of the back-end of the nuclear fuel cycle, including ultimate disposal, have to be added to the cost of electricity generated in nuclear power stations according to the "polluterpays-principle".

As an immediate response to the above mentioned governmental instruction, the specially founded company, Deutsche Gesellschaft für Wiederaufarbeitung von Kernbrennstoffen (DWK) - a subsidiary of 12 German electricity utilities operating or planning to operate nuclear power plants - decided in early 1977 to build an integrated nuclear fuel cycle center at Gorleben in Lower Saxony with an annual reprocessing capacity of 1500 tons of irradiated uranium. This site was chosen because of its potential to house an underground repository in a salt dome underneath. However, after the well-known "Gorleben-Hearing" in 1979, the project was not pursued further on political grounds.

Soon afterwards, i.e. in the beginning of 1980, a revised concept for closing the back-end of the fuel cycle was proclaimed by the state authorities. The concept of an integrated closed fuel cycle including recycling of fissile materials has been maintained, but does no longer require to have all the individual sectors located at the same site. Further, the reprocessing capacity was reduced to around 350 tons per year since it was recognized that such a plant should be built primarily to develop and demonstrate advanced know-how and not as a means for supplying necessary resources for nuclear energy [2].

Waste Management Options

The two principal philosophies for radioactive waste disposal are "dilute and disperse" and "concentrate and contain". The latter is generally admissible in the FRG. The radioactive release limits for nuclear facilities are regulated in the German Radiation Protection Ordinance[3] according to the so-called 30 mrem principle (0.3 mSv/a for whole body exposure) and set up individually for each installation by the competent licensing authorities.

For final waste disposal, three options are in general available all over the world, namely sea dumping, shallow land burial, and deep underground burial. In the FRG, deep underground burial of properly conditioned waste in a geologically suitable host rock has been considered to be most attractive ever since the early days of nuclear energy. Deep underground disposal means placing a waste repository in a stable geological formation of salt, granite, basalt, clay, etc. at a depth of 300 - 1000 meters, so that any risk from re-exposure due to an accidential human intervention or natural disturbance is reduced to a negligible level. The only conceivable route by which radionuclides in the waste would return to man is movement in groundwater which eventually reaches the surface to enter the food chain.

The current German concept for the management of spent fuel is based on three different options as shown in Fig. 1, which are highlighted as follows:

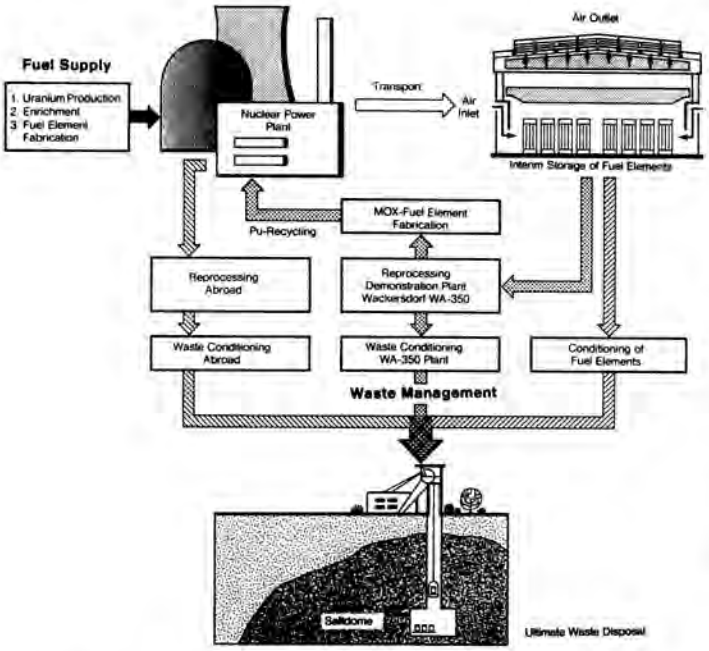

Fig. 1. Alternative routes for closing the back-end of nuclear fuel cycle in the FRG

1. Reprocessing abroad in the plants of COGEMA in France and BNFL in Great Britain
   This route does not require away-from-reactor interim storage of spent fuel elements. However, all wastes arising from reprocessing have to be taken back, incorporated in suitable solid matrices, and confined in appropriate waste containers, ready for final disposal in a national repository. This option should only be applied for a transition period of time.

2. Reprocessing, refabrication of mixed oxide fuels, and waste treatment to be performed in an integrated industrial complex at the same site
   Accordingly, on February 4, 1985, the German nuclear power industry has decided to build a reprocessing facility at Wackersdorf in Bavaria with an annual throughput between 350 and 500 tons of UO_2. In september 1985 the decisive construction license has been granted. However, it took another 4 month to solve the legal problems and start the construction work at Wackersdorf in December 1985.

   First, the dry interim storage facility for spent fuel elements shall be constructed and commissioned, the schedule for its completion is 1990. The reprocessing plant will follow one to two years later. The hot start-up of the reprocessing plant is scheduled for 1994.

3. *Direct* disposal of properly packed spent fuel elements in a stable container without any reprocessing in a repository constructed in a salt mine
   A recently completed assessment on the question whether or not ultimate disposal of spent fuel elements offers decisive safety advantages in comparison to a waste disposal with previous reprocessing has revealed that no meaningful differences exist [4]. The radiological impact of direct disposal is generally less than for reprocessing and the waste management belonging to it. The difference in the total collective dose is rather small however, and is negligible compared to the natural radiation exposure.

Cost estimates using current valid parameters indicate 30 to 40 % lower specific costs for the direct disposal route [5]. However, this cost advantage is marginal compared with total electricity generation costs. An estimate of the break-even point is forecasted for the beginning of the next century.

According to government recommendation, the necessary technology should now be developed and demonstrated. The DWK company has just completed a conceptual design for an appropriate facility. A licensing application is expected to be filed soon. Most likely, Gorleben will be taken into consideration as a suitable site.

A yet unresolved problem in realizing an ultimate direct spent fuel element disposal concept may arise from safeguarding requirements.

Apart from the above 3 options, long-term interim storage in cast-iron containers applying air-cooling may represent a prudent interim solution until a reliable appraisal of future development of nuclear power in general, and fast breeder utilization in particular, is possible.

This however, does not imply that an immediate construction and startup of the Wackersdorf reprocessing plant could become futile, since in the event of an unevitable requirement of recycling technology at the beginning of next century, adequate experience and expertise would be available.

Status of Interim Spent Fuel Storage

Table I illustrates that there will be sufficient interim spent fuel storage capacity available for German nuclear power stations at least until the year 2005. The forecast is based on an optimistic installed

Table I. Spent LWR fuel element storage facilities and possibilities in the FRG

Plant	Storage Capacity (tons of U)	Status of Project
Interim storage facility Gorleben	1500	facility completed, waiting for operating license
Interim storage facility Ahaus	1500	facility under construction, completion expected 1988
Interim storage facility Stade	240	application for construction license filed November 1982
Interim storage facility Wackersdorf (entry storage facility to reprocessing plant)	1500	construction permission license September 1985; startup of construction summer 1986; completion expected 1988
Interim storage and reprocessing abroad		actual existing contracts
Cogema/France	3000	
BNFL/England	760	
Interim storage in nuclear power plants (densified water pool storage)	6000	related to the situation in the year 2000
Total interim storage capacity (year 2000)	14,500	
Cumulative spent fuel element arising (year 2000)	12,000	

electricity capacity of 35,000 MWe at the end of this century, though a more realistic figure would be 30,000 MWe.

SUMMARY OF CURRENT WASTE CONDITIONING PRACTICE

A range of choices exist in most nuclear installations for a partial or total removal of radionuclides from liquid and gaseous effluent streams. For wastes that arise in liquid form, appropriate treatment is generally required to convert the waste to a stable solid form suitable for storage and disposal. The type of treatment will depend on the volume of the waste, its chemical and physical characteristics, the content of radionuclides and the role the waste matrix is expected to play during storage and ultimate disposal.

The Various Waste Streams

Around 99 % of the total radioactivity generated in a nuclear cycle will appear in the various waste streams of the fuel reprocessing plants as shown in Fig. 2.

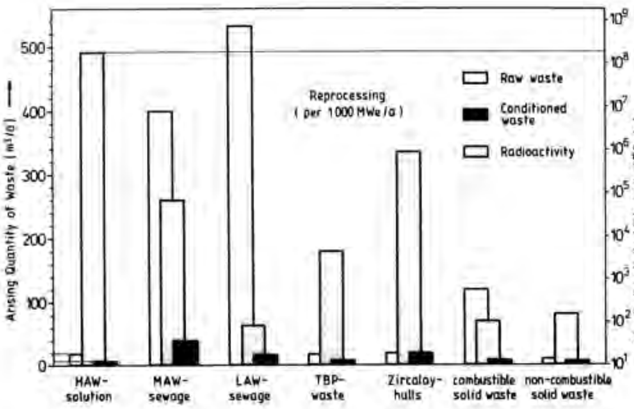

Fig. 2. Overview of waste arisings from a reprocessing plant

The remaining 1 % is distributed in the wastes arising in nuclear power plants, the details of which are depicted in Fig. 3. By volume of the raw waste, power plants generate quite a large fraction.

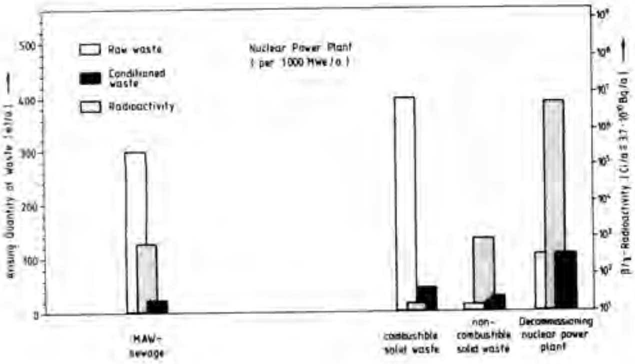

Fig. 3. Overview of the waste arisings in a nuclear power plant

It is obvious that by far the most important waste stream is the liquid concentrate of the aqueous raffinates from the reprocessing extraction cycles. These contain up to 1 % of the uranium and the plutonium, practically all of the fission products and the other transuranium elements, e.g. Np, Am and Cm.

Conditioning Procedures

The raw wastes have to be transformed into an appropriate physical and chemical form to obtain waste packages which satisfy the requirements of transport, handling, interim storage and ultimate disposal. Waste packages should have high mechanical and chemical stability and the minimum possible volumes[6]

An overview of the various product forms assessed in respect to their suitability in the FRG is shown in Table II.

Table II. Overview of suitable waste products for final disposal

Category	Waste Loading [max]	Product	Remarks		
Low Temperature Process	50	**Bitumen**	production simplicity	lack of thermal stability	radiolysis of hydrocarbons and possibly nitrates
	20	**Cement or Concrete Composites**	production simplicity	lack of thermal stability	radiolysis of water and nitrates
	80	**Calcines**	production simplicity	lack of thermal stability	disperable form
	30	**Polymeric Materials**	production simplicity	lack of thermal stability	radiolytic instability, swelling
Glass	35	**Phosphate**	low viscosity melt	glass melt highly corrosive	devitrification tendency
	25	**Borosilicate**	medium viscosity melt	advanced technology	good radionuclide host, resistancy against radiation and transmutation effects, hydrothermal attack at elevated temperature and pressure
	20	**Nepheline-Syenite**	high viscosity melt	high melting temperature	
Ceramic	25	**Glass ceramic**	largely crystalline	complicated technology	no appearant advantage compared to glass
	50	**Supercalcine**	predominantly crystalline	tolerable technology	thermodynamically stable product particulate form dispersable
	10 – 20	**Synroc**	purely crystalline	sophisticated technology	thermodynamically stable product waste partioning required?

Borosilicate glass for HLLW, bitumen and concrete composites for MAW and LAW, respectively, have been selected as suitable waste forms. In addition, high integrity nodular cast iron containers are proposed as waste canisters which, due to their excellent mechanical stability, do not require a waste fixation. The container barrier provides the equivalent of a stable solid matrix.

Polymeric materials were used for solidification of limited amounts of radioactive wastes only. A typical example is the fixation of spent bead resins in polystyrene. The underlying basic idea is that the incorporation of the resins into the same material as the resin itself shows decisive advantages, which however, turned out to be not that important.

Composite waste forms like embedding calcines or small glass particles into a metal matrix or improving the product barrier by coating the primary products with a layer of pyrocarbon, silicon carbide, etc., have never been considered seriously since the necessary technology is difficult and costly. The achievable benefits, however, are not worth the extra effort.

It is mandatory in Germany to convert the high-level and transuranic waste into a stable solid form, such as glass. It is a common understanding that the waste form should be made as chemical resistant as reasonably achievable in order to provide a maximum possible protection within an acceptable cost range.

Crystalline waste forms, commonly known as SYNROC, certainly exhibit better hydrolytic behavior than glass as well as lower leach rates if properly prepared, but they are difficult to produce.

Their availability on a technical scale is not yet proven. Since from an overall safety point of view they are not required, the FRG concept relies on HAW borosilicate glass which is completely satisfactory. The PAMELA pilot plant at the former EUROCHEMIC site at Mol/Belgium has proven its reliability as a suitable vitrification process during recent hot operation campaigns.

Up till now, tritium and krypton-85 are released to the atmosphere in all existing fuel cycle plants. Enforced environmental protection standards may require a partial retention in the FRG. Promising techniques are under development [7].

Deep well injection of tritium-containing waste waters into isolated porous geologic formations is the reference procedure. Binding the water using hydraulic cement and encapsulating the cured cement blocks in gas-tight containers is the back-up solution.

There are three efficient krypton removal processes from the dissolver off-gas under development. They are based on

- cryogenic distillation,
- absorption in halogenated solvents like freon, and
- adsorption on charcoal or molecular sieves.

The best way for a final disposal of the separated krypton is not yet decided.

Waste Canisters

Container integrity under normal and accidental conditions must be guaranteed during the interim storage and in addition to the operational period of a repository. That means a time span between 40 and about 80 years. Still open is the question whether for a very few special waste categories a longer lifetime for their integrity may be a necessary requirement. The time period in question could be in the order of 200 years until the nearfield convergence of the salt rock has encapsulated the waste canisters in such a way that solutions cannot attain access. The required answer will become evident from a site-specific safety analysis.

The instantaneous status in the FRG is founded on a standardized waste container system including:

- 400 liter steel drums with or without a sacrificial concrete shielding
- cylindrical canisters of concrete or cast iron (net volume 50 to 500 l)
- box type containers with different gross volumes (4 - 11 m^3) made of sheet iron
- high level activity glass block canisters made of stainless steel with a net volume of 60 liters (≅ 250 kg glass).

Quality Control of Waste Products and Canisters

Radioactive wastes properly conditioned and packed have to meet acceptance criteria specific to a particular repository. The operator of a nuclear installation will make all efforts to satisfy these requirements since he is hold responsible for its compliance. Therefore, the plant operator in his capacity as the waste conditioner has to install an adequate control system.

The PTB carries the responsibility for the disposal of radioactive waste and supervises the operating company of the repository called DBE (= Deutsche Gesellschaft zum Bau und Betrieb von Endlagern) as well as the appointed quality control section, which has an independent status [8].

The quality control group supports the observance of the waste acceptance requirements by the following measures:

1. qualification of conditioning processes,
2. control of the conditioning processes and inspections,
3. random tests, and
4. checking of the documentation.

Controls at the repository entrance are dictated by health physics as well as legal accident prevention regulations. They comprise:

1. visual inspection of the waste package,
2. control of dimensions and weight,
3. measurement of the dose rate, and
4. control of surface contamination.

The corresponding reports are included in the waste package information to fulfill the waste acceptance conditions. All important data arising at the waste producers, the control points and the repository are documented in a central data library.

FINAL DISPOSAL

Land disposal in a suitable geologic formation deep underground is the favored route in the FRG. The general requirements are based on the German Atomic Act[1], the Radiation Protection Ordinance[3], taking account of the ICRP radiation protection standards and criteria, as well as national mining regulations. The Federal Ministry of the Interior has published in 1983 "Safety Criteria for the Disposal of Radioactive Wastes in a Mine" [9]. This covers all relevant items of safety and techniques in an overall manner and are valid for all types of radioactive wastes. The safety criterias are applicable to normal operation of a repository, to incidents during the operational phase as well as to long-term aspects after decommissioning of the repository.

Necessary details will be established in the legal licensing procedure called "Planfeststellungsverfahren" in Germany. The safety of a repository has to be demonstrated with a site-specific safety analysis which takes due account of the prevailing overall geological situation, the technical concept of the repository and the waste packages to be disposed of. The German strategy in isolating nuclear waste is to impose a series of engineered and natural barriers between the radioactive material and the biosphere.

The mobilization of radioactivity from waste canisters during the entire period of repository operation by water or brine contant is avoided by appropriate preventive measures. In the post-operational phase, even in the very unlikely event of a transport of radionuclides via a water path, individual dose rates are not allowed to exceed the limiting values specified in § 45 of the German Radiation Ordinance, e.g. 30 mrem/a, at any time. There are several good reasons for limiting the time of performing safety analysis to approximately 10,000 years, based upon comparison with natural analogues of radiation levels [10]. Consideration of consequences beyond this time range are largely speculative and therefore arguable.

Retrievability of waste canisters is not even foreseen during the operational period of the repository. Following closure of a deep underground repository, surveillance may continue for a period of time, but should be restricted to ordinary environmental protection requirements since the assurance of health and safety of the public must not depend on any surveillance measures. Disposal systems shall be marked and their locations documented in all appropriate state records.

Status of FRG Waste Repositories

A number of practical disposal options exist for all types of radioactive wastes in the Federal Republic of Germany, an overview of which is given in Table III. In the course of a recently completed assessment by PTB it was found that approximately 95 volume percent of the total waste arisings in the FRG can be accomodated in the KONRAD mine[11].

Table III. Overview of German nuclear waste disposal options

Underground Laboratory Salt Mine ASSE

Location: Remlingen, Lower Saxony
R&D-installation for the demonstration of disposal techniques in operation since 1967;
various disposal techniques for LAW and MAW demonstrated during 1967–1978
Retrievable borehole storage experiments for HAW- and MAW-containers in preparation
Experimental programmes on mining geology, heat transfer, geothermal behaviour, near- and far-field phenomena in progress, investigations on radiation stability, brine migration tests, radionuclide migration investigations

Underground Waste Repositories
KONRAD Iron Ore Mine

Location: Salzgitter, Lower Saxony
Suitable for wastes with negligible thermal impact on the host rock and decommissioning wastes
Licensing procedure in progress; license for construction and operation expected for the years 1987 and 1989, respectively

GORLEBEN Salt Dome

Location: Gorleben, Lower Saxony
Suitable for all kinds of solidified wastes
Survey from surface mainly by a drilling program finished; results indicate promise of suitability as a repository site
Exploration shaft drilling under way; results from exploration by mining methods available in 1992
Installation of repository 1995–1999; startup of disposal expected in the year 1999

Additional Activities in the FRG

Participation in the Swiss Grimsel Project regarding disposal in granite formations
Reconnaissance of potential granitic host rocks in the FRG

Safety Analysis

Practically the only relevant mechanism of radionuclide release from a disposal site and its transport to the biosphere is via a water path. Models developed from theoretical scenarios, which are increasingly verified by in-situ experiments, allow the prediction of radionuclide transport taking account of various phenomena, the most important of which are sorption, ion exchange, precipitation, diffusion, dispersion and radioactive decay.

In view of the complexity of these phenomena, all predictions are associated with a considerable uncertainty. However, a normal release scenario can be identified leading to a reasonably predictable radiation exposure pattern. On the contrary, the appearance of other processes, like seismic or tectonic events and influences which may alter the water flow are rather probabilistic.

Contradictory opinions are expressed about the usefulness and suitability of probabilistic safety methodology. Since no statistical experience and thus no relevant data are available, the assignment can be made solely on the basis of a subjective judgement. It is mainly for this reason why in the FRG only a deterministic approach is recommended in performing a safety analysis for a waste disposal system.

The situation is different for the waste conditioning and handling installations with active systems. Here, a probabilistic safety assessment may be helpful.

Merits of Waste Partitioning

A question which needs further comprehensive R&D work before it can be answered reliably is the incentive of applying chemical separations to alter the character of the wastes. Selective isolation of some actinide and other long-lived radioelements may be a goal worthwhile for development in the future. Its expected merit lies in the potential of decreasing the long-term radiological hazard. Further, one can remove the principal heat-producing isotopes Sr-90 and Cs-137 which, however, would be attractive only if these radionuclides find large scale utilization as irradiation or heat sources[12].

CONCLUSIONS

Considerable progress has been made towards establishing safe procedures and practical solutions for conditioning and ultimate disposal of all kinds of radioactive wastes. Different waste treatment methods have been developed and have attained a high technical standard.

Pilot plant operation has successfully demonstrated the safe handling and suitability of waste products and packages for final disposal in deep underground repositories. Models are available for short and long term safety assessments of such repositories. The positive picture is based on substantial scientific and technical results from the past the ongoing R&D activities as well as from experience already available.

The final waste repository projects Gorleben and Konrad are progressing according to schedule. The Asse mine is being used for supplementary R&D investigations.

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