SCANDINAVIAN WORK ON DISPOSAL OF WASTE FROM REACTOR OPERATION

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ABSTRACT

Waste is now being produced from the 8 power reactors in operation in Scandinavia, a number which is planned to increase to 16 within short. Solidified waste is stored at the reactor sites, but the waste management scheme cannot be optimized as long as methods for disposal have not been decided. In a joint Nordic project a reference waste handling system is analyzed in order to show the interrelation between the various steps: intermediate storage, transportation, and disposal, to correlate them with the necessary waste product specifications, and to show how possible radiation doses to humans can be assessed. Factors to be taken into account in the safety analysis are evaluated in the project. A considerable amount of development work in relation to reactor waste is also performed in the Nordic countries.

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THE NORDIC SCENE

Today eight power reactors are in operation at six sites in the Nordic countries (Fig. 1). Two more reactors are coming into operation in Finland in the next few months. The future of additional six Swedish reactors depends on the outcome of a referendum in March 1980. Solidification plants for incorporation of reactor waste in concrete or bitumen have been built at five reactor sites where the solidified waste is stored at present.

The Nordic radiation protection authorities have jointly worked out guidelines concerning waste management, and the reactor safety authorities have developed a special competence in this field. The authorities accept new waste plants on a case-by-case basis and also specify certain conditions; thus in Sweden a central data based book-keeping system is now required to keep track of all stored activity. The Finnish authorities request that funds be set aside to make sure that later occurring expenses can be met.

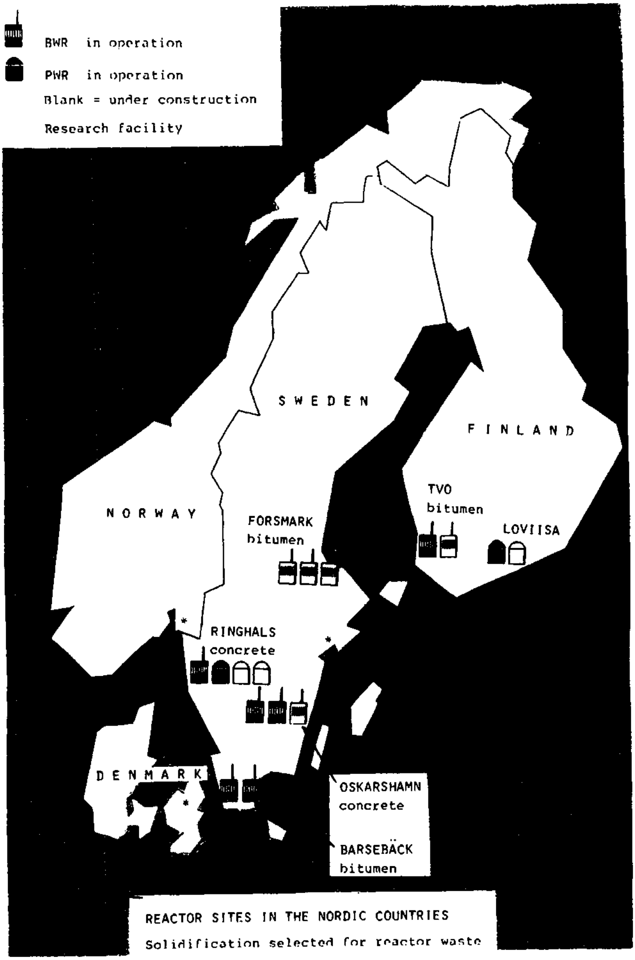
In the essential area of formulating criteria for waste products and disposal sites, Scandinavian authorities do not wish to take faster steps than their colleagues in other countries.

Equipment for waste solidification is installed at several Nordic research establishments, permitting to handle the waste from research reactors, including the Halden reactor.

The public debate on nuclear waste has been very intensive in the Nordic countries, but it is focused on waste from reprocessing.

In accordance with their general environmental policy, none of the Nordic countries permit sea disposal. Therefore, nuclear utilities pursue their work on incorporation of reactor waste, trusting that the products will be suitable for the disposal alternatives under consideration. One may suppose that the technical solutions actually being used represent a sub-optimization, since the product requirements (leaching, mechancial properties, fire resistance, etc.) cannot as yet be correlated to the handling sequence(intermediate storage, transporatation, underground storage) which the waste product will eventually be subjected to.

At present the utilities are extending their on-site storage facilities to permit reactor waste produced up the second half of the 80's to be stored locally (Fig. 2). Development work on solidification continues, and recently the Swedish national council for radioactive waste has terminated its "ALMA" conceptual study of a


BWR in operation
PWR in operation
Blank = under construction
Research facility
SWEDEN
FINLAND
NORWAY
DENMARK
TVO
bitumen
LOVIISA
FORSMARK
bitumen
RINGHALS
concrete
OSKARSHAMN
concrete
BARSEBÄCK
bitumen
REACTOR SITES IN THE NORDIC COUNTRIES
Solidification selected for reactor waste

<u>Figure 1</u>

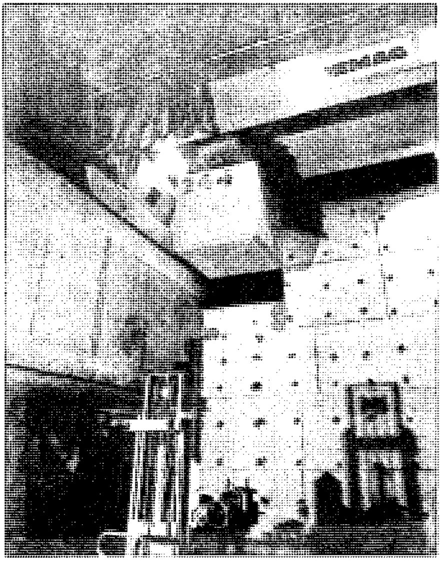


Figure 2: A 60,000 m^3 rock cavern for local storage of all waste from the Oskarshamn reactors up to the second half of the 1980'es. Commissioned recently at a cost slightly above $5 million.

complete handling scheme including disposal in an underground rock cavern[1].

In Denmark and Norway no decision has been taken about the introduction of nuclear power, but R&D efforts continue.

R&D on reactor waste in the Nordic countries is now being extended to used core components. There is also a different line of action towards a drasical reduction of the volumes of active waste to be finally disposed of. Thus, transfer of radioactivity from spent resins to inorganic ion exchangers (zeolites, titanates) is now being demonstrated. The spent resins could subsequently be burned in a fluidized bed incinerator also being developed, while the active ion exchangers would be sintered into small volumes of ceramic or glass so that the matrix itself provides the main barrier against leaching.

The utilities see it as an important task to present all-round solutions on waste management. Swedish and Finnish utilities this year invest $2.5 million on reactor waste R&D while $0.7 million are allowed by national funds.

Also the international Stripa project in Sweden, where Finland is a participant, will provide information useful in the design of disposal facilities for reactor waste.

THE NORDIC PROJECT

The Nordic cooperation in the nuclear waste field, previously presented at this conference[2], has since 1977 been centered around a joint project which is a part of a $2 million safety research program financed by the Nordic council of ministers. There is also a Finnish-Swedish utility waste group where questions of mutual interest are being discussed.

The Nordic waste project deals with the relation between the different parts of the handling scheme for solidified waste, their relative safety, the corresponding product quality requirements, and the overall safety analysis of the handling and disposal scheme. This working area was selected in view of the well-known dilemma that no firm rules or requirements exist for reactor waste management but that all installations have to be built with an adequate safety margin.

One important purpose is also to adapt R&D results, calculation models, and operating experience reported from elsewhere to the specific conditions prevailing in the Nordic countries. Thus, part of the project work consists in a careful evaluation and correlation of results reported - also from previous conferences in Tucson.

The working method of the project is to go through a number of typical "reference" cases[3], shown in Table 1 and Fig. 3. As reference waste spent reactor water clean-up resin from a BWR is selected[4], incorporated in cement or bitumen. Road and sea transportation are considered between the intermediate storage to either of three disposal sites: case 1 is shallow burial, case 2 is a concrete structure a few meters below the surface, and case 3 is a rock cavern. As many typical factors as possible are evaluated, drawing on the experience of the utilities and the viewpoints of safety authorities, both of whom are represented in the project steering group.

The project will be terminated at the end of 1980, so in this paper main emphasis is put on discussing the working procedures while the final findings will be reported later.

The result will permit those who have to perform the safety analysis in acutal cases to refer to reference calculations and to know which factors can be expected to be important in the evaluation. Another purpose is to establish interrelations between the system conditions and the required quality of the waste products and their packages. In establishing a catalogue of existing knowlege, the project will also give an indication of further R&D work needed.

Previous calculations have shown that the activity released as a result of normal reactor waste management is very low[5]. In the safety analysis the abnormal events are therefore those which are of the greatest interest. In Table 2 some of the scenarios are shown which are considered in the project as having some - although low - probability of occurrence in the Nordic countries.

RADIATION DOSES FROM WASTE MANAGEMENT

Consequences from waste management - normal and abnormal events - are measured by their possible radiation doses.

According to the radiation protection authorities in the Nordic countries both individual doses and collective doses have

Table 1

REFERENCE SYSTEM

Waste type: bead resins from BWR clean-up system.

Waste amount: 12 m^3/y for a 500 MW reactor = 2400 kg dry resin/year.
6 reactors (3 GW) are assumed to operate over 30 years.

Activity content after 5 years from the operation of one reactor for one year:

nuclide	Cs-134	Co-60	Sr-90	Cs-137	Ni-63	C-14	Ni-59	Tc-99	Cs-135	J-129	total
TBq/year	0.15	0.5	0.2	0.4	1	0.5	$5x10^{-3}$	$2x10^{-6}$	10^{-6}	$6x10^{-7}$	2.8
Ci/year	4	14	6	11	27	14	0.1	$5x10^{-5}$	$3x10^{-5}$	$2x10^{-6}$	76

Waste packages:

1) 220 l drums with cement-resin mixture (20%).
2) 220 l drums with bitumen-resin mixture (50%).
3) concrete moulds 1.2 m on each side with 100 respectively 300 mm wall thickness with cement-resin mixture.

Intermediate storage: 5-50 years in three types of local storage facilities.

1) A concrete structure with storage holes for 24 bitumen-drums each.
2) A concrete structrue above ground for storage of concrete moulds.
3) A 100 m long excavated near surface rock cavern.

Transportatation:

1) By lorry in transport containers of 50 respectively 100 mm steel. Number of containers per load: 14 cement drums or 7 bitumen drums or 4 concrete moulds.
2) By boat with roll-on roll-off loading of ALMA* containers with 72 drums or 12 moulds.

Final storage:

Case 1 - shallow burial.
2.5 x 0.6 km area slightly below grade.
Case 2 - concrete structure.
6 m deep 5 x 5 m concrete storage units with metal liner.
Case 3 - rock cavern. See [1].

*See [1].

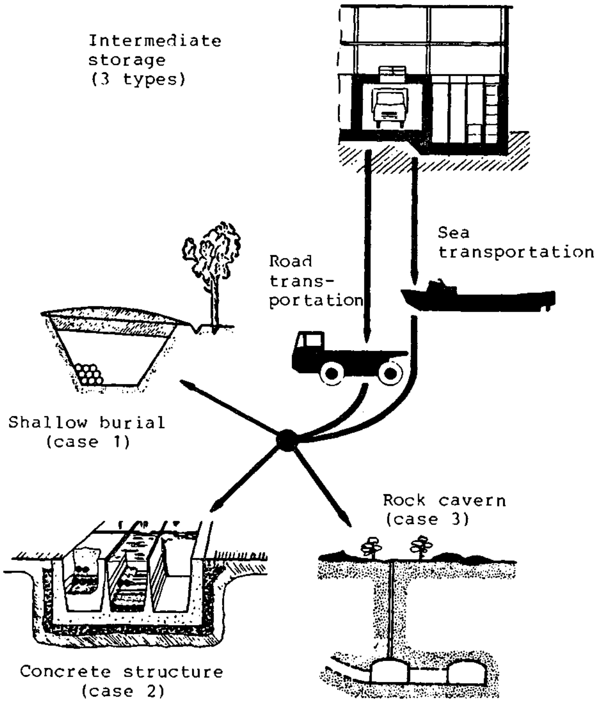


Figure 3: Reference system for handling of reactor waste.

Table 2

EXAMPLES OF ABNORMAL EVENTS

	probability	possible consequence	action
Intermediate storage			
plane crash with fire	very small	considerable	none
fire (bitumen)	small	considerable (aerosols)	evaluation
drop	larger	small (low heights)	none
disintegration of waste package	larger	only local	none
Transportation			
boat collision	small	small	ref. ALMA [1]
fire in boat accident	small	considerable	ref. ALMA
fire in road accident	small	considerable	evaluation
accident on bridge	small	local contamination	evaluation
Final Storage			
land uprise	exists	1 m/100 years	none
earthquake	small	none	none
distortion of barriers	exists	small	evaluation
inadvertant intrusion	exists after 50-100 years	considerable	evaluation
uncovering (wind, erosion)	small except for ref. case 1	small	none
war	-	smaller than other war consequences	none

to be considered[6]. Based on the ICRP recommendations the authorities state that individual doses due to waste management must be kept below those which are accepted for the planning of other steps in the fuel cycle. This is valid even if the doses occur in the future. In the calculations the accumulation of long lived radionuclides in the biosphere has to be considered. The possible risks from waste management must not significantly alter the risk situation of the fuel cycle.

While no fixed dose limits have been set by the authorities as yet, they do specify that individual doses are to be expressed by the weighed whole body dose equivalent. If release continues year after year, the dose limit should apply to the calculated dose commitment. In the case of a singlular abnormal event, the resulting dose is to be calculated over a human lifetime ("committed dose equivalent").

The concept of collective dose commitment will be used to judge the influence of waste management on various generations now and in the future. It appears that 10,000 years may be a reasonable span of time; this corresponds to the interval between glacial periods in the Nordic countries.

One way of judging the acceptability of abnormal events in waste management could be to calculate the resulting doses and multiply them by their probability, provided that the maximum calculable doses turn out not to give any acute effects. If the product of dose and probability does not influence the total picture of doses from normal operation in the fuel cycle, this could be an argument for their acceptability.

RADIOACTIVITY FROM DISPOSAL SITE

An essential case to analyze is the leakage of activity from the final storage. Values selected for the reference study are shown in Table 3.

In calculations made so far it was assumed that the total stored activity will become available by leaching at a constant rate into the surroundings after a certain number of years. The next step is to make calculations on the events taking place inside the storage facility. Here the integrity of the various barriers and the mechanism of water flow have to be considered[7]. In Fig. 4 an example is shown which gives the background for a mathematical treatment of activity distribution inside the concrete

Table 3

VALUES SELECTED FOR REFERENCE SITE CALCULATIONS

Hydraulic gradient	m/m	5×10^{-2}
Rainfall	mm/y	600 (with 20% evaporation)
Distance from recipient	m	well: 100, lake: 500, sea: 200

		rock	fissured rock	sandy till	clay till
Permeability	m/d	10^{-1}	5×10	10^{-2}	10^{-4}
Porosity	%	3×10^{-2}	5×10^{-2}	40	20
Ground water velocity	m/d	0.1		10^{-3}	10^{-5}
Ion exchange capacity	meqv/ 100 g			5	9
Distribution coefficient	cm^3/g				
Co-60		200		200	1000
Sr-90		10		20	100
Cs-137		100		100	500
Tc-99		0		0	0
I-129		0		0	0
Ni-63		100		100	1000
C-14		0		0	0

Diffusion constant m^2/y

	Co	Sr	Cs
concrete	4×10^{-9}	7×10^{-6}	3×10^{-4}
clay till	8×10^{-6}	7×10^{-5}	2×10^{-4}

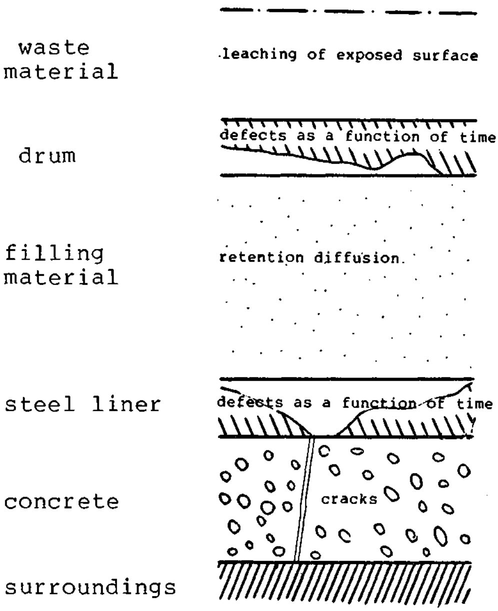


Figure 4: Calculable barrier system inside concrete structure.

structure, case 2 in Table 1. A data program handling such cases is expected to be available within a few months in Sweden.

Only extremely small amounts, if any, of the "normal" radionuclides contained in reactor waste escape from the reference storage facilities in the abnormal situations considered. In Fig. 5 the maximum activity of Sr-90 released through a concrete barrier in one of the reference storage cases is shown together with the decay curve. Assuming a diffusion constant ten times smaller than the reference value, no Sr at all will leave the zone before decay.

In the safety analysis longlived radionuclides (I-129, C-14, and others) are checked although their possible effect can be shown to occur only after 1000 years or more. Simplified calculations can be made because it is of no importance whether the storage structures remain intact during say, 100 or 500 years. It appears that better information is needed concerning the actual presence of these nuclides in the primary reactor circuit and about where and in which form they end up in the waste. Also the knowledge about their retention and diffusion needs to be improved. It is likely that such improved knowledge will demonstrate that the importance of these nuclides has so far been overestimated.

The values proposed in Table 3 turn out to have a major influence on the results. This is illustrated in Fig. 6 showing a parametric study of various reference values and storage alternatives[8].

From the figure it is also seen how the presence of clay may reduce the requirement for the leak resistance of the waste product itself.

CEMENT AND BITUMEN FIRE

According to Table 2, a fire either in the intermediate storage or during transportation needs to be evaluated with respect to its possible consequences.

In order to assess the amount of activity that can be released full-sixe fire tests were performed using the ISO standard fire curve[9]. With a 60 minutes fire and a cemented waste drum with 15% inactive resin the weight loss was 6%. A surface layer of 10-15 mm had become charred. Assuming that resin had been present in the surface layer and had burned, the maximum release fraction would have been around 10% for volatile nuclides (Cs), smaller for other nuclides.

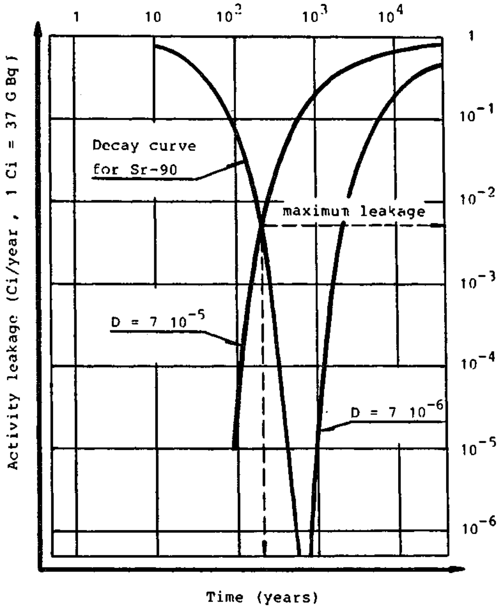


Figure 5: Maximum leakage of Sr-90 with two different assumptions for the diffusion constant D.

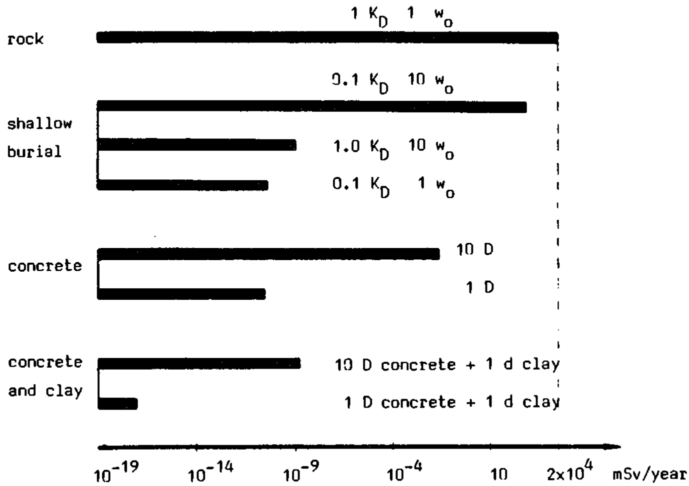


Figure 6: The influence of different assumptions for K_D (distribution constant), w_0 (ground water velocity) and D,d (diffusion constants) in relation to the reference values in Table 3.

Full-scale tests on bitumen-filled drums showed that a strong fire began after 20 minutes heating to 800°C. Bitumen irradiated experimentally responded faster due to the presence of radiolytic gases. The maximum weight loss was 85%. In a small scale fire experiment with resin contained in bitumen, 70% of the released particles were found to have diameters smaller than 10 μm so that they will be easily dispersed with air. Certain assumptions have to be made for the particle size and activity distribution, mainly because the volatility of Cs depends on the fire temperature. The individual doses depend on the heat content in the fire as well, because the dispersion varies with the emission height apart from other local factors such as land surface characteristics, wash-out phenomena, the weather at the time of release, etc. The codes presently used to calculate atmospheric distribution are based on reactor accidents and perhaps need to be modified to cover the waste situtation.

Calculations of the resulting doses for two different weather types show the order of magnitude for individual committed doses if one assumes an activity release of 0.3 TBq (8 Ci) of Co-60 (Fig. 7). In the figure the results of fire in the intermediate storage and in transport are compared[10]. The corresponding collective doses were calculated over a radius of 500 km yielding aproximately 0.5 person Sievert (50 personrems). Note that the weather type only influences the collective doses slightly.

It must be borne in mind that these dose calculations are made for postulated accidents and do not take their probability into account. The fire tests have shown considerable difficulties in igniting the drums so that it is hard to devise an accident scenario in the reference handling sequence which would lead to a bitumen fire.

ROAD COLLISION

In order to evaluate the consequences of road collisions, drop tests were performed using small blocks and full scale drums with resins incorporated in concrete and bitumen. The experiments showed that only minor damage is observed at the worst test conditions (9 meter puncture test). The increase of surface area of a small test block without a steel container has been expressed as $0.6 \times \sigma^{-0.7} \times h$, where σ is the compressive strength (MPa) and h the effective dropping height (m)[11]. In the worst case 0.06% of a full-size drum was released and the total mass of fine particles

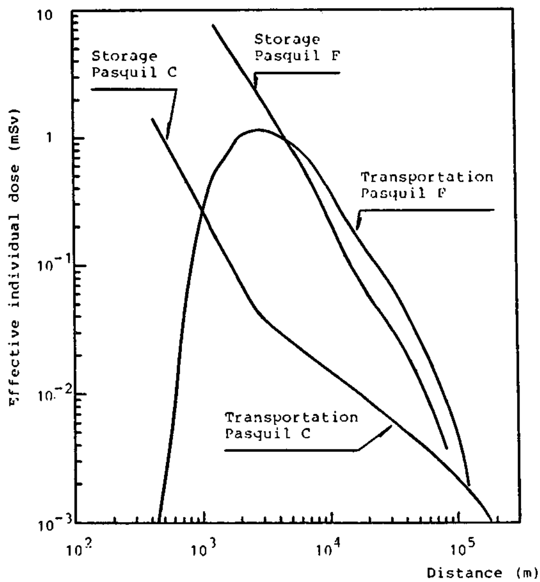


Figure 7: Doses as a result of bitumen fire during storage and transportation. The maximum dose occurs at different distances depending on the weather type.

(< 44 μm) was smaller than 0.2%. For bitumen drums no release of activity is indicated except perhaps at temperatures below the break point. This will be verified experimentally and might perhaps lead to some restriction on transportation during the cold season.

Assuming that the activity content involved in the accident is 0.07 TBq (2Ci) of Cs-137 and corresponding amounts of other nuclides (Table 1), then the activity released would be of the order of 0.5 MBq (1.5×10^{-5} Ci), giving negligible individual and collective dose commitments (10^{-8} resp. 2×10^{-7} person Sievert). Futhermore, the probability of a traffic accident causing such violent mechanical damage has been found to be very low: 10^{-10} per km for one drum or mould transported.

The above seems to indicate that this accident need not normally be considered in the analysis. But the place of collision may become contaminated, and if the package falls into a lake or a water stream, this may result in a different scenario, perhaps with swelling of the contents resulting in breakage of the package. Such a scenario remains to be analyzed.

PRODUCT CHARACTERISTICS

The above analysis of a road collision is an example of how a correlation between the safety analysis and the product characteristics can be established.

Repeating the dose calculations as above but assuming an increase by several orders of maginitude in the "acceptable" dose commitments, then a much lower compressive strength would be required than what is normally aimed at.

A similar procedure could be used to go through all abnormal events in order to examine which nominal value for each individual product characteristic would be sufficient.

One important product property is leachability. Leach rates are of significance at various points in the safety analysis, especially in the calculations of radionuclide migration and concentration around the underground disposal facilities. Unfortunately, leach tests are not easy to reproduce, and results from different laboratories are seldom consistent. The chemical composition of the leak water is of considerable importance (Fig. 8). In the

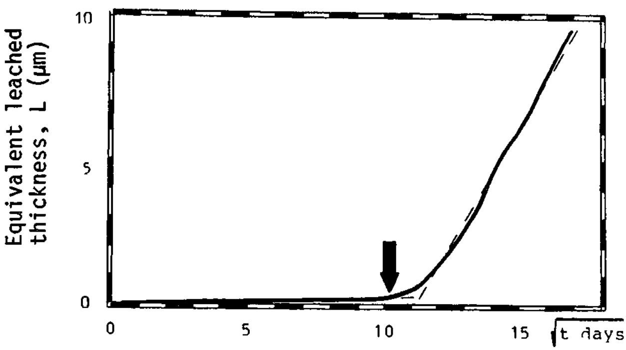


Figure 8: Leaching curve for a 40% resin-bitumen mixture with Co-60 tracer. The arrow indicates a change in the leach fluid from 14 meq Na_2SO_4/l to 33 meq $MgSO_4$/l.

Nordic work use of different water qualities for testing have been evaluated (demineralized water, "Nordic" ground water, Baltic sea water). However, with the static testing methods presently in use, the water composition may change rapidly according to the composition of the test piece itself, thereby affecting the leach rate. This situation will also exist around an actual storage facility.

Within the project it is intended to establish a catalogue of relevant methods to test waste incorporated in bitumen and cement. Three lines of tests will ultimately be needed: - process control tests (to be used at the reactor site), - quality control tests, - laboratory tests for R&D.

So far the safety analysis has identified only few important process control tests, among which a screening test for water resistance appears to be particularly useful. Here product samples are immersed in water under standardized conditions and classified according to their stability. Samples which can be visually identified in the test turn out to have high leach rates (about ten times the normal). Other attempts to reduce lengthy leach rate determinations use correlations with visual inspection, microscopy, and electron microscopy combined with microprobe analysis.

Present leach tests are mostly based on Cs-tracer studies. The application of similar leach rates for other nuclides may render the results of safety assessments far too pessimistic. The following leach rates (cm x day^{-1}) are assumed, and it is planned to check their importance through sensitivity analysis:

	Cs	Sr	Co
Concrete/resin	10^{-3} - 10^{-4}	10^{-4} - 10^{-5}	< 10^{-5}
Bitumen/resin	10^{-4} - 10^{-5}	10^{-4} - 10^{-6}	10^{-5} - 10^{-6}

The laboratory work is closely correlated to full-scale experience to avoid overlooking major effects such as cracking, swelling, bacterial growth, temperature rise during curing, etc.[12]. A major work of development has recently been performed in Sweden to explain the behavior of resin-cement mixtures and recommend processing conditions (low water/cement ratio, limited resin content in cement) in order to reduce the risk for cracking.

The Nordic work includes studies concerning compatiblity between resins and cement[13]. A better compatibility is obtained for cation than for anion exchangers, better for spent than for new resins. It is highly dependent on resin manufacturer, cement type, and additives. Excellent results are reported from Norway with small amounts of a commercial additive to the cement paste.

CONCLUDING REMARKS

The results from the Nordic project will be reported about one year from now. Even if its reference calculations were to show that, for example, no solidification of resins or no metal casing would be needed to surround solidified concrete mixtures, it is hard to believe that this would change the accepted handling sequences in the short run. However, the system analysis approach can help in selecting the appropriate conditions for disposal and in keeping the required safety precautions at a reasonable level.

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Note that the AO-project working documents are mainly in Scandinavian languages.