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Encapsulation of spent nuclear fuel - safety analysis

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ENCAPSULATION OF SPENT NUCLEAR FUEL-
SAFETY ANALYSIS

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This report concerns a study which was conducted for SKBF/KBS. The conclusions and viewpoints presented in the report are those of the authors and do not necessarily coincide with those of the client.

A list of other reports published in this series during 1983, is attached at the end of this report. Information on KBS technical reports from 1977-1978 (TR 121), 1979 (TR 79-28), 1980 (TR 80-26), 1981 (TR 81-17) and 1982 (TR 82-28) is available through SKBF/KBS.

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ENCAPSULATION OF SPENT NUCLEAR FUEL - SAFETY ANALYSIS

SUMMARY

The radiological safety of the encapsulation procedure planned for the unprocessed nuclear fuel is analysed. Two methods of encapsulation are studied, both including a copper canister. In one process the copper canister with the spent fuel is filled with copper powder and pressed to solid copper metal at high pressure. In the other process lead is cast around the fuel before the canister is sealed by electron beam welding.

The activity decay of the fuel has been going on for 40 years before it arrives to the encapsulation station. This is the basic reason for expecting less activity release and less contamination of the plant than would be the case with fuel recently taken out from the reactors.

In analysing the plant safety, experience from the nuclear power plants, from the planning of the swedish central storage facility for spent fuel (CLAB) and from La Hague has been used. In those cases where experience is missing, conservative values have been chosen. The analyses is also based on experience of today's technology, although it should be possible to improve the encapsulation process further before time has come to actually build the plant.

The environment activity release will be very low, both at normal operation and following accidents in the plant. Using very conservative release rates also the most severe anticipated accident in the plant will induce a dose to critical group of only 3 μSv.

The staff dose can also be kept low. Due to remote handling, fuel damage will not primarily give staff dose. Of the totally anticipated staff dose of 150 man mSv/year the greatest portion will come from external radiation during repair work in areas where fuel containing canisters by failure can not be taken away.

The hot isostatic pressed (HIP) canister process contains more operations than does the lead casting and welding procedure. It is therefore expected to give the highest activity release and staff dose unless extra measures are taken to keep them low.

Using remote operation and adequate equipment the encapsulation station with any of the two processes can be built and run with good radiological safety.

CONTENTS

ENCAPSULATION OF SPENT NUCLEAR FUEL-SAFETY ANALYSIS

1 INTRODUCTION

This report will deal with the radiological aspects of the handling and encapsulation of the unprocessed spent nuclear fuel. The analysis covers the fuel handling from the arrival of the transport flasks to the encapsulation station until the fuel canisters have been placed in the final repository. It also covers the handling of fuel boxes from BWR:s and boron glass rods from PWR:s. The encapsulation station may also be used for encapsulation of other core components, e g neutron detectors and control rods. That is not included in this safety analyses.

The study covers the plant internal radiological safety as well as release of radiocative material from normal operation and from possible accidents in the plant.

The report is a background report to the KBS-3 project. Thus basic data, e g amounts of fuel handled and fuel activity contents are taken from the KBS-3 reference data.

Regarding release of activity from the fuel, experience from swedish nuclear power stations and from the planning of the swedish central facility for storage of spent fuel (CLAB) has been used to the greatest possible extent. Also Studsvik expericence from hot cell handling of spent fuel has been applied. As the fuel to be handled in the encapsulation station will be stored for 40 years before encapsulation, some parameters can not be based on experience. In these cases, conservative estimates have been used.

The detailed design of the equipment used for handling and encapsulation is not yet settled. Parts of the process, e g welding of copper canisters, lead casting and hot isostatic pressing have been studied, mostly though to find the parameters for the long term behaviour of the canister. These studies of course have also shown many of the important problems from the practical point of view. In other cases this safety analysis is based more on a general experience of what is possible to do with today's technical tools than on a detailed analysis of an actual design. This is natural as the plant is not to be built until about year 2020.

2 ACTIVITY AND DOSE RATES

The plant is designed to handle the spent fuel from the 12 swedish nuclear power plants operated for 30 years. The total amount of uranium in this program is 6000 tons. The fuel will be stored for 40 years before it is taken to the encapsulation station, which will be open for a period of 30 years. Thus 200 tons of uranium will be handled annually.

In addition to the fuel bundles, 23.000 BWR fuel boxes and 200 boron glass rod bundles will be handled in a separate line and stored in a separate repository.

2.1 ACTIVITY CONTENT OF THE FUEL

The spent fuel activity depends on numerous parameters, e g initial enrichment, reactor type and burn-up. The spent fuel in this plant will come from the swedish reactors, 9 BWR:s and 3 PWR:s. The amount of uranium will be 4.550 and 1.450 tons respectively. The burn-up of the fuel has been rather low in the first cores of the reactors but is believed to increase. It is generally higher in PWR - than in BWR-reactors.
The content of the most important fission products and transuranic elements are shown in table 1. It is calculated for fuel that has been stored for 40 years after discharging. The half-life of the different isotopes are also shown in the table. Many of the isotopes important in connection with accidents in the power plants have after 40 years decayed to negligable values, for example I 131 and Cs 134.

Many of the isotopes are of little importance from the point of view of release from the encapsulation station. The most important ones are tritium, krypton, iodine and cesium, most of the other can be released only as aerosols.

The decay heat in the fuel is very low after 40 years. For the PWR fuel, as shown in table 1, the decay heat is 740 W/tonU. This value is somewhat higher than the average value for the fuel in each canister, but has been used for calculating fuel temperature rise in vaccum.

The fuel element structural material contain certain amounts of induced activity, chiefly cobalt and nickel. This is of minor importance in this study, as the complete fuel elements are encapsulated in the canisters. Also the fuel boxes, which are handled separately, contain induced activity, the principal isotopes are shown in table 2, where the activity is given in GBq per ton of box material. The calculation is based on actual material specifications and on the assumption that each fuel box has been irradiated during one fuel cycle.

Table 1

Fission products and heavy nuclides in spent PWR-fuel after 40 years decay

Fission product	Half life years		Acitvity Bq/tonU		Heavy nuclide	Half life years		Activity Bq/tonU	
H 3	12,3		2,5	E12	Ra 226	1,6	E3	155	E3
C 14	5,7	E3	4,8	E6	Th 229	7,3	E3	14	E3
Se 79	65	E3	17	E9	Th 230	80	E3	18	E6
Kr 85	10,8		28	E12	Pa 231	32,5	E3	1,2	E6
Sr 90	29		1,1	E15	U 233	162	E3	3	E6
Zr 93	1,5	E6	74	E9	U 234	247	E3	52	E9
Nb 93M	13,6		63	E9	U 235	710	E6	518	E6
Tc 99	210	E3	555	E9	U 236	23,9	E6	10	E9
Pd 107	6,5	E6	5,2	E9	U 238	4,5	E9	12	E9
Cd 113M	14,6		407	E9	Np 237	2,1	E6	16	E9
Sn 126	100	E3	34	E9	Pu 238	89	E0	100	E12
I 129	15,9	E6	1,33	E9	Pu 239	24,4	E3	11	E12
Cs 135	2,3	E6	14	E9	Pu 240	6,8	E3	14	E12
Cs 137	30,2		1,74	E15	Pu 241	14,6	E0	925	E12
Pm 147	2,6		122	E9	Pu 242	379	E3	107	E9
Sm 151	93		10	E12	Am 241	433	E0	178	E12
Eu 154	8,5		20	E12	Am 243	7,7	E3	1,2	E12
Eu 155	4,8		1,1	E12	Cm 244	18,2	E0	35	E12
					Cm 245	8,3	E3	13	E9
					Cm 246	4,7	E3	3,3	E9

Table 2

Induced activity in fuel boxes, GBq/ton

Specimen	Half life years	Activity GBq/ton
C 14	5735	62
Co 60	5,25	200
Ni 59	80·E3	33
Ni 63	92	3900
Zr 93	1,5 E6	20
Nb 93 m	13,6	18

The amount of crud activity on the outside of the fuel elements is more difficult to calculate. Analysis of crud films have been done as well on BWR- as on PWR-fuel rods from swedish plants. The BWR-fuel has shown generally thicker crud layers, which should also be expected as the water is boiling. Apart from this also the water chemistry, the power density, the burn-up and the reactor primary system material composition influence the amount and composition of the crud. As a basis for this study, conservative crud activity estimates used in the CLAB-project (Ref 2) have been recalculated to 40 years activity decay. By this time Co-58 and Zn-65 have principally vanished and Co-60 is dominant from the radiation aspect. Also this isotope has decayed to 22 GBq/tonU, which is only 0,006 of the activity when the fuel is taken to CLAB. When disussing crud release to the atmosphere also Ni-63 may be of interest. As it is a β-emitter it is though easily shielded.

The crud in some cases also contain heavy nuclides, normally from damaged fuel in the reactor but to some extent also from the fuel manufacturing. Measurements in Oskarshamn 1 indicate that 400-700 MBq/reactor year could accumulate on the fuel. Measurement of the isotope composition has not been made. As some part of this α-activity will have a short half-life, e g. Cm 242, and also Oskarshamn 1 has had more fuel-rods damaged than should be generally expected we have reduced this measured value but only with a factor of 4, or to 8 MBq/tonU.

The crud activity is given in GBq/tonU for practical reasons.

2.2 RADIATION LEVELS

The radiation from different components, handled in the encapsulation station, has been calculated by ASEA-ATOM (Ref 3). The calculations will be used as a design basis for the shielding in the plant and has also been used to calculate dose commitments for those cases when manual action close to the components is necessary, e g after failure of the remote handling systems.

In table 3 an example of dose rate from unshielded fuel in air is given. These figures are chiefly of interest to check that equipment in the hot cell is not damaged by the radiation, staff entry to the cell will not be possible when it contains unshielded fuel.

Table 3

Radiation from an unshielded BWR fuel bundle
(E = 33 MWd/kg u, t = 40 y)

Distance (m)	Dose rate (Sv/h)
Contact	72
1	3.1
2	1.2
5	0.3

In table 4 the dose rates from a copper canister containing complete fuel elements as described in section 3 are shown. Calculations have been made as well for the welded canister as for the pressed one, and the radiation levels both in the mid radial and downward axially direction are presented. Dose rates in the upward axially direction are found to be an order of magnetude smaller. The neutron dose rate is listed separately as the neutrons call for specially shielding materials.

Table 4

Radiation levels (mSv/h) outside copper canisters containing BWR fuel elements

Type of canister		Welded		Pressed	
Direction		Mid radial	Axially down	Mid radial	Axially down
Contact	gamma	1.9	0.12	9.0	0.48
	neutrons	1	0.2	1	0.2
1 m	gamma	0.53	0.02	2.5	0.10
	neutrons	0.3	0.04	0.3	0.04
2 m	gamma	0.30	<0.01	1.4	0.03
	neutrons	0.2	0.01	0.2	0.01
5 m	gamma	0.10	<0.01	0.46	<0.01
	neutrons	0.05	<0.01	0.05	<0.01

Finally, table 5 shows radiation levels in two directions outside a concrete block containing BWR fuel boxes

Table 5

Radiation levels (mSv/h) outside a concrete block containing BWR fuel boxes

Direction	Mid radial	Axially down[1)]
Contact	2.9	7.4
1 m	1.1	2.2
3 m	0.42	0.45

1) Without extra iron-shield.

The calculations are described in more detail in references 3 and 3a. Used source terms are in accordance with the radioactivities given in section 2.1 above.

2.3 ACTIVITY RELEASE FROM FUEL

When handling the fuel in the encapsulation station, activity may be released to the handling and storage pools and to the plant atmosphere. The amount of activity released naturally depends on the history of the fuel element in the reactor and to some extent in the intermediate storage. When discussing release it is important to draw a distinction between intact and damaged fuel rods.

It is also important to note that the highest possible release from a fuel rod is far less than from a reactor accident, as also a lot of the gaseous fission products are bound in the fuel pellets as long as the fuel is cold and has a "normal" power history in the reactor.

As far as possible the choice of activity release figures in this report is based on experience from handling of spent fuel in power plant pools and also in hot-cells and during transport in shielding flasks. From this basis, conservative figures have been chosen, when calculating activity release from normal operation and misadvantures.

2.3.1 Release from intact fuel elements

From the intact fuel only crud activity can escape. The release to the fuel storage pools has been measured in the Oskarshamn and Ringhals 1 plants. The release rate is in the order 0,5-1% per year of storage. As part of the crud is very hard burned to the fuel cladding a constant release rate after 40 years is not very probable. If yet we just calculate for activity decay the crud release rate to the pools would be 6 Bq/s,tonU of Co-60 and 2,2 mBq/s,tonU of heavy nuclides.

When handling the fuel for transport or encapsulation the crud release will be greater. Not only shaking the fuel element but also eventual heating during transport in dry flasks will incerase the crud release. In the CLAB safety report normally 1% and at most 10% of the crud inventory is assumed to come loose. Although the decay heat of the fuel is much lower the same percentages are used in this analysis. Thus a normal fuel transport would leave 220 MBq/tonU of Co-60 and 80 kBq/tonU of actinides in the transport flask or fuel pools. On certain occations the values could be a factor of 10 higher.

The experience of dry fuel handling in Sweden is limited. In the Studsvik hot-cell laboratories a few fuel-rods and elements have been treated, but no direct measurements have been made of natural crud release to the cell. In one set of experiments crud composition was measured (Ref 5). This experiment also showed that most of the crud is hard burned to the cladding. Another applicable experience is from the dry flask fuel transports to the fuel reprocessing plant in La Hague. During this transport the fuel is heated to the order of 350°C.

When the fuel in the encapsulation station arrives some crud may get losse due to handling and transport heat-up, although not to 350°C. When then the fuel is lifted up to be placed in the canisters, it is assumed that 10% of what was released during transport is released to the dry-cell atmosphere. That gives a figure of 22 MBq/tonU of Co-60, on some occasions up to 200 MBq/tonU. If a fuel element by mishap falls on the floor it could be still higher, the additional amount will though principally deposit directly on the floor.

2.3.2 Release from earlier damaged fuel

The relaese of crud activity will of course be the same as discussed in 2.3.1.

Measurements have been made of the activity release from damaged fuel stored in the fuel pools in Oskarshamn 1 and Ringhals 1. The dominant isotope for releasse after 40 years is Cs-137, which would give 200 Bq/s per damaged fuel rod. For the CLAB-project it is assumed that 0,01-0,1% of the total cesiuminventory in damaged rods will be released during transport. This assumption is based on La Hague-experience and is probably very conservative as it is based only on PWR-rods, which operate at considerably higher temperature than BWR-fuel thus releasing more fission products from the fuel pellet lattice structure. The same assumption applied for the encapsulation station would give a release of 0,5-5 GBq per damaged fuel rod. Assuming 0,1% of the fuel rods are damaged this would in turn induce a cesium release to the pools in the order of 100 GBq/year.

The cesium release will only go to the water at ambient temperature. Cesium can be released to the atmosphere only as aerosol unless the temperature is risen over 350°-450°C or to greater extent over 800°C (Ref 6).

2.3.3 Release from handling damage

When a fuel rod gets damaged in the encapsulation station, the greatest activity release occurs if the rod is previously intact. The basis for the release is again the power and burn-up history of the fuel. Regarding BWR-rods with normal load history, measurements (ref 4) show that generally less than 1% of krypton, iodine and cesium has got loose from the pellet lattice to the cladding gap or fission gas volume. Tritium is released from the lattice to a somewhat greater extent depending on the small size of the tritium atoms.

The krypton stays gaseous in the cladding gap while iodine and cesium to a great extent forms cesium iodide, the cesium also e g -molybdate and -uranate. The tritium, finally, forms water or binds to the cladding as hydride. All these reactions act to reduce the activity release following a fuel rod damage at ambient temperature. Experiments show for example that gaseous cesium can be formed in very small fractions (<1%) at temperatures of 350-450°C but principally over 800°C.

As cesium and iodine merely follow the water, and some water droplets may be released to the atmosphere, the atmosphere release figures in table 5 are chosen conservative.

For PWR-fuel with higher specific power and higher burn-up, the fission product detachement from the lattice is higher, we have conservatively assumed a factor of 10.

Following more severe fuel rod damage, also fuel pellets can break and release aerosol activity.

The result of the above discussion is shown in table 5, the figures beeing used in the safety analysis in sections 4 through 6.

Table 5

Release rates from damaged fuel rods

Element	% released to gap		% released from gap to plant atmosphere		Released per damaged rod	
	BWR	PWR	<300°C	300-500°C	BWR	PWR
H 3	5	50	10	50	35-170 MBq	0,2-1 GBq
Kr 85	1	10	100	100	0,8 Gbq	5 GBq
I 129	1	10	20	100	7,5-50 kBq	50-250 kBq
Cs 137	1	10	1	10	0,5-5 GBq	3-30 GBq

3 ENCAPSULATION PROCEDURE

The encapsulation station is situated on ground level and has a vertical elevator connection to the final repository area, see fig 1. The fuel arrives to the site by rail or by a special lorry, in both cases packed in the same type of transport-flasks as are used for transport from the power plant to CLAB.

This section briefly describes the fuel handling from the arrival of the flask until the fuel canisters are placed in the final repository. A more thorough description is given in ref 7. Fig 2 illustrates the flow of the fuel through the encapsulation station.

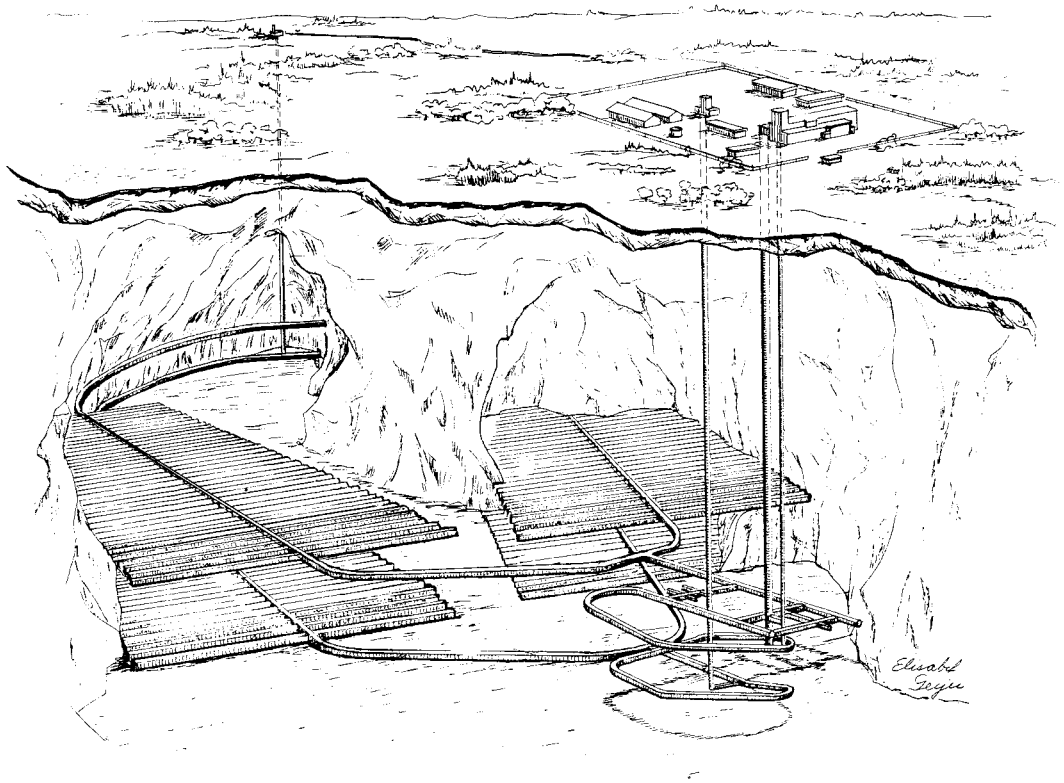


Fig 1. Encapsulation station and final repository

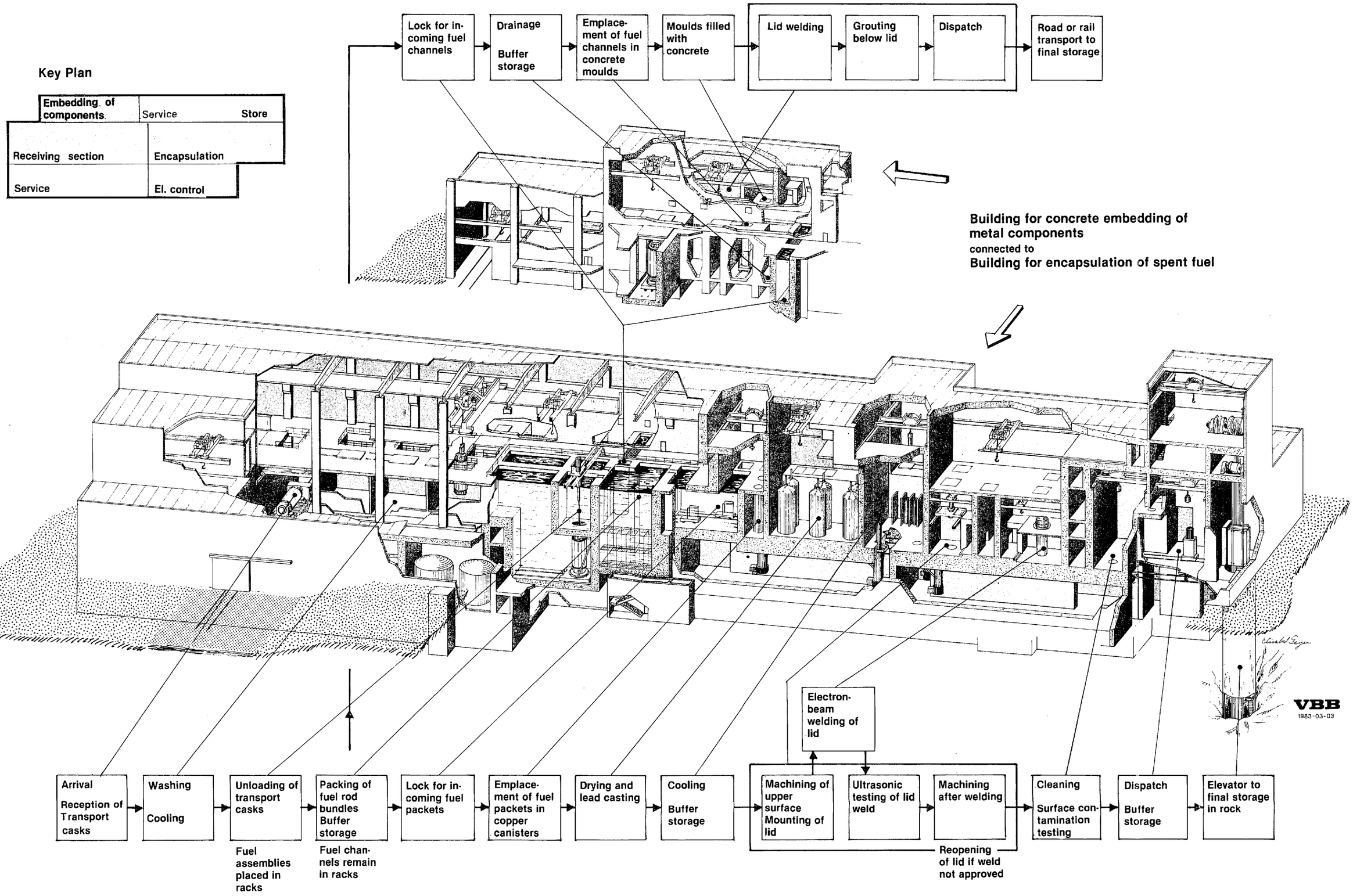


Fig 2. Encapsulation station
with process scheme.

3.1 FUEL RECEPTION AREA

The fuel reception area is functionally similar to the corresponding one in CLAB. Differences are due to the decay of the fuel activity and to the separation of the BWR-fuel from the fuel boxes.

The fuel is unloaded from the flask in a reception pool. Damaged fuel is transported in a special container in the flask. The damaged fuel is placed in a clean container before it leaves CLAB.

In a pool adjacent to the flask unloading pool there is space for a buffer storage of fuel for 3 month's use in the encapsulation station. In this pool the fuel elements are also separated from the fuel boxes and put into casettes suited for the copper canisters.

Each canister casette will contain 9 BWR- or 2 PWR- and 4 BWR-elements, se fig 3.

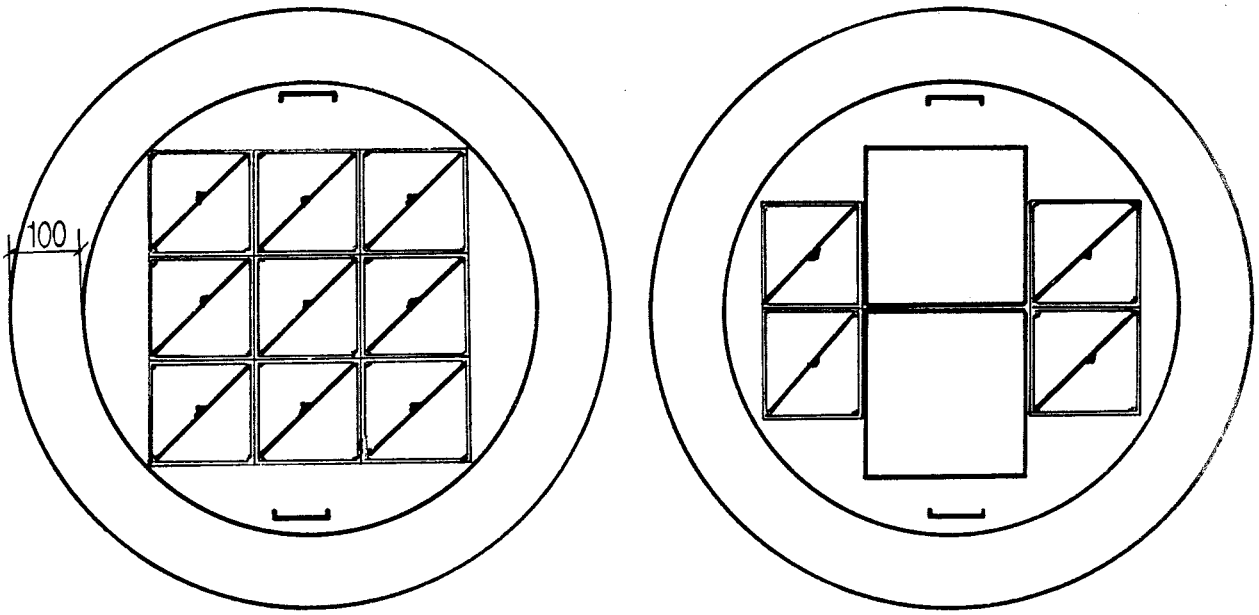


Fig 3. Canister with 9 BWR- or 2 PWR- and 4 BWR-elements

The configuration of fuel is chosen to be safely subcritical also with new fuel. From the casette loading the casettes are taken into a lock pool, from which they are transferred to the encapsulation line. The fuel boxes are treated in a separate concrete blockmaking line.

All the fuel handling in this section of the plant is made under water as thus well shielded. The pools are provided with water clean up systems principally similar to those in CLAB but adapted to the activity decay of the fuel.

3.2 ENCAPSULATION IN WELDED CANISTERS

The encapsulation is done in a dry treatment line and is remotely controlled from a separate control room. There are two parallell treatment lines. The transfer of the canister between the different treatment cells is done with a transport machine running below the treatment cells. The transport line is split up in two halves, the cooling down cell after lead casting beeing connected to both transport lines.

The fuel element casettes are lifted out of the lock pool by an overhead crane. When the water has drained off, the casette is placed in the canister, which has previously been fitted to the dry side of the canister loading cell. This is the only cell in the line where the fuel bundles are treated completely unshielded. The walls of the cell are accordingly thick and much attention is payed to the reliability of the lifting device. Thus the crane can be withdrawn to a shielded service area. The ventilation of the cell is, as for the other cells where contamination may occur, separated from the normally accessible areas by pressure difference. The ventilation exhaust air is filtered for aerosols and iodine in filter banks.

The charged canister is transferred with the transport machine to the next cell. It is elevated into the lead casting furnace. The fuel and canister are heated in vaccum to dry the fuel and prepare for the lead casting. The lead is molten in a separate furnace and pumped into the canister when this has reached a temperature of 380-400 °C. Lead is filled well over the top plates of the fuel. The canister is partly cooled down by circulating nitrogen before it is taken out of the furnace and transferred to the cooling down and buffer cell, which is also used for the input of empty canisters to the treatment line.

In the next step, the back-end transport machine operates the canister to the machining cell, where the lead surface is turned plane and the welding surface of the canister is polished. This cell is also used to redo canisters, that have failed in the final quality control of the sealing weld. All the machines are remotely operated and the cuttings are taken care of with respect to its contamination.

The lid is fitted to the canister, which then is transferred to the welding cell. The electron beam welding is done in vaccum and tested by ultrasonic. Finally the canister is checked for outside contamination and if necessary cleaned before beeing placed in the output canister buffer.

3.3 ENCAPSULATION BY HOT ISOSTATIC PRESSING, HIP

As an alternative to the above described procedure, the copper canisters may be filled with copper powder and sealed by hot isostativ pressing. The principal layout of the treatment line is similar to the one described in 3.2 but the number of operations is greater, see fig 4.

Thus after loading the fuel casette into the canister a covering lid is placed on the canister and it is thoroughly filled with copper powder. An inner lid is mounted and welded to the canister.

To make a good result of the HIP-procedure, the oxide layer of the canister and copper powder must be reduced. This is done by hydrogen conditioning in a special furnace operated at 350 ^{o}C. Hydrogen gas, saturated with water at ambient temperature, is filled in the canister and then evacuated to vaccum. This treatment is repeated a number of times until the oxide layer is reduced. After cooling down, the hydrogen connection is sealed and a second, outer and redundant, lid is welded to the canister.

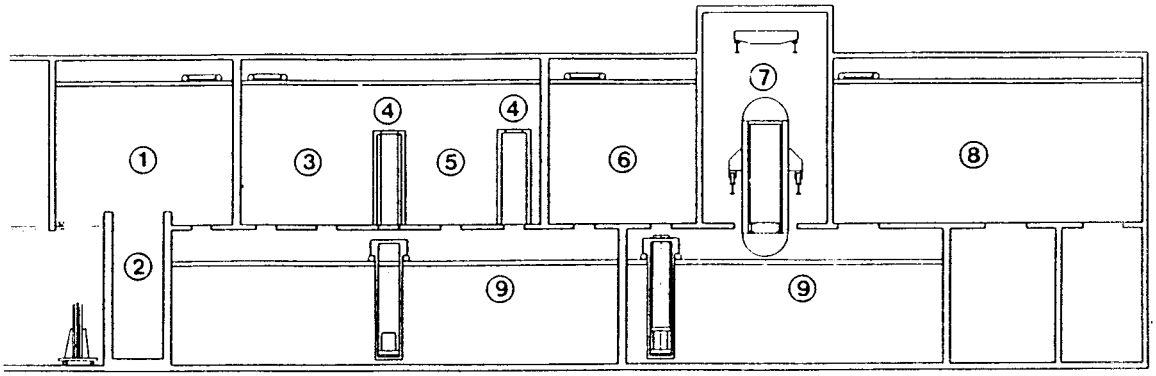
After another hydrogen conditioning, the canister is prepared for the so called Quintus furnace. In this, the furnace is filled with argon to about 1000 atm kept constant for a few hours, the pressure beeing about 1500 atm. Under this pressure the canister is slightly deformed and the copper powder turns to homogeneous copper metal, very efficiently sealing the fuel rods for the final repository.

Also in this case the treatment line is finalized by quality control an eventaul decontamination. Although the number of tools, necessary for this method of encapsulation, is greater than for the welding procedure, all the operations may by built with existing technique. The details of hydrogen conditioning and pressing are dealt with inter alia in ref 7 and 8.

3.4 TRANSPORT OF CANISTERS TO FINAL REPOSITORY

The transport area for the canisters is a non-contaminated area. The lifting and transport of the canisters is remote operated as the dose rate from the canister in rather high unless transport shielding is applied.

The canister stands upright in the elevator (see fig 5), which is shielded upwards to ensure access to the top of the elevator. Also the wagon carrying the canister into the elevator is shielded.



1. Drying cell
2. Drying chamber
3. Copper powder storage. Application of inner seal lid
4. Hydrogen conditioning furnaces
5. Application of main and outer seal lids
6. Transfer cell
7. Quintus press for hot isostatic pressing
8. Exit and control
9. Transport ducts

Fig. 4 Fuel and canister handling in the encapsulation part.
Canister for hot isostatic pressing.

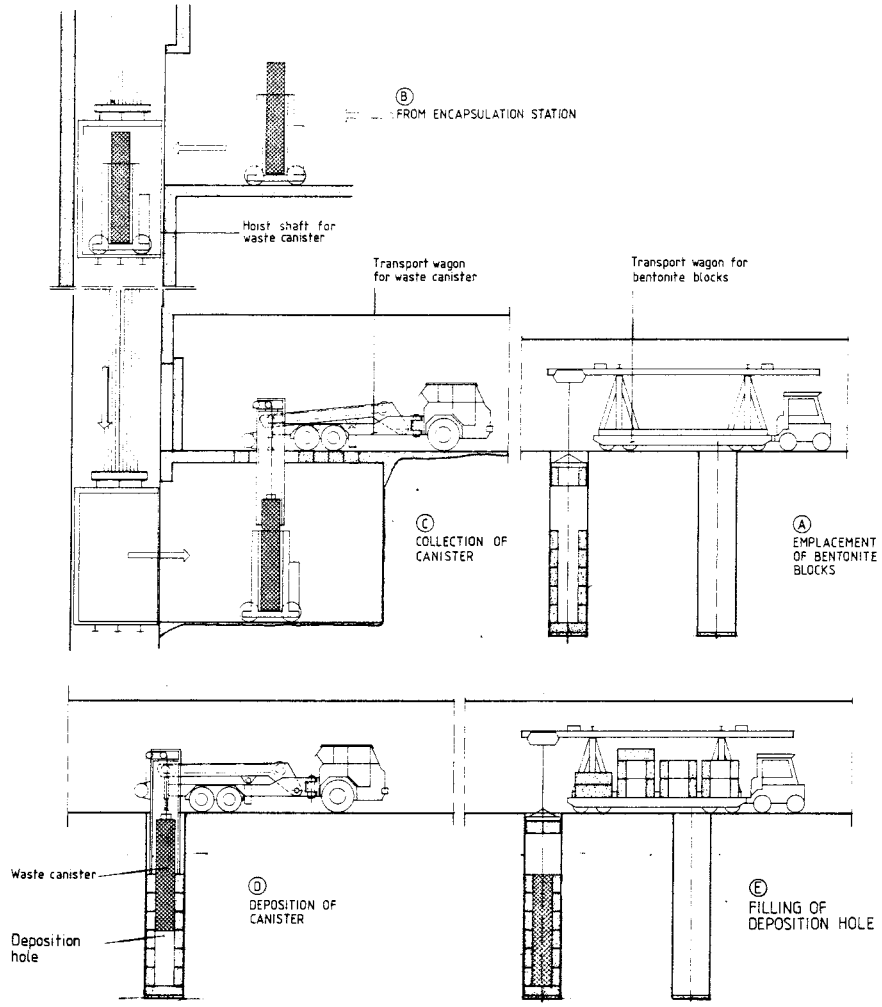


Fig 5. Transport of canisters to final repository

Coming out of the elevator, the canister is placed under a hole in the transport tunnel. The transport truck lowers its radiation shield hydraulically over the canister, lifts it up and tilts it before starting the transport to the storage tunnel. In the storage position, the shield is raised and lowered into the storage hole with the canister before the canister is further lowered to its final position. Wheather the transport wagon will be railbound or not is subjuct to later decision.

3.5 FUEL BOX CONCRETE BLOCKS

The fuel boxes a treated in a separate line from the pool where the fuel is taken away.

In this line (see fig 2) a prefabricated concrete container is placed in a separate loading cell into which the fuel boxes are transferred from the lock pool to the concrete container. 7 x 7 boxes fill the container. Instead of fuel boxes the central 3 x 3 space can be used for boron glass rods from PWR or for other medium active components.

The dose rate from the boxes is of course far less than from the fuel, but there is still need for remote operation and ventilation system underpressure. Most of the activity originates from Co-60 and Ni-63.
After loading the goods into the concrete container, the container which is placed on a wagon is transferred to a concrete pouring cell. Here, concrete is poured, a lid is mounted and welded to the container and concrete is finally injected under the lid.

The quality demands on the concrete are not very stringent. Therefore the concrete is easy to handle and no special quality control is required. After hardening the container and the waste form a concrete block 1,25 m square and 5,3 m high.

The concrete block is tilted, hoisted with a crane into a shielded transport container and taken by truck to the medium active waste depository not far from the plant. Here it is unloaded from the transport container, raised into the elevator and hoisted down to the depository level, where again it is laid down, (see fig 6). By aid of a straddle crane it is transferred into the storage tunnel and placed in the depository.

The concrete blocks are stacked five blocks wide and three blocks high in the tunnel, which is successively filled with concrete around the blocks.

The blocks give a dose rate up to 30 mSv/h radially but much less in the lid direction. This induces that short access to the straddle crane is possible if it gets stuck in the tunnel, as the lid is turned backwards during this transport. This is also the direction in which the blocks are piled in the tunnel.

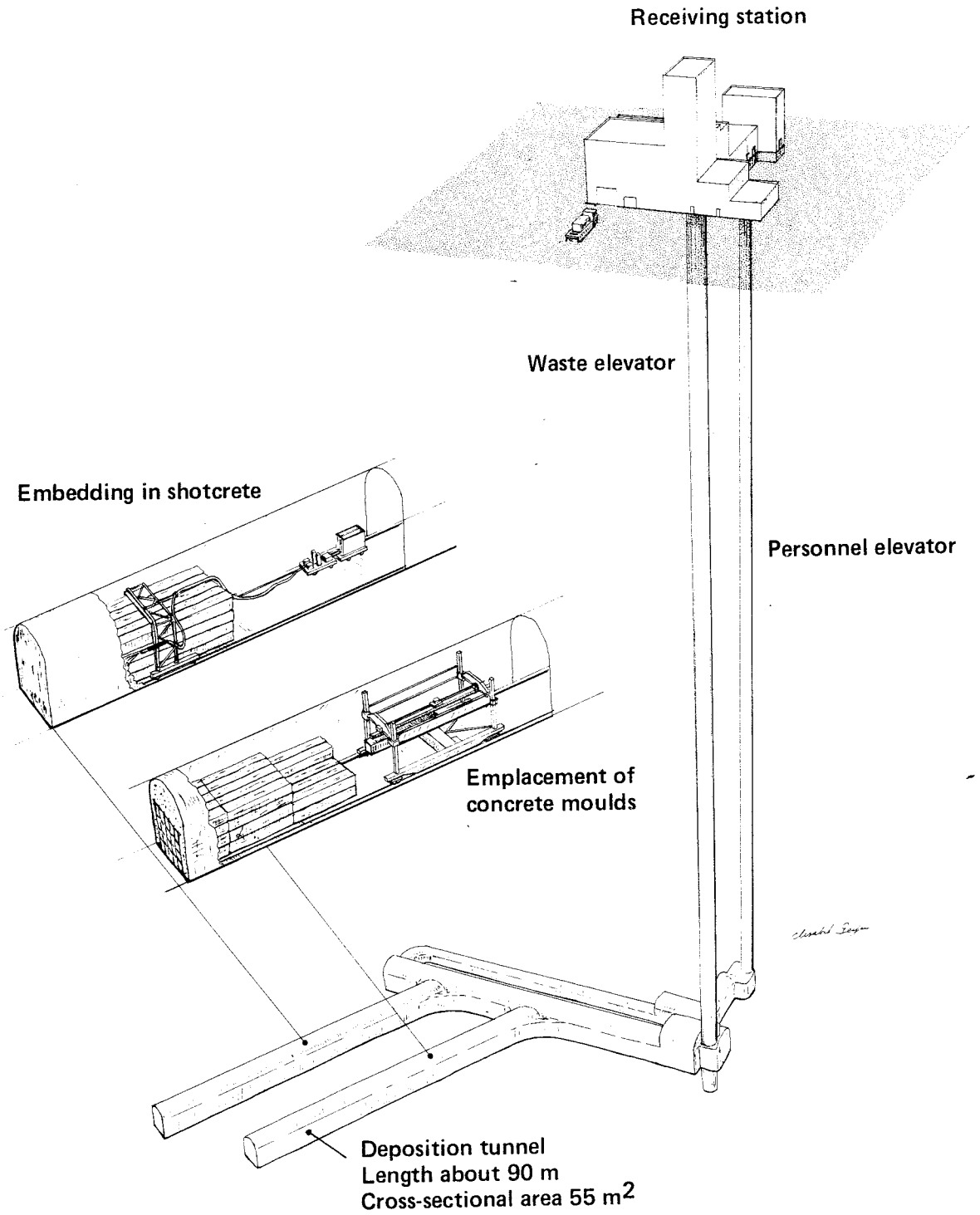


Fig 6. Final repository of fuel box containers

4 SAFETY IN THE RECEPTION AREA

The principal difference between the reception area of the encapsulation station and of the CLAB plant is the decay activity of the fuel. When arriving to CLAB, the activity has decayed one year compared to 40 years for the fuel arriving to the encapsulation station. The fuel inventory of e g Cs-134 and the crud inventory of e g Co-58 and Zn-65 have almost vanished. The activity of Cs-137 is reduced to 40%, of Kr-85 to 8% and of Co-60 to 0,6%.

From these figures it is obvious that, using CLAB experience, the reception area can be designed to meet all requirements both on normal operation and accidents. The matter to decide upon at a later stage will be which simplifications can be done in comparison to CLAB within the frame of acceptable dose to staff and to enviroment. For this reason a detailed safety analysis is not given in this report.

5 SAFETY ANALYSES OF THE WELDED CANISTER LINE

All the fuel handling in this line is remotely operated. Efficient shielding of the hot cells and the controlled ventilation air flow keeps the normal operation doses to the staff low.

5.1 NORMAL OPERATION

When lifting the fuel out of the lock pool, before placing it into the waiting canister, normally very little activity will be released to the cell atmosphere. The hold up time in the cell is kept low, only enough for the water to drain off. When the fuel is placed in the canister it is still wet. The release of crud and eventually fission products is expected to take place during and after the drying procedure. The drying should take place in the canister, to minimize the plant contamination.

If 10% of the crud release, discussed in section 2.3.1 above, is estimated to contaminate the canister charging cell, the contamination will be < 5 GBq/year (4 GBq Co-60, 0,016 GBq actinides, 0,025 GBq Cs).

The transport area under the hot cells, in which the canister is transferred between the different cells, will during normal operation be still less contaminated.

When the canister is placed in the lead casting furnace, vaccum is established and heating is started. The last water on the fuel evaporates and some activity will be released. Most of this activity will deposit in the canister. The furnace and the filter in the pipe to the vaccum pump will only get a small proportion, 5-10 GBq/year.

When handling already damaged fuel some fission products will be released during the heating. Estimating the conservative figures of sec 2.3.7 and that 0,1% of the fuel rods are damaged, the contamination would be 50-100 GBq/year of Cs-137. As the furnace will be designed for decontamination even this amount could be handled without problems.

When casting the lead crud, released from the fuel or deposited in the canister, will float on top and follow the lead layer, that is turned off before mounting the lid. The amount will be < 0,5 GBq/canister or 50 GBq/year. Most of it will follow the cuttings, a small proportion will contaminate the machining cell.

After the machining, the lid is welded to the canister. As the welding surface has been polished after the lead casting, no contamination should be expected during this operation.

When the canister is taken out after casting and welding it may be to some extent contaminated. Assuming 1/3 of the activity released to the atmosphere outside the canister has deposited on the canister surface, a canister with intact fuel would have 15 MBq of Co-60 on the outside. A canister containing 10 damaged fuel rods could additionally have 2 GBq of Cs-137. This contamination is easily reduced to acceptable levels in the decontamination position which ends the welding line.

5.1.1 Staff doses and stack release

3 persons will operate the line. They will work in shielded areas with a dose rate < 0,25 µSv/h. The maximum annual dose will thus be 1,35 man mSv. Normal maintenance of equipment including exchange of filters etc could be done within a total dose of 20 man mSv/year (200 h, 0,1 mSv/h).

The enviroment release of activity will be low, a few GBq/year. This is due to the filters in the ventilation system.

5.2 FAILURES AND ACCIDENTS IN THE WELDING LINE

Failures during the encapsulation procedure could be characterized as

- leading to activity release
- leading only to external staff dose

Both types of failure can of course be more or less probable. The release of activity will of course induce staff dose, but due to remote operation this can be kept low. Repair of equiment and loosening of stuck radiating components will give the main part of staff dose.

A detailed failure analysis is at this stage not possible. The plant is only briefly designed. Detailed studies have been done in some fields, but also there more to find long term safety parameters than to show the reliability of the equipment. The analysis in this section is based on general experience of what is possible to achieve with today's technical measures and not on a detailed analysis of a specific design. This way of working could also point out areas of interest for further investigation during the design phase.

5.2.1 Failures leading to activity release

This type of failure is tied to mishaps in fuel handling leading to either abnormal crud release or to fuel cladding damage and thus the release of fission products and eventually actinides.

Experience so far indicate that the frequency of fuel damage during transport and handling is very low. From La Hague as a matter of fact no incident at all has been reported. A number of parameters will influence the risk for fuel damage, some of which are not included in existing experience. The fuel is stored 40 years before transport to the encapsulation station. The lower decay power and the storage time are expected to be of minor importance. Higher burn up rates may increase the risk for cladding damage.

In spite of the good experience it should be assumed that a number of fuel rods are damaged every year due to minor handling mishaps.

During the transfer of the fuel from the lock pool to the canister the bunch of 9 BWR- or 2 PWR- and 4 BWR-elements is lifted. Assume that 5 times a year (out of 150) one fuel rod is damaged and part of the gap activity is released to the cell. Using table 5, the release from a PWR-rod would then be 5 GBq Kr-85, 3 GBq Cs-137, 0,2 GBq H-3 and 45 kBq I-129. If on one occasion each year is assumed that the rod is broken some aerosol activity could additionally be released. If $3 \cdot 10^{-3}$ % of the broken rod (·0,018g) is released as aerosol this would give 60 MBq Sr-90 and 16 MBq Pu-241.

The most severe accident would be if the whole bunch of fuel elements falls. The lifting height in the cell is limited. A bunch of fuel elements could turn over on the floor or it could fall down into the canister. The risk that the bunch falls through an empty canister hole and down to the bottom of the transport area can be neglected, as there will be an interlocked shutter in the hole.

A fall of a bunch of elements, that could lead to the release of activity from 10-20 rods, has a very low probability and is not included in the annual dose calculation. On this accident it must be assumed that some pellets are crushed and the activity release is increased. If conservatively a factor of 4 is applied on the above figures, the release to the cell would be 200 GBq Kr-85, 120 GBq Cs-137, 8 GBq H-3 and 2 MBq I-129.

In the lead casting furnace the canister and the fuel are heated in vaccum. The fuel decay power heats the fuel about 9°C/h, which indicates that overheating of the fuel could occur. Calculations show that radiation cooling of the fuel will restrict the maximum temperature to 420-430°C with the canister at 400°C, the consequences of the possible overheating are thus moderate. The heating of the rods leads to gas pressure increase. Although moderate this could lead to cladding failure on single rods, preferably PWR-rods as these have higher internal gas pressure than BWR-rods. Most of the activity, with the exception of krypton and tritium, will stay in the canister, furnace and vaccum filter.

A number of rods could be simultaneously damaged in case of failure in focusing the welding beam. The release would be less than from the furnace as the fuel avarage temperature is less. This accident can be hindered by appropriate design of the equipment.

When the canister is sealed, quality control is done on the weld before decontamination and placing in the transport position. In case the canister is failed in the quality control it is taken back to the machining cell where the lid is cut off and a new one is applied. The lead casting is left intact and release, other than discussed above, should not be expected.

A fire accident has a very low probability, but could of course occur. The amount of combustible material in the different cells is low. The risk of a fire that heats the fuel to temperatures leading to high activity release could be neglected. In the canister loading cell there is hardly any combustible material and in the rest of the line the fuel is contained in the canister which has a high heat capacity. The problem of a fire accident would merely be failure of the canister handling equiment. A fire accident might lead to the failure of ventilation pressure differences and to the release of minor amounts of activity to adjacent areas. Additionally a fire in the filter system might lead to the relase of part of the activity stored in the filter. The environment consequence of this is restricted to the amount of activity stored in one filter.

5.2.2 Failures leading to staff dose

Due to the remote operation the accidents described in sec 5.2.1 do not directly lead to staff activity exposure. Indirectly, during decontamination and repair work, the staff may be exposed to radiation.

Following an accident in the canister loading cell, when fuel or fragments of fuel rods are left on the floor of the cell, a remotely operated tool must be avaiable to take the fuel away. It may be put down into the canister or into the lock pool. The alternative method, to fill the cell with water for shielding, is for practical reasons probably not attractive. Using adequate tools, that can be taken into the cell through the overhead crane service area, also the unprobable event of a bunch of 9 BWR-elements fallen on the floor should be possible to handle with a staff dose < 0,1 man Sv. This includes internal dose from airborn activity during work and decontamination. Annually expected incidents are estimated to give a total dose of 0,02 man Sv. Damaged fuel rods are packed in copper cans, which are put into the canister in an empty fuel position.

In the canister transport area, below the loading-, casting-, machining- and weldingcells, failures in the transport wagon way occur. In most cases it can be manually drawn to a position where the canister can be lifted up to a shielded position by the overhead crane in one of the cells. The transport area is then accessible. A few accidents may occur when the wagon is stuck with a canister in low position. The wagon could be built with a certain amount of shielding (2-3 cm of iron) so as to reduce the dose rate in the vicinity to about 1 mSv/h. Access is then possible during shorter periods.

The most severe accidents in the transport area are

- a canister falling before lead casting
- a canister is stuck half way into e g the lead casting furnace

Both cases should have very low probability with appropriate equipment design. The first case is dealt with by flooding the area to a level necessary for shielding. Then the canister can be taken back and hoisted into the canister loading cell or even out through the lock pool. In the second case, the work must be done using temporary built up shields. Assume for example, that a foreign object during the lead casting falls outside the canister and then during the lowering of the canister squeezes the canister half way out. The wagon of the parallell line can then be used to take all other canisters away from the lead casting cell. One position may though be blocked by the stuck wagon. The furnace over the squeezed canister must be dismantled and lifted off so that the overhead crane can lift the canister or the foreign object can be taken away.

The staff dose from such an operation depends on if there is space enough for temporary shielding and for efficient work. It is also important that the furnace is designed for rapid dismantling. Also under rather good circumstances such an operation could give a dose in the order of 20-40 man mSv.

In the lead casting cell, the risk for contamination is small. Upon need, the canister can be taken away and the equipment is accessible for service at rather low dose rate.The need for remote operated decontamination of the furnaces should be investigated. Possible failures in the lead casting are firstly overfilling with lead in the canister. The furnace must be designed so that overfilled lead can be drained. Lead must not be left in the furnace in such a way that the canister transport is hindered. Should this, for some reason, still be the case, the furnace and also the adjacent one has to be dismantled and the canister lifted out through the adjacent hole.

Possible accidents in the cooling-, machining- and welding cells are of the same nature and have the same type of consequences as the ones described above.

5.2.3 Summary of release and staff dose

Due to failures and accidents activity is released from the fuel. All active gases pass the filters, iodine is assumed to have a penetration factor of 10^{-3} and other activity 0,1. On this basis the annual release to the environment from a number of incidents and the greatest anticipated release have been calculated, the result is shown in table 6.

Table 6

Environment release of activity

Element	Annual release	Greatest anticipated release
H3	10 GBq	80 GBq
Kr 85	75 GBq	200 GBq
I 129	0,05 MBq	0,2 MBq
Cs 137	30 MBq	120 MBq
Sr 90	0,01 MBq	0,6 MBq
Pu 241	0,03 MBq	0,2 MBq

In table 7 the estimated staff dose from failures and accidents in the different parts of the line has been summarized.

Table 7

Staff doses in different areas

Area	Annual dose man mSv	Greatest anticipated dose man mSv
Canister load cell	20	100
Transport area	20	40
Lead casting cell	15	30
Cooling cell	5	20
Machining cell	10	30
Welding cell	10	30
Total	80	Max = 100

6 ENCAPSULATION BY HOT ISOSTATIC PRESSING

This alternative method of encapsulation is briefly described in section 3.3 above. The procedure includes many operations similar to the ones used in the welded canister process. There are also some important differences. This safety analysis is done with reference to the welded canister process.

Basically the HIP-process includes a greater number of operations than the welding process. Also the maximum fuel temperature is higher. This should lead to higher release and higher staff doses.

6.1 NORMAL OPERATION

Drying of the fuel with warm air in the canister loading cell will imply an increased release of crud to the cell. From damaged fuel where water may have gone into the cladding gap small amounts of fission products may also contaminate the cell. The normal operation release will remain small, 10-20 GBq/year, but could be further reduced if the drying is accomplished with the fuel in the canister, e g combined with the hydrogen conditioning.

The pouring of copper powder into the canister is also an operation that might lead to release of crud activity. Also this release could be reduced if the fuel is wet during the powder filling. The crud release from powder filling with dry fuel should be of the same order as for the drying i e 10-20 GBq/year.

When the inner sealing lid is mounted and welded the canister treatment cells cannot be contaminated any more during normal operation. Activity release will occur only ot the vaccum system in the hydrogen conditioning process The thermal conditions are similar to those in the lead casting process, but the hydrogen atmosphere is reducing. Experiments show that this has only minor influence on e g iodine and cesium release at these temperatures. Conservative use of the high release rates of table 5 still restricts the iodine release from damaged fuel rods to < 10 GBq/year. After sealing the inner lid no more activity can be released.

6.1.1 Staff dose and activity release from normal operation

As the process is remote operateted the staff dose will be greater than in the welded canister process only in proportion to greater staff for handling and equipment maintenance. As the cell contamination may be somewhat higher, dose commitment for decontamination may also be higher. On the other hand, measures could be taken to keep the staff dose down.

Comparing the two processes the HIP-encapsulation would give staff doses < 30-40 man mSv/year, most of which comes from decontamination, maintenance and filter exchange operations.

The enviroment release of activity can due to ventilation filters, be kept very low. In addition to minor amounts of Kr-85 and tritium the release will be < 1 GBq/year of I-129 and < 0,1 GBq/year of Co-60 and Cs-137.

6.2 FAILURES AND ACCIDENTS IN THE HIP-LINE

As the principles of fuel handling are similar in the two alternative processes, possible failures described in section 5 may occur also in this process, with minor variations. Only differences and additional failure modes will be discussed in this section.

6.2.1 Failures leading to activity release

The handling of unshielded fuel is similar in both processes as is risk of mishaps and failures.

The welding of the inner lid is a step which could fail. Theoretically, bad focusing could induce the puncture of a number of fuel rods. As the canister contains complete fuel elements, the distance from the welding zone to the fuel rods is big. The risk could thus be neglected.

The hydrogen conditioning process is designed to minimaze the risk for combustible H_2-/O_2-mixtures. An external fire in the cell or a pipe rupture during the conditioning could in an improbable case still lead to a local explosive mixture. As the fuel is embedded in copper powder, only a few rods would be expected to puncture and the activity release would be restricted.

The important addition to release risk is the pressing operation in the Quintus-press. Under normal conditions no release will occur. Should however the filling of copper powder have failed without detection, the release could be severe. The pressing includes high temperature 550-600°C and a plastic deformation of the canister. With lack of copper powder the deformation will be extreme and both numerous fuel rods and the canister itself might fail. In table 8 is shown the activity released after such an accident and assuming 50% of the fuel rods are damaged by the pressing.

Table 8

Activity released from failed HIP-canister

Element	Release
H 3	300 GBq
Kr 85	1300 GBq
I 129	60 MBq
Cs 137	8000 GBq

Most of the activity will probably stay in the canister, but the furnace contamination could still be severe. Due to ventilation filters the environment release can be restricted.

Minor release can occur if the canister fails in the final quality control and is taken back in the line for machining and repeated treatment. If the sealing during these operations is broken, deliberately or by accident, most of the activity has already settled in the canister and mostly Kr-85 is released.

6.2.2 Failures leading to staff dose

The risks of handling failures are similar to those described in sec 5.2. The greater number of operations of course increases the number of minor mishaps and the probability of accidents. Also the quality control will show failed welds and force repeated treatment of some canisters. The consequence of this basic statement should primarily be to increase the emphasis in equipment design so as to keep staff dose low. Only in the second place an increased staff dose should be accepted.

With respect to the more heavy components in the Quintus-furnace as compared to the welding equipment longer repair times have to be assumed in case of dismantling, e g when a canister is stuck half-way out of the Quintus-furnace. For the most severe case the staff dose should thus be assume higher than for the welded canister process, say 50% up.

6.2.3 Summary of environment release and staff dose

Using the same assumptions as for the tables in section 5.2.3 the annual and the maximum anticipated release respectively are summarized in table 9.

Table 9

Failure and accident release from the HIP-line

Element	Annual release	Maximum anticipated release
H 3	12 GBq	300 GBq
Kr 85	100 GBq	1300 GBq
I 129	50 kBq	6 MBq
Cs 137	40 MBq	8 GBq
Sr 90	0,01 MBq	0,6 MBq
Pu 241	0,03 MBq	0,2 MBq

The annual staff dose could be kept within 0,1 man Sv, which is also the assumed dose from the most severe anticipated accident.

7 TRANSPORT OF CANISTERS TO THE FINAL REPOSITORY

The dose rate from the canister is shown in table 4, section 2. The various vehicles used for canister transport will be furnished with shields to keep the radiation at acceptable levels. As the encapsulation station contains a decontamination cell the canisters will be clean and no contamination is expected in the canister transport areas. The transport is shortly described in sec 3.4.

7.1 NORMAL OPERATION

From the output buffer of the encapsulation station the canister is lifted down into a wagon, which takes it into the elevator. The canister is shielded on the wagon, which is remotely operated into and out of the elevator.

Coming out of the elevator the canister is fetched by another transport wagon, which takes it to a free repository position. This wagon is more heavily shielded as it is manually operated. Normal operation thus gives low staff doses and no activity release.

7.2 FAILURES AND ACCIDENTS DURING TRANSPORT

Possible accidents could be that the canister turns over or falls during transport or that the elevator fails and the canister falls with the elevator. Finally a fire e g in a transport vehicle could occur or some part of the transport area could be flooded.

The canister is mechanically very stable. The result of a fall of the canister will normally be only local deformation, the tightness of the canister will not be violated. Only in the elevator shaft to the storage area the possibility to break the canister exists. Should the elevator with the canister fall, in spite of safety measures, the canister could break and the activity from a great number of fuel rods escape. As compared to failure in the HIP-pressing described in 6.2.2 the release would be moderate, as the fuel is cooled down in this case. The height of the elevator shaft would substatially reduce the aerosol release from the plant, mostly tritium and krypton would come out. Staff working close to the elevator especially on the lower level could get a high radiation dose if there was no automatic radiation alarm to warn them.

Failure in the transport vehicles normally gives moderate staff dose as the canister during transport is kept within a transport shield. The most severe accident will be similar to the one described in sec 5 when a canister is stuck half way into a shield. In most positions there is space enough for temporary shielding and the dose commitment should thus be smaller than assumed in sec 5.2.2.

The risk of a fire accident is low. If the transport vehicle is run on diesel fuel, this would be the greatest amount of combustible material in the vicinity of the canisters. The heat capacity of the canister with surrounding radiation shield is high. The combustion of 100 l of diesel fuel could heat the canister only 50 °C, the canister would not be damaged.

A flooding of some part of the transport area could not violate the tightness of the canister or lead to the release of activity.

8 TREATMENT OF FUEL BOXES AND BORON GLASS RODS

The principal part of the activity in fuel boxes and boron glass rods is induced activity, bound in the material. The activity is shown in table 2, sec 2. Surface crud activity is much less than on the fuel elements. If crud activity conservatively is assumed to be 10% of the crud density on the fuel cladding, this would mean 0,2 GBq on each fuel box. The risk for severe contamination of the box handling cells is thus very low.

Co-60 is the most important element in the boxes, especially with respect to shielding. The Ni-63 only emits weak β-radiation, which though may be important when handling unshielded fuel boxes.

8.1 NORMAL OPERATION

Also this handling is done with remote operation in the different cells for loading containers, pouring concrete and lid mounting and welding.

The bottom part of the box contains more activity than the rest of the box. Therefore the container is furnished with an extra radiation shield during handling and transport. This shield is dismantled only just before placing the container in the final repository.

Only the transport from the encapsulation station to the concrete repository site is done with a manually operated vehicle. This transport is done in a shielding container which is loaded by aid of an overhead crane.

The remote operation and the restricted contamination ensures low staff doses from the box treatment line. The normal operation should contribute to the total staff dose with 1-2 man mSv/year, the number of box containers also being low.

8.2 FAILURES IN THE BOX TREATMENT LINE

Possible mishaps during loading of the concrete container is that a fuel box or a boron glass rod is lost and falls on the floor. With respect to the restricted weight of these components it is rather easy to move them either into the container or back into the lock pool. Adjacent to the lock pool in the loading cell, there is a buffer pool, which could be water filled or drained. This pool can also be used after losing components on the floor. Small amounts of crud activity, < 20 MBq Co-60 could contaminate the cell atmosphere.

When pouring concrete, overfilling with concrete could occur. Should the mishap be observed at a late stage when the concrete is already hardened, manual operations might be necessary on top of the container. As the dose rate upwards is low, < 0,2 mSv/h this risk can be accepted.

During transport the container may fall or, in some cases, turn over. The reinforcement will keep the container together but the concrete may crack and increase the dose rate from the container. Local dose rates higher than 30 mSv/h call for the use of a shielded or remote operated straddle truck, which could handle the container without using the lifting eyes of the container. As the concrete container is only a transport package, it may be deposited without further treatment.

The activity release from a fallen and cracked container is expected to be < 0,1 GBq Co-60. The use of a makeshift to handle a damaged container may induce a staff dose of 5-10 man mSv. The risk for such an accident should be very low, only 20-25 containers are to be produced each year.

9 CONCLUSIONS

The estimates of activity release from different steps of the fuel treatment in the encapsulation station have generally been conservative. In spite of this the environment activity release from normal operation and from possible accidents are shown to be very low. A comparison has been made with release calculations for a 20 m high ventilation stack at Forsmark nuclear power plant. Also the most severe accident release discussed in this report would induce a dose to most exposed person at 0,5 km of about 3 µSv.

The primary reason for the comparatively low activity release is of course the activity decay of the fuel during the 40 year storage.

In spite of the rather complicated fuel treatment also the staff dose can be kept rather low, within 150 man mSv/year. In comparison to earlier described process, when the fuel elements were dismantled, the encapsulation is now simplified by storing the complete fuel elements. On the other hand the HIP-process is somewhat more elaborate than the lead-casting and welding process.

Using remote operation, adequate equipment and a thorough ventilation system design keeps the staff dose low. The figure given above, 150 man mSv/year, is of course very much dependent on the detailed design of the equipment and the plant lay-out. There is no doubt however that the plant can be built and run safely.

REFERENCES

1. Edlund, O: Calculation of the burnup dependant parameters of the fuel in a PWR and BWR using the computer code ORIGEN 2. KBS 83-

2. Jonsson, L: CLAB-Aktivitetsinventarier i system och komponenter. ASEA-ATOM PM KPC 82-27.

3. Lundgren, K: KBS3-Kopparkapsel för direktdeponering - Strålningsnivåer. ASEA-ATOM PM KPC 83-19.

3a. Carlquist, L: KBS-Slutdeponering av komponenter innehållande neutroninducerad aktivitet - Strålskärmsberäkningar. ASEA-ATOM PM KPC 83-20.

4. Forsythe, R: Studsvik Energiteknik: Private communication.

5. Roy, R, Crudkarakterisering: Studsvik Energiteknik ET 81-113.

6. Liljenzin, J-O, Chalmers Institute of technology: Private communication.

7. Lönnerberg, Larker, Ageskog: Encapsulation of spent nuclear fuel for final disposal. KBS TR 83- preliminary edition.

8. Larker, Tegman, Burström: HIPOW-inneslutning av använt kärnbränsle i koppar. ASEA TR KX3 83-002.

LIST OF KBS's TECHNICAL REPORTS

1977-78

TR 121 KBS Technical Reports 1 - 120.
Summaries. Stockholm, May 1979.

1979

TR 79-28 The KBS Annual Report 1979.
KBS Technical Reports 79-01--79-27.
Summaries. Stockholm, March 1980.

1980

TR 80-26 The KBS Annual Report 1980.
KBS Technical Reports 80-01--80-25.
Summaries. Stockholm, March 1981.

1981

TR 81-17 The KBS Annual Report 1981.
KBS Technical Reports 81-01--81-16
Summaries. Stockholm, April 1982.

1983

TR 83-01 Radionuclide transport in a single fissure
A laboratory study
Trygve E Eriksen
Department of Nuclear Chemistry
The Royal Institute of Technology
Stockholm, Sweden 1983-01-19

TR 83-02 The possible effects of alfa and beta radiolysis on the matrix dissolution of spent nuclear fuel
I Grenthe
I Puigdomènech
J Bruno
Department of Inorganic Chemistry
Royal Institute of Technology
Stockholm, Sweden January 1983

TR 83-03 Smectite alteration
Proceedings of a colloquium at State University of New York at Buffalo, May 26-27, 1982
Compiled by Duwayne M Anderson
State University of New York at Buffalo
February 15, 1983

TR 83-04 Stability of bentonite gels in crystalline rock - Physical aspects
Roland Pusch
Division Soil Mechanics, University of Luleå
Luleå, Sweden, 1983-02-20

TR 83-05 Studies of pitting corrosion on archeological bronzes
Åke Bresle
Jozef Saers
Birgit Arrhenius
Archeological Research Laboratory
University of Stockholm
Stockholm, Sweden 1983-02-10

TR 83-06 Investigation of the stress corrosion cracking of pure copper
L A Benjamin
D Hardie
R N Parkins
University of Newcastle upon Tyne
Department of Metallurgy and Engineering Materials
Newcastle upon Tyne, Great Britain, April 1983

TR 83-07 Sorption of radionuclides on geologic media - A literature survey. I: Fission Products
K Andersson
B Allard
Department of Nuclear Chemistry
Chalmers University of Technology
Göteborg, Sweden 1983-01-31

TR 83-08 Formation and properties of actinide colloids
U Olofsson
B Allard
M Bengtsson
B Torstenfelt
K Andersson
Department of Nuclear Chemistry
Chalmers University of Technology
Göteborg, Sweden 1983-01-30

TR 83-09 Complexes of actinides with naturally occurring organic substances - Literature survey
U Olofsson
B Allard
Department of Nucluear Chemistry
Chalmers University of Technology
Göteborg, Sweden 1983-02-15

TR 83-10 Radiolysis in nature:
Evidence from the Oklo natural reactors
David B Curtis
Alexander J Gancarz
New Mexico, USA February 1983

TR 83-11 Description of recipient areas related to final storage of unreprocessed spent nuclear fuel
Björn Sundblad
Ulla Bergström
Studsvik Energiteknik AB
Nyköping, Sweden 1983-02-07

TR 83-12 Calculation of activity content and related properties in PWR and BWR fuel using ORIGEN 2
Ove Edlund
Studsvik Energiteknik AB
Nyköping, Sweden 1983-03-07

TR 83-13 Sorption and diffusion studies of Cs and I in concrete
K Andersson
B Torstenfelt
B Allard
Department of Nuclear Chemistry
Chalmers University of Technology
Göteborg, Sweden 1983-01-15

TR 83-14 The complexation of Eu(III) by fulvic acid
J A Marinsky
State University of New York at Buffalo, Buffalo,NY
1983-03-31

TR 83-15 Diffusion measurements in crystalline rocks
Kristina Skagius
Ivars Neretnieks
Royal Institute of Technology
Stockholm, Sweden 1983-03-11

TR 83-16 Stability of deep-sited smectite minerals in crystalline rock - chemical aspects
Roland Pusch
Division of Soil Mechanics, University of Luleå
1983-03-30

TR 83-17 Analysis of groundwater from deep boreholes in
Gideå Sif Laurent
Swedish Environmental Research Institute
Stockholm, Sweden 1983-03-09

TR 83-18 Migration experiments in Studsvik
O Landström
Studsvik Energiteknik AB
C-E Klockars
O Persson
E-L Tullborg
S Å Larson
Swedish Geological
K Andersson
B Allard
B Torstenfelt
Chalmers University of Technology
1983-01-31

TR 83-19 Analysis of groundwater from deep boreholes in Fjällveden
Sif Laurent
Swedish Environmental Research Institute
Stockholm, Sweden 1983-03-29

TR 83-20 Encapsulation and handling of spent nuclear fuel for final disposal
1 Welded copper canisters
2 Pressed copper canisters (HIPOW)
3 BWR Channels in Concrete
B Lönnerberg, ASEA-ATOM
H Larker, ASEA
L Ageskog, VBB
May 1983

TR 83-21 An analysis of the conditions of gas migration from a low-level radioactive waste repository
C Braester
Israel Institute of Technology, Haifa, Israel
R Thunvik
Royal Institute of Technology
November 1982

TR 83-22 Calculated temperature field in and around a repository for spent nuclear fuel
Taivo Tarandi
VBB
Stockholm, Sweden April 1983

TR 83-23

TR 83-24 Corrosion resistance of a copper canister for spent nuclear fuel
The Swedish Corrosion Research Institute and its reference group
Stockholm, Sweden April 1983

TR 83-25 Feasibility study of EB welding of spent nuclear fuel canisters
A Sanderson
T F Szluha
J Turner
Welding Institute
Cambridge, United Kingdom April 1983

TR 83-26 The KBS UO_2 leaching program
Summary Report 1983-02-01
Ronald Forsyth
Studsvik Energiteknik AB
Nyköping, Sweden February 1983

TR 83-27 Radiation effects on the chemical environment in a radioactive waste repository
Trygve Eriksen
Royal Institute of Technology
Stockholm, Sweden April 1983