MANAGEMENT OF DECOMMISSIONING WASTES: THE MANAGEMENT OF HIGH ACTIVE WASTE AND THE RECYCLING OF LOW ACTIVE METALS AND CONCRETE

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ABSTRACT

The dismantling of a nuclear facility creates a great variety of materials that have to be removed from the site. The materials vary by their physical characteristics (concrete, liquids, carbon steel, plastics,...), radiochemical characteristics (activation products, contamination from fuel leakages, isotope vector..) and geometrical form (tubes, valves, tanks,...). For the management of these materials, the dismantler has the choice between several routes, each with their own and sometimes overlapping requirements. These routes are nuclear waste, industrial waste, recycling for reuse in the nuclear industry or recycling in the non nuclear industry (free release).
This great variety of materials, the different routes with their overlapping requirements and the used decontamination techniques for decategorisation of the materials, lead to a complex management of these materials.

SCK•CEN started the dismantling of the BR3 PWR research reactor in 1989 and has therefore yet several years of experience with this complex materials management.
High level radioactive waste disposal and recycling of decontaminated materials are important issues for the public and hence for the dismantler. Therefore this paper will highlight SCK•CEN's experiences in three areas of this materials management: the management of high and medium active waste, the recycling of radioactive materials in the nuclear sector and the free release of materials.

The first part of the paper describes how SCK•CEN manages the high and medium active waste from the dismantling of the reactor internals and reactor pressure vessel. Due to the high dose rate and the associated radiation exposure risks, all the manipulations are carried out under water and remotely. Also special attention is drawn to the conditioning of this waste, namely the conditioning of the cut pieces and of the swarfs which must follow severe waste packaging requirements. The swarfs are the result of the cutting process itself (secondary waste) and demand a specific conditioning basket and method.

The second and third part of the paper deal with the recycling of low active metals and concrete or their free release. First, the different possible routes and the associated decontamination techniques are described. The free release of dismantled pieces requires to follow a strict procedure comprising historical data, identification throughout dismantling, free release procedures and free release measurements. Experience on the management of the materials stream will be given along with special attention to the required low level measurement. Finally, the paper describes the R&D project that SCK•CEN started together with BBRI (Belgian Building Research Institute CSTC/WTCB) on the recycling of cement grout or concrete. This "radioactive' grout or concrete can be used for conditioning of radioactive waste e.g. conditioning of metallic waste or conditioning of sludges or ion exchangers.

INTRODUCTION

BR3 is a small 10 MWe PWR shutdown in 1987 after 25 years of operation. It was selected as one of the four pilot projects of the EU for its R&D programme on Decommissioning of nuclear installations. The decommissioning project started in 1989. In 1991, a Full System Decontamination of the primary loop reduced the dose rate in the vicinity of the primary loop by a factor 10. The same year, a first high active internal, the 5.4 t thermal shield was dismantled underwater by 3 different dismantling techniques, the EDM cutting, the milling cutter and the plasma arc torch. Mechanical cutting, essentially milling cutter and band saw, were selected for the further dismantling of the two sets of internals; the original Westinghouse internals ("33 years decay") and the Vulcain internals ("7 years decay"). This allowed to compare deferred dismantling with immediate dismantling.

The next important step is the cutting of the 28 t Reactor Pressure Vessel. All the preparatory work is finished and the real cutting operations are foreseen from mid 99. Dismantling of some contaminated circuits was also performed using mostly hands on cutting techniques. Minimizing the amount of radioactive waste and free release of the dismantled materials have always been our main objectives. Recycling of slightly radioactive metallic materials could also be performed thanks to an agreement with a nuclear foundry. For concrete, an R&D programme has been started to recycle radioactive concrete in the radioactive waste conditioning sector. Progress was also made on the establishment of free release limits and procedures and on the development of decontamination techniques for metals and concrete.

WASTE MANAGEMENT OF HIGH AND MEDIUM ACTIVATED WASTE [1]

The high and medium activated metal waste and the secondary waste, produced by the cutting of the internals (see fig. 1), were transferred from the BR3 site to the Belgoprocess site where they were conditioned. The evacuation and the conditioning were performed in respect with the Belgian waste conditioning requirements and the specific internal infrastructure requirements.

It is the National RadWaste Authority (ONDRAF/NIRAS) who sets up the different acceptance criteria on waste types and waste packages. Concerning the solid waste (big pieces), there are three important groups of waste and the distinction between these groups is based on the contact dose rate. These three groups are: Low Level solid Waste (LLW) with a contact dose rate < 2 mSv/h, Medium Level solid Waste (MLW) with a contact dose rate between 2 mSv/h and 0.2 Sv/h and High Level solid Waste (HLW) with a contact dose rate > 0.2 Sv/h. For HLW, MLW and LLW, the reference conditioning scenario in Belgium is direct cementation of the radioactive waste in 400 l drums. This is performed in the BELGOPROCESS installations (Cilva for the LLW and hot cells for the MLW and HLW).

Up to now, 12.7 tons were transported as HLW or MLW to BELGOPROCESS, the Belgian waste conditioner, conditioned there and placed in an interim storage facility. The remaining part of 11 t was LLW and was conditioned also at Belgoprocess in the CILVA installation. Table I gives the weight for the different types of waste for the three main internals.

Table I: Amount of waste produced during the dismantling of the internals

	HLW (t)	MLW (t)	LLW (t)	Total (t)
Thermal Shield	5.5	0	0	5.5
Vulcain internals	2.8	1.0	7.4	11.2
Westinghouse internals	3.4		3.3	6.7
Total	12.7		10.7	23.4

Remark: The difference between HLW and MLW was not made during the evacuation of the Westinghouse internals for reasons of waste volume optimisation

For the conditioning and transport, the waste has to fulfil different requirements from NIRAS/ONDRAF, the legal Safety authority and the waste producer himself.

For NIRAS/ONDRAF and Belgoprocess, the main requirements are:

- The waste must be put in a basket which fits in a 400 l-drum.
- The weight of each unconditioned drum may not exceed 800 kg. (to respect the 1500 kg limit for the conditioned drum).
- There must be a free space of three centimetres between the waste pieces and the inner side of the drum and 5 cm between the waste pieces and the top/bottom of the drum.

For the Safety Authority:

- The transported pieces may not move during the transport.
- The national transport limits may not be exceeded (the contact dose rate of the container must be less than 2 mSv/h and the dose rate at 1 metre of the transport vehicle must be less than 0.1 mSv/h.
- The worker's dose uptake has to be minimised (ALARA principle).

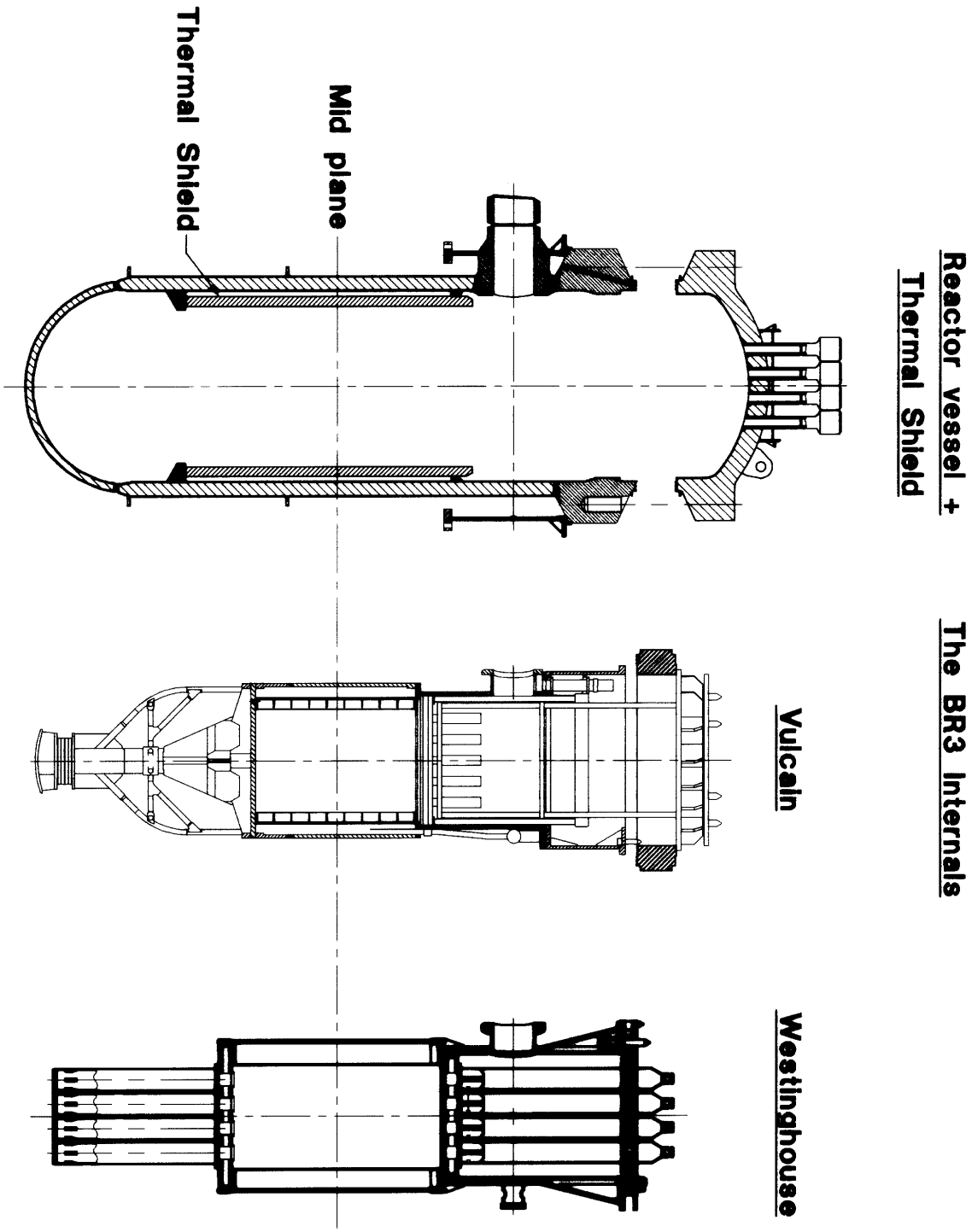

Fig. 1: A view of the different BR3 internals, cut by the BR3 dismantling team

For the waste producer:

- The waste manipulations must be carried out under shielding (in this case water) for protecting the workers against the high radiation.
- The filling must be optimised in order to produce as less number of drums as possible (economic optimisation).

– The cut pieces had to fit in a small internal transfer container. Indeed, the only location to fill the transport container was in the spent fuel pool. The cut pieces had to be transported from the reactor pool, where the dismantling actually took place, to the spent fuel pool. A 13 tons lead shielded transfer container, which had to pass the small personnel air lock, had been designed and fabricated to transport the waste to the spent fuel pool. Due to this transfer container (inner side dimensions: 800 mm x 500 mm x 500 mm), the dimensions of the cut pieces were limited.

To fulfil all these requirements, the export system consisted of a carrier structure and two baskets, which fit over the carrier structure. Two baskets and a carrier structure formed a waste package (see fig. 2). A special lifting device was designed to manipulate the waste package for loading and unloading the transport container at respectively BR3 and the waste conditioner (see fig. 3) The whole system was of course designed for working under water, but also in hot cells.

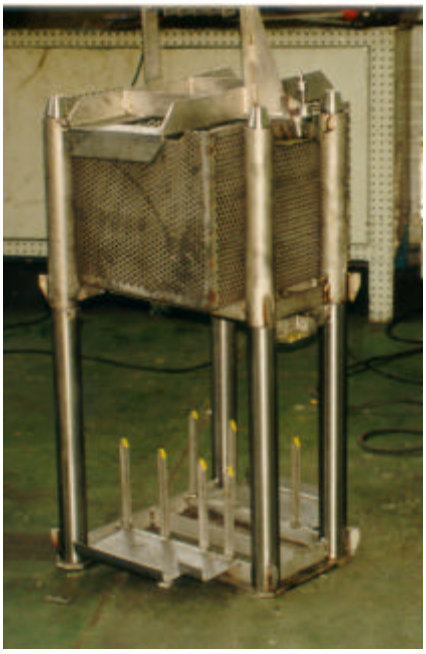

Fig. 2: The waste package for the transportation and conditioning of high activated nuclear waste. The upper rack is from a basket type (for small pieces), the lower rack is for the cut pieces.

For the transportation of the waste, there were two types of baskets. The first type was the open basket (see fig. 2) and was used for the transportation of big pieces. For the evacuation of small pieces like bolts and thermocouples, a second type (see fig. 2) in the form of a closed basket was used. This basket had the same base structure as the open basket but was totally surrounded with a fine metal plate perforated with 10 mm holes. This to keep the small pieces in the basket and to let drip out the basket.

A totally different evacuation system was used for the swarfs. During the cutting of the thermal shield, the cutting equipment was equipped with a separate filtration system. This filtration system had a metal strainer (mesh 40µ) as a prefilter. During the whole cutting campaign of the internals, the team used this filtration system for the collection of the swarfs. The strainer, which could easily be taken out of the filtration system, was than covered with a plastic cover plate to avoid the release of swarfs during the manipulations under water. Fifteen strainers at once could be transported to Belgoprocess with a special evacuation structure (see fig. 3) With this structure, self made filter bottles, containing high or medium radiating filters, could also be transported to Belgoprocess.

The transports were carried out with a special transport container of TRANSNUBEL containing each time one waste package (support structure and two baskets). Totally, 40 transports were carried out for the evacuation of 12.7 tons high and medium activated waste.

Fig. 3: The special structure for the transportation of the strainers filled with swarfs. The picture shows also a filter cask for the transportation of high radiating fine filters.

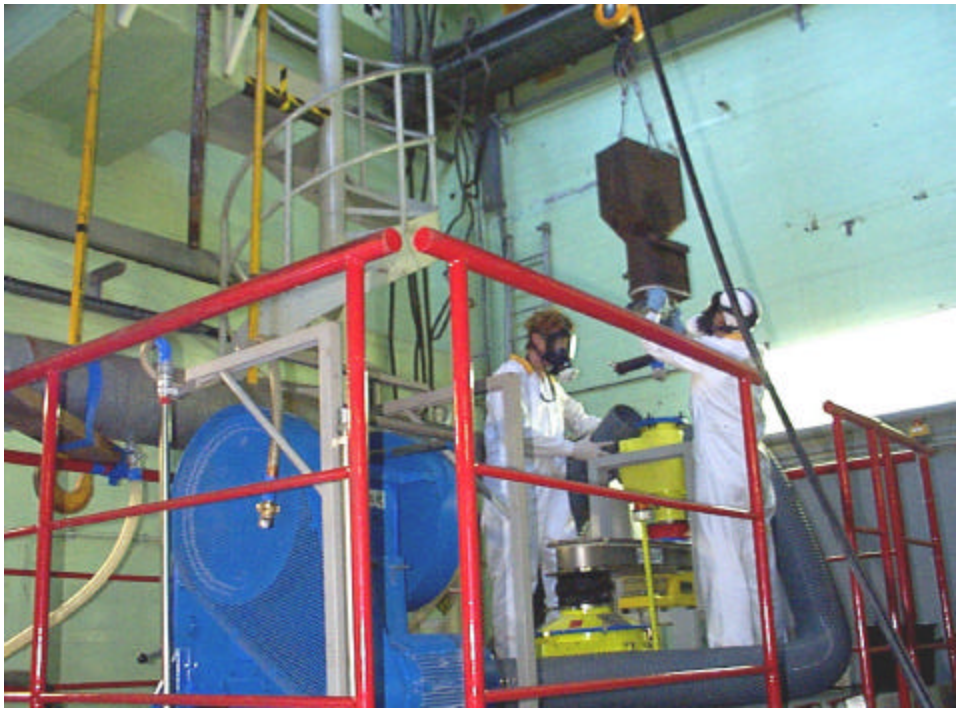

Fig. 4: The KEMA mobile installation for the crushing and sieving of the radioact ive concrete

Some lessons learned:

- Two types of internals were cut and evacuated. This allowed to compare deferred dismantling with immediate dismantling. No significant radiological, technical or economical profit was gained for the dismantling of the old internals (33 years decay) because due to the still high dose rate of 2 to 3Sv/h at mid plane, remote underwater cutting is still required. But for transportation, the "young" internals (7 years decay), mainly the core baffle ones gave us some trouble due to the high dose rate of 50 to 70 Sv/h and to the absence of self-shielding effect.
- For the massive solid wastes, the mean capacity was only 350 kg/drum (apparent specific gravity < 1 !) with variations between 88 (core baffle) and 672 (thermal shield) in function of the geometry of the pieces. In fact, for the conditioning of 12.7 tons, we produced 16 m^3 of conditioned HLW waste.
- The evacuation through the personnel lock is rather complicated so that a simpler evacuation way was realized by cutting openings in the bioshields of the operating deck and of the plant container.

Dismantling of the reactor pressure vessel

During the dismantling of the internals, the difference between high and medium activated waste was not made. This was done due the relatively small amount of medium active waste. On the contrary, the dismantling of the reactor pressure vessel leads to the production of a high proportion of medium active waste.This means that it is economically worthwhile to transport the medium waste separately. Even if this waste has also to be put in 400 l drums, the main advantage is that the transport is done with a special shield truck which can contain, depending on the total activity, seven or fourteen 400l waste drums. Unnecessary to say that in this way the transportation costs will decrease a lot.
For the RPV cut pieces, it is foreseen to carry out 1 or 2 medium active waste transports (for a total of 12 drums) and 9 high active waste transports (for a total of 9 waste packages).
Based on the lessons learned during the manipulation of the waste from the internals, the waste evacuation system has undergone small changes.
First of all, the whole system has been made lighter in order to reduce the "dead" weight of the baskets. This allows the team to transport more waste per waste package. This was done by an optimisation of the dimensions of the material of the waste racks and by changing the support structure by a bolting system in the upper rack.
Also the old filtration system with the strainer has been replaced by a cyclone filtration system. This filtration system allows the dismantling team to put the collected swarfs into a bigger basket than the old strainer. Again there is a gain in the transported weight and volume of the swarfs per transport.

RECYCLING OF RADIOACTIVE MATERIALS IN THE NUCLEAR SECTOR [2]

Recycling of metallic radioactive materials

Low level radioactive materials can be recycled in the nuclear world. The melted materials can be used for the fabrication of shield blocks or for the fabrication of radioactive waste containers. SCK•CEN has an agreement with GTS-Duratek in the USA; the recycled materials are used as shielding for the DOE facilities. The materials must respect composition and radiochemical criteria. The secondary wastes are conditioned and disposed off by Duratek. Up to now, we have sent, in agreement with all the competent authorities, 26 t of mild and stainless steel arising from the dismantling of very low contaminated or activated pieces.

Recycling of concrete radioactive materials

The dismantling of the activated bioshield around the Reactor Pressure Vessel and in the refuelling pool will lead to the production of about 650 t of slightly activated concrete. The activation products present are mainly ^{133}Ba, $^{152,154}Eu$ and ^{60}Co with activation levels lower than 100 Bq/g. A R&D programme has been started with a specialized center in the building industry (CSTC/WTCB, Scientific Center for the Building Industry) to study the possibility to reuse the concrete as raw materials for the conditioning of radioactive wastes. Crushing and sieving tests have been performed first on real but inactive heavy barytine concrete from the BR3 reactor. The fine fraction and the aggregates were separated. Tests are underway to optimize the composition of mortar and concrete fabricated using new cement and recycled concrete either as aggregate or sand. The quality of the recycled mortar or concrete must follow strict specifications among which compression strength and the workability are the most important. The first orientative results are very encouraging and a demonstration with active concrete has been performed in October 1999. The crushing and the sieving were done in a mobile pilot unit of KEMA (see fig. 4). The objective is to demonstrate that recycled heavy concrete can be re-used after adequate treatment as sand or aggregates for conditioning of heterogeneous or homogeneous radioactive wastes. The next step now is to perform a qualification programme in collaboration with the waste conditioner and the responsible authority (ONDRAF/NIRAS).

FREE RELEASE OF RADIOACTIVE MATERIALS [2][3]

The steadily increase of the conditioning and disposal costs as well as environmental concern and public perception are pushing the nuclear sector to decrease the amount of radioactive waste and hence is a strong incentive to the development of thorough decontamination processes and procedures for the free release of obsolete radioactive materials and their reuse in the industrial sector or their evacuation as industrial waste. The free release of radioactive materials requires a combination of factors to be successful:

- Procedures and well defined free release criteria: a consensus is not yet achieved on international level and generally a case by case management is still applied. IAEA, EU, OECD are progressively converging towards some harmonisation. In our case, procedures and limits are being set by the Health Physics department under supervision of the Competent Authority. This procedure is still a "case by case" practice and is applied currently for the free release of materials from the BR3 dismantling.
- A strict follow-up of the dismantled materials comprising origin of the materials, treatment performed, characterization results..
- The traceability of the materials must be guaranteed at each step: this can only be achieved with a strong Quality Assurance programme.

Free release processes

Several processes can be used to free release materials

- Melting in a nuclear foundry
- Free release without specific treatment
- Manual decontamination: simple washing techniques
- Decontamination with a wet abrasive process
- Decontamination with a hard chemical process
- Decontamination of concrete by scabbling and shaving.

The characterization of materials to be free released is still a difficult topic. Materials subject to free release without melting can be subdivided into:

- Materials of simple geometry for which a 100 % surface measurement is possible using hand held β monitors. For these materials, surface specific free release values are established and the procedures are well known. The values used are 0.4 Bq/cm^2 $\beta\gamma$ and 0.04 Bq/cm^2 for α.
- Homogeneous materials such as concrete rubble for which only volumic or mass measurement is possible. For these materials, mass specific guide lines are generally followed and measurement procedures are available (e.g. γ spectrometry of the whole amount in a 200 l drum or statistical sampling after homogenization). There are for the moment no fixed values for the free release of such bulk materials; the health physics consider this still on a case by case basis. Their decision depends not only on the measured level but also on the origin of the piece, its history and its final destination (e.g. recycling as scrap materials or disposal as industrial waste).
- For materials of complex geometry and or heterogeneous (pipes internally contaminated, pumps, valves.), the question is: how to prove that the activity level is lower than the current free release guidelines? Presently a procedure based on a double measurement method is worked out.

Main equipments and methods that can be used to measure materials at low level

We can use:

- Hand held monitors for direct surface measurements
- For volumetric measurements
  - Spectroscopy HPGe detectors: Q2 Canberra System-220 l waste barrels
  - Spectroscopy NaI(Tl) detectors: adapted Q2
  - Versatile spectrometry with modular NaI(Tl) or HPGe detectors: Nanospecs, Isocs
  - Gross gamma counting with scintillation detectors.

The procedures followed are:

- Hand held monitors for easy to measure materials; 100% of the surface measured twice at a max 3months interval for materials submitted to a decontamination treatment (sweating effect).

- For homogeneous materials, we actually use the Q2 spectrometer for measurements of 200 l drums.
- For heterogeneous materials, we have two possibilities:
  - The materials are sent to a nuclear foundry which allows a further decontamination and a reliable measurement thanks to the homogenization.
  - We combine two measurements techniques:
    - A gross gamma counting with scintillation detectors for measurements of individual pieces or of small batches (1/10 of a 200 l drum).
    - A Q2 spectrometer for the determination of the specific activity per individual gamma nuclide.

Selection of a gross gamma counting system

This system must allow the measurement of small batches of material or of individual pieces. These batches are produced by the various dismantling and decontamination processes and the producer wants to rapidly know if these materials can be free released.

The constraints imposed by the materials producer were the following:

- The measuring system must be installed in the vicinity of the decontamination installations to allow immediate measurement and to allow the producer to take immediate corrective actions. The producer, in case of negative result (measurement above the clearance limits), can take a decision on the new destination route for this material. He can decide that this material can be re-routed e.g. to a second chemical decontamination, to the melting and reuse route or to the melting and free release route. This decision will be taken in function of the residual contamination measured, the geometry and the type of materials. This decision takes also into account the economic impact of a second treatment on the overall free release cost.
- The measuring system will be installed in a separate room between the decontamination room and the outlet of the controlled area. The room is protected by a shielding wall. The actual background radiation in the room is now less than 1 µSv/h. The BG will probably increase due to the increase of the activity in the decontamination room. A doubling of the background is expected. The measuring system should then be insensitive to this BG increase.
- The detection limit should be low enough to comply with the guiding values. For a mass of 20 kg and a guiding value of e.g. 0.3 Bq/g for ^{60}Co, this corresponds to an MDA (Minimum Detectable Activity) of 6000 Bq.
- Most of the cut pieces have a length between 60 and 80 cm and weight less than 60 kg.
- The measurement time should be low enough to allow a sufficient throughput. A value of 1 min for the measurement has been considered as a target value which corresponds to a throughput of min 300 to 400 kg/h taking into account the handling of the pieces.
- The mass of the materials must be known so the pieces must be weighted and the results expressed in Bq/g.
- It is advantageous to have a system which determines the ^{60}Co/^{137}Cs ratio to use the correct efficiency.

The different solutions proposed were:

- 4 π plastic scintillation chambers (closed chamber with 6 detectors, lead shielding, automatic or manual door)
- tunnel system with plastic detectors with or without the CCM technology (Cobalt Coincidence Method) with an open chamber with 4 detectors and without lead shielding needed in the case of the CCM technology.
- 4 π plastic scintillation chambers with CCM technology: closed chamber with 6 detectors, lead shielding, automatic or manual door.

The 4 π plastic scintillation system with the CCM technology and with lead shielding was finally selected. The limit of detection claimed are far below the guiding values for free release; values as low as 75 to 110 Bq for ^{137}Cs and ^{60}Co are announced. These values will have to be controlled for real cases with shielding effect, geometry effect.... The tunnel system has been rejected to avoid any influence from a variable background. The CCM technology without shielding has been rejected because we need also to measure pieces mainly contaminated with ^{137}Cs. Moreover this installation will also be used for characterization of materials from other sources such as hot cells, glove boxes... so that the nuclides inventory could be completely different from the reactor inventory where Co dominates. The CCM technology has been selected together with the gross gamma counting because it can facilitate the interpretation of the results when the nuclide vector is simple e.g. dominated by ^{60}Co or by ^{137}Cs.

Free release by melting

Some dismantled materials are either very low contaminated, very difficult to measure or not homogeneously contaminated. For these materials, it can be advantageous to sent them to a nuclear foundry. Melting offers several advantages:

- It decontaminates the metals by volatilization of some nuclides (e.g. ^{137}Cs) or by transfer to the slag (e.g. heavy nuclides such as alpha emitters).
- It allows an accurate determination of the radionuclides content thanks to the homogeinity of the metal melt.
- The amount of secondary waste (dust, slag) is rather low.

This practice has already been used in Belgium for dismantled waste. SCK•CEN is actually preparing a transport of very low radioactive materials comprising secondary reheaters with copper tubes, a carbon steel massive plinth and a variety of CS and SS small pieces stored in 200 and 400 l drums. About 18 t with an average activity of 0.26 Bq/g of ^{137}Cs and 0.15 Bq/g of ^{60}Co are considered for the first transport.

Free release of materials without specific treatment

A certain amount of materials can be free released immediately either on the basis of its origin (e.g. tertiary systems) eventually combined by radiochemical measurements (e.g. secondary equipment considered as suspect material). Up to now, 94 t could so be unconditionnally free released and sent for recycling to the metal scrap industry (50 t) or to the construction industry for the concrete (44 t).
Recently, we have free released (on the radiological basis) about 3 t (24.4 m^3) of asbestos containing insulation material. This waste will be conditioned and disposed as hazardous materials in an approved disposal site.
It must be noted that during the dismantling of the secondary circuit, some unexpected contamination has been discovered. The main contaminant in that case is always ^{137}Cs which seems to be preferentially fixed at some particular locations. In any case, for each batch of materials, a strict documentation is required and the free release must be approved by the Health Physics even for materials situated outside the controlled area.

Free release of materials after manual decontamination by washing

Materials which are slightly contaminated on the external surface by some contaminated dust can be free released by simple decontamination by washing manually either before dismantling or after size reduction. The washing of the dismantled pieces is done in a reservoir using some detergents. The decontaminated materials are then free released either by direct surface measurement of 100% of the surface (double measurement; the second one is performed after 3 months) or by mass measurement using the Canberra Q2 200 l drum gamma spectrometer at the condition that the material can be considered as homogeneous.
Up to now, 60 t have been so free released: 15 t of metals sent to the scrap industry and 45 t of concrete sent to the construction industry.

Free release of materials after decontamination in Wet Abrasive Decontamination unit

Painted, rusted or stainless steel pieces which are contaminated at a higher level e.g. up to 1000 Bq/cm^2 and which can not be cleaned by simple washing can be treated by abrasion to remove the contaminated layer. This technique is however limited to pieces of simple geometry because all the treated surface must be accessible for the abrasive jet. A wet abrasive unit ZOE has been built at BR3 and allows to treat pieces up to 3t and 3m long. The abrasives and the water are continuously recycled to minimize the amount of secondary waste.
Up to now, 21 t of materials were treated in the ZOE unit: 10 t were immediately free released, 6.3 t are still slightly activated (the decontamination was performed to reduce the dose rate and to eliminate the loose contamination), a massive plinth of 2.8 t was partially decontaminated to remove most of the alpha contamination present and allow its free release after melting. In general about 10 to 20% of the materials could not be free released either due to difficult to remove contamination or to difficult to measure locations. The materials not free released are normally sufficiently low to be sent to a nuclear foundry for free release by melting.
It is interesting to note that contaminated lead bricks could also be easily decontaminated by wet abrasion. Lead is a special waste which is quite difficult to evacuate as radioactive waste.

Free release of materials after decontamination in a Chemical decontamination unit [4][5]

Carbon steel or stainless steel pieces heavily contaminated e.g. up to 20,000 Bq/cm^2 $\beta\gamma$ can be treated by thorough chemical decontamination processes. For stainless steel, a process called MEDOC (MEtal

Decontamination by Oxidation with Cerium) has been developed at SCK•CEN and an industrial installation has recently been built. The process is based on the strong oxidation potential of Ce^{4+} in sulfuric acid medium at 80°C. The oxide layer (the so-called crud layer) and the base metal as well are removed. Pilot tests with real active pieces have shown that the removal of 10 to 20 µm allows to completely remove the contamination layer even for strong contaminated pieces from the primary loop. During the process, the Ce^{4+} is reduced to Ce^{3+}; in order to recover the process efficiency it is necessary to regenerate the Ce^{3+} to Ce^{4+}. In our process, this is realized by reaction with ozone gas in a special gas-liquid contactor (static mixer). The installation comprises also a rinsing unit with ultrasonics and an effluent storage tank. Up to now 4 t have been treated. For most of the pieces, the procedure followed will be a combination of the gross gamma counting piece by piece or batches (typically 20 kg) followed by a gamma spectrometry with the Q2 system on a 250 kg batch.

Free release of concrete

In BR3, 28 anti-missile heavy concrete slabs were installed in the refuelling pool above the reactor pressure vessel. The characterization studies have shown that all the slabs were contaminated and that some were activated. Decontamination of 22 slabs representing 247 t was performed using mainly scabblers, shavers and jack hammer. After treatment, 205 t could be unconditionally free released and sent for recycling in the construction industry and 42 t still slightly activated are kept for further conditioning. Sludges from cutting operations of potentially contaminated or suspect concrete blocks by cable sawing or by a diamond saw were free released by measurement of the dried sludges with the Q2 system.

CONCLUSIONS

The management of the waste from dismantling operations does not fundamentally differ from the management for operating plants. However, due to the strong importance of the waste cost in the overall decommissioning cost, the incentive is put on the minimization of the volumes of primary and secondary radioactive waste and on the development of alternative evacuation routes for dismantled materials. Recycling in the nuclear sector or free release for recycling in the industrial sector are therefore developed and encouraged for economic and also for ethic reasons. This choice however implies the set-up of a strong QA programme to ensure the traceability of the materials and pushes also the industry to develop more cost-effective decontamination techniques.

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