Workshop on Radiological Characterisation for Decommissioning
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Radiological Characterisation Experience with Magnox Reactors

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Abstract

At the end of generation, power reactors will be decommissioned. Whether decommissioning is prompt or deferred, knowledge of the radioactive inventory of plant and structures is needed to develop and underpin the decommissioning strategy. As decommissioning progresses the level of detail required for the radioactive inventory increases as more specific and detailed questions need answering. Failure to adequately characterise will result in increased costs and project overruns due to missing optimal solutions, over pessimistic assumptions or unforeseen problems and regulatory issues.

Radiological characterisation for decommissioning of Magnox power stations in the UK has been in progress for over a quarter of a century. Firstly measurements and calculations were carried out to develop a strategy. These have been followed by measurements to determine radioactive inventories of waste streams and packages or to allow decontamination of structures and most recently for partial delicensing of sites.

Some examples of the work carried out for the Magnox stations will be given, ranging from the neutron activation calculations to estimate the radioactive inventory within a bioshield and measurements to validate them. Various plant and structures where the radioactive inventory is due to contamination have been characterised by measurements and examples for boilers and cooling ponds will be discussed. Various routine and ad-hoc measurements and shielding assessments have been performed on waste forms to help satisfy conditions for acceptance for disposal or exemption, which will be reviewed. Finally the measurements for delicensing and the successful application of Data Quality Objectives will be addressed.

Introduction

Magnox reactors are graphite moderated and CO_2 cooled, using natural uranium fuel clad in a magnesium alloy. All but two sites have steel reactor pressure vessels: Oldbury and Wylfa were built last and have concrete pressure vessels. The first Magnox reactor was Calder Hall which started generation in 1956 with 11 stations being constructed in the UK in the period to 1971 when Wylfa started generation. Most stations consist of two reactors apart from Calder Hall and Chapelcross which have four each. Currently only one reactor at Wylfa is operating with the rest of the fleet in various stages of defueling and decommissioning.

Nuclear power stations have similar hazards as other industries but it is radioactivity that sets it apart from the rest. Knowledge of radioactivity is essential to:

- plan the optimum decommissioning strategy and understand the costs
- know how much waste is going to be produced and how it will be treated
- consider the health and safety of the general public and workers
- reduce the chance of unforeseen problems and delays during decommissioning

Therefore the radioactivity inventory has to be known with some confidence before any of these factors can be addressed.

Presented is some of the work done to assess the decommissioning radioactive inventories for Magnox reactors after they have been defueled.

After final shutdown, removing the spent fuel off-site takes with it more than 99.9% of the radioactivity. Therefore on an operating site the radioactivity of the plant was never significant compared with that of the fuel. The only occasions when the radioactive inventory was of interest was when items were being disposed of as waste. For decommissioning everything is waste of some sort therefore the radioactive inventory of the plant is now important.

There is a great variety of radiological characterisation for decommissioning but the examples here will be restricted to:

1. measurements to verify the inventory neutron activation calculations within the bioshields

2. contaminated plant such as the gas circuits and fuel route which cannot be predicted and must be found entirely from measurement

3. vaults and tanks containing activated and contaminated solid waste and contaminated resins and sludges

4. clearance monitoring of land and buildings for site delicensing

1 Validation of Neutron Activation Calculations

Once the fuel has been removed there is less than 0.1% of the radioactivity remaining on-site and most of this is to be found within the bioshield.

The radioactivity is mainly due to neutrons from the fission process activating stable atoms into radioactive species. For example stable ^{59}Co, found at trace concentrations in steel is activated to ^{60}Co which is radioactive with a half-life of 5.27 years and gives rise to much of the gamma dose rate within the bioshield in the medium term. There is also contamination present but this is usually insignificant compared with the inventory due to activation. The radioactive inventory due to neutron activation of the structure is readily calculated as the structure is well known and immobile and the activating neutron fluxes at various locations can be calculated. The initial calculations for Magnox reactors started in the late 1970s and were developed over time to produce an inventory for each individual station. A simple zoning model was used which allowed the activation calculations to be carried out on spreadsheets using a two group neutron flux.

For all components within the bioshield, shortly after final shutdown, ^{55}Fe is the most significant radionuclide in terms of the inventory. Its short half-life of 2.7 years means that it soon decays so that after 20-30 years the longer lived ^{63}Ni becomes the most significant radionuclide. Its 100 year half-life results in a much slower reduction in the inventory with time. The other significant radionuclides are ^{3}H, ^{60}Co, longer lived ^{41}Ca from the concrete and ^{14}C from the graphite. In a few decades the inventory will have decayed by approximately two orders of magnitude after which the rate of change is much slower.

Of those radionuclides that emit gamma radiation and so are important to radiological protection for workers, ^{60}Co is the most significant for approximately 100 years when it decays to the level of other gamma emitting radionuclides such as ^{94}Nb, ^{152}Eu and ^{108m}Ag. In

this time the ^{60}Co inventory will have decreased by approximately five orders of magnitude hence a large reduction in dose rate. Exact dose rates will depend upon the components but in simple terms, deferral of the dismantling of the core allows a useful reduction in dose rates for decommissioning and waste treatment even though the reduction in total radioactive inventory is much smaller.

The calculated radioactive inventory needs validating to ensure it is representative.
The factors affecting the calculations are:

- Neutron Fluxes
- Modelling
- Elemental compositions of the structural materials
- Nuclear data

Elements that could be important were investigated in the early 1980s and of all the possibilities only about 27 elements needed to be considered. Elements were included that, when activated, would give rise to radionuclides of half-life greater than a year at a level exceeding the exemption order. At the time this was 0.4 Bq g^{-1} total specific activity but these have recently changed to be nuclide specific so the elements of importance may need review. This selection process (with the inclusion of daughter in-growth calculations) will bound the list of radionuclides requiring consideration for long term disposal.

The major constituents of the materials will be known with good certainty and any small variation will not be significant. Where the greatest uncertainties lie is in the trace elemental constituents where elements present at low concentration such as Co might give rise to significant amounts of radioactivity. These constituents were generally not well characterised if at all. Therefore a programme of work was initiated to identify significant trace constituents in reactor materials and measure their concentrations in nearly 200 samples. For example Co is important to the dose rate in the short to medium term, Ni is important to the long term inventory, Ag and Nb are important to the long term dose rate.

The results of the measurements showed that the data used in the calculations for significant items such as the pressure vessel and graphite core were good. Where materials were not well specified, the trace constituents were generally assigned very conservative concentrations in the calculations, which could now be given more realistic concentrations. The work has led to a refinement of the calculated inventories rather than a radical change and has improved the level confidence.

For more direct validation of the inventories some radiometric measurements have been made.

Wigner probes are invar (Fe 64%/Ni36%) rods used to measure any dimensional changes in the graphite core. They extend from the core reflector to the outer edge of the bioshield on the core mid plane. They have been irradiated by neutrons along their length and become radioactive proportional to the activating flux. Therefore it can be used as a means to verify the flux in a radial direction on the core midplane from the core edge all the way through the bioshield. ^{59}Fe was measured as it is a good gamma emitter produced from ^{58}Fe (0.33% abundant) so that the elemental concentration is well known and with a short half-life meant it was free from the effects of irradiation history. The probe was measured using in-situ high resolution gamma spectrometry as it was withdrawn from the reactor and at the end of the

measurements was reinserted so avoiding any waste disposal problems. The results of the Wigner probe measurements are in good agreement with activities calculated from the inventory neutron fluxes. Similar measurements were also made on steel surveillance specimens held below the core. Agreement between measured and calculated ^{59}Fe mass specific activities was within a factor of 2, adequate for validation purposes.

Two full length concrete cores were taken from the bioshield after final shutdown, one from near the core mid plane and the other at the sub core level. These allowed direct comparison of the measured and calculated activities for a wide range of radionuclides through the 2.5 m thick concrete. Samples were taken along the lengths of each core and assayed by high resolution gamma spectrometry. All samples from one core were analysed for tritium and a number of samples were assayed using radiochemical separation techniques for a range of other radionuclides not detectable by gamma spectrometry.

There is an exponential decrease in activity with distance which reflects the attenuation of the neutron flux. Tritium is the most abundant radionuclide followed by ^{152}Eu which have half-lives of 12.3 and 13.5 years. The next most abundant is ^{41}Ca with a half-life of 103,000 years. Tritium from the most active inner samples cross contaminated less active outer samples which precluded observing possible tritium diffusion in the bioshield. Cross contamination was not an issue for other radionuclides. Comparing measured with calculated activity gave generally good agreement, but with the measured activities lower than those calculated meaning that the attenuation of the neutron flux through the bioshield was underestimated in the calculations.

Gamma dose rates at various locations within the inner radius of the bioshield were calculated. Comparing these calculated values directly with measured values tests the whole calculational model. There are a number of access points on the pile cap where a dose probe can be inserted and measurements made between the reactor pressure vessel and the inner bioshield for the full length of the reactor. Additionally for one reactor, measurements were possible through the axis of the core. The agreement between measured and calculated dose rates on the core midplane was very good, but for other locations, above and below the core agreement was not as close. These measurements highlighted some over simplifications in the model. This resulted in the inventory being reassessed using Monte Carlo codes developed for the steel pressure vessel safety cases.

Similar calculations have been performed to assess the radioactive inventory of Miscellaneous Activated Components (MAC) held in shielded vaults. Validation of the inventory has also been performed by dose probe measurements.

2 Contaminated Plant and Structures

For plant which has been contaminated there is no easy way to predict the inventory so the radioactivity must be measured. The inventories of contaminated plant are a tiny fraction of the activity remaining on site but are still significant as they are associated with large quantities of material so must be assessed for waste disposal.

2.1 Primary Coolant Circuit

Examples of contaminated plant are the gas circuits and boilers, contaminated by corrosion products activated in the core being carried out of the core and redeposited on all surfaces. The boilers are large structures weighing upwards of 300 te and consist of a shell containing a number of tube banks. For the steel pressure vessel reactors the gas is fed to the boilers along approximately 1.5 m diameter ducts incorporating a number of bends and bellows units. For the concrete vessel reactors the boilers lie within the bioshield therefore there are no gas

ducts. In-situ high resolution gamma spectrometry measurements were made on tube banks, and inside or on the outside of gas ducts to obtain an inventory for ^{60}Co. This was the key radionuclide and the only one reliably measured. Access for measurements was by a combination of manned entry, lowering detectors into ducts or by measurements on the outside of ducts. The contribution from the boiler shell in comparison to the tube banks is difficult to assess for an operating site. To do this, a small lead shield was produced which contained a piece of photographic film used in film badges. The film was exposed to the boiler shell internal surfaces for a period and the film developed in the normal way. From the measured dose, the ^{60}Co photon flux could be estimated and therefore the area specific activity. For a decommissioned boiler, samples of the shell could simply be removed and direct measurements made. To assess the non-gamma emitting radionuclides, smear samples were removed from the internal surfaces and subjected to radiochemical analysis. In this way a total radioactive inventory was calculated.

In the short term the radioactive inventory is dominated by ^{55}Fe but this has a half-life of 2.7 years so other radionuclides such as ^{3}H and ^{60}Co become most significant until in the longer term when ^{14}C controls the inventory. The gamma dose rate is predominantly due to ^{60}Co and therefore will reduce by an order of magnitude every 17.5 years from cessation of generation.

The boiler tubes have the highest mass specific activity as they have a large surface area but low mass, the internal structure has a smaller surface area and mass but the shell has a relatively small surface area and large mass therefore the lowest mass specific activity. As a consequence the boiler components apart from the shell will not decay to below the low level waste exemption level on any reasonable decommissioning time scale.

The gas ducts closest to the reactor can be activated by neutrons. This was identified by removing specimens through the thickness of the duct after the end of generation. The specimens were cut into slices and analysed by gamma spectrometry. Additionally, high ^{3}H specific activity was noted at the inside surface of the duct reducing rapidly through the thickness in the steel.

The inventory data was used at Berkeley to identify some of the non-activated duct work for decontamination and release. This was achieved relatively easily but the steel needed to be baked to remove ^{3}H. Decontamination of the tube banks was not found to be economical. Five of the boilers have recently been shipped to Studsvik for metal melt recycling.

2.2 Fuel Cooling Ponds

The fuel cooling ponds and associated plant are contaminated mainly by fission products. Fuel once discharged from the reactor is stored underwater in all but two of the sites before being shipped to Sellafield for reprocessing. The two sites that do not have ponds are Calder Hall, which is on the Sellafield site so ships directly, and Wylfa which has dry stores. The ponds are large concrete structures, different at all stations, but are about 7 m deep with a floor area in the range of 500 m^2. Radioactive contamination has diffused into the concrete surface from the pond water and contaminated the inner layer. The ponds structures are a potentially large radioactive waste stream therefore it is vital to understand the radioactive inventory to develop the most appropriate decommissioning strategy.

To measure the penetration of activity into the concrete several cores were taken. The cores were 25 mm in diameter by about 100 mm long and were cut into 10 mm slices, crushed and analysed using gamma spectrometry in the lab. The most abundant gamma emitting radionuclide in this case was ^{137}Cs. Some of the paint layer and near surface samples were subjected to radiochemical analysis to obtain a fingerprint of other radionuclides relative to ^{137}Cs to allow the full inventory to be calculated.

The inventory and radionuclide fingerprint of the paint layer and underlying concrete are very different. The paint layer was successful in stopping much of the penetration of radionuclides into the concrete. The paint layer contains a significant contribution from ^{90}Sr and actinides as well as ^{137}Cs. However, ^{137}Cs did significantly penetrate the paint layer and is the only man made radionuclide detectable at depths greater than approximately 10 mm. ^{137}Cs is also the most significant contributor to the gamma dose rate therefore there is little reduction in dose rate in the medium term and no advantage in deferring decommissioning. The small thickness of the paint results in a high mass specific activity but this has to be compared with the total inventory. Therefore by removing the paint layer most of the inventory of the actinides is removed but the reduction of gamma dose rate is small. The radioactivity reduces with depth in the concrete approximately exponentially. This has consequences to decommissioning strategy in that the boundary between Intermediate Level Waste and Low Level waste occurs at approximately 10 mm depth and the exemption level occurs at approximately 40 mm. The inventory data allow the optimum end state, to be selected that satisfies a wide range of criteria including reducing the volume of waste produced.

3 Waste Vaults and Tanks: ILW Characterisation

Bi-products from reactor and pond operations have been placed in various waste vaults and tanks on site, for example:

- fuel element debris (FED);
- miscellaneous activated components (MAC), e.g. control rods;
- resin and sludge from pond water and active effluent filtering, ion exchange and cleaning processes.

Retrievals from waste vaults and tanks are now underway or imminent at Magnox sites. Many of these wastes are classified as Intermediate Level Waste (ILW)[1]. It is necessary to characterise these wastes in order to segregate any waste that may be disposed of via an alternative route e.g. classified as Low Level Waste (LLW). ILW is packaged for on-site storage and eventual disposal to the Geological Disposal Facility (GDF) and requires a reliable and justifiable radionuclide inventory. The radionuclide inventory forms a mandatory part of the submission to the Radioactive Waste Management Directorate (RWMD) to obtain an agreement in principle (Letter of Compliance) for disposal of the particular waste stream.

To produce the radionuclide inventory, generally, in-situ measurements of key gamma emitters are combined with the results of prior representative sampling and radiochemical analysis. The sampling results are used to produce fingerprints, which are sets of ratios of each radionuclide to its related key gamma emitter, i.e. fission products are ratioed to ^{137}Cs and activation products are ratioed to ^{60}Co. Radionuclides can also be ratioed to waste mass where this is judged to be more representative.

In-situ measurements are often dose rates where the gamma signal is dominated by one radionuclide, e.g. ^{60}Co in activated components; or using gamma spectrometry when a number of radionuclides contribute to the dose rate and need to be characterised separately.

Specific examples of ILW characterisation projects are provided below, with a discussion of techniques, outcomes and lessons learnt.

[1] LLW-ILW boundary is 12 GBq/te beta/gamma activity or 4 GBq/te alpha activity

3.1 Fuel Element Debris (FED)

Figure 1: FED Vault

FED results from the fuel desplittering process and consists of Magnox metal plus nimonic (Co/Ni alloy) springs from the fuel element end fittings and some small fragments of fuel. The nimonic springs are highly activated in the neutron flux and give rise to a high dose rate due to ^{60}Co.

The FED is successively retrieved from the vault, assayed in a 30 L tray, and transferred to a disposal container (3m^3 unshielded box which is later grouted with cement). Initially a custom designed low-resolution gamma-ray spectrometry assay system was used, which is now being replaced by a high resolution system to improve throughput.

There are a number of complexities in assaying the radionuclide inventory of this waste. Firstly the ^{60}Co signal from the activated nimonic springs dominates the ^{137}Cs signal from the general contamination of the Magnox metal. This requires that the detector must be optimised for high peak to Compton scatter ratio to allow the ^{137}Cs peak, which sits on the ^{60}Co Compton continuum, to be more easily discriminated. This demands a larger, more efficient detector however, for ILW, a larger detector would 'saturate' due to the high count rate. Placing a shield in front of the detector would reduce the count rate, but at the expense of the peak to Compton (signal to noise) ratio. The selected design therefore employs a large detector crystal with a very narrow V-shaped collimator, as this retains the signal to noise benefit whilst reducing counts to the detector. This provides improved performance in terms of the ^{137}Cs limit of detection against the high ^{60}Co spectral background. Approximately 30 litres of waste on a tray can currently be assayed in around 300 to 600 seconds, depending on the ^{60}Co activity on the tray, achieving a ^{137}Cs limit of detection of around 3 GBq/m^3 for the 3m^3 waste package (the reporting level specified by RWMD). There is also an option, for exceptionally active trays of waste, to insert a steel filter to reduce the count rate to the detector. This does degrade the ^{137}Cs peak to Compton ratio, although not as much as lead would, and the total count time may need to be increased in this case to achieve the ^{137}Cs limit of detection.

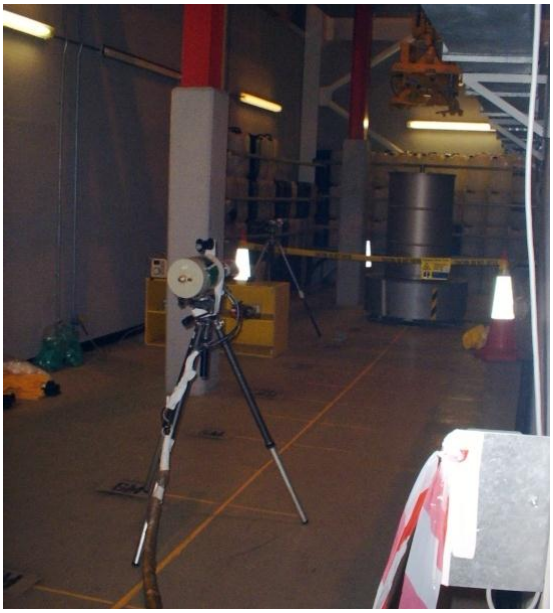
Accounting for fuel fragments to provide a realistic inventory is achieved by selecting a threshold signal below which the ^{137}Cs is assumed to be general contamination of the Magnox

and above which it is assumed to be a fuel piece. This threshold equates to a few grams of fuel, with ^{137}Cs activity ~3x10^{8} Bq. Once a piece of fuel has been detected, a best estimate correction is made for the self-shielding of the uranium in order to calculate the ^{137}Cs activity. Other radionuclide activities are then derived using the code "FISPIN" and ratioed to the ^{137}Cs activity. The fingerprint from contaminated Magnox was derived from radiochemical analysis on samples, with the radionuclides ratioed to waste mass. The nimonic springs contain ^{60}Co and ^{63}Ni from neutron activation. Hence a separate fingerprint for the springs was produced with only ^{63}Ni ratioed to ^{60}Co.

Figure 2: Magnox Samples from FED Vault

So far, Cs^{137} package activities ranged from 4x10^{8} Bq to 2x10^{11} Bq: Co^{60} package activities ranged from 1 x10^{10} Bq to 5 x10^{11} Bq (at reference date 01/04/2011).

In addition to the vault FED, 104 drums of FED were also produced and stored in a shielded facility on site. These drums were used when the FED vaults became full. The drums were assayed on a turn-table using a stand-alone gamma spectrometer.



The activities of the key radionuclides were calculated by modelling the drums in a point kernel shielding code (MicroShield). Similar assumptions to those used for the FED assay system were made in order to estimate the presence and activity of fuel pieces and to fold in the radionuclide fingerprints. These calculations were performed in a spreadsheet rather than automatically in the assay system software. This adhoc, bespoke method is suitable for smaller quantities of waste where the expense of having assay experts on site and performing the calculations is lower than that of developing/purchasing an assay system which can then be run day to day by 'non-experts' on site.

Figure 3: FED Drum Gamma Spectrometry

3.2 Ion Exchange Resin

Ion exchange resin is used to remove ^{137}Cs and other radioactive species from pond water and active effluent. It is currently stored in large tanks on site and is generally characterised by sampling and radiochemical analysis, in a similar way to the FED vaults, by taking samples from different depths and then bulking. Dose rate measurements down the outside of resin tanks and down the inside of a pipe pushed into the resin have been made. The dose rate is dominated by ^{137}Cs and providing any background signal from other plant in the measurement area is low, in theory it is possible to create a simple radiation transport model which relates measured dose rate to ^{137}Cs activity. However, large uncertainties can result due to the activity distribution assumptions. Each dose rate measurement will realistically only assess resin in the nearest 20 cm or so. Layering of different activities and ages of resin occurs as the tank is filled and unevenness and piling of the layers has occurred during previous retrievals. Some attempts to mix the resin to homogenise it have been made, although the bottom layer of resin is often compacted and difficult to penetrate.

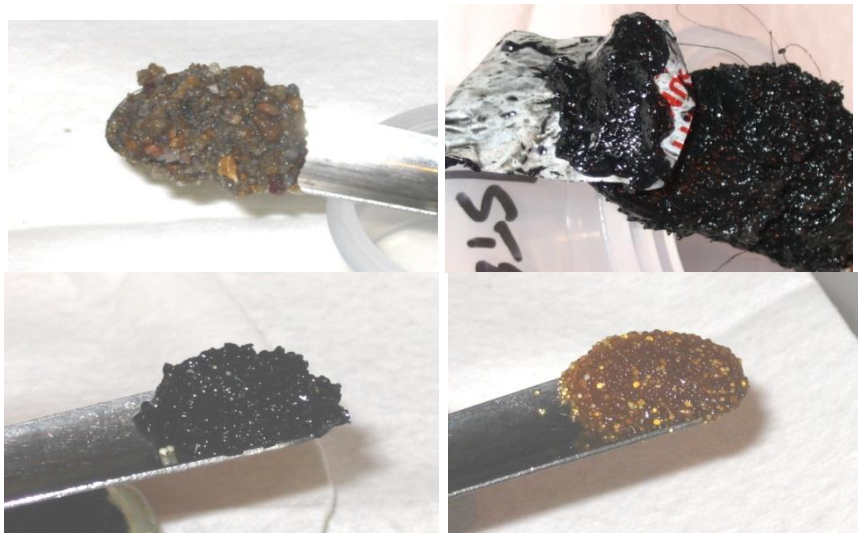


Figure 4: Resin Samples

The resin is retrieved and grouted in shielded drums. The characterisation data has been used to predict the drum shielding thickness required to keep operator doses to acceptable levels. Once grouted, the maximum contact dose rate on the external surface of each drum is recorded and the ^{137}Cs activity is calculated using a simple model. The fingerprints produced from sampling are then folded in to produce radionuclide inventories for each drum.

4 Delicensing and the DQO Process

Two Magnox sites have been partially delicensed to date, Berkeley and Oldbury.

4.1 Berkeley

Berkeley site includes the power station site and Berkeley Centre. The power station operated from 1962 to 1989. Berkeley Centre was formerly Berkeley Nuclear Laboratories and consisted of research facilities for Magnox Generation including radiochemical and instrument testing laboratories, a shielded cave and cell area and office accommodation.

The approach to delicensing was agreed with the regulator prior to data collection. The up-front work included:

- an initial statement of Magnox's understanding of the regulatory expectations and proposals for delicensing criteria;
- a site study which included the history of the site and potential sources of radioactivity on-site;
- a breakdown of different zones of the site, depending on likelihood of contamination, e.g. no history of active works, potentially contaminated and known to be contaminated;
- a proposed in-situ-measurement, sampling and analysis plan.

The delicensing criteria were based on the regulations and guidance in existence at the time the delicensing proposal was made, and are as follows:

- The Substances of Low Activity (SoLA) Exemption Order under the Radioactive Substances Act (RSA93) stated an exemption level of 0.4 Bq/g total activity;
- Annex 1 of the Basic Safety Standards Directive (Euratom 96/29) allows member states to exempt a practice where doses to members of the public are of the order of 10 μSv or less per year;
- The guidance to regulators discusses the requirement of "no danger" from any residual radioactivity on the site above the average natural background, which equates to a risk level of 1 in a million per year. It was agreed that the activity and dose levels quoted above were consistent with this risk level.

Radiological characterisation measurements included the following techniques:

- 600+ gamma spectra in-situ in buildings
- 300 particulate samples analysed on site by high-resolution gamma-ray spectrometry (HRGS)
- 600 smear samples analysed on site by HRGS
- Health Physics Surveys
- Land survey HRGS measurements over most of the land
- In-drain LRGS measurements over 1 km of drains
- Samples for radiochemical analysis

Gamma spectra were acquired in every room of every building; and off-site background measurements were also made to calculate ambient dose rates. HRGS was used because it is nuclide specific and sensitive enough to detect the 10 uSv/y target quickly. Heath Physics instruments do not distinguish between man-made and natural radionuclides and the latter dominate dose in this situation.

The Health Physics surveys used relatively insensitive gross counting techniques which were undertaken on exposed surfaces (floors, walls, work-benches, etc.) in potentially and known contaminated areas, to indicate localised elevated areas of contamination which could then be remediated.

Smear and particulate samples were collected from ventilation ducts and places where dust may have accumulated over the years in areas that were potentially or known to be contaminated.

Samples were collected, based on professional judgement, to develop radionuclide fingerprints, which were applied conservatively to provide a total activity to key radionuclide (^{137}Cs or ^{60}Co) ratio. The derived total specific radioactivity was then compared against the 0.4 Bq/g limit.

The delicensing safety case was accepted by the regulator and partial site delicensing was achieved in 2006.

4.2 Oldbury

Oldbury Nuclear Licensed Site consisted of the power station and adjacent land and buildings. The power station began generating in 1967 and remained operational at the time of the partial delicensing. The area to be delicensed, around 32 ha, was situated outside of the site security fence.

A similar approach to the Berkeley Delicensing Project was adopted but the planning of characterisation campaigns used Data Quality Objectives, DQO. In addition, the regulators now favoured dose-risk based radionuclide specific exemption values based on IAEA RS-G-1.7 (Application of the Concepts of Exclusion, Exemption and Clearance) rather than the SoLA limit.

The area to be delicensed consisted of grassland, woodland, roads, car parks, buildings, drains, a silt lagoon and standing water (pools and ditches). The area was divided according to terrain type, as this can significantly affect contaminant levels and sampling / monitoring techniques, and then sub-divided based on the data review.

The most pertinent findings of the review of existing data were as follows:

- The area to be delicensed was situated outside the power station site security fence and had no history of radioactive works.

- Authorised aerial and liquid discharges from the power station were considered to be the main potential mechanisms of contaminating the area.

- Transport of radioactive materials across the area to be delicensed was considered to be another possible route for contamination e.g. spent fuel flasks.

- The buildings within the area to be delicensed were used for training, visitors and security. The training centre previously contained a sealed source store and the visitor centre had sealed displays containing naturally occurring radioactive material. All sources had been removed.

Based on the data review, all zones were categorised as having the potential for contamination but at levels unlikely to be above the delicensing limit.

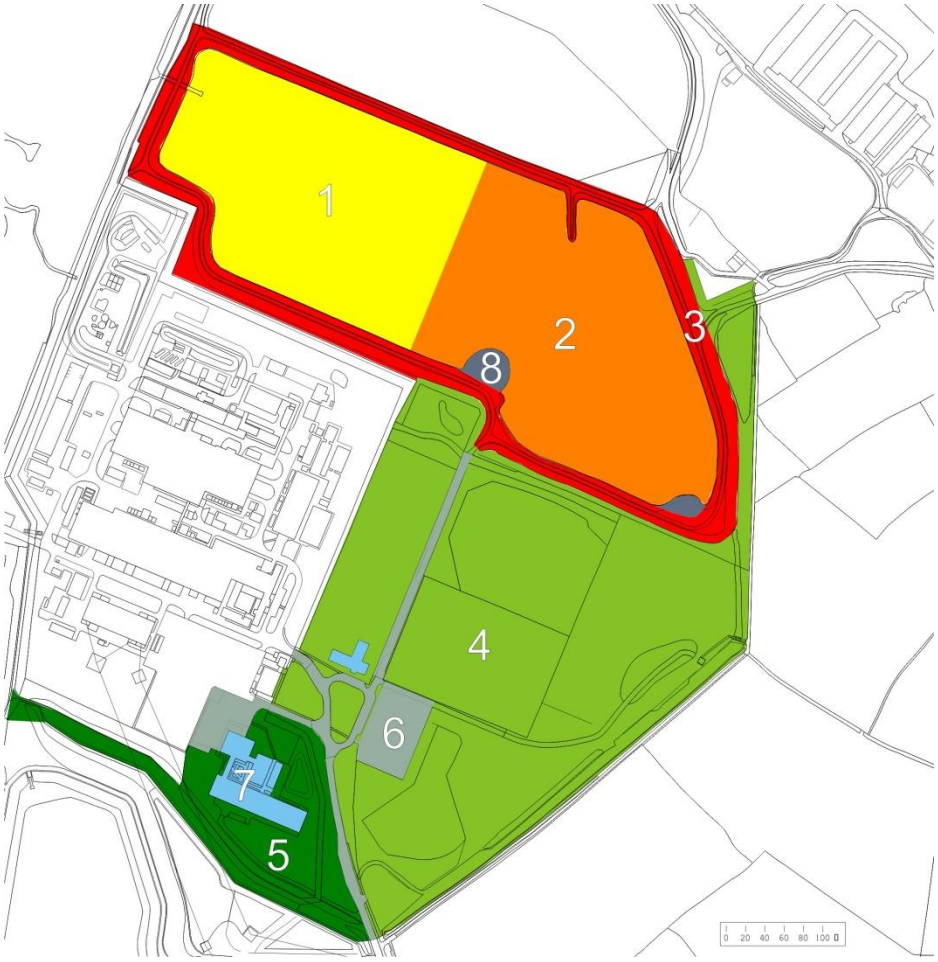


Figure 5: Delicensing Area Zones

The grassland zone (4) was considered to be potentially contaminated by authorised gaseous and particulate aerial discharges, which were expected to be limited to the top few centimetres of soil only and uniformly distributed. Surface soil samples were considered appropriate for assessing the activity within this zone.

Ground which had been disturbed since the power station became operational was considered as separate zones (3 & 5), in which potential contamination from aerial discharges could have been transferred to a greater depth within the soil. Surface and sub-surface samples were considered appropriate for assessing activity within these zones.

Silt Lagoon 2 contains material dredged from the adjacent river and therefore potentially contains contamination from liquid discharges. Silt Lagoon 2 was further subdivided into two zones (1 & 2), due to a mechanism for transport of fine sediment in surface water following heavy rainfall. This could lead to the deposition of fine sediments containing elevated radionuclide concentrations nearer the outfall structure. Surface and sub-surface samples were considered appropriate for assessing the activity within this zone.

Building internals (7) were assessed in the same way as those at Berkeley. Uncollimated high-resolution gamma-ray spectrometry measurements were acquired in the centre of each room. Smear samples and Health Physics contamination probe measurements were taken local to the locations of source stores and displays.

The roads, car parks (6) and externals of the buildings were believed to be more adequately represented by storm drain silt samples, as the rain water run off would have washed any

contamination in to the drains. Areas of standing water were also believed to be best represented by sediment samples from the bottom.

Prior to devising the sampling plans, gamma spectrometry surveys of the area were performed to provide preliminary data and confirm that the zoning was appropriate:

A high-resolution gamma-ray spectrometry (HRGS) survey using an uncollimated hyper pure germanium detector at one metre above the ground was performed. A total of 1564 spectra were reported with 168 of these on tracks over the silt lagoon. A target Minimum Detectable Activity of 0.02 Bq/g was set for ^{137}Cs and ^{60}Co; this led to a counting time at each position of 150 seconds. The averaging area of each measurement was assumed to be a circle of radius 7 m with measurements being taken every 10 m where accessible.

A Low Resolution Gamma Survey was also carried out. The equipment consisted of an array of 3" x 3" sodium iodide detectors. A total of over 329,000 measurements were taken. The system was used in gross counting mode to provide a more rapid survey. The total area covered was considered to be around 12.7 hectares. High count rate alarms were investigated using the HRGS to identify the radionuclide(s) responsible.

Most ^{137}Cs detections occurred on the silt lagoon: thus it was appropriate to zone the silt lagoon separately. Other elevated count rates were found to be due to the routine release of ^{41}Ar from the reactor, and naturally occurring ^{214}Bi on the track around the silt lagoon.

Prior to the main sampling campaign, up to five samples were collected from each of the zones to be sampled: the grassland, silt lagoon, drain silt and standing water sediment. These were analysed using gamma spectrometry and for total alpha and beta activity. Portions of samples from each zone were also bulked and analysed for individual radionuclides considered likely to be present based on routine environmental monitoring during power station operations, plus dominant naturals e.g. ^{40}K, ^{214}Bi. Bulking was used to save analysis costs. A relatively small number of samples were considered sufficient since the site history indicates potential widespread uniform contamination rather than 'hotspots'. Preliminary radionuclide fingerprints relative to ^{137}Cs and ^{60}Co for each zone were produced. Cross-checks were made between total alpha and beta activities and individual radionuclides.

The drain silt and standing water sediment samples were collected from every drain inspection chamber and area of standing water: it was therefore decided that a statistical approach was not required, as it was easy to sample a large proportion of the area.

The statistical sampling plan design software "Visual Sample Plan" was used to generate the sampling plan for grass land and the silt lagoon. Since ^{137}Cs dominated the fingerprint, the calculations were based on this radionuclide. A non-parametric distribution was assumed as a pessimistic assumption. The IAEA RS-G-1.7 limit of 0.1 Bq/g was used with an upper confidence level of 95%. The probability that a false positive is generated was selected as 10%, balancing sampling costs against the required level of certainty. 32 samples were collected from each zone requiring surface sampling. 5 confirmatory boreholes were also drilled from depths of 3 m to 6 m in zones requiring sub-surface sampling. Samples were collected at 1 or 2 m intervals depending on depth. "Targeted" samples were also collected for reassurance from small piles of construction material (8) on the silt lagoon and a small area of soil at the edge of a car park where rain water run-off collects. Samples were analysed by an external accredited laboratory.

Figure 6: Borehole Sampling

The borehole results were compared with the surface sample results to determine whether stratification of activity existed at depth. Sample results were compared with the IAEA RS-G-1.7 limits (the quotient rule is used to sum activity results where more than one radionuclide is present).

The maximum level of ^{137}Cs determined at the 95% upper confidence level within the area for delicensing was 0.065 Bq/g. Other radionuclides did not make a significant contribution. This is well below the Exemption Order value and the indicative value for regulatory concern of 0.1 Bq/g.

Delicensing was granted in 2011. Nearly 80 acres of land at Oldbury has been removed from nuclear regulation, representing the single biggest plot to be delicensed in the UK at one time.

Conclusions

The techniques and systems developed for radiological characterisation in decommissioning Magnox reactor sites has allowed Magnox to assess the following:

- How much radioactivity remains after defueling
- How the radionuclide inventory varies with time

Data of appropriate quality reduces uncertainties in predictions and conservatism in assumptions. This provides the following environmental, safety and cost benefits:

- More accurate prediction of waste arisings
- More effective segregation of waste for re-use, recycling, exemption, etc.
- Better informed strategy and prediction of dose to workers
- Reduced overall project risk