EVALUATION OF RADIONUCLIDES TRANSPORT IN CONCRETE USING SPECTROMETRIC DATA MEASURED IN THE IFIN-HH VVR-S RESEARCH REACTOR BUILDING*

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1. INTRODUCTION

The VVR-S nuclear research reactor, owned by Horia Hulubei – National Institute for Physics and Nuclear Engineering (IFIN-HH), was the first research reactor using the VVR-S type soviet design it was built between 1955 – 1957, in Romania. The role of the reactor was for research and radioisotope production. VVR-S means that is a thermal neutrons reactor model S moderately cooled and reflected with distilled water, fuelled with enriched uranium 10% in the beginning and 36% subsequently.

The research reactor operated until 1997 and was permanently shut-down in 2002. During his life time, it was functional for a period of 113467 h, including 2000 h at 3.0–3.5 MW power. The total power output up to 1997 was 9.59 GWd. It had a utilization factor of 65 % (approximate 9510 effective days of operation) with an average of 1 MW thermal power.

The decommissioning of the reactor has started in 2010 and is planned to be finalized in 2020, being the first nuclear research reactor that has started the decommissioning project from the South-East of Europe [1].

Radioisotope production generated a significant contamination in the reactor main building, ventilation system and radioactive leakage drainage, overflow and collecting system. Major radioactive contaminants generated by this activity, with the half life higher than one year, are: Co-60, Cs-134, Cs-137, Sr-90, Eu-152, U-238 and Am-241 [1].

The reactor was designed to provide experimental activities for research and radioisotopes production in a thermal neutron flux of max. 2x1013 n/cm2sec, at an operating power level of 2 MW.

The Construction of the VVR-S Nuclear Reactor began in 1955 and criticality was attained on July 29, 1957 with a fuel loading of 4.5 kg of 235U. Its last shut down was on July 1997. All of the fuel (10% 235U until 1984, and 36% 235U after this date) was removed from reactor core to the Cooling Pond (Reactor Hall) and the basins of Spent Nuclear Fuel Storage (Building No. 20).

In 2002, the Romanian Government approved, by Government Decision, the final shutdown of the VVR-S reactor and its decommissioning.

National Institute for Research & Development for Physics and Nuclear Engineering-Horia Hulubei (IFIN-HH) is a state company that is subordinated to The Ministry of Education and Research. The main activity, according to CAEN classification is Research & Development (73)-fundamental and applied research, technological development, activities and services in the field of physics and nuclear engineering.
VVR-S Nuclear Reactor for Research and Radioisotope Production is a public property under the administration of IFIN-HH.

1.1. FACILITY DESCRIPTION

IFIN-HH is located in the Romanian Plain, in Magurele Village, about 3 km southeast of the built borderline and 8 km straight of downtown Bucharest. In the south-southwest, at 0.8, 1.2 and 5 km are Ciorogarla, Sabar and Arges rivers. In the south and southeast is Jilava Village. The location is shown in Fig. 1

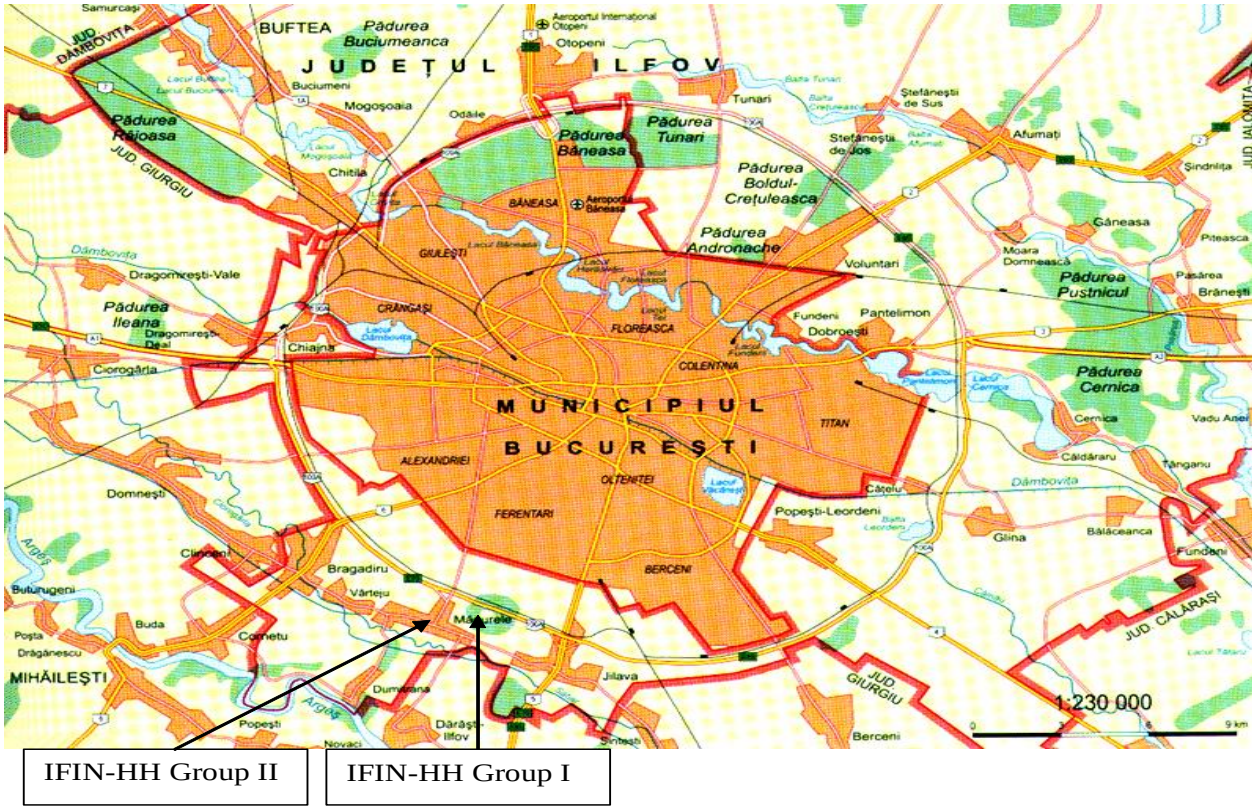

Fig. 1 Location of IFIN-HH

VVR-S Nuclear Reactor is situated in the IFIN-HH GROUP I (Fig. 2) located in the neighborhood of Magurele Village, in the west and next to Jilava Village in the south and southeast. IFIN-HH GROUP I have access to the road Bucharest-Magurele, through a way with one traffic bands on each direction (concrete with asphalt of 9 m width). As well as, the IFIN-HH GROUP I premise has

access to the railways networks through a connection to the borderline railway from the south of Bucharest.

The Group I of IFIN-HH is surrounded by 200 m protective forest ring, VVR-S Nuclear Reactor being the center point of this ring shows the layout of Group I of IFIN-HH.

The neighborhoods of VVR-S Nuclear Reactor facility are:

- Radioisotopes Production Center–CPR (Buildings no. 11, 12, 13, 14, 38, 39);
-Radioactive Waste Treatment Plant-STDR (Buildings no. 15, 16, 17, 40, 41);
-Nuclear Engineering and Vacuum Department (Buildings no. 18, 19, 21);
-Heavy-Ions Van de Graaff-Tandem Accelerator (Building no. 30);
-Zero-Power Nuclear Reactor (Building no. 24);
-Thermal Plant (Buildings no. 25, 26, 27, 28, 29).

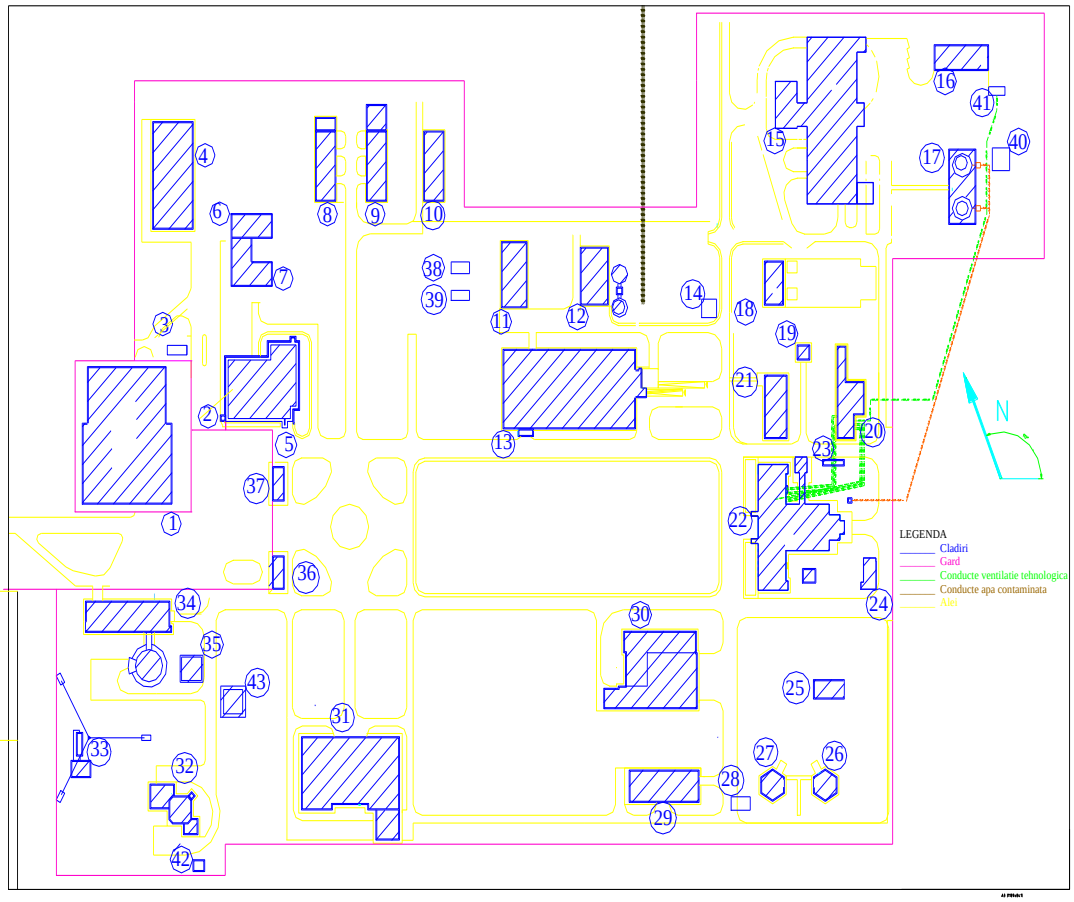

Fig. 2 Location of IFIN-HH Research reactor inside IFIN-HH premises.

The VVR-S reactor equipment was set on in many buildings, as following:

- The main building (no. 22)
- Technological ventilation station, that has a 40 m tower (as part of the building no. 20);
- Secondary circuit pumps building (building no. 29);

- Cooling tower, float cabin, and valve of the cooling tower (building no. 26);
- Underground emplacement containing the 30 m^3 tank for contaminated water;
- Metallic barrack for the material storage (building no. 23) ;
- Underground emplacement containing two tanks of 300 m^3 each for contaminated water (building no. 17);
- Underground system of ventilation pipes
- Radioactive liquids. sewers system
- Power Transformators (next to the Reactor Hall, Laboratories Building and Zero-Power Nuclear Reactor Building;
- Spent Filter Storage Building (building no. 41).
- Away From Reactor (building no. 20);
- Radioactive liquid treatment plant (building no. 25);

Decommissioning activities of VVR-S Nuclear Reactor (Fig. 3) will take place on a 7500 m^2 surface. Almost half of this surface, around 3400m^2, is occupied with buildings, metallic structures for material storage, underground emplacement and pipes.

The decommissioning project of VVR-S Nuclear Reactor takes into account buildings with activated or contaminated structures, including those in which nuclear spent fuel or radioactive waste arising from operational period are stored.

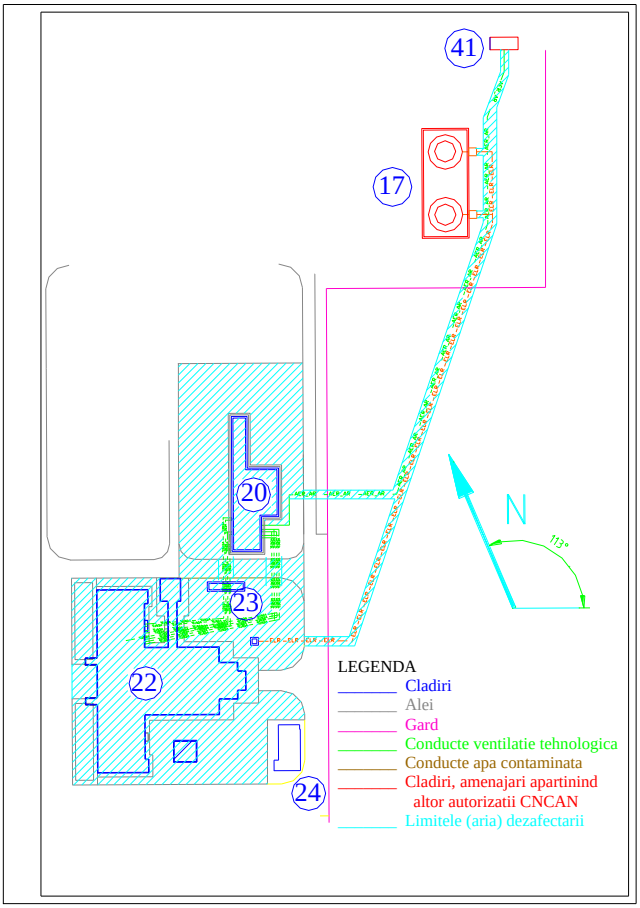

Fig. 3 Reactor Decommissioning Limits

1.2. FACILITY OPERATION HISTORY

The construction of the VVR–S research nuclear reactor took place from 1950 to 1955, with a presumed lifetime of approximate 40 years, from nuclear security reasons. The facility operated 113467 h at a power of 2 MW and 2000 h at a power of 3.0-3.5 MW. The total thermal energy produced was 9510 Mwd.

After the start-up of reactor, the main functions and services ensured were the followings:

- Irradiation for radioisotopes production for diagnosis and therapy in nuclear medicine. The most used radioisotope was ^{131}I;
- Irradiation for radioactive source production with applications in industry (level gauge, welding fault detection with ^{192}Ir);
- Sample irradiation and the use of the neutron flux for research in physics, biophysics, biochemistry, radiochemistry;
- Diodes' irradiation for commutation time shortening.
- Doping with silicon for electronic industry;
- Control equipment's calibration of the reactor and NPP devices in neutron field and in neutron – gamma field
- Maintenance of the two national standards of thermal and intermediate energy neutron density (one of these standards is also used for the characterization of semiconductor devices irradiated with fast neutrons).
- Achievement of internal column with thermal neutrons flux increased for some special irradiation.
- Utilization of the reactor for didactical purposes.

Major events:

- Two floods in the pumps room (overflow from the 300 m^3 tank into the buffer tank and the pumps room);
- A technetium contamination on large surface;
- Mechanic distortion of two fuel assemblies (one EK-10 and one C-36), without fissuring the fuel rods;
- Cutting of a rod cell with iridium pill during the handling and lifting from the reactor tank;
- Melting of a 235uranium capsules in the helium cooled irradiation loop.

Minor events:

- Dropping objects into the core (aluminum sheet boards, tools, pencils etc.);
- Sheet boards blockings into the channel between the reactor and hot cells;
- Detachments of some aluminum sheet boards lids;
- Primary circuit's contamination with substances for irradiation;
- Over irradiation or under irradiation of some sheet boards caused by wrong handling.

An abnormal increase of the radioactivity of the primary cooling agent was recorded only once due to a fuel assembly. This assembly was taken out from the device and sent to the cooling pond. Afterwards, it was reintroduced in the reactor without recording any increase of the radioactivity. At the cooling primary circuit some leakage were noticed, especially at the main traffic pumps.

As concerns the system for collecting the radioactive liquid waste, a flooding of the floors took place in the rooms at the lower level, through traps, as well as in the cooling pond because of the filling and discharging of the buffer tank.

History of the Radioisotope Production

1. After reactor's shutdown, operations related to the radioisotope production were no longer carried out.
2. Between 1.01.1990 and 1.01.1998 – radionuclides were produced through the irradiation of targets encased in aluminum blocks with dimensions of ϕ 37 x 140 mm and ϕ 22 x 140 mm, in wet channels as it follows:
   – Medical products: ^{131}I approx. 100 Ci/year; ^{99}Mo approx. 50 Ci/year; ^{198}Au approx. 50 Ci/year; ^{192}Ir approx. 50 Ci/year;
   – Irradiated silicon for the electronic industry: between 10 and 30 blocks;
   – Various radiochemical products: maximum activity 5 Ci/year;
   – Sources of ^{60}Co for furnaces and other industrial applications approx. 2 Ci/year.

3. Between 1957 and1990:
   – Medical products: ^{131}I approx. 100 Ci/ year; ^{99}Mo approx. 50 Ci/ year; ^{198}Au approx. 50 Ci/ year; ^{192}Ir approx. 50 Ci/ year; ^{90}Y approx. 2 Ci/ year;
   – Radio-chemical products: ^{82}Br approx. 2 Ci/year; ^{42}K approx. 4 Ci/ year; ^{32}P approx. 3 Ci/ year; ^{35}S approx. 300 mCi/ year, various radiochemical products max. 5 Ci/ year;
   – Sources of ^{60}Co for furnaces and other industrial applications approx. 5 Ci/ year;

4. Between 1970 and 1983:
   - A surplus of sources ^{192}Ir was produced for gammagraphy with approx. 15 – 20 Ci/source and total activities between 1000 and 8500 Ci/ year.

5. Between 1980 and 1985:
- small quantities of uranium were irradiated for research purposes, attempting to achieve the molybdenum separation to obtain (^{99}Mo - ^{99m}Tc) generators.

A preliminary cleaning was made in room 18 and in the operators 'rooms 21, 22, 23 and 24, as well as in the transfer room 20. The corridor 17 was cleaned, remaining contamination on the corridor 19 and in the operators' rooms. Regarding the room 22, the linoleum was cut off and replaced with a new one. In 2001 the remaining contamination level was of maximum 200 imp./cm^{2}min.

2. RADIOLOGICAL CHARACTERIZATION OF ROOM 30 AND 31

2.1 Sampling of radionuclides

The main activation products found when the reactor was shut down:

- ^{55}Fe, ^{60}Co, ^{59}Ni, ^{63}Ni, ^{39}Ar, ^{94}Nb in cast iron and iron,
- ^{3}H, ^{14}C, ^{41}Ca, ^{55}Fe, ^{60}Co, ^{152}Eu, ^{154}Eu in concrete,
- ^{3}H, ^{14}C, ^{152}Eu, ^{154}Eu, in graphite.

Regarding the iron and cast iron, ^{55}Fe and ^{60}Co are the most important part of the radioactive inventory for the first ten years after shut down. Most of these isotopes are disintegrated during the

next fifteen years. The main activation component is given by ^{59}Ni, ^{63}Ni and ^{94}Nb, which have greater half-life. Regarding the concrete and graphite, the most important contribution for long term is given by ^{14}C, ^{41}Ca and the europium isotopes.

The specific types and numbers of measurements are specified for every area listed below. The alpha, beta and gamma activity is measured for all the samples (in a fixed geometry) in order to establish if an analysis is necessary to identify radionuclides and to determine the specific activity. The radionuclides will be identified only for those samples for which the activity exceeds the minimum detectable activity –MDA-specified.

The large samples can be divided into sub-samples. The concentration of each radionuclide is determined only for one or two sub-samples. For the other sub-samples only the gross activity is determined. The ratio of the nuclide's specific activity concentrations to the gross activity can be used to infer the activity concentrations for all the other sub samples.

Where alpha contamination is found, the laboratory will perform an alpha spectroscopy analysis to determine the alpha emitting radionuclides. Where beta activity is observed, the gamma spectroscopy results will be compared to the results of the gross beta analysis.

Where beta activity detected is not associated with gamma activity, samples will be analyzed to determine the activity concentration of the following beta emitting radionuclides: ^{3}H, ^{14}C, ^{36}Cl, ^{99}Tc, ^{63}Ni, ^{55}Fe, ^{90}Sr, ^{90}Y.

The size of samples (volume, mass, geometry) must been selected in such a way as to give results of laboratory measurements with a minimum 95% confidence level.

Two floods occured in the pumps room in 1972 due to an overflow from the 300 m^3 tank into the buffer reservoir and the pump room. The influence of these floods is pointed out by the results of the radiological characterization for rooms 31 and 30

Room 30 is fully and highly contaminated. Contamination penetrated into the wall and the depth penetration is about 2 mm.

- Nuclide vector: Cs-134, Cs-137 and Co-60. Cs-134 is the predominant radionuclide.
- Contamination level: max. 103.3 Bq/cm^2.

This room must be decontaminated taking into account the contamination depth penetration.

Room 31 is highly contaminated. Contamination penetrated into the floor and the depth penetration is about 2 cm. Contamination is concentrated behind pumps and their pedestal.

-Nuclide vector: Co-60 and Cs-137. Co-60 is the predominant nuclide.

-Contamination level: max. 23.8 Bq/cm^2.

-Bulk contamination: max. 20.7 Bq/g.

This room must be decontaminated taking into account the contamination depth penetration.

2.2 Types of devices and instruments

Mechanical saw HILTI WSR 1200-PE *(*see. Fig. 4.) is an electrical tool-machine used to cut the samples from wood, plastic, metal and composite, with the following technical characteristics: power – 1150W, revolution: 1200-2600 rot/min, cutting 32mm.

Fig. 4 Mechanical saw - HILTI WSR 1200-PE

Diamond drill equipment for concrete core extraction - *DD200 (*<u>see. Fig. 5)</u> is a tool-machine used for concrete sampling through coring, with following technical characteristics: power-25000W, revolution: 320/550/1120 rot/min, core max.diameter–25/250mm.

Fig. 5 Diamond drill equipment for concrete core extraction - *DD200*

Device for sampling through stamping, type *TRUMPH Nibbler N-1000-0 (*see. Fig. 6*)*, is a tool-machine electrically driven, used for sampling from steel, Al, non ferrous, hard metals and plastics through stamping, with following technical characteristics: power-2000W, cadence: 369 cycles/min, width of the cut material 4-12 mm.

Fig. 6 Device for sampling through stamping, type *TRUMPH Nibbler N-1000-0*

Discs for smears sampling (see. Fig. 7) made of Watman filter paper with a diameter of 47 mm.

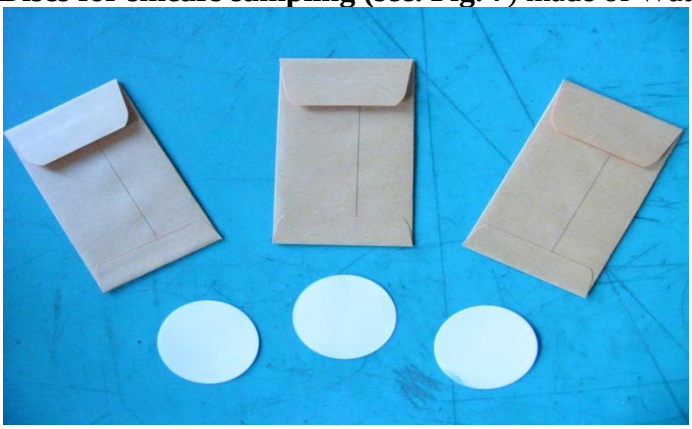

Fig. 7 Discs for smears sampling

Vacuum-cleaner with *HEPA- NILFISK filter* (see. Fig. 8) is equipment used for clean-up and decontamination operations with the following technical characteristics: power-1200W, air rate – 38 l/sec., sack capacity-6.25l.

Fig.8 Vacuum-cleaner with HEPA- NILFISK filter

The contamination and/or alpha, beta, gamma activity and dose rate measurements will be made in situ with the following instruments:

Universal Monitor, LB 123 "UMo", type EG&G BERTHOLD (see Figure 9) with the following components:

a) For contamination measurements:
   - LB 1233 with Alpha-Beta-P10 gas flow- counter tube LB 6359;
   - LB 1231 with Beta-Gamma, Xenon detector LB 6357;

b) For activity measurements:
   - LB 1238 proportional end-window detector (window diameter = 29 mm)

c) For beta-gamma dose rate measurements:
   - LB 1236 with proportional counter tube LB 6006;

Fig. 9 Universal Monitor, LB 123 "UMo", type EG&G BERTHOLD

EBERLINE E-600 **Monitor** (see Figure 10) used for alpha, beta and gamma measurements.

- Operating ways: rate-meter, integration, scaler and 'peak hold'
- Great selecting variety of units for dose and dose rate.
- Total and net measurements
- Memory -500 measurements

- Automatic settings when it is used with the " intelligent" probes
- Internal storing of the parameters for the "non-intelligent" probes
- Can be configured for any type of detector.

Fig. 10 EBERLINE E-600 Monitor

EBERLINE Monitor for Beta aerosols, type AMS-4 (see Fig.11).
It is an instrument used for detection of the beta emitting aerosols. It consists in a basic processor and a sampling cap.Detector type: Sealed Proportional Detector filled with Argon –CO2.

Fig. 11. EBERLINE Monitor for Beta aerosols, type AMS-4

SPECTROMETRY ANALYSIS INSTRUMENTS

Spectrometric analysis will be done in situ and/or in the laboratory. These consist in measurements of the radionuclides spectrum and element's composition of samples.

Radionuclides measurements will be made by γ spectrometry using:

In laboratory: Gamma Spectrometer Object Counting System with large hyper pure germanium (HPGe) detector, EG&G ORTEC (CPR laboratory).

b) In situ: γ Portable Spectrometer - ORTEC, DigiDART (see Figure 12), with NaI detector.

Fig. 12 γ Portable Spectrometer - ORTEC, DigiDART

Characterization Tasks List

In this stage, characterization tasks list is:
- Under Reactor Room (room No. 30);
- Pumps Room (room No. 31);

The work planning for sampling and measuring will be performed in compliance with preliminary radiological data provided in Figure 13, All areas identified with activities higher then local background will be characterized to identify the contaminants, their activities and to obtain their limits and the depth of contamination.
Before beginning the work, air samples will be weekly taken from the Reactor Hall Room, room under Reactor, Pumps Room, Hot Cells, Uranium Processing Laboratory and the Technological Ventilation Building.
During work, air samples from the above areas will be daily taken.

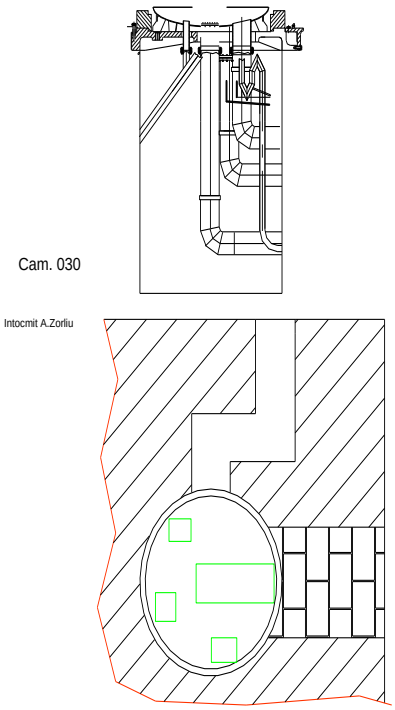

Fig.13 Preliminary radiological characterization in room 30

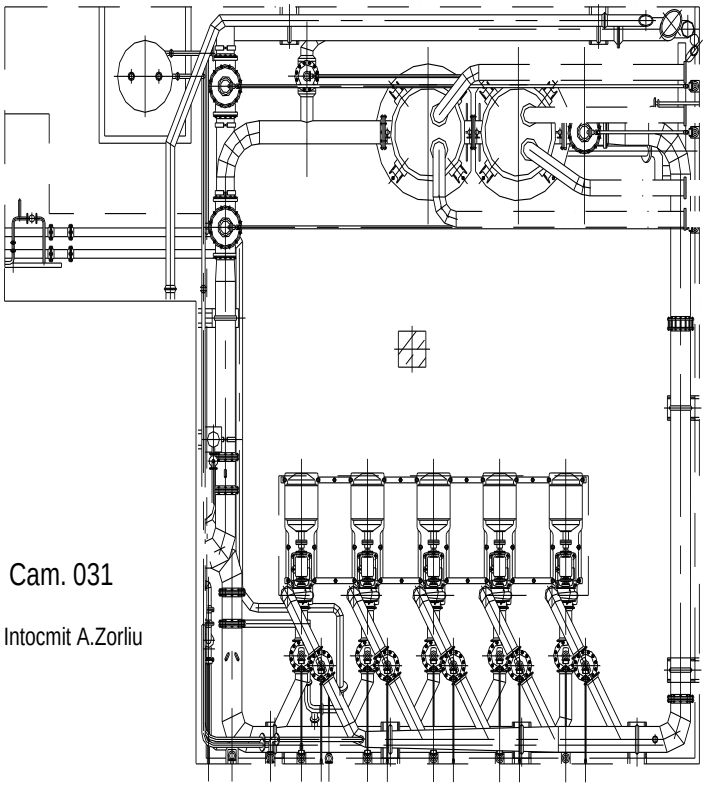

Fig.14 Preliminary radiological characterization in room 31

2.3 Operation procedures chosen for measurements

Room under Reactor (room 30) (Fig. 15)

- Scan the floor and walls.
- Record the average values.
- Identify the hot spots; take measurements of the dose rate with LB 6006 and in situ measurements.
- Survey alpha, beta, gamma gross activity for contaminated areas.
- Smear a 100 cm^2 surface where contamination is found (alpha, beta, gamma gross activity).
- Take two smears at random from the floor and walls to determine alpha, beta, gamma gross activity. If hot spots are found, take samples from these areas.
- Scan contaminated areas to determinate alpha activity. If alpha activity is found, take in situ measurements.
- If $\beta\gamma$ activity is found, submit one smear sample for γ spec. If alpha contamination is found, submit one smear sample for α spec.
- Scan and smear the ceiling (made of cast iron sustaining plate) and take 4 smears at random.
- Drill the plate and submit some samples from the location with the highest activity for γ spectrometry.
- Scan the pipes and take smears from the flanges area.
- Measure lead in paint with XRF.

Pumps Room (room 31) (Fig. 16)

- Scan and smear the floor and walls of the access corridor (up to the 2 m above the floor).
- Scan and smear the room's floor.
- Scan and smear the room's equipment.
- Record the areas with the highest dose rate.
- Scan the walls and ceiling.
- Take a smear from each wall and the area with the highest activity.
- Open the pipes flanges from the five water pumps of the primary circuit.
- Take smears and samples from waste if found.
- Prepare a complex sample and analyze it by γ spectrometry.
- Take a sample from the ion exchange resin. Analyze it to detect heavy metals by γ spectrometry.
- Take an oil sample from the gearing of the pumps. Analyze it to detect PCB, heavy metals by γ spectrometry.
- Check the flanges' clamps and walls anchors of the pipes for asbestos.
- Take a sample from the stainless steel drain sewer. Analyze it.
- Drill a 50 mm diameter core with a depth of 100 mm from the most contaminated fixed area from the pumps concrete bedplate.
- Drill a 50 mm diameter core with a depth of 100 mm from the most contaminated fixed area from the concrete of the drain sewer.
- Drill a 50 mm diameter core with a depth of 100 mm from the most contaminated fixed area from the pumps room's floor.

- Scan and smear the cores. Mark the location with the highest dose rate.
- Take paint samples from walls, ceiling and equipment,
- Measure lead in paint with XRF.

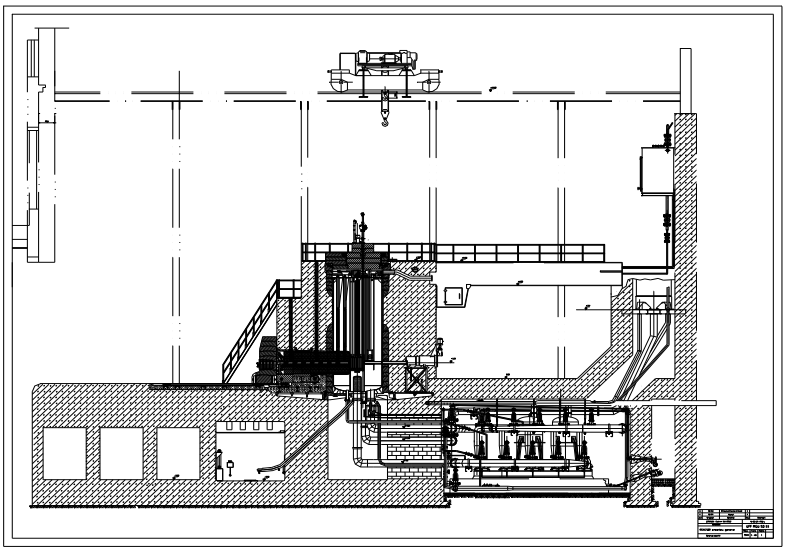

Fig. 15 Room 30 position in Nuclear Reactor – profile view

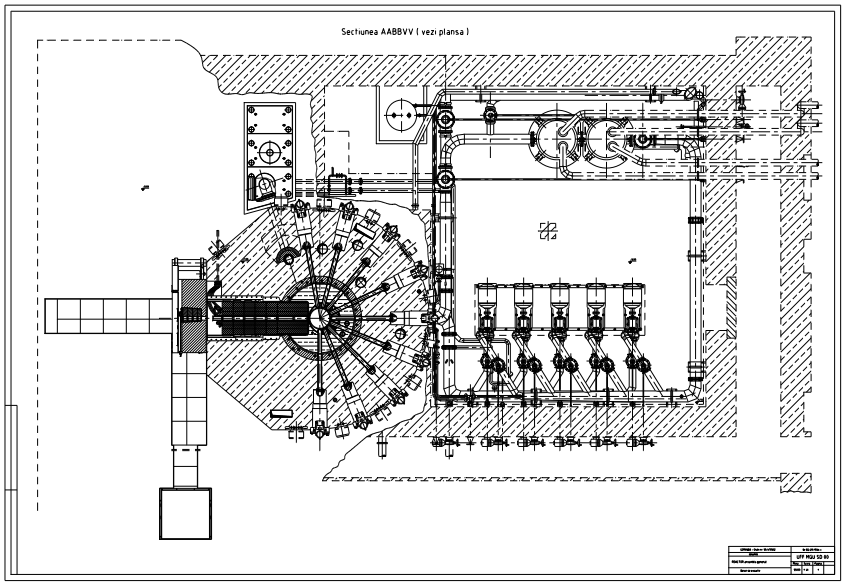

Fig. 16 Room 31 position in Nuclear Reactor – above view

3. METHODOLOGY OF MEASUREMENTS

3.1 Data Conversion

Radiological survey data is usually obtained in units, such as counts per unit time, which have no intrinsic meaning relative to the guideline values. Therefore, the survey data from field and laboratory measurements are converted to units, which will enable comparisons. Standard units used for expressing survey findings are:

- Surface Contamination - Bq/cm^2 (disintegrations per second per cm^2)
- Radionuclide Concentration - Bq/g (disintegrations per second per gram)
- Dose Rate - μSv/h (microsievert per hour)

3.2 Surface Activity

A measurement for surface activity is performed over an area, represented by the sensitive surface area of the detector. If the instrument display is in count rate, i.e. counts per second, the conversion to Bq/cm^2 is performed by:

$$A_s = (n - n_B) / (E \bullet W) \quad (1)$$

$$[Bq/cm^2] = [counts/second]/ ([counts/decay] \bullet [cm^2])$$

where

A_s = surface activity, in Bq/cm^2;
N = count rate, in counts per second;
n_B = background count rate, in counts per second;
E = efficiency, in counts per disintegration (counts/decay);
W = surface of the detector window, in cm^2.

For a technique using an integrated count on a digital instrument the conversion is:

$$A_s = (N-N_B) / (t \bullet E \bullet W) \quad (2)$$

$$[Bq/cm^2] = [counts] / ([seconds] \bullet [counts/decay] \bullet [cm^2])$$

where

N = total counts recorded by the measurement;
N_B = count during recording period, due only to background levels of radiation;
t = time period over which the count was recorded, in seconds.

The level of removable activity collected by a smear is calculated in the same manner:

$$A_s = (N-N_B) / (t \bullet E \bullet S \bullet F) \quad (3)$$

$$[Bq/cm^2] = [counts] / ([seconds] \bullet [counts/decay] \bullet [cm^2])$$

where

S = the area smeared, in cm^2;
F = removal factor (unitless).

3.3 Radionuclide Concentration

To determine the radionuclide concentration in Bq/g, the calculation performed is:

$$A = (N-N_B) / (t \bullet E \bullet M) \quad (4)$$
$$[Bq/g] = [counts] / ([seconds] \bullet [counts/decay] \bullet [g])$$

If the analytical procedure includes a wet chemistry separation, it will also be necessary to correct for the fractional recovery (R), determined by a spike or tracer added to the sample.

$$A = (N-N_B) / (t \bullet E \bullet M \bullet R) \quad (5)$$
$$[Bq/g] = [counts] / ([seconds] \bullet [counts/decay] \bullet [g])$$

where

M = weight of sample, in grams

3.4 Dose Rate

If an instrument, such as a pressurized ionization chamber or a dose rate meter is used for measuring exposure rate, the instrument reading will be directly in the desired exposure rate units (μSv/h). A gamma scintillation or GM detector with a count rate or digital scaling instrument provides data in units of counts per second or per some preset time, respectively. Conversion to μSv/h is accomplished, using calibration factors developed for the specific instrument and survey site. The background exposure rate is then subtracted from their total to determine the net level. This net level is compared with the guideline value.

$$D = (n–n_B) \bullet C \quad (6)$$
$$[\mu Sv/h] = [counts/second] \bullet [[\mu Sv/h]/[counts/second]]$$

or

$$D = (N–N_B) \bullet C / t$$
$$[\mu Sv/h] = [counts] \bullet [[\mu Sv/h]/[counts/second]] / [seconds]$$

where

D = dose rate, in micro sievert per hour;
C = calibration factor.

For UMo LB 123 monitor and dose rate probe LB 1236, the calibration factor is, for example, 0.20 μSv/h per 1 counts/second.

3.5 Measurement Uncertainty

It is recommended that each reported value include an assessment of its uncertainty. The rate of radioactive decay is not constant with time and is therefore described by a Poisson probability distribution. Based on such a distribution, the best estimate of the standard deviation (s) on a number of counts (N) is the square root of the counts, i.e.

$$s = \sqrt{N} \qquad (7)$$

and the standard deviation in a count rate over time (t) is therefore:

$$s_r = \frac{\sqrt{N}}{t} \qquad (8)$$

The ratio of the standard deviation to the total count (s/N) obviously decreases with the total count; in other words, the greater the number of counts recorded, the less the relative uncertainty in the measurement.

For the majority of measurements, the number of counts due only to background will be a significant portion of the total count. The background also has an uncertainty associated with it, which is taken into consideration by:

$$s_r = \sqrt{\frac{N}{t^2} + \frac{N_B}{t_B^2}} \qquad (9)$$

where t_B is the time period over which the background was determined.

The standard deviation or uncertainty in the count or count rate is converted to the same standard units used to express the measurement value, by use of the equations provided in Section 6.1. For survey and laboratory analytical data, the uncertainty is usually given at the 95 % confidence level, which requires multiplying the standard deviation value, by a factor of 1.96.

Unfortunately, the uncertainty described above, commonly referred to as the "counting error", is only that due to the uncertainty in the decay process. Other sources of uncertainty will be present in the measurement and in other parameters used in the conversion calculations. Examples include the counting time, distance and area measurements, instrument efficiency, laboratory weights and physical measurements (e.g. pipetting), and chemical recovery factors. The total uncertainty associated with a particular type of measurement can be determined empirically by performing repeat (6 to 10 recommended) measurements of several selected locations and determining the average and standard deviation of the data. This will provide an estimate of the upper bound on the magnitude of systematic uncertainties.

3.6 Minimum Detectable Activity

The **detection sensitivity** of a measurement system refers to the statistically determined quantity of radioactive material or radiation that can be measured or detected at a preselected confidence level. This sensitivity is a factor of both the instrumentation and the technique or procedure being used. Detection sensitivity has been defined as that level above which there is less than a 5 %

probability that radioactivity will be reported present when it is really absent (Type I error) or reported absent when it really is present (Type II error).

Two terms used when referring to detection sensitivity are the lower limit of detection and the minimum detectable activity. The **lower limit of detection** is an *a priori* estimated detection capability, related to the characteristics of the instrumentation. **Minimum detectable activity (MDA)** is an *a priori* estimate of the minimum activity level, which is practically measurable with a specific instrument and sampling and/or measurement technique. The basic relationship for estimating the MDA is:

$$MDA = K(2.71 + 4.65 s_b) \qquad (10)$$

where K is a proportionality constant relating the detector response (in counts) to an activity concentration and s_b is the standard deviation of the background.

3.7 Surface Activity Measurement

For integrated measurements over a preset time, MDA for surface activity is given by:

$$MDA = \frac{2.71 + 4.65\sqrt{B_R . t}}{t.E.A} \qquad (11)$$

where: MDA = activity level in Bq/cm^2
B_R = background rate in counts/second
t = counting time in seconds
E = detector efficiency in counts/disintegration
A = active probe area in cm^2

The MDA of a counter instrument for surface activity measurements can be approximated by taking twice the time constant of the meter as the counting time and using the relationship:

$$MDA = \frac{4.65\sqrt{\frac{B_R}{2.t_c}}}{E.A} \qquad (12)$$

where t_c is the meter time constant in seconds.

3.8. Scanning

The ability to identify a small region or area of slightly high radiation during surface scanning is dependent upon the surveyor's skill in recognizing an increase in the audible output of the instrument. Experience has shown that a 25% to 50% increase may be easily identifiable at ambient background levels of several thousand counts per minute. At ambient levels of a few counts per minute, a two to three fold increase in the audible signal is required before a change is readily recognizable. The detection sensitivity of scanning is dependent upon a number of other factors, such as detector speed, size of high activity region, level of activity, detector/surface distance. Therefore, the ability to detect a region of high activity using a particular survey scanning technique should be determined empirically. A rough estimate of the MDA can be calculated by substituting the audibly discernable increase in count rate for the numerator in equation 6-12.

3.9 Laboratory Analyses

Additional factors may be introduced into the calculation for estimating detection sensitivities for laboratory analyses. Examples of these factors are: chemical recovery, sample size, and emission abundance for specific radiations of interest in the analytical process. An example of a calculation for a typical lab procedure would be:

$$MDA = \frac{2.71 + \sqrt{B_R . t}}{t.E.S.Y} \qquad (13)$$

where MDA = activity in Bq/g
B_R = background rate in counts/second
t = counting time in seconds
E = detector efficiency in counts/disintegration
S = samples size in grams
Y = other factors such as percent chemical recovery and number of emissions of radiation being measured per disintegration of the radionuclide.

3.10. General Considerations

In applications, the system should be capable of measuring levels below 75%, and preferably at or below 10%, of an established guideline value. Many of the radiological instruments and monitoring techniques used for applied health physics activities in an operating facility may not provide the detection sensitivities necessary to demonstrate compliance with the guideline levels.

As described above, parameters which will determine the detection sensitivity of a system are: background level, detection efficiency, measurement (or counting) time and sample size or area.

The detection sensitivity for a given application can be improved, (i.e. lowered) by:
(1) selecting an instrument with a higher efficiency or a lower background;
(2) increasing the counting time;
(3) increasing chemical recovery;
(4) increasing the size of the sample or the effective probe's area.

Increasing efficiency, recovery, and sample or area size has the effect of lowering the MDA in direct proportion to the amount of change. For example, selecting a detector with twice the active probe area will decrease the MDA by a factor of two (assuming all other parameters remain unchanged). Changes in background rate or counting time affect the MDA proportional to the square root of the change. If the background rate is increased by a factor of two and all other parameters remain unchanged, the MDA will be increased by a factor of 1.414. Doubling the counting time has the net effect of lowering the MDA by a factor of 1.414.

For the purposes of a careful data presentation the minimum detectable activity (MDA) for each measurement procedure (and each instrument if more than one instrument is used for a given procedure) is calculated. Data from surveys will often be close to the background levels and/or below the detection sensitivity (MDA) of the procedure. Therefore, negative data will be a frequent result of calculations. Use of the MDA for data that has a value less than the MDA is a common practice. This approach enables the surveyor to significantly reduce the number of calculations; however, use of the MDA, instead of actual data when calculating averages, will bias the results on the high side and the true conditions of the site will not be described. Substituting MDA for actual data will also result in overestimates of source inventory and dose assessments, possibly leading to

decisions for further actions that may not be justified. Finally, when evaluating data distributions, i.e. in a normalcy test, use of MDA will result in a skewed distribution and may lead to an incorrect conclusion regarding the distribution. To avoid the pitfalls associated with use of MDA, it is recommended that actual data be presented and used for calculation purposes. One exception to this approach might be the use of MDA for averaging site activity levels, when the MDA is small in comparison to the applicable guideline (small may be considered as less than 10% of the guideline value).

3.11 Format for Data Presentation

All data from surveys should be presented in a format which provides (1) the calculated surface activity or specific radionuclide concentration value; (2) the estimated uncertainty at the 95 % confidence level for that value; and (3) the estimated MDA for the measurement. An example of such a format would be:

Sample ID	Radionuclide Concentration (Bq/g)		
	Activity	Uncertainty (95% confidence level)	MDA
001			
002			
003			

In expressing survey results, the number of significant figures is also important. The reason is that data should be reasonable and should not mislead or imply a false level of accuracy in reported values. The appropriate number of digits in a value depends upon the magnitude of the uncertainty attached to that value. In general, final survey data, which are usually in the range of environmental data, seldom justify more than two or three significant figures for the value and one or two significant figures for the uncertainty. The number of significant figures in the uncertainty is first determined and the value is stated to the last place affected by the uncertainty term.

During arithmetic calculations all decimals should be retained and the results rounded to the desired number of significant figures. Rounding is done by increasing the last digit with 1, if the value that is dropped is equal to or greater than $^1/_2$; if the value is less than $^1/_2$, the last digit is left as it is.

3.12 Removable Activity

Data for removable activity levels are compared directly to the guideline values. The limit for removable activity is 20% of the guideline value for total surface activity. If that level is exceeded, remediation and resurvey are necessary.

3.13 Areas with High Activity

Levels of residual activity, which exceed the guideline value, are compared directly to the guideline. Residual activity exceeding this limit should be remediated and followed-up through new surveys.

Areas with high activity between one and three times the guideline value are then tested to ensure that the average surface activity level within a contiguous 1 m^2 area containing the area with high activity, is less than the guideline value. To evaluate whether this condition is satisfied or not, additional measurements are taken to determine the activity level and the size of the area with high activity .The average (weighted average) in the 1 m^2 area is then calculated, taking into consideration the relative fraction of the 1 m^2 occupied by the area(s) with high activity, using the relation below:

$$\overline{\Lambda}_{sw} = \frac{1}{n_s}\sum_{i=1}^{n_s}\Lambda_{si}\left(1-\sum_{k=1}^{n_k}A_k\right)+\sum_{k=1}^{n_k}\Lambda_{sk}A_k \qquad (14)$$

where $\overline{\Lambda}_{sw}$ = weighted mean of the surface activity including area(s) with high activity
Λ_{si} = systematic and random measurements in point "i" from low activity area
n_s = number of systematic and random measurements
Λ_{sk} = elevated area activity in area k
A_k = fraction of 1 m2 occupied by area k with high activity
n_k = number of areas with high activity.

3.14 Exposure Rates

Exposure rate levels are compared directly to the guideline value. The maximum exposure rate may not exceed two times the guideline value, above background. If the level is above that value, the area should be remediated and surveyed again.

3.15 Calculation of Average Level

Usually, the guideline values of surface activity, soil activity, and exposure rate are average values, above background, established for areas of survey unit surfaces (surface activity). To enable comparison of the survey data to those guidelines, the average of measurements in each of the survey units is calculated using all measurements n within that area:

$$\bar{x} = \frac{1}{n}\sum_{i=1}^{n}x_i \qquad (15)$$

3.16 Comparisons

Average levels, calculated following the procedures in Section 3.14, are compared to the guideline values and conditions. If the averages exceed the applicable guideline values and/or conditions, further remediation is required and follow-up measurements are performed to verify the effectiveness of the actions. After the averages satisfy the guideline values and conditions, the results are further evaluated to determine whether the data for each survey unit (i.e. group of contiguous grids or regions with the same classification of contamination potential), provides a 95% confidence level that the real average activity level meets the guidelines.
The test is performed by calculating the average (equation 3-12) and standard deviation of the data for a particular radiological parameter in each survey unit using all measurement locations; the standard deviation is calculated by:

$$s_x = \sqrt{\frac{\sum_{i=1}^{n}(x_i - \bar{x})^2}{n-1}} \qquad (16)$$

The following equation is recommended for testing data, relative to a guideline value, with a certain level of confidence:

$$\mu_\alpha = \bar{x} + t_{1-\alpha,df}\,\frac{s_x}{\sqrt{n}} \qquad (17)$$

where $t_{1-\alpha, df}$=95% confidence level (NUREG/CR-5849, Appendix B, Table B-l);

df=degrees of freedom (n-1);

α=false positive probability, i.e. the probability that μα is less than the guideline value if the true mean activity level is equal to the guideline value;

$\bar{x}$=calculated mean from equation (3-12);

s_x=standard deviation from equation (3.13);

n=number of individual data points used to determine x and s_x.

The value of μ_α is compared to the guideline value; if μ_α is less than the guideline, the area being tested meets the guideline at a 95 % confidence level. This means that the probability is less than 5 % that μ_α will pass the test, when the true mean activity level exceeds the guideline value. Areas for which μ_α is smaller than guideline values by this testing procedure are considered acceptable and no further survey actions are required. If the mean value exceeds the guideline value, the area is not acceptable and further cleanup is required. If the mean value is less than the guideline value, but the test of confidence is inconclusive, i.e. $\bar{x}$ < guideline value < μ_α, either (1) further cleanup with follow-up measurements /sampling or (2) additional measurements/sampling may be conducted.

3.17 Background Survey

Because guidelines for radioactivity are presented in terms of radiation levels or activity levels above normal background for area or facility, it will also be necessary to perform a background survey. This survey will require measuring both direct radiation levels (usually gamma exposure rates) and concentrations of the potential radionuclide contaminants in the vicinity of the site. Where only gamma emitting contaminants are found, it may be adequate to perform only background exposure rate determinations.

Background is determined by measurements and/or sampling at locations on site or in the immediate vicinity of the site, which are unaffected by site operations. Preferable locations for interior background determinations are within on-site buildings of similar construction, but with no history of licensed operations. Background direct radiation levels within buildings may differ from those in open land areas, because of the presence of naturally occurring radioactive materials in construction materials and the shielding effect of those construction materials. Background samples and measurements for land areas should be collected at locations which are unaffected by effluent releases and other site operations. Locations of potential overflow from areas with surface contamination should also be avoided.

Because the background levels will be subtracted from the total radiation or radioactivity levels to determine the net activity, it is necessary that background be determined with a detection sensitivity and

accuracy at least equivalent to data from which it will be subtracted. This can be achieved by using the same instruments and techniques for background surveys as those used in surveying site conditions.

The degree to which the average background is representative of the true background level is a factor in determining the number of background measurements required. Experience has indicated the variance in the average background value from a set of 6 to 10 measurements will usually not exceed $\pm$ 40% to 60% of the average at the 95% confidence level. It is recommended that 6 to 10 measurements for each parameter of concern should be initially performed and the average and 95% confidence level should be determined. If the upper 95% level bound on the background average is less than 10% of the guideline value for that parameter, variations in background may be considered insignificant and no further determination are necessary. If the upper 95% level bound on the background average is greater than 10% of the guideline value, the background data should be tested to assure that the average represents the true mean to within $\pm$ 20% at the 95% confidence level. If necessary, additional background determinations should be performed to satisfy this level of representativeness. The total number of background measurements necessary to satisfy the objective is calculated by:

$$n_B = \left(\frac{t_{0.975,df} \cdot s_x}{0.2.\bar{x}_B} \right)^2 \qquad (18)$$

where n_B= number of background measurements required

$\bar{x}_B$= mean of initial background measurements

s_x= standard deviation of initial background measurements

$t_{0.97.5,df}$ =t statistic for 97.5% confidence at df = n-1 degrees of freedom (NUREG/CR-5849, Appendix B, Table B-l), where n is the number of background survey initial data points.

Subtracting the number of data points already collected (n) from this total calculated number (n_B), you determine the number of additional measurements or samples which will be required to demonstrate the desired confidence level of the data. If this calculation indicates that additional background data are necessary, it is recommended that they should be collected uniformly over the area, using the same sampling methodology as used for the initial samples. The average background is then recalculated using all data points. The flow diagram for interpreting and comparing survey data with guideline values is presented in Fig. 17.

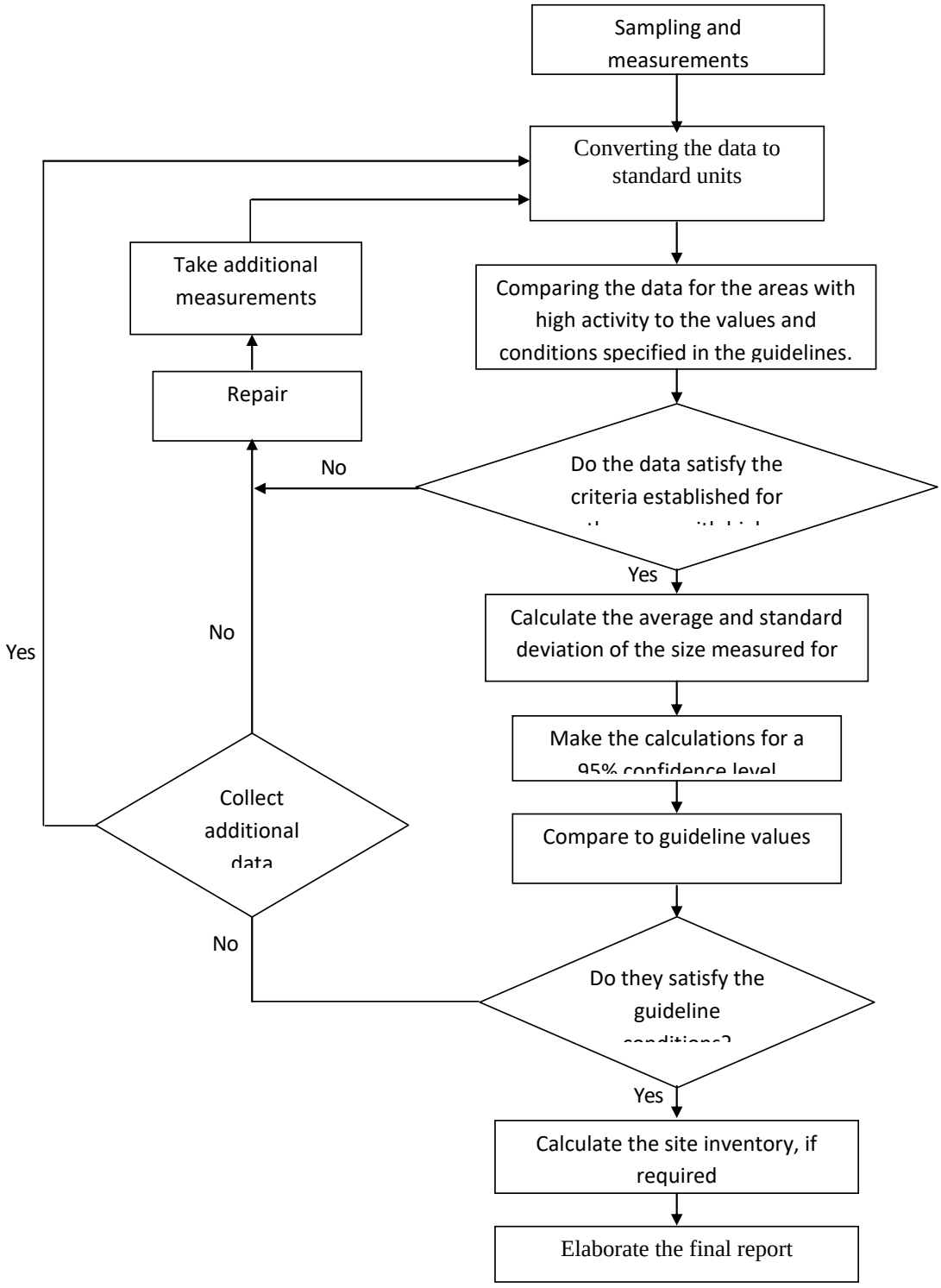

Fig. 17 Diagram of the process for interpreting and comparing the monitoring data to the release levels

4. DATA MEASUREMENTS RESULTS

We determined the depth penetration of the contamination. Thus, we extracted concrete cylinders (diameter=5cm) from the floor in room 31. Samples were cut from cylinders and then measured by gamma spectrometry. Measurement results are given in Table 1 As it is shown in this appendix, the depth penetration is about 2 cm.

In room 30, layers were removed from the wall. After each layer removal, we measured the surface activity. Results are given at the end of Table 2. In this case, the depth penetration is about 2 mm.

Table 1. Contamination depth penetration-Room 31

Point no. (see App.17)	Sample code	Depth (cm)	Specific activity (Bq/g)		
			Co-60	Cs-137	Cs-134
CDP1	1	0.00	0.00	0.00	0.00
CDP2	2	0.00	0.00	0.00	0.00
CDP3	3	0.00	13.10	1.69	0.00
CDP4	4	0.00	0.18	0.05	0.00
CDP5	5	0.00	20.68	4.39	0.40
CDP6	6	0.00	11.25	2.48	0.00
CDP7	7	0.00	0.00	0.00	0.00
CDP8	8	0.00	0.00	0.00	0.00
CDP9	9	0.00	0.00	0.00	0.00
CDP10	10	0.00	0.00	0.00	0.00
CDP11	11	0.00	0.00	0.00	0.00
CDP12	12	0.00	0.00	0.00	0.00
CDP5	136	10.0	0.00	0.00	0.00
CDP5	388	2.00	0.00	0.00	0.00
CDP5	389	4.00	0.00	0.00	0.00
CDP5	390	7.00	0.00	0.00	0.00

Table 2 Contamination depth penetration –Wall of the Room 30

Radiation Monitor E-600 SN02996–Probe HP-210T SN727425				
Depth (mm)	Background (cps)	Counting rate (cps)	Contamination (Bq/cm^2)	Remarks
0	2.7	22.4	6.57	Point W9 (Middle of cell-9, High=1.5m)
1	2.4	6.0	1.20	Layer of 1 mm was removed from the wall.
2	2.3	3.5	0.4	Another layer of 1 mm was removed from the wall.

REMARKS

- Dose-rate measurements were performed at 1 m above the floor.
- Measurement points from walls correspond to the middle of the cell at 2±0.5 m hight.

From Table 3, we identified 10 (ten) areas of elevated activity (surface activity higher than exclusion level) in the Room 30.

Table 3. Areas of elevated activity in the Room 30

Point no.	Surface (cm^2)	Smear code	Nuclide vector	Surface activity (direct measurements) (Bq/cm2)	Surface activity (smears) (Bq/cm^2)	Removal factor
F6	~100	-	Cs-134, Co-60,Cs-137	4.71±0.9	-	-
F7	~100	-	Cs-134, Co-60,Cs-137	3.52±0.7	-	-
F8-1	~100	349	Cs-134, Co-60,Cs-137	11.90±2	0.99±0.2	0.08
F9-1	~75	-	Cs-134, Co-60,Cs-137	41.20±8	-	-
F9-2	~75	-	Cs-134, Co-60,Cs-137	80.00±16	-	-
F9-3	~75	463	Cs-134-85%, Co-60-15%	103.33±20	12.25±2.4	0.12
F10	~100	350	Cs-134, Co-60,Cs-137	6.10±1.2	1.42±0.28	0.23
F11	~100	-	Cs-134, Co-60,Cs-137	10.66±2	-	-
F12	~150	351	Cs-134-79%, Co-60-18%, Cs-137-3%	22.09±4	12.48±2.4	0.56
W9	~50	459	Cs-134, Co-60,Cs-137	6.57±1.3	0.1±0.02	0.02

In the Room 31, we identified 5 (five) areas of elevated activity (surface activity higher than exclusion level). These areas are characterized in Table 4

Table 4. Areas of elevated activity in the Room 31

Point no.	Surface (cm^2)	Smear code	Sample code	Specific activity (Bq/g)		Surface activity (direct measurements) (Bq/cm2)	Surface activity (smears) (Bq/cm^2)	Removal factor
				Co-60	Cs-137			
F5.1-1	~200	362	-	-	-	4.17±0.9	0.96±0.2	0.23
F4.1-2	~200	363	-	-	-	23.77±4.6	0.55±0.1	0.02
CDP3	-	-	3	13.10±2.6	1.69±0.4	-	-	-
CDP5	-	-	5	20.68±4.2	4.39±0.8	-	-	-
CDP6	-	-	6	11.25±2.2	2.48±0.5	-	-	-

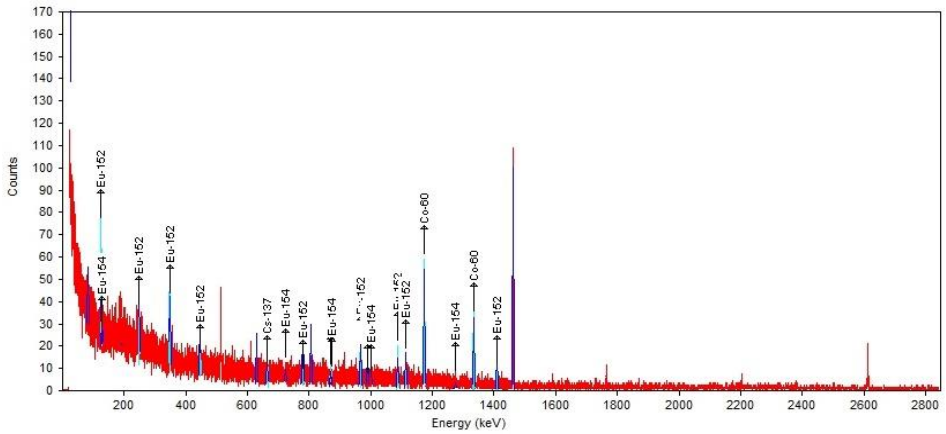

Figure 18 – Probe 1 spectrographic analysis from concrete floor drilling in room 31

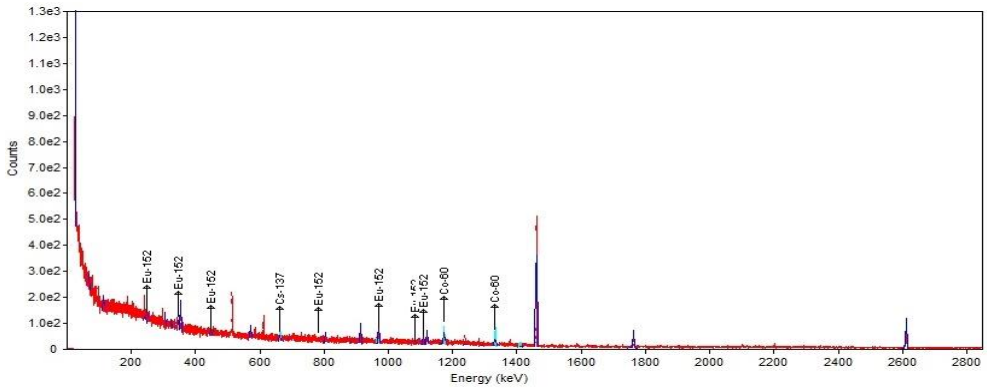

Figure 19 – Probe 2 spectrographic analysis from concrete floor drilling in room 31

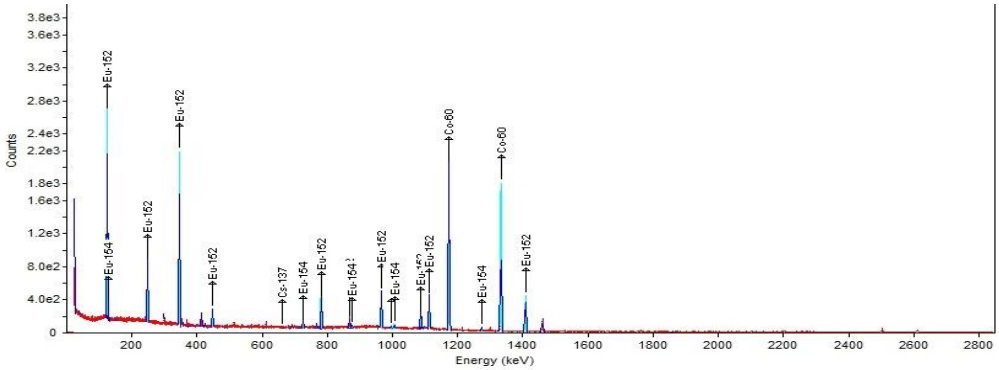

Figure 20 – Probe 3 spectrographic analysis from concrete floor drilling in room 31

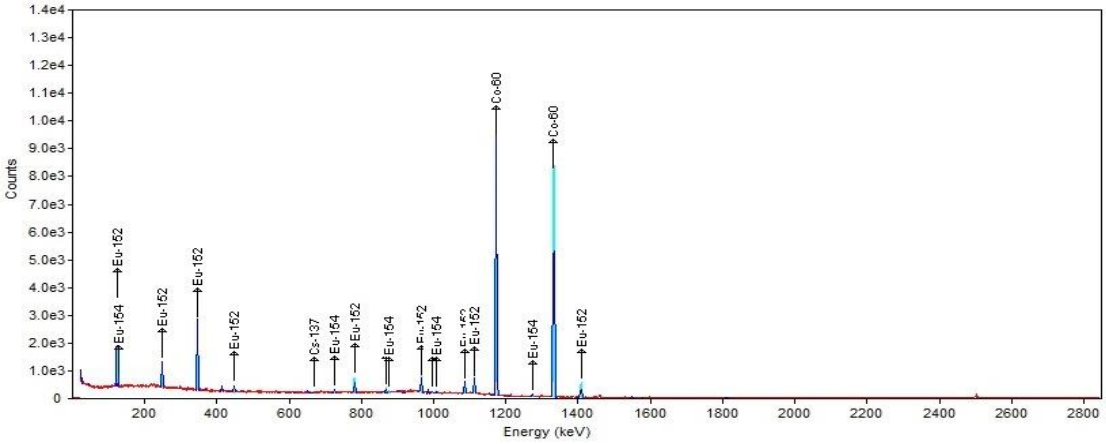

Figure 21 – Probe 4 spectrographic analysis from concrete floor drilling in room 31

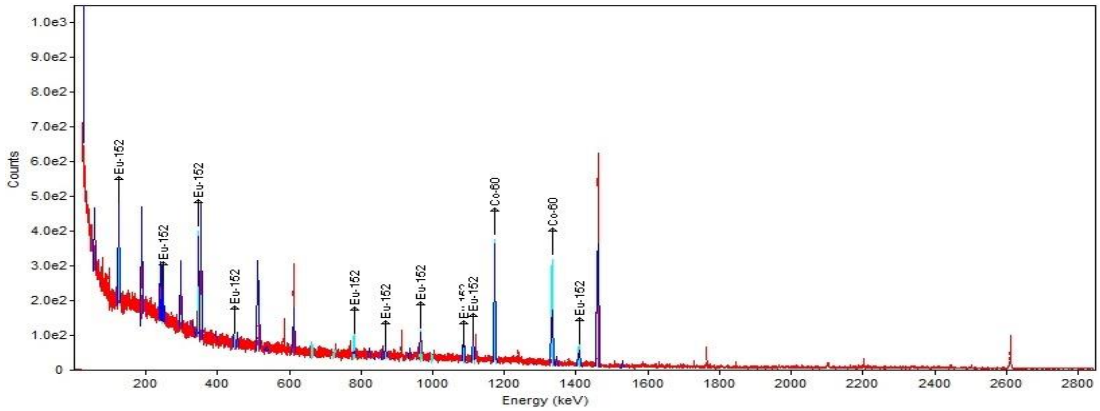

Figure 22 – Probe 5 spectrographic analysis from concrete floor drilling in room 31

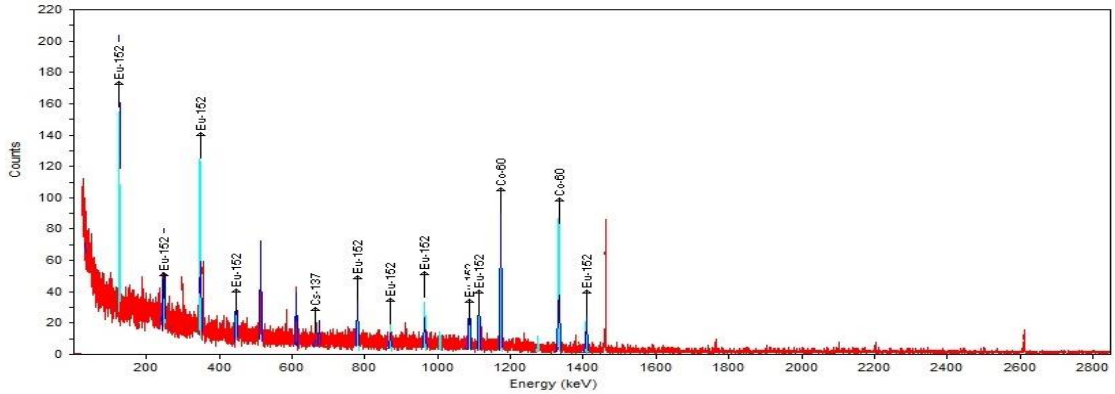

Figure 23 – Probe 6 spectrographic analysis from concrete floor drilling in room 31

5. DOSE AND RISK MODELING USING RESRAD COMPUTER CODES PACKAGE

5.1 RESRAD Build 3.5 Code

RESRAD is a computer code designed to estimate radiation doses and risks from RESidual RADioactive materials (Fig. 5) [8]. The only code designated by Department Of Energy for the evaluation of radioactively contaminated sites.

United States National Regulatory Commission (NRC) has approved the use of RESRAD for dose evaluation by licensees involved in decommissioning, NRC staff evaluation of waste disposal requests and dose evaluation of sites being reviewed by NRC.

RESRAD has been applied to over 300 sites in the U.S. and other countries. Environmental Protection Agency (EPA) Science Advisory Board reviewed the RESRAD model. EPA used RESRAD in rule-making on radiation site cleanup regulations.

RESRAD code has been verified and has undergone several benchmarking analyses, and has been included in the IAEA's VAMP and BIOMOVS II codes to compare environmental transport models. RESRAD training workshops have been held at DOE, NRC, and EPA headquarters. Around 800 people have been trained at these workshops and RESRAD has been used by several universities as a teaching tool as well.

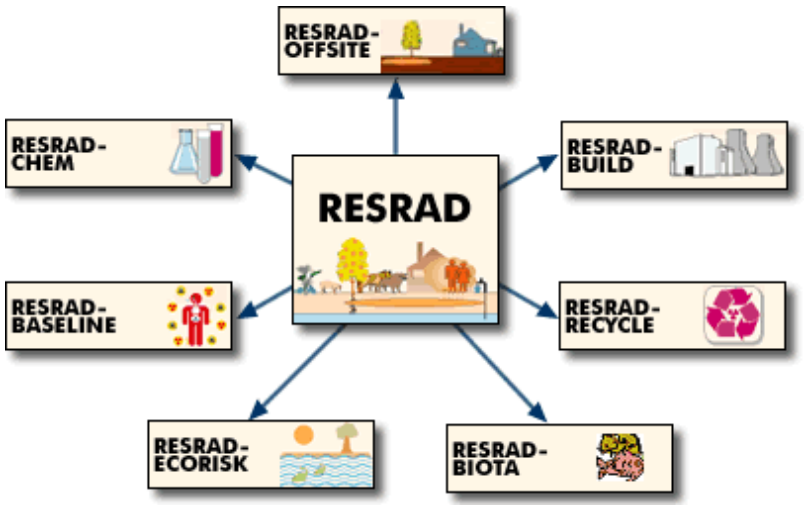

Fig. 24 – RESRAD computer code package [

The RESRAD Build Code is a model for analyzing the radiological doses resulting from the remediation and occupancy of buildings contaminated with radioactive material with the following features (Fig. 6) [9]:

- Considers external exposure, inhalation of dust and radon, and ingestion of soil/dust.
- Up to 10 sources and 10 receptors can be modeled.
- Sources geometry can be point, line, plane, or volume.
- Building can be any structure composed of up to three compartments.
- Radioactive contamination can be on surface or in building material.
- Exposure scenarios considered include building occupancy (residential use and office worker) and building remediation (decontamination worker and building renovation worker).

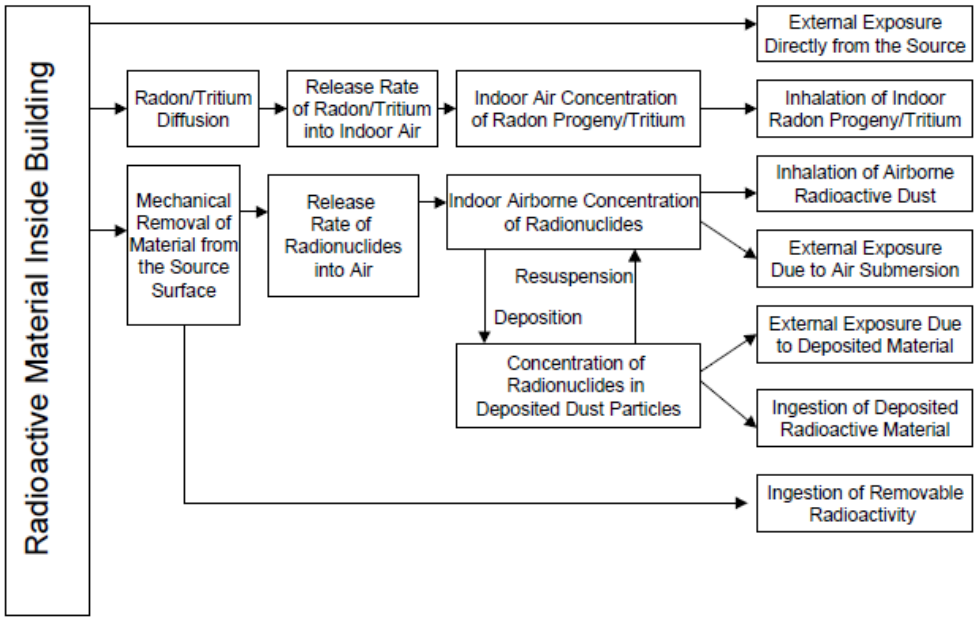

Fig. 25 - Exposure Pathways Incorporated into the RESRAD-BUILD Computer Code [9]

Table 4 Key parameters used in the building occupancy and building renovation scenarios [10]

Parameter	Unit	Building Occupancy[a]	Building Renovation[b]	Remarks
Exposure duration	days (d)	365.25	179.00	To match the occupancy period of 365.25 days in NUREG/CR-5512 building occupancy scenario (Beyeler et al. 1999 [8]) and renovation period of 179 days in NUREG/CR-5512 building renovation scenario (Wernig et al. 1999 [9]).
Indoor fraction	_[c]	0.267	0.351	To match the 97.5 d/yr time in building in NUREG/CR-5512 building occupancy scenario (Beyeler et al. 1999) and 62.83 days spent in the building during renovation period in NUREG/CR-5512 building renovation scenario (Wernig et al. 1999 [9]).
Receptor location	m	0, 0, 1	0, 0, 1	At 1-m from the center of the source.
Receptor inhalation rate	m^3/d	33.6	38.4	For building occupancy scenario it matches with 1.4 m^3/h breathing rates in NUREG/CR-5512 (Beyeler et al. 1999 [8]) and for building renovation scenario it matches with 1.6 m3/h breathing rate of moderate activity given in the EPA Exposure Factor Handbook (EPA 1997 [9]).
Receptor indirect ingestion rate	m^2/h	$1.12x10^{-4}$	0	Value for the building occupancy scenario is the mean value from the distribution and for the building renovation scenario it is assumed the ingestion is only from the direct contact with the source.
Source type -	_	Area	Volume	For building occupancy scenario it is assumed that contamination is only on the surfaces, whereas for the building renovation scenario is volumetric

Direct ingestion rate	1/h (area) g/h (volume)	3.06×10^{-6}	0.052	Calculated from the default ingestion rate of 1.1×10^{-4} m^2/h in NUREG/CR-5512 building occupancy scenario (Beyeler et al. 1999 [8]). The effective transfer rate from NUREG/CR-5512 building renovation scenario for ingestion of loose dust to the hands and mouth during building renovation (Wernig et al. 1999 [9]).
Air release fraction	_	0.357	0.1	For the building occupancy scenario, it is the mean value from the parameter distribution. For the building renovation scenario, a smaller fraction is breathable.
Removable fraction	_	0.1	NR[d]	10% of the contamination is removable (NUREG/CR-5512 building occupancy scenario default [10]). The parameter is not required for the volume source.
Time for source removal or source lifetime	d	10,000	NR	Value for the building occupancy scenario is the most likely value from the parameter distribution. The parameter is not required for the volume source.
Source erosion rate	cm/d	NR	4.1×10^{-4}	For the building renovation scenario, it is assumed that the total source thickness of 15 cm can be removed in 100 years of building life.

[a] Parameter values used in the building occupancy scenario.
[b] Parameter values used in the building renovation scenario.
[c] A dash indicates that the parameter is dimensionless.
[d] NR = parameter not required for the analysis.

5.2 Calculation of the intake rates and time integrated cancer risks

RESRAD is able to calculate lifetime cancer risks resulting from radiation exposure. The radiation risk can be computed by using the U.S. Environmental Protection Agency (EPA) risk coefficients with the exposure rate (for the external radiation pathways) or the total intake amount (for internal exposure pathways).

The EPA risk coefficients are estimates of risk per unit of exposure to radiation or intake of radionuclides that use age- and sex-specific coefficients for individual organs, along with organ-specific dose conversion factors (DCF). The EPA risk coefficients are characterized as best-estimate values of the age-averaged lifetime excess cancer incidence risk or cancer fatality risk per unit of intake or exposure for the radionuclide of concern. Detailed information on the derivation of EPA risk coefficients and their application can be found in several EPA documents [11] [12]. The methodology used in the RESRAD code for estimating cancer risks follows the EPA risk assessment guidance (EPA 1997) and is presented very briefly in the following section.

Intake rates calculated by the RESRAD code are listed by radionuclide and pathway and correspond to specific times. Intake rates for inhalation and ingestion pathways are calculated first for all of the principal radionuclides and then multiplied by the risk coefficients to estimate cancer risks.

For inhalation and soil ingestion pathways (p = 2 and 8, respectively), the intake rates (Bq/yr or pCi/yr) can be calculated by using the following Equation (1) [13]:

$$(Intake)_{j,p}(t) = \sum_{i=1}^{M} ETF_{j,p}(t) \times SF_{i,j}(t) \times S_i(O) \times BRF_{i,j}, \quad (1)$$

where:
$(Intake)_{j,p}(t)$ = intake rate of radionuclide j at time t (Bq/yr or pCi/yr),
M = the number of initially existent radionuclides,
$ETF_{j,p}(t)$ = environmental transport factor for radionuclide j at time t (g/yr),
p = primary index of pathway,
$SF_{ij}(t)$ = source factor,

i,j = index of radionuclide (i for the initially existent radionuclide and j for the radionuclides in the decay chain of radionuclide i),
$S_i(0)$= initial soil concentration of radionuclide i at time 0,

The cancer risk at a certain time point from external exposure can be estimated directly by using the risk coefficients, which are the excess cancer risks per year of exposure per unit of soil concentration. Because the risk coefficients are derived on the basis of the assumptions that the contamination source is infinite both in depth and lateral extension and that there is no cover material on top of the contaminated soil, it is necessary to modify the risk coefficients with the cover and depth, shape, and area factors to reflect the actual conditions. Non-continuous exposure throughout a year also requires that the occupancy factor be considered when calculating the cancer risks.

Consequently, the RESRAD code uses the environmental transport factor for the external radiation pathway, along with the risk coefficient and exposure duration, to calculate the excess cancer risks as seen in Equation (2) [13]:

$$(Cancerrisk)_{j,1}(t) = \sum_{i=1}^{M} ETF_{j,1}(t) \times SF_{ij}(t) \times S_i(O) \times BRF_{i,j} \times RC_{j,1} \times ED, \quad (2)$$

where:
$RC_{j,1}$ = risk coefficient for external radiation (risk/yr)/(pCi/g),
ED = exposure duration (30 yr).

For the inhalation and ingestion pathways, the cancer risks at a certain time point are calculated as the products of intake rates, risk coefficients, and exposure duration. Unlike the intake rates, cancer risks from inhalation of radon and its decay progeny are reported as total risks that include radon and progeny contributions. Therefore, the radon risks are the sums of the products of the individual radon or progeny intake rates and their risk coefficients.

5.3 Modeling results

In the present study the dose rate and the associated risk related to the clean-up and decontamination of pump room 31 was evaluated. Using the measured doses, the activity of these sources and the associated risk was estimated, due to the difficulty of the spectroscopy analysis in order to determine the exact nuclide sources and their activity. The presumed time for clean-up and decontamination operation is 1 month for each hot cell, and therefore the estimation was be made for this period. The dose rate and the associated risk related to the clean-up and decontamination of room 30 and 31 were calculated using RESRAD Build 3.5 code.

In the model 1 receptor was considered and 3 nuclide sources were taken into account: Co-60, Cs-137, Sr-90, because these nuclides are most probable the pum rooms. The activity of these sources was estimated by using Rad Pro 3.26 software models [15], starting from the measured doses. This was done, due to the lack of a difficult to perform spectroscopy analysis (inside the hot cell) in order to determine the exact nuclide sources and their activity.

In Fig. 26, a graphic is presented, showing the evolution of modeled hourly dose rate at the beginning of the operations, then after 1, 2, 3 and 4 weeks of clean-up operations when only the external pathways are taken into account. At the beginning of clean-up operations, the calculation revealed an initial equivalent hourly dose rate of 7 mSv (700 mrem). The relative high dose rate is due to the fact that receptor was modeled without a biological lead protection. The dose decreases in time towards insignificant values, as the hot cell no. 4 is cleared and decontaminated.

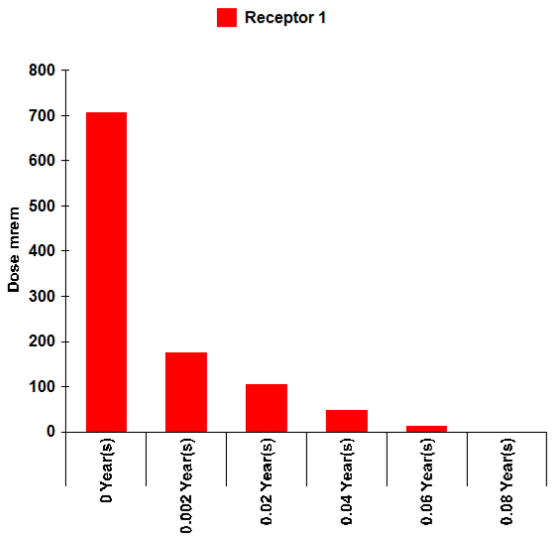

Fig. 26 – Dose received by time for summed nuclides and external pathways

As it can be seen in Fig. 27, in the unlikely scenario in which the exposure pathways are summed (inhalation, ingestion, immersion, external, deposition and radon) the equivalent dose rate becomes extremely high. The corresponding cancer risk, presented in Fig. 28, is higher than 100 for this particular scenario. Due to high dose/risks involved, complying with ALARA principle, the utilization of a robot is proposed, instead of a human operator.

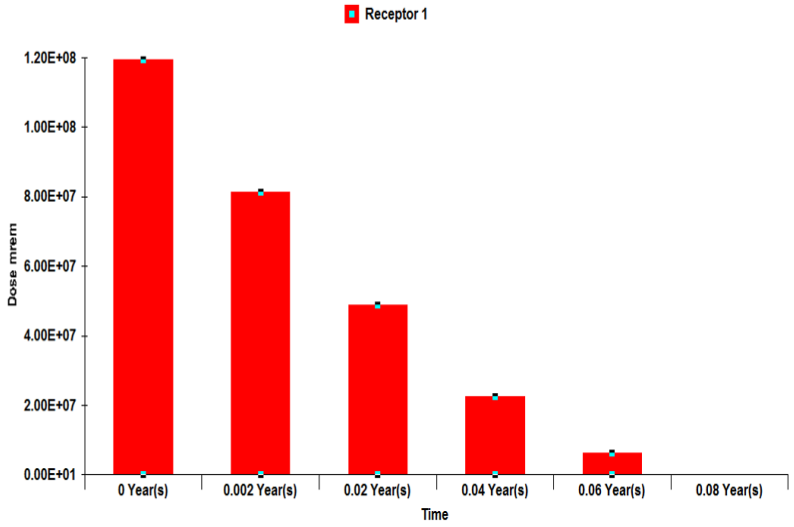

Fig. 27 - Dose by time for summed nuclides, summed sources and summed pathways

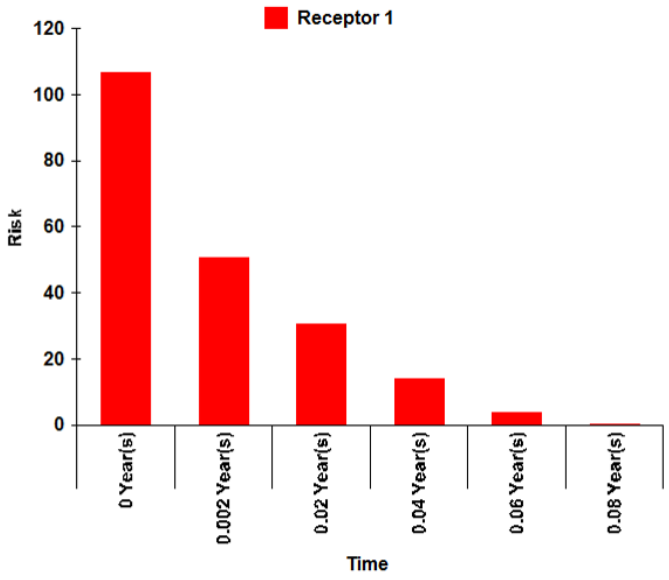

Fig. 28 - Risk by time for summed nuclides, summed source and summed pathways.

Furthermore, the modeling revealed each nuclide contribution to the overall risk. In the proposed reference scenario, the greatest risk is presented by Sr-90 followed by Cs-137 and Co-60 (see Figure 29)

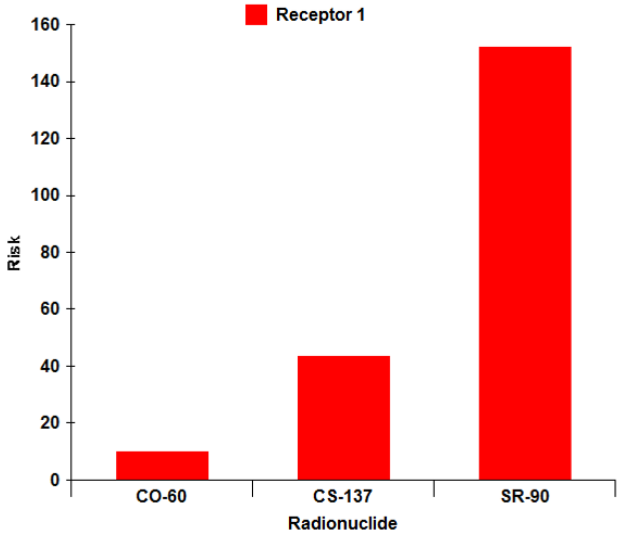

Fig. 10 (29) - Risk by each nuclide for the reference scenario

In Fig. 30, the risk for all the exposure pathways is presented. At the beginning at the operations, both the risks from aerosols inhalation and ingestion are presented, then only the risk from ingestion remains high. This is due to the fact that the modeling did not take into account any breathing equipment for the operator. The risks from external exposure, deposition, immersion and radon are considerable smaller.

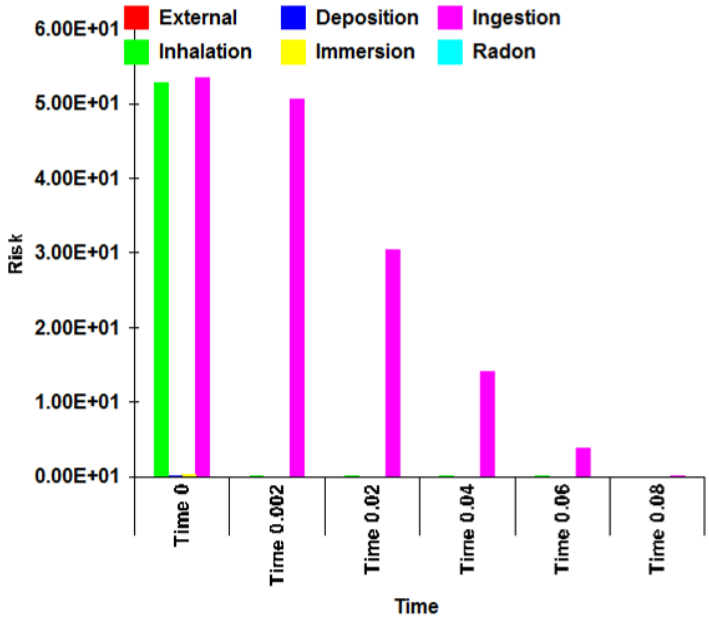

Fig. 30 - Risk by time and pathway for summed nuclides and summed sources

6 CONCLUSIONS

Data input:

- Potential radioactive contaminants: beta-gamma emitters (especially Co-60, Cs-134, Cs-137)
- Guideline value for beta-gamma emitters:
  Surface activity: 3 Bq/cm^2 (Co-60, Cs-137, Cs-134)
  Specific activity: 1.0 Bq/g-Co-60; 0.8 Bq/g-Cs-137; 0.5 Bq/g-Cs-134.

Comparison

- Room 30 is fully contaminated with Cs-134, Co-60 and Cs-137. We identified 10 (ten) areas of elevated activity in this room.
- Room 31 is highly contaminated with Co-60 and Cs-137. Contamination is concentrated within area located behind pumps (cells 1.1, 2.1, 3.1, 4.1, 5.1). In this area, we have bulk contamination due to the migration of nuclides into concrete.
- The evaluated dose rates are in a good agreement with corresponding in-situ measurements.

The rooms 30 and 31 are highly contaminated and therefore they must be decontaminated.
In room 31 a concrete layer of about 2 cm must be removed from the floor. Also, a concrete layer of about 2 mm must be removed from the wall of room 30.

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