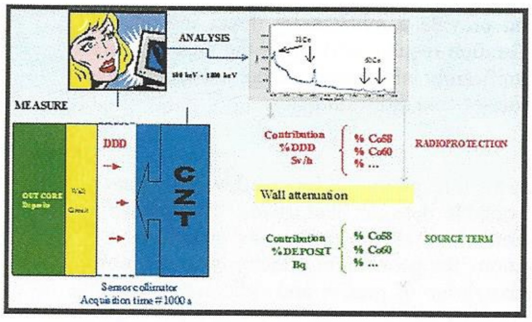


Figure 27: CZT principle diagram

The equipment is shipped with 3 interchangeable CZT probes of varying sensitivity:

— 60 mm^3: 0.5 mGy/h - 10 mGy/h;
— 20 mm^3: 5 mGy/h - 100 mGy/h;
— 5 mm^3: 20 mGy/h - 150 mGy/h.

The gamma energy spectrum ranges from 100 to 1800 keV for exposures from 0.5 to 150 mGy/h, depending on the used probe. The spectral resolution is from 15 keV (at 600 keV) up to 25 keV (at 1300 keV). Approximately 15 minutes is necessary for the acquisition of a spectrum with an exposure of 1 mGy/h, without any probe collimator. The first feedback analysis (comparison with EMECC more accurate but also more difficult to handle) shows that the CZT device is able to satisfactorily quantify the main radionuclide contribution to equivalent dose rate.

The spectral resolution of the CZT detector is greater than that of NaI and less than that of ultra-pure Ge and meets industrial requirements for radiation protection issues.

4.4 ~~Advantages and Disadvantages~~Comparison between High Purity Germanium and CZT

Comment [HBO37]: A draft Table to be discussed within the group. Is it needed to include other techniques?

Added but units need to be adjusted

Table-12 summarizes the characteristics of a germanium to a CZT detector. Both of these detectors are applied in the industry and Table-12 provides a quick overview of technical specification and comparisons.~~Include a table here to summarize resolution, time demand and dependence, work requirements, precision, output, etc. for various measurement techniques~~

Table 12: High Purity Germanium to CZT detectors

Characteristic	HP Germanium	CZT
Output	Radionuclide contribution to deposited activity	Radionuclide contribution to dose rates
Resolution	Less than 3 keV (at 1.3 MeV)	From 15 keV (at 600 keV) to 25 keV (at 1.3 MeV)
~~Output range~~Background Limitations	~~From 1 MBq/m2 to 100 GBq/m2~~	From 0.5 mGy/h to 150 mGy/h
Activity Measurement Range	From 1 MBq/m^2 to 100 GBq/m^2	Not Yet Applicable
Energy range	From 10 keV to 5 MeV	From 100 keV to 1.8 MeV

Comment [E38]: Verify units

EDF to verify the units and limiting number by September 1st

Acquisition time	2 hours	15 minutes
Work requirements	Very heavy device Liquid Nitrogen cooling Collimator needed	Very easy to handle No liquid Nitrogen cooling
~~Precision~~Nuclide Identification	~~Very accurate~~Most gamma emitters	~~Only the main radionuclides can be identified~~Limited by resolution

5. Measurement Locations and Indices

There are several key aspects to be considered for source term tracking mechanisms. These factors should be developed for specific plant and included in plant procedures. These factors are as follows:

- Survey locations - the points should be identified and remain consistent over time. The points should be selected based on accessibility and associated systems. This enables better tracking and trending of source term over time. Consideration should be given to providing a form of unique identifier of the survey location.
- Time after shutdown - where possible, the time the surveys are taken after shutdown should be consistent. This minimizes any error due to decay corrections for short lived radionuclides.
- Instrumentation - consider the limitations of the instrument and where possible use the same instrument over time to reduce the influence of instrument errors.
- Survey conditions - for greatest accuracy the condition of the survey location should be noted with special attention to the following which can influence survey results:
    - Insulation present and thickness
    - Pipe wall thickness
    - System full of water or drained

These are a few of the key items to consider and the following approaches are example methods of how these can be addressed.

5.1. PWR
5.1.1 EDF methodology

Reactor Coolant System (RCS) Index

This historical RCS index, carried out on the French fleet since the startup of all units, is adapted from the SRMP measurement program proposed by EPRI[9]. This program contains 3 points per loop located on hot leg, cold leg and cross-over leg (figure 28). The RCS index is calculated as the average over 9 points for a 900-series unit (3 points × 3 loops) or over 12 points for a 1300-series unit (3 points × 4 loops). RP department performs these measurements before oxygenation (internal EDF procedures). Post oxygenation measurements are optional if required for source-term measurements.

[9] *Application of the EPRI Standard Radiation Monitoring Program for PWR Radiation Field Reduction.* EPRIi Palo Alto, CA: 2007. 1015119.

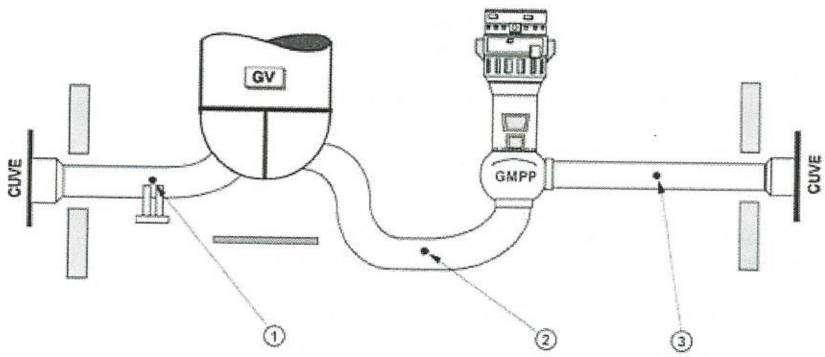


Figure 28: Localization of RCS index measurement points

Reactor building index

Comment [GRx39]: Proposition : Move this section in the outcome chapter ?

Author happy camper – committee 7/12/2013

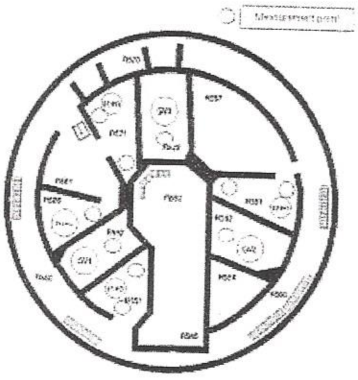
Management of source term is a key element of any ALARA action plan. Important efforts have been undertaken at the beginning of the 2000's so as to clean up reactors showing highest collective dose due to numerous hot spots and contamination issues. Following these dedicated actions, it has appeared that a tool for the detailed follow-up of reactor state of cleanness in order to detect long-term trends for the overall facility radiological state as well as for single system was missing.

RCS index, followed since the first startup of each French NPPs, is particularly useful to compare the dose rates near primary pipes between several units but does not give any information about the fleet radiological state of the reactor building.

Based on this, an index of the radiological state of the reactor building has been developed and experienced on all EDF nuclear plants between 2010 and 2011. This index, which is mainly based on a similar tool that has been experienced for more than 10 years at the Blayais NPP, provides an average ambient dose rate (taken at 50 cm or 1 m from the measured point) on the different levels of the reactor building. This average value is based on cartography with approximately 50 measurement points. It has been developed for the various sister units of the fleet and allow the following of the radiological state of the reactor building as a whole and the main systems (RHRS, CVCS, RCS, VDS, PZR, SIS, Steam Generator and RFCTS - Reactor and Fuel pools Cooling and Treatment System).

In order to be able to compare this index between the different reactors and within the same plant for different times, the mapping must be achieved in the same conditions on all the fleet and for each shutdown. It is thus requested to perform the mapping just after the shutdowns of the reactor building so as to follow the real state of the systems due to the operation.

As an example, figure 29 shows the measurement program for 900-series at level +11m .



n°	Location	Monitored system
34	Facing pressurizer at 1m high	PZR
35	On the footbridge, between SG1 and wall	SG
36	At 50 cm, facing the primary pump #1	RCS
37	at 50 cm from the valve, facing wall	RCS
38	On the footbridge, between SG2 and wall	SG
39	At 50 cm, facing the primary pump #2	RCS
40	at 50 cm from the valve, facing wall	RCS
41	On the footbridge, between SG3 and wall	SG
42	At 50 cm, facing the primary pump #3	RCS
43	at 50 cm from the valve, facing wall	RCS

Figure 29: Example of cartography at level +11m (900-series)

The implementation of this new index on every shutdown will allow the evolution of the dose rate in time monitoring and will allow detecting as quickly as possible any derivates of pollution. In that case, corrective actions could be taken (on chemistry, on the hot spots, filtration or decontamination of system, etc.).

First years show that this index meets its initial objectives, allowing sites and corporate staffs:

— To compare quickly and easily the different units of a power plant in order to identify pollution levels,
— An analysis in time and through the operation cycles of the evolution of the radiological state of all the nuclear power plant units,
— The implementation of corrective actions.

Contamination characterization

In addition to dose rates measurements, contamination characterization is also achieved by gamma spectrometry (EMECC campaigns and CZT program).

~~Figure 25: EMECC device~~

Figure 30: EMECC Device

EMECC campaigns, germanium detector

EMECC campaigns (performed by CEA), given in figure 25 have been commissioned for more than 30 years on French fleet units in order to better characterize contamination mechanisms. At the same time, EDF has also commissioned and financed EMECC campaigns on foreign units (Doel, Sizewell, Trillo during the 4 last years) with the contribution of several European operators in order to compare different good international practices.

The EMECC program has to be defined before the beginning of each campaign according to its specific aim. As an example, a typical EMECC program performed in 2009 in a 4-loop unit is presented in table 13 and figure 26.

Table 13: Example of an EMECC program performed in 2009 on a EDF 4-loop unit

Measurement points	Before oxygenation	After oxygenation
RCS – Hot leg – Loop 1	1	2
RCS – Hot leg – Loop 2	3	4
RCS – Crossover leg – Loop 1	5	
RCS – Crossover leg – Loop 2	6	
RCS – Cold leg – Loop 1	7	8
RCS – Cold leg – Loop 2	9	10
RCS – SG hot side – Loop 1	11	12
RCS – GG cold side – Loop 1	13	14
RCS – SG hot side – Loop 2	15	16
RCS – SG cold side – Loop 2	17	18
RCS – Bypass line – Loop 1	19	20
CVCS – NRHE	21	

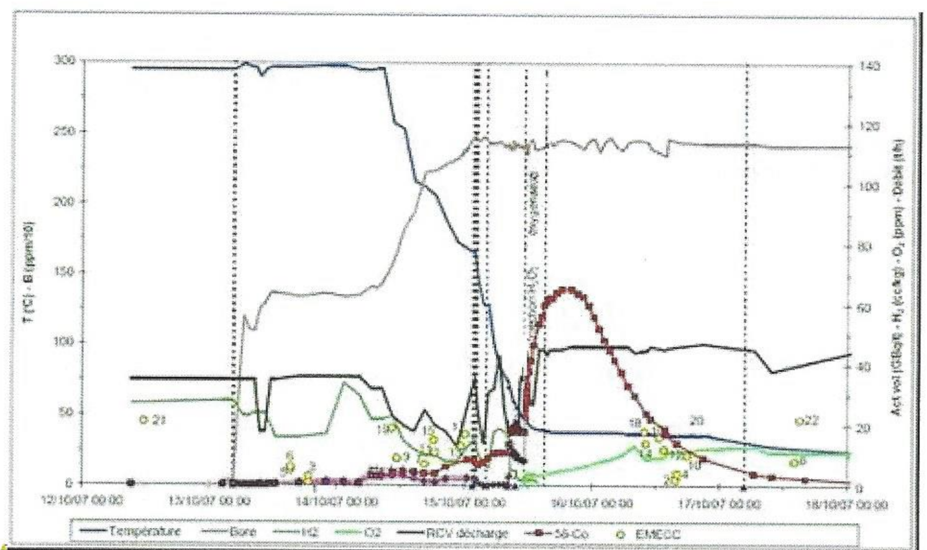


Figure 26: Measurement point localization during shutdown

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As an illustration, the evolution over cycles of ^{60}Co surface activity deposited on hot legs and steam generator tubes is shown respectively on figure 27a and 27b for one unit representative of each EDF fleet sub-series.

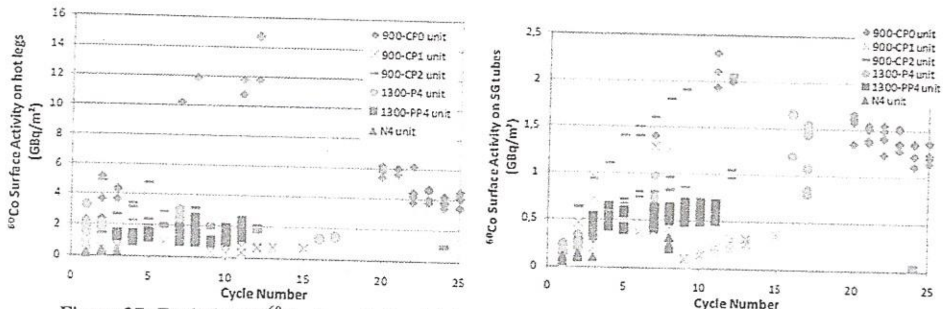

Figure 27: Evolution of ^{60}Co deposited activities on hot legs (24a) and steam generator tubes (24b)

CZT detector

As a matter of fact, EMECC campaigns are a very accurate way to characterize contamination in primary circuit but it clearly appears that the campaign number per year cannot exceed 10. There are 2 reasons explaining this limitation: in one hand, the CEA staff restricted capacity and in the other hand, a significant cost of each EMECC campaign.

Therefore, it is not possible to perform an EMECC campaign for every unit and every outage and this kind of characterization is necessarily dedicated to specific major issues for EDF (impact on contamination of SG replacement, primary pump stopping criteria, pre-oxidation and acid-reducing cleaning after SG replacement or new plant first start-up) and particularly those with undertaking toward Authorities (zinc injection, fuel management impact for instance).

In order to give a supplementary operational way to each Radiation Protection Departments in each unit, EDF have been carrying out a new dose rate measurement program since 2006 based on a semi-conductor CZT probe (Cadmium-Zinc-Tellurium).

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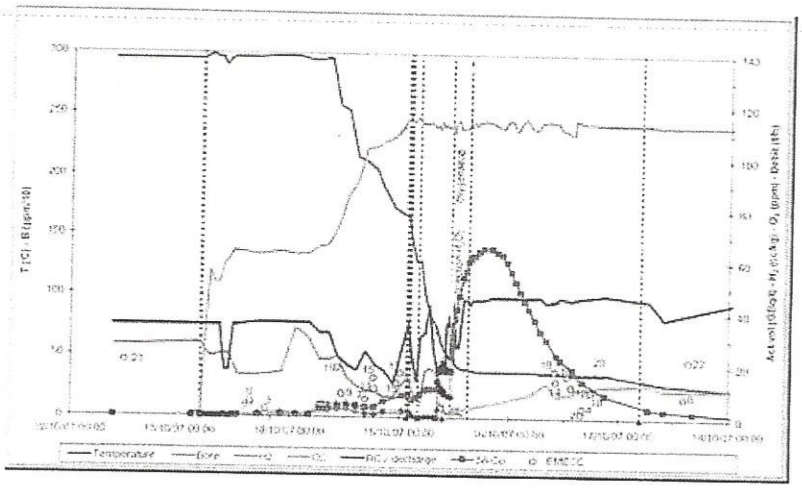


General objectives of the CZT gamma spectrometer consists in allowing each nuclear plant:

— to characterize the radionuclide contribution to the dose equivalent rates in order to take the relevant action with regard to reducing staff exposure doses (radiation protection),
— to produce a "point zero" contamination diagnosis (source term),
— to monitor the evolution of contamination from one cycle to the next,
— to identify as soon as possible any penalizing pollutants with regard to over-contamination risks,
— to assess the cleansing remedies efficiency.

Furthermore, the CZT "routine" program has been optimized recently in order to give an efficient basis for the contamination mechanism understanding. This program, containing 16 measurement points located on RCS, CVCS, SIS and RHRS (table 14), was proposed to all units in 2010.

Table 14 : CZT optimized program

P1a	CVCS	Before purification system	Power operation
P1b			After fuel download
P2a	CVCS	After purification system	Power operation
P2b			After fuel download
P3a	CVCS	Exchanger	Power operation
P3b			After fuel download
P4a	RCS	Crossover leg	Hot shutdown
P4b			Pool flooding beginning
P5a	RCS	Hot leg	Hot shutdown
P5b			Pool flooding beginning
P6a	RCS	Cold leg	Hot shutdown
P6b			Pool flooding beginning
P7a	SIS	Valve	Hot shutdown
P7b			Pool flooding beginning
P8a	RHRS	Exchanger	Hot shutdown
P8b			Pool flooding beginning

5.1.2 EPRI methodology

The Standard Radiation Monitoring Program (SRMP), sponsored by EPRI, was first instituted in 1978, as part of a more general program with the major emphasis on improving plant reliability and availability. The objectives of this program in 1978 were as follows:
— To provide a meaningful, consistent, and systematic approach to monitoring the rate of PWR radiation field buildup and to provide the basis for projecting the trend of those fields.
— To provide a reliable set of radiation field data for each participating plant, from which comparisons can be made.
— To monitor certain plant parameters that affect or may affect observed radiation fields.
— To use the information from this program to identify plant design features, material selection, and operational techniques that present opportunities for radiation control.

The objectives of the SRMP have not changed. From 1983 to 1996, EPRI published reports as a result of the SRMP program listing the factors that affect plant dose rates and quantitatively evaluated the effect of these factors. The most important factors at that time were found to be operational coolant chemistry and variations in cobalt input based on Inconel fuel grids.

Comment [O40]: TO DAN, Could it possible to prepare a new appendix on EPRI methodolgy to keep details there and general information in this sub section?

D Perkins – I would agree if that is the plan for all programs. Capture an overview in the discussions here and details in the attachments

Comment [HBO41]: Attachment

Disposed to leave as completed – 7/12/2013

The SRMP program had consistent data collection efforts for Westinghouse and Combustion Engineering plants through 1985 and 1996, respectively. Afterwards, SRMP data collection had been limited primarily to plants that had implemented elevated primary coolant pH, zinc injection or replaced steam generators with Alloy 690 tubing.

In 2005, adverse industry trends in Radiation Protection were a key factor in the development of the NEI/EPRI/INPO RP 2020 Initiative that had the stated goal of 'Taking Radiation off the Table.' EPRI was charged with taking the technical lead for Radiation Source Term Reduction. In response to this initiative, the EPRI Chemistry and LLW Technical Advisory Committee strongly recommended that EPRI restart PWR radiation field data collection efforts to help quantify the effects of plant changes such as replacement steam generators, core uprating, adverse radiological incidents, and various changes in shutdown and normal chemistry procedures. These changes have caused unpredictable fluctuations in dose rates throughout the out-of-core surfaces, and a more fundamental understanding is required.

In 2007 the program was reinstated and currently 129 units have submitted data to the program. Several projects beginning in 2007 have used the collected data to evaluate a consider the effect of parameters such as plant age, chemistry control methodology, effective full power year (EFPY), coolant chemistry, cobalt source terms, and startups and shutdowns. These factors have been evaluated and published in other EPRI reports.

The collection of isotopic gamma spectroscopic data at the SRMP points was not part of the original program even though some data exist from these plant locations Defined procedures for the collection of this data will be developed in a 2012 EPRI project. The project will also define additional data collection points outside of the reactor coolant recirculation system.

Procedures

The SRMP survey procedures define the methodology needed to collect radiation surveys at well-defined locations and to record pertinent plant conditions. The data gathered in the surveys give a better understanding of the parameters that influence RCS radiation fields. This information will, in turn, provide the potential for reducing plant radiation fields.

~~Specific procedures for the collection of data can be found elsewhere (/ref ?)~~

Survey Point Priority

Several concerns about worker safety and ALARA were considered in the development of the program and lead to the prioritization of the survey points. The survey locations were defined as 'Required Points,' 'Highly Recommended Points,' 'Recommended Points,' and 'Optional Information.' The definitions of these terms are below:

— Required points are those that must be taken.
— Highly Recommended are those that are strongly requested, but may be skipped in only cases of personnel safety, poor accessibility, or significant ALARA impact. The points have significant research value and the plants are asked to make the best possible effort to obtain them.
— Recommended points are those that are requested, but may be skipped in cases of personnel safety, poor accessibility, or significant ALARA impact.
— Optional information is information that is requested if available.

The procedures provide a controlled measurement program for assessing radiation field trends of RCS components.

The radiation surveys are conducted during plant shutdowns and collect dose rate readings at permanent markers located on the outside surfaces of RCS components. Surveys are also specified for the internal surfaces of the steam generator channel heads when maintenance or inspection activities are performed.

Survey Points

The following section discusses the survey points and requirements of the radiation survey procedures for Westinghouse designed plants. Equivalent points have been identified in Combustion Engineering and Babcock and Wilcox designed PWRs [??].

Comment [E42]: Reference

101519, Application of EPRI Standard Radiaiton Monitoring Program for PWR Radiation Field Reduction

Reactor Coolant Loop Piping Survey Procedure

The reactor coolant loop piping survey locations for a Westinghouse designed PWR are given in and are summarized in figure 28.

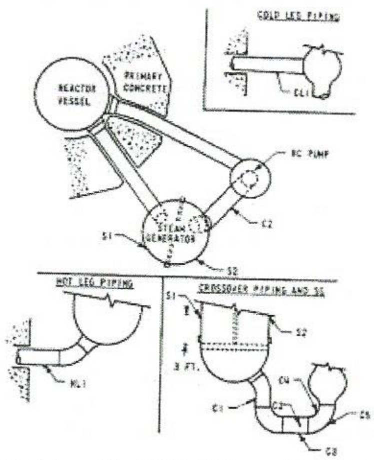


Figure 28: Typical Westinghouse 4-Loop Plant With Piping and Steam Generator Survey Points Marked.

Required Points

- C2 - Straight section of crossover piping, side of pipe (generally away from primary concrete shield)
- HL1 - Bottom of hot leg piping between steam generator inlet and reactor vessel shield
- CL1 - Bottom of cold leg piping between reactor coolant pump and reactor vessel shield
- S1 & S2 if taken previously (See below)

Recommended Points

- C1 - Above crossover piping elbow, midway along vertical section of piping from the steam generator
- C3 - Straight section of crossover piping, bottom
- C4 - Crossover piping elbow to RCP, midway along inside radius
- C5 - Crossover piping elbow to RCP, midway along outside radius

- S1 - Outside of steam generator hot leg side, approximately 1 meter above top of channel head tube sheet and approximately midway between secondary side hand-hole cover and hot leg piping (90 degrees radially from the tube lane)

— S2 - Same as S1 but approximately midway between secondary side hand-hole cover and cold leg piping (90 degrees radially from the tube lane)

Optional Information Points

Note: Specify location of measurements, e. g., on letdown piping, one foot downstream of regenerative heat exchanger

— Letdown piping
— CVCS heat exchanger (on the shell)
— RHR piping
— RHR heat exchangers (on the shell)
— Refueling water surface

Steam Generator Channel Head Survey Procedure

If access to the steam generator channel head(s) occurs during the shutdown period, the results of the channel head survey are to be recorded on an appropriate survey form included in procedure. The Westinghouse designed steam generator channel head survey locations are summarized in figure 29.

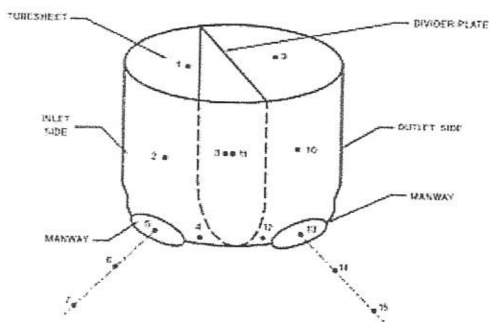


Figure 29: Westinghouse Plant Channel Head Survey Points

Required Points

— Midpoint of Tubesheet (Hot Leg & Cold Leg ,points 1 and 9)
— Channel Head Center (Hot Leg & Cold Leg, points 2 and 10)
— Center Divider Plate (Hot Leg & Cold Leg, points 3 and 11)
— Bottom of Channel Head (Hot Leg & Cold Leg, points 4 and 12)

Recommended Points

— Manway Entrance (Hot Leg & Cold Leg, points 5 and 13)
— 30 centimeter from Manway (Hot Leg & Cold Leg, points 6 and 14)
— One meter from Manway (Hot Leg & Cold Leg, points 7 and 15)

5.1.3 Hot Spots

In most cases, hot spots are due to particles of cobalt activated by a neutron flux (^{60}Co) mainly from hard facing surfaces equipments (Stellite®, rich in cobalt) in the RCS (valves, pumps, internals, etc.). The contribution of hot spots to shutdown dosimetry may appear to be marginal in French PWR reactors (2 to 4 %), but becomes more significant (15 to 25 %) for the units affected. This excess dose has to be taken into account, particularly for the most exposed workers. Approximately ten French PWR units have been affected by this phenomenon over the last 15 years.

Surveillance is designed to inform the site as early as possible, of the presence of hot spots (mapping) in order to take the appropriate measures to prevent their propagation and/or to eradicate them. During unit operation, most hot spots will remain fixed to the fuel. Others may fall, by gravity, to the bottom of the pool or the low points of the primary coolant system or be trapped in the special devices. The most common locations are as follows:

- Thermal sleeves of the pressuriser,
- Steam generator packing glands,
- Valves of the primary cooling system, etc.

After the Residual Heat Removal System (RHRS) is placed in service, some hot spots may migrate into this circuit and be deposited or fixed. The most common locations are: the pumps, heat exchangers and valves of the circuit. An underwater pool cleaner should pass through the pool out after discharging. In this case, particularly high equivalent dose rates, equal to or greater than 1 Sv per hour, measured in contact with the filters, represent the last indicator of the possible presence of hot spots, before draining of the pools. Since no warning signs have been identified yet, to indicate the occurrence of hot spots, it was decided to concentrate on preventive filtering, trapping hot spots as close as possible to their source to eliminate them.

Comment [E43]: To be moved to a new home on the final reading to find out where it fits. Crud traps are...

5.2 VVER

5.2.1 Dose rate measurements

An IAEA Regional Technical Co-operation Project RER/9/63 on Improving Occupational Radiation Protection in Nuclear Power Plants in Central and Eastern Europe and in Republics of the former Soviet Union was launched in 1997, having as one of its principal objectives to facilitate information exchange between Health Physics in VVER and RBMK nuclear power plants. In this forum a Working Group on Standardization of Dose Rate Measurements in VVER reactors presented its first report in November 1998, when an agreement on a scheme for measurements was also reached. Pre-defined measuring positions, as shown on figure 30, were used to measure dose rates in uncolimated arrangement 24-48 hours after reactor shutdown. It must be noted that as this measurement is performed shortly after shutdown, activity of short lived radionuclides like ^{51}Cr has higher impact to the result than some other longer lived radionuclides that have higher impact to dose rates later during outage. In November 1999, information from all VVER reactors was collected and presented for the first time by the members of the Working Group, and especially those who registered very low dose rates, to go back and investigate what may have had a significant impact on the dose rate. Comparison of VVER plant data for years 2000, 2001s is presented at figure 31.

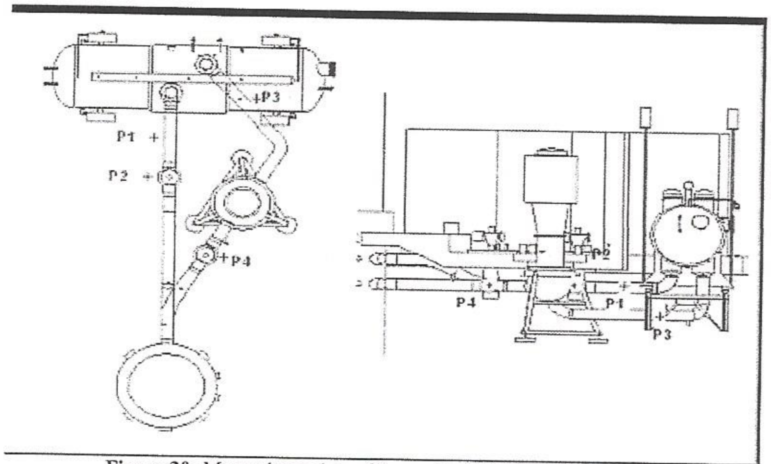


Figure 30: Measuring points of dose rates at VVER-440 reactors

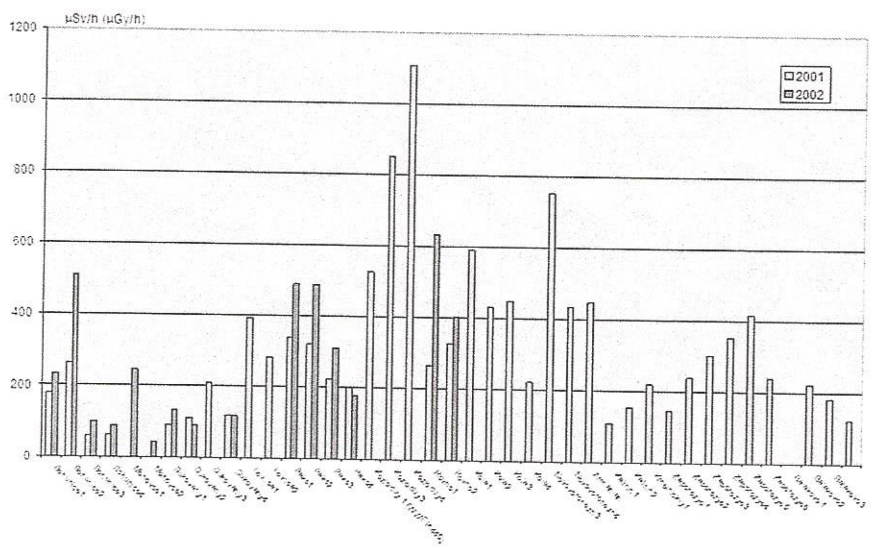


Figure 31: Comparison of primary loop dose rate averages for VVER reactors

Despite of fact that IAEA project was terminated in 2002, some plants still continue in these measurements, but data collection/comparison on the international basis does not continue. Results of of these measurement for NPP Bohunice Unit EBO-3 over last 12 years is shown at figures 32 and 33 and for Paks-1 unit on figure 34.

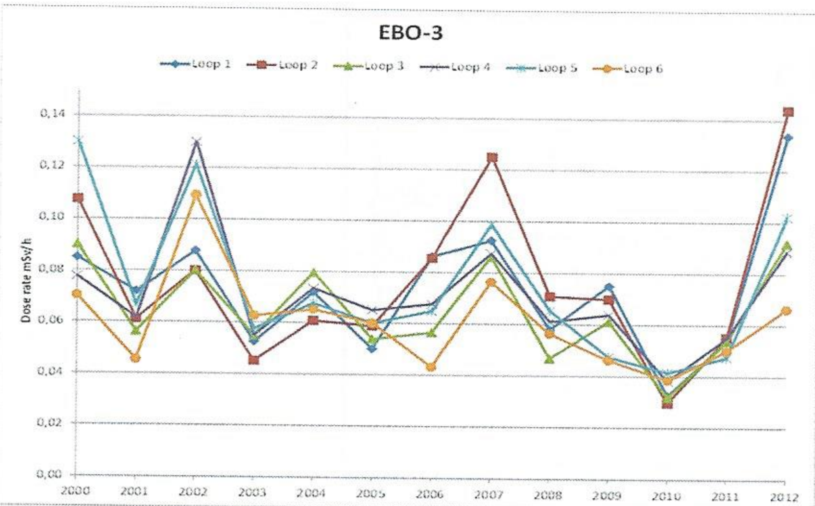


Figure 32: NPP Bohunice primary loop dose rates in period 2000-2012 by loops

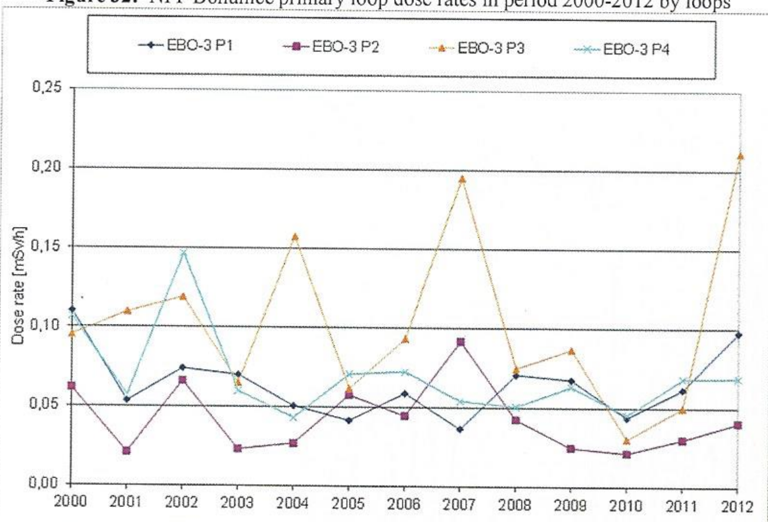


Figure 33: NPP Bohunice primary loop dose rates in period 2000-2012 by measurement points

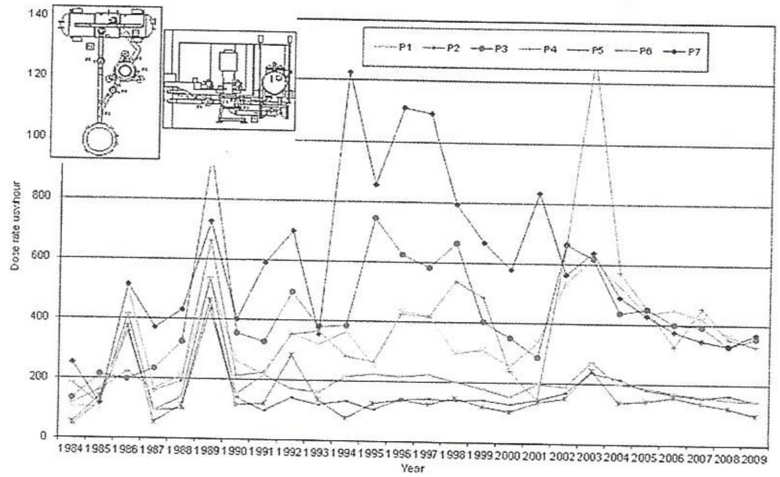


Figure 34: NPP Paks Unit 1 primary loop dose rates in period 1984-2009 by measurement points

5.2.2 In-situ gamma spectrometry

This measurement technique for primary loop surface activities was first developed at the NPP Loviisa and NPP Paks [25] and later it was implemented also at other VVER NPPs - in Slovak and Czech Republic. At the beginning measurements were carried out at hot and cold leg of primary loops and later this technique was used at NPP_Loviisa, Paks, Dukovany, Bohunice and Temelin also for monitoring of steam generators from outside [26], activity profile of vessels with ion exchange resins, titanium sponge in high temperature filters in order to optimize resin replacement/regenerations. Some plants are performing this measurement with their own staff and equipment but there are also specialized companies capable to provide this service for majority of plants.

Measurement is made by portable LN2 cooled HPGE detector with collimator installed at reference points. Typically two measurements are made - one with plugged collimator hole and one with open hole to compensate environmental radiation in the vicinity of measurement locations. Efficiency calibration is made either by calculation (e.g. Monte Carlo modelling, Canberra ISOCS model) or by direct calibration using large surface type calibration source.

Measurement points are not well standardized among VVER plants, many plants measure at straight part of hot and cold leg and also at crossover leg of all primary loops as shown at fig 34, NPP Loviisa performs measurement at 8 positions of one loop and 4 position of 2 additional loops (at this plant due to the space problem in the Steam Generator Compartment a full scan of all loops is almost impossible).

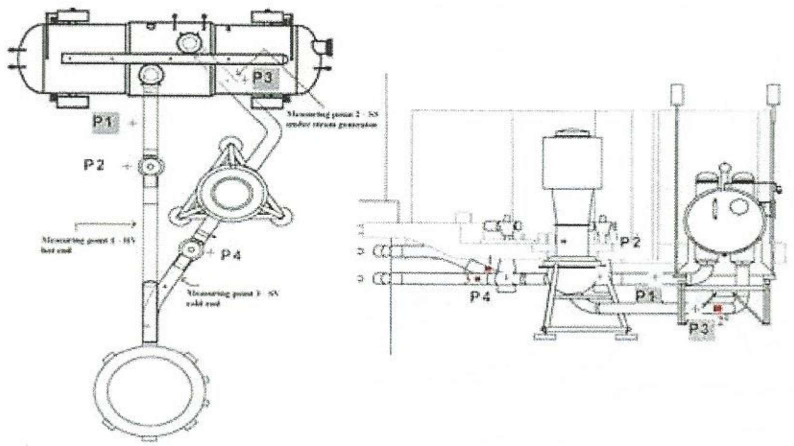


Figure 34: Measurement points scheme at Czech and Slovak NPPs Bohunice and Mochovce, Dukovany

Examples of measurement arrangements are shown at following figures:

Figure 35: Primary loop piping mock up made purposely for NDT calibration/validation which was used for real efficiency calibration at Slovak NPP Bohunice.

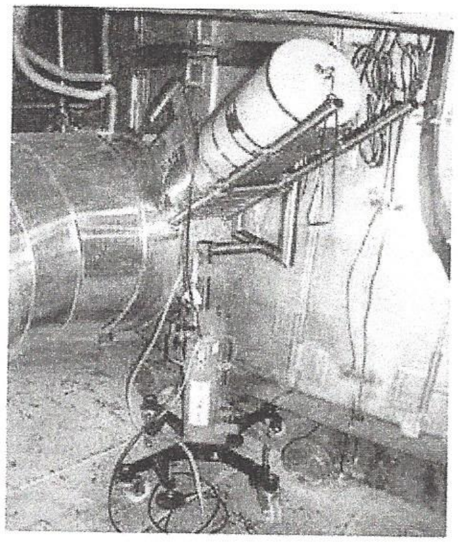


Figure 36: Real measurement arrangement at Czech NPP Dukovany

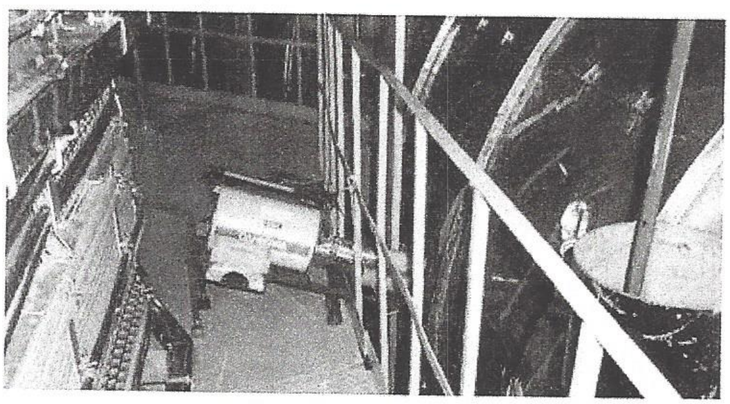


Figure 37: Steam generator measurement at Hungarian NPP Paks

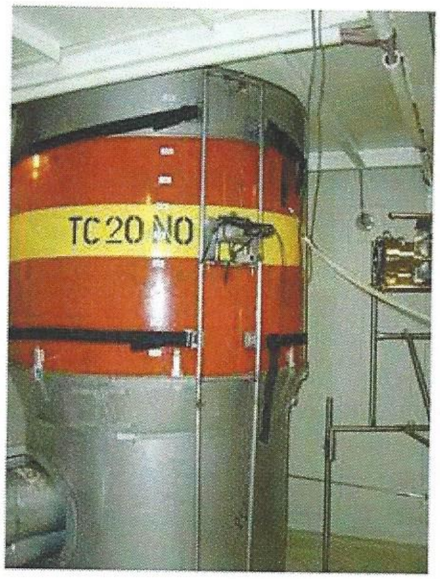


Figure 38: Activity profile measurement of primary cleanup filter at Czech NPP Temelin

Examples of surface activity trends

Comparison of measurement results over past 10-20 years for NPP Bohunice, Loviisa and Paks are shown at Figures 35-37. From these results it can be shown that in some cases variation of particular isotope activity are high - as it is for Loviisa and Paks, while in other case activity of is relatively stabilized - as for the NPP Bohunice unit EBO-3 case. Comparison with dose rate measurements for EBO-3 shows some correlation of dose rate measurement and loop surface activities data.

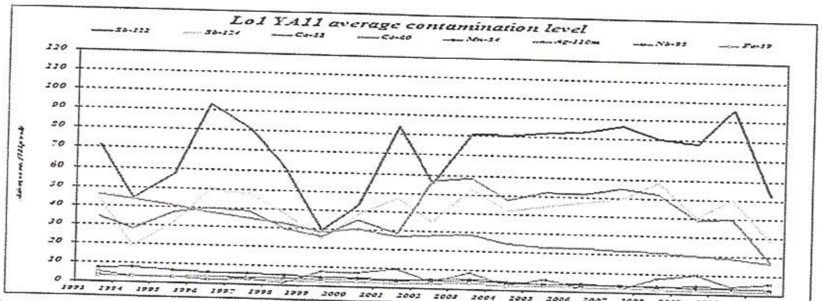


Figure 39: Average contamination levels on the 1st loop of Loviisa 1 (8 points) - values in kBq/cm^2.

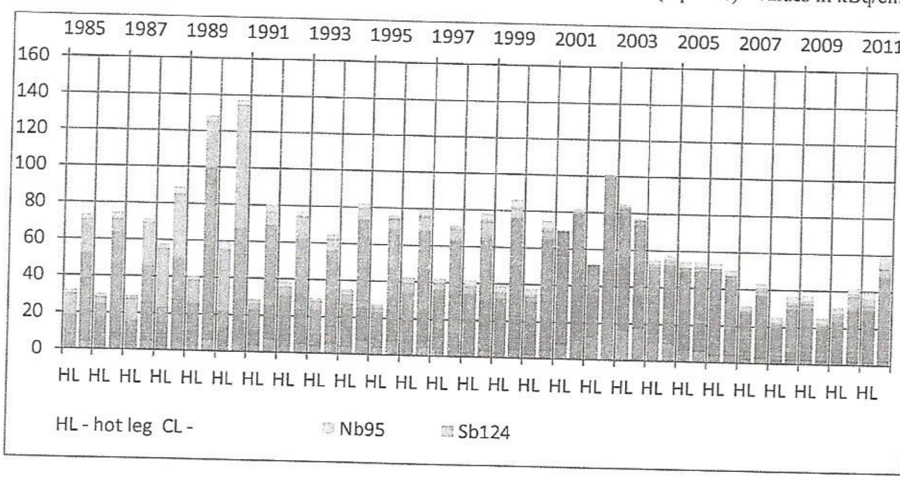


Figure 40: Average primary loop contamination levels on the PAKS-1 unit - values in kBq/cm^2.

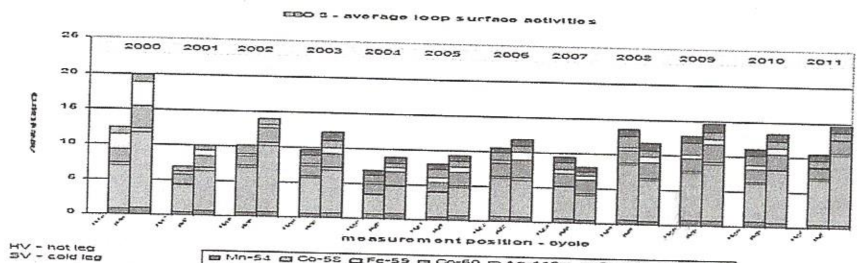


Figure 41: Average primary loop contamination levels on the Bohunice-3 unit - values in kBq/cm^2.

5.3 PHWR

Comment [HBO44]: Colin (Canada)

David Miller, to meet with Colin and get added by September 1st or we drop.

5.4 BWR

5.4.1 EPRI Methodology

EPRI BWR Radiation Level Assessment and Control Program

The BWR fixed point radiation field survey program, commonly referred to as BRAC (BWR Radiation Level Assessment and Control), is discussed in General Electric document NEDC-12688, which was issued in 1977 based on work sponsored jointly by General Electric and EPRI []. The intent of the BRAC program is to establish a consistent set of fixed survey points in order to monitor radiation buildup, review plant operational and design factors for effect on dose rates, and to provide reference data input to radiation buildup modeling. The BRAC program specifies locations, frequency, timing, and instrumentation for periodic fixed-point radiation dose rate surveys of BWR primary system components in order to provide consistent and comparable data.

Plants participating in the EPRI BWR Radiation Level Assessment and Control (BRAC) program have classically used the Eberline HP220 A (E-530N) detector/shield housing assembly; however, other instruments that have been similarly calibrated may be applied. The HP220A detector consists of a small Geiger-Mueller detector inside a hemispherical tungsten shield, which provides a 7 to 1 attenuation front to back for ^{60}Co gamma emitters. A digital readout ratemeter is preferred, but analog models are acceptable. Instruments that switch to a second, internal detector when on the highest scale should not be used for directional measurements. The collection of isotopic gamma spectroscopic data at the BRAC points has been routinely collected using plant and task specific procedures. Defined procedures for the collection of this data will be developed in a 2012 EPRI project.

Survey Points

Survey points are specified throughout the primary system, and include the suction and discharge piping of the recirculation pumps, suction and discharge piping of the reactor water cleanup pumps, the main steam lines, the inlet and outlet of the regenerative and non-regenerative heat exchangers, and points on the heat exchangers themselves. Figure 42 shows the BRAC sample points on the recirculation system of a typical BWR plant. The BRAC average values used throughout the summary reports for a set of measurements from a given plant, the average of the recirculation suction and discharge contact dose rate readings.

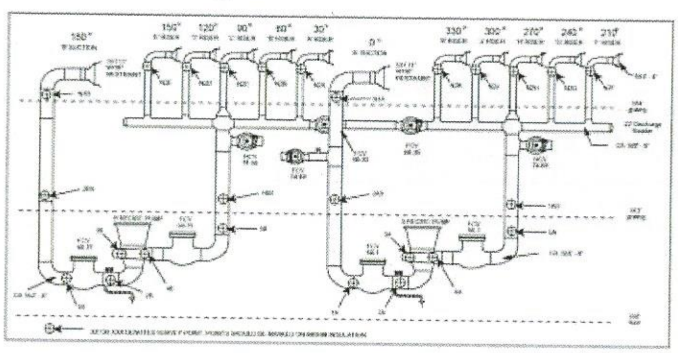


Figure 42: BWR radiation sampling points for typical BWR recirculation piping

Surveys should be conducted with the component in its normal configuration; for example, with any insulation in place and liquid-filled. The BRAC program does not specify a distance between the target survey point and the system components (e.g., a valve in the pipe) nor does it recommend a length for straight run of pipe. It only specifies that each unit should be consistent with its selected point. Differences in plant design, access platforms, etc. also contribute to inconsistency in the exact location of the survey points, but these are unavoidable.

Survey Timing and Plant Configuration

Surveys should be conducted during each refuelling outage and during other outages that are long enough to permit a meaningful survey. The surveys should be conducted between 7 and 14 days after shutdown, with the 7-day minimum to allow short lived isotopes to decay. Variability in the time at which surveys are taken may complicate the interpretation of the results; particularly in the absence of piping gamma scan results.

Surveys may be conducted when systems are drained or with insulation removed. Many units have changed insulation types over the years, which would change the effective standoff distance or the radiation shielding value of the insulation. Several plants have added permanent shielding to the BRAC components. Performing the survey with systems drained will most likely result in higher readings than if the system was full.

Time after shutdown when the survey is taken can also vary significantly. Plants obtain BRAC surveys for trending, even in short mid-cycle outages. The continued compression of outages may make collecting consistent BRAC data in the specified 7 to 14 days range more difficult. Shielding may be installed on components well before the 7-day minimum. Plants then must either take the BRAC survey early or wait past the 14-day recommendation, when the system is restored. In addition, if a plant experiences a fuel failure during the cycle, 7 to 14 days may actually be insufficient to eliminate the unique contribution from the additional iodine-131.

5.4.2 ASEA Atom methodology at Vattenfall

In the late seventies ASEA Atom developed a methodology for the fleet of plants that was built by the company – in total nine stations at the Oskarshamn, Ringhals, Forsmark and Olkiluoto sites in Sweden and Finland. The methodology was named MADAC (Mobile Analyser for the Detection of Activity in Crud) and is based on collimated measurements with a shielded germanium detector.

The MADAC program has been further adapted at the Ringhals site to also become applicable on the Westinghouse PWR units at the site (Ringhals 2, 3 and 4). With time the name of the methodology has changed, and the MADAC designation is no longer used. Current names in use are NYMF at Forsmark, SAM at Ringhals, NSSAM at Oskarshamn.

A portable low efficiency HPGe detector (ca 4 %) is placed on a cart with shielding, as shown in figure 40. Another cart is equipped with a digital MCA and laptop. Data are collected to the laptop and copied to an office computer to correct for background, efficiency and decay since the beginning of the outage.

Typical measurement points during a campaign are [27, 28]:

- On each of the two pipes that lead water from the reactor tank to the shutdown cooling system. This system also feeds water to the reactor water cleanup system (RWCU).
- On the pipes and selected heat exchangers along the the RWCU system: before heat removal, after heat removal but before filtration, after filtration and after final regenerative heat exchange. The temperature dependence of the surface contamination can be observed.

— On a pipe for the system that supplies water to the hydraulic scram function and to the crud removal flow through the control rod guide tubes. This water is a partial flow of the filtrated water from the RWCU system.
— On a pipe of the cooling and clean-up system for the fuel pool water.
— On two of the steam lines close to the high pressure turbines.

The outcome of the measurements is the radionuclide specific contamination inside pipes and heat exchangers, given in Bq/m^2. In order to obtain a correct value, great care has to be taken as the efficiency of the measurement is calculated. The efficiency calibration is based on the reference measurement of a certified planar source of ^{152}Eu. A correction is then done for the actual conditions at the measurement point, taking into consideration the materials and dimensions of the pipes or heat exchangers, whether the system is water filled, the amount of insulation present and the size of the collimator.

At the Swedish Vattenfall sites (Forsmark and Ringhals), as well as the E.ON site (Oskarshamn) there is a measurement campaign for each station during the annual outage. The results and conclusions are reviewed and spread within the organization. Typical applications are:

— Assessment of which radionuclides that contribute to the total dose rate. The dose rate is mostly dominated by ^{60}Co with occasional large contributions by mainly ^{110m}Ag, ^{58}Co or ^{124}Sb.
— Trend analysis and assessment of the causes for trend development.
— Assessment of radioactive inventory in waste, either directly or the results may be used as data input for radionuclide vectors.

An obvious strong point of the measurement of surface contamination is that the result will be consistent even as insulation is removed or the system is drained. The result is also not affected by for example the thickness of a pipe. This is not the case for dose rate measurements where the result will be higher for a drained or de-insulated component, or for a thinner pipe. As results are compared between measurement points or plants, consideration must be given to what is to be compared: to compare surface contamination will yield information about the nature of the source term while the comparison of any dose based measurement will yield information about the effect of the source term.

Figure 43: Portable HPGe detector with collimated shielding for a BWR MADAC based measurement campaign, here measuring on a BWR-75 steam pipe (Forsmark 3).

6. Conclusions

Comment [HBO45]: First draft will be provided by EDF.

Several factors clearly linked with dose rate evolution

- Type of the unit (NSSS design)
- Manufacturing – careful material selection, surface conditioning
- Properly designed and executed Hot functional tests
- Operation
  - impurity control, FME, pH(T) strategies and adherence
  - pH(T) limitations related to use of Li at PWR – potassium alternative ?
  - modifications - careful material selection
  - role of purification ?

Remedial activities

- Shutdown chemistry – developed on the basis of need
- Injection of zinc, iron NMCA
- Decontamination – mature processes with proved dose rate benefit
- Decontamination – not only remedial but in some cases also radiation buildup promotor
- Hot spots removal ……

Monitoring

- Many approaches adopted (national, utility, owners clubs,…)
- Several measurement techniques from simple to sophisticated instrumentation
- Different workforce demand – different quality of data
- Preferred approach: Low workforce/low dose – maximum output (value of information obtained by measurement)
- Long term systematic monitoring highly recommended (identification of problems, analysis of potential reasons, evaluation of measures implemented, water chemistry quality evaluation,)
- Monitoring strategies developed for large utilities (also for reasons of comparability, sharing of effective measures, ….)

Appendix-1

Typical primary materials for PWRs

Table 1: Weight Percent Composition of Structural Alloys

	Steam Generator Tubing			**Structural**		**Cladding**		
Element	**Alloy 600**	**Alloy 690**	**Alloy 800**	**304 SS**	**316 SS**	**Zircaloy-4 (ASTM R60804)**	**Standard ZIRLO™**	**M5™**
C	0.01–0.05	0.015–0.025	<0.03	≤0.08	≤0.08			
Co	0.015–0.10	0.015–0.10 (≤0.015 for tubing)	<0.10			≤0.0020		
Cr	14.0–17.0	28.0–31.0	20-23	18-20	16-18	0.07-0.13		
Cu	<0.50	<0.50	<0.75			≤0.0050		
Fe	6.0–10.0	7.0–11.0	balance	Balance	balance	0.18-0.24	0.09-0.13	~0.0350
Mn	<1.0	<0.50	0.4–1.0	≤2.00	≤2.00	≤0.0050		
Mo					2.0-3.0	≤0.0050		
Nb						≤0.0100	0.80-1.20	0.80-1.20
Ni	>72.0	>58.0	32.0–35.0	8-11	11-14	≤0.007		
O						0.09-0.160	0.10-0.15	0.110-0.170
P				≤0.04	≤0.03			
S				≤0.03	≤0.03	≤0.0270		0.0010-0.0035
Si				≤0.75	≤0.75	≤0.0120		
Sn						1.20-1.70	0.80-1.10	
Zr						balance	balance	balance

Table 2 from Reference [1] provides an example of hard faced cobalt materials composition while Table 3 provides examples related to hard faced nickel material composition.

Table 2: Weight Percent Composition of Cobalt-Based Hardfacing Alloys

Alloy	Weight %									
	Co	**Cr**	**Ni**	**Fe**	**C**	**Mn**	**W**	**Si**	**B**	**Mo**
Co-156	bal	29	3	0.75	1.6	1	4.5	1.2		1
Haynes 36	bal	18.5	10	2	0.4		15		0.03	
Stellite 6	bal	33	3	3	1.1		6			
Stellite 6B	bal	30	3	3	1.1	2	4.5	2		1.5
Stellite 21	bal	27	2.8		0.25					1

Table 3: Weight Percent Composition of Nickel-Based Hardfacing Alloys

Alloy	Weight %								
	Ni	**Cr**	**Fe**	**C**	**W**	**Si**	**B**	**Mo**	**Other**
Colmonoy 4	bal	10	2.5	0.4		2.8	2.1		
Colmonoy 5	bal	13.8	4.8	0.45		3.3	2.1		
Deloro 50	bal	12	3	0.35		3.5	2.5		
Metco 19E	bal	16		0.5				2.4	Si+B+Fe=4
Nucalloy 488	bal	17.5	5.5	0.3	1	6.8	1		Sn=0.7
Trialoy T700	bal	15.5		0.08		3.4		32.5	Co+Fe=< 3

Appendix 2: Strategy for implementing an optimized CZT programme

Comment [RBCN46]: New draft Ludovic and Alain

New draft for committee to review.

CZT measurement programme to be carried out during shut down

Starting in 2006, an initial CZT measurement programme was proposed to each NPP in complement of dose rate cartographies. The initial programme to be carried out systematically during shutdown includes 8 points of measurement to determine the contribution to the dose rate of radionuclides deposition in the out-of-core circuits: 3 points in the nuclear auxiliary building, 1 in the fuel building and 4 in the reactor building. Table 1 below gives the precise location of the measurements to be made.

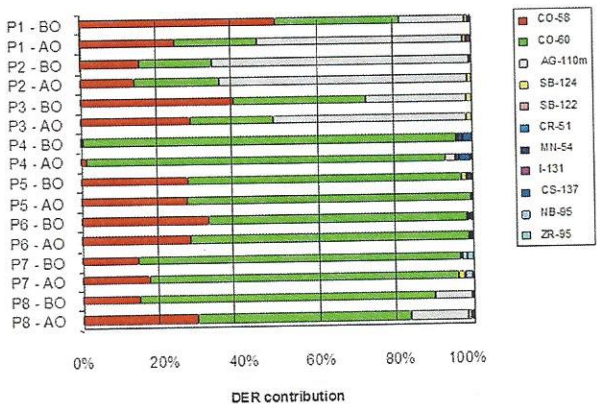
The harmonization of these measurements on all EDF units is used to establish inter-unit comparisons and, for a single unit, to monitor the contamination over a period of time. The measurement system is also used to check the correct operation of oxygenation of the primary fluid during cold shutdown of the reactor.

Table 1.Shutdown initial programme - CZT measuring points before and after oxygenation

Measuring points in the Nuclear Auxiliaries Building
P1: Chemical and Volume Control System (CVCS): Upstream for purification **P2:** CVCS: Downstream for purification **P3:** CVCS: Non regenerative heat exchanger
Measuring point in the Fuel Building
P4: Spent Fuel Pit Cooling and Treatment System (SFPCTS) at the junction of the drainage lines
Measuring point in the Reactor Building
P5: Reactor Coolant System (RCS): Hot Leg **P6:** RCS: Cold Leg **P7:** Safety Injection System (SIS) - After the RCS valve **P8:** Residual Heat Removal System (RHRS) - Heat exchanger

Statistical analysis of CZT measurements

The results obtained on all PWR plants between 2006 and 2007 (380 measurements) are shown below.



**BO: Before oxygenation - AO: After oxygenation*

This graph shows that the radionuclides that contribute to almost all the dose rate are Co60, Co58 and Ag110m. The relative contributions of the 3 radionuclides are highly variable depending on the different units and circuits.

<u>a) Measuring points in the Nuclear Auxiliaries Building (CVCS): *P1, P2 and P3*</u>

In the above example, the measurements made on the chemical and volume control circuit highlight that in addition to Co58 and Co60, the chemical and volume control system often shows a high level of contamination by Ag110m.

C V C S

The analysis of the results shows that there is a wide dispersion.

The non-regenerative heat exchanger and the downstream part of the purification chain usually present similar behavior although in different proportions. In fact there is a wide variety of behaviors depending on shutdowns: Ag110m can be present in varying quantities and radio-cobalt can increase or decrease in proportion after oxygenation. These elements prove that the measurements before and after oxygenation should be considered separately.

It is the comparison before and after oxygenation that is of interest and can be used to check that there is no significant recontamination.

<u>b) Measuring point in the Fuel Building (SFPCTS): *P4*</u>

S F P C

On the pool side, when the unit is shut down, the handling of fuel releases hot spots (Co60 particles from deterioration of the stellite) that were fixed to it. In the absence of a containment barrier, they are dispersed in the circuits by the movements of water. The analysis of the feedback on pool drainage points show highly variable dose rate from one plant to the next (from 0.07 to 100 mSv/h), around 94% of which are generated on average by Co60. Since this measurement point is almost always dominated by Co60, it can be eliminated from the systematic measurement programme.

<u>c) Measuring point in the Reactor Building (RCS, SIS, RHRS): *P5 to P8*</u>

R C S

The measurements made on the hot and cold legs of the primary circuit are similar and are not normally impacted by oxygenation. We recommend comparing them in order to make the measurements more reliable.

The average of the CZT measurements on the primary legs show contributions to the dose rate of 30% for Co58 and 70% for Co60; this is in the range of expected values from the EMECC feedback.

Some units stray from this average and show a higher Co58 contribution. This is especially the case for units in which the steam generator was recently replaced.

S I S

The analysis for the point of measurement downstream from the reactor coolant system valve highlights a high contribution of Co60 to the dose rate averaging 81%, the rest coming from Co58. These results are not much different from the reactor coolant system loops (P5 and P6) but with a higher Co60 contribution. In fact there is less circulation and the Co58 depositions are lower here. They are quite even in all the units with a standard variance of 15%.

R H R S

The Residual Heat Removal System shows the behaviour between that of the Reactor Coolant System circuits (P5 and P6) and that of the CVCS circuit downstream from the purification chain (P1).

- Before oxygenation, Co60 is the main, and almost exclusive contributor to the dose rate (the other radionuclides having decreased) with an average of 75%.
- After oxygenation, Co58 and Ag110m recontamination is sometimes observed (which depositions are mostly on the cold parts). Co60 remains however the main contributor to the dose rate with an average of 55%.

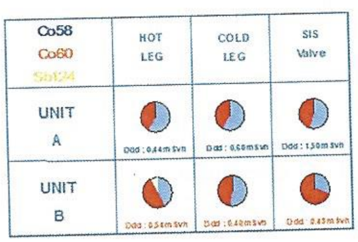
The same remarks as for the CVCS circuit apply, i.e., on the one hand high dispersion in the measurements and on the other the interest in comparing before/after oxygenation measurements. In the analysis this point should be used when considering evolution of the CVCS.

Analysis and interpretation of the CZT results from nuclear plants

Besides monitoring of the state of contamination in the units, the systematic CZT measurement programme is used to define the impact of optimized operating conditions and/or specific malfunctions. The examples described below are the result of preliminary feedback from the plants and are a concrete illustration.

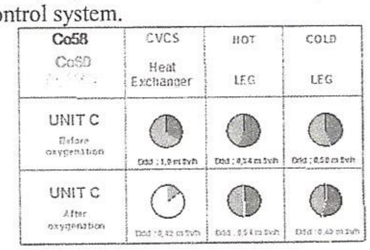
Optimization of primary chemistry and purification in operational phase

The results obtained on two NPPs equivalent age and design, plant A and B, showed that the depositions can be different; this can be explained by the different operating conditions in both plants. In this particular case, plant A operates with a higher primary pH and a purification circuit with a finer filter; this can partly explain the differences found in Co60 content.
The analysis of these differences forms part of a global study within EDF to optimize practices and reduce doses in all units.



Malfunction: by-passing purification during reactor shutdown

The following table highlights the recontamination by Ag110m of the CVCS circuit, in unit C of a 900 MW unit, due to the by-pass before the oxygenation phase of the filter and resin purification on the primary circuit during reactor shutdown. The differences in the before and after shutdown spectrums highlight an over-contamination by Ag110m of the heat exchanger on the auxiliary purification circuit in the chemical and volume control system.



It is also noted that contamination of the reactor coolantsystem primary circuit loops is not affected by this malfunction. This confirms that Ag110m has a special affinity for the auxiliaries heat exchangers.

Use of CZT device for the purification of polluted power plants units

EDF Cleaning Engineering Unit proposes to set up a CZT measurement programme in addition to the programme carried out during shutdown in order to monitor the radiological characterization of the units pollution as well as the efficiency of the decontaminations implemented.

The CZT measurements are input data used to elaborate decontamination processes. For the units needing purification, target circuits RHRS/CVCS, NVDS (nuclear island vent and drain system) and LWPS (non-recycled liquid waste treatment system) tanks, pools have been identified. It has been shown that they can be efficiently decontaminated which leads to significant gains in terms of doses to personnel in the concerned zones.

Conclusions

The CZT spectrums obtained by the radiation protection department are indispensable in the diagnosis of circuit contamination as are the analyses of water conducted by the chemists. The first results obtained in the plants show the relevance of the tool in the understanding of contamination phenomena, the investigations to be conducted for prevention, the impact of malfunctions on over-contaminations (pollution). This tool allows improving good practices and providing better process control because of its good availability and ease of use.

Prospects

From 2011, during each shutdown on the EDF fleet, radiation protection department has to perform an optimised programme of CZT measurements. This programme integrates the feedback of both initial CZT programme and CZT measurements for the cleaning of the most polluted units. The strategy of this programme, the description of the measurements and the link made with the radiological conditions (dose rates) are developed in the Appendix at the end of the present document.

Appendix 3: ISOE Programme Information

ISOE was created in 1992 to improve the management of occupational exposures at nuclear power plants through the collection and analysis of occupational exposure data and trends, and through the exchange of lessons learned among utility and national regulatory authority experts. Since then, the system has grown continuously and now provides participants with a comprehensive resource for optimising occupational exposure management at nuclear power plants worldwide.

Membership in ISOE includes representatives from nuclear electricity utilities and national regulatory authorities who participate under the ISOE Terms and Conditions. The ISOE programme includes the participation of utilities and regulatory authorities in 29 countries. The ISOE database itself contains information on occupational exposure levels and trends at 470 reactor units worldwide (396 operating units; 74 in under decommissioning), covering about 91% of the world's operating commercial power reactors. To find out more about the ISOE programme: www.isoe-network.net

ISOE is jointly sponsored by the OECD Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA). ISOE operates in a decentralised manner. A Management Board of representatives from all participating countries, supported by the joint NEA and IAEA Secretariat, provides overall direction.

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Four ISOE Technical Centres (Europe, North America, Asia and IAEA) manage the programme's day-to-day technical operations, serving as contact point for the transfer of information from and to participants.

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Appendix 4: ISOE Expert Group on Water Chemistry and Source-Term Management (EGWC)

Comment [HBO47]: UPDATE (by the end of preparation period)

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