ISBN 978-0-9989666-5-6

IRPA 14

50 years

Proceedings of the 14th International Congress of the International Radiation Protection Association

Cape Town, South Africa
9 – 13 May 2016

Volume 5 of 5

10- Emergency Preparedness and Management

11- Decommissioning, Waste Management and Remediation

12- Societies

Organised in collaboration with:

European Commission

IAEA

ICNIRP International Commission on Non-Ionizing Radiation Protection

ICRP

ICRU

International Labour Office

International Organization for Medical Physics IOMP

ISRRT International Society of Radiographers & Radiological Technologists

NEA Nuclear Energy Agency

Pan American Health Organization

UNEP

United Nations Scientific Committee on the Effects of Atomic Radiation

World Health Organization

The Proceedings of the 14th International Congress of the International Radiation Protection Association are divided into 5 volumes, as follows:

Volume 1 of 5

Area 1: Fundamental Science
Area 2: Policy, Standards and Culture

Volume 2 of 5

Area 3: Medical
Area 4: General Ionising Radiation Protection

Volume 3 of 5

Area 5: Optimisation and Design of New Facilities
Area 6: Radiation Detection and Dosimetry

Volume 4 of 5

Area 7: Environment and Natural Background
Area 8: Transport / Sealed Source Management
Area 9: Non-ionising Radiation

Volume 5 of 5

***Area 10*: Emergency Preparedness and Management**
***Area 11*: Decommissioning, Waste Management and Remediation**
***Area 12*: Societies**

Proceedings of the 14th International Congress of the International Radiation Protection Association

EDITORIAL TEAM
**Christopher Clement, Jack Valentin,
Haruyuki Ogino, Devon Foote, Julie Reyjal,
Laila Omar-Nazir**

Published by
The International Radiation Protection Association

www.irpa.net

ISBN 978-0-9989666-5-6

Table of Contents

Note:
Submissions marked with Ⓡ represent entries published in a Special IRPA 14 issue of *Radiation Protection Dosimetry*.
Submissions marked with Ⓨ represent entries by contestants for the Young Scientists and Professionals Award.

The Proceedings of the 14th International Congress of the International Radiation Protection Association
Volume 5 of 5

Area 10: Emergency Preparedness and Management

Study on the Protection Planning Actions and Response to Nuclear or Radiological Emergency

Perez, Clarice F.A.[a], Sahyun, Adelia[b,c], Freitas, Kenia A.M.[d]

[a]Centro Tecnológico da Marinha em São Paulo, - CTMSP, São Paulo, SP, Brasil.
[b]Atomo – Radioproteção e Segurança Nuclear S/C Ltda, São Paulo, SP, Brasil.
[c]Instituto de Pesquisas Energéticas e Nucleares – IPEN-CNEN/SP, São Paulo/SP, Brasil.
[d]Navele Empreendimentos Ltda, Rio de Janeiro, RJ, Brasil.

Abstract. Nuclear or Radiological emergencies can have as a consequence the rise of Deterministic effects, in the population involved, and/or Stochastic effects due to their doses. In these situations, protective actions need to be done in order to keep the doses in the affected population below the levels of deterministic effects and protective actions that might reduce the risk of stochastic effects should be adopted, minimizing the doses to reasonably achievable levels. This work presents a comparative study between the publication of IAEA Safety Series 109 and the document of the International Atomic Energy Agency GSG-2 "Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency" regarding the effective dose value system to be used as a basis to trigger protection actions in the Planning and Response to Nuclear or Radiological Emergencies that can reduce the risk of stochastic effects.

KEYWORDS: GILS; Response Actions to Exhibitions Emergency Situations.

1 INTRODUCTION

In this work we compared the publications IAEA Safety Series nº109[1] and IAEA GSG-2 "Criteria for Use in Preparedness and Response for Nuclear or Radiological Emergency"[2] with regard to the effective dose values system to be used as a basis to trigger protection actions in the Planning and response to Nuclear or Radiological Emergencies that can reduce the risk of stochastic effects.

2 APPROACH GSG-2 "CRITERIA FOR USE IN PREPAREDNESS AND RESPONSE FOR A NUCLEAR OR RADIOLOGICAL EMERGENCY"[2]

The recommendations presented in the document address health consequences due to external exposure and internal exposure of specific target organs, for which the generic criteria were developed. This use of generic criteria meets the need for a common term for the system of values that would be used as the basis for the implementation of protective actions and other response actions. Table 1 provides a set of generic criteria expressed in terms of the dose that has been projected or the dose that has been received. The set of generic criteria expressed in terms of the projected dose is compatible with reference levels within a range of 20–100 mSv. According to this publication, taking protective actions at this level of dose will allow the occurrence of all deterministic effects to be avoided and the risk of stochastic effects to be reduced to acceptable levels. If a protective action is implemented effectively, the majority of the projected dose can be averted. The concept of averted dose is therefore useful for the assessment of the efficiency of individual protective actions or their combination.

Table 1: Generic Standards for Protection Actions and Response Exhibition on Emergency Situations to Reduce the Risk of Stochastic Effects

	Generic Criteria	Examples of protective actions and other response actions
Projected dose that exceeds the following generic criteria: Take urgent protective actions and other response actions		
$H_{Thyroid}$	50 mSv in the first 7 days	Iodine thyroid blocking
E H_{Fetus}	100 mSv in the first 7 days 100 mSv in the first 7 days	Sheltering; evacuation; decontamination; restriction of consumption of food, milk and water; contamination control; public reassurance
Projected dose that exceeds the following generic criteria: Take protective actions and other response actions early in the response		
E H_{Fetus}	100 mSv per annum 100 mSv for the full period of in utero development	Temporary relocation; decontamination; replacement of food, milk and water; public reassurance
Dose that has been received and that exceeds the following generic criteria: Take longer term medical actions to detect and to effectively treat radiation induced health effects		
E	100 mSv in a month	Screening based on equivalent doses to specific radiosensitive organs (as a basis for medical follow-up); counseling
H_{Fetus}	100 mSv for the full period of in utero development	Counselling to allow informed decisions to bemade in individual circumstances

Note: H_T — equivalent dose in an organ or tissue T; E — effective dose.

3 APPROACH THE IAEA SAFETY SERIES 109[1]

According to the document the intervention in nuclear or radiological emergencies should be based on the Generic Intervention Level System (GILs) and Generic Action Levels (GALs) adopted in order to guide the implementation of the various protection measures proposed for avoid or reduce the population's exposure to radiation.

Intervention levels are expressed in terms of dose which can be avoided in a time period, ΔT, corresponding to the duration of a specific protective action associated with the intervention, that is, the dose which the individual would be subjected in the absence of measurement, integrated in the ΔT period, minus the integrated dose which would be subject to the application of a protective measure. The action levels are expressed in radionuclide activity concentrations in water, milk and other foods. Intervention Levels (GILs) recommended for emergency protection measures are presented in Table2.

Table 2: (GILs) Generic Intervention Levels for Urgent Protection Actions

Protective Actions	Generic intervention level (dose avertable by the protective action)
Sheltering	10 mSv
Evacuation	50 mSv
Iodine prophylaxis	100 mGy
Temporary relocation	30 mSv in first month 10 mSv in a subsequent month
Permanent resettlement	1Sv in lifetime

4 CONSIDERATIONS GILs

The Generic Intervention Level for Urgent Protection Actions (GILs) provided in the publication IAEA Safety Series109/1/ were calculated using optimization. The following will show, briefly, the optimization calculation developed by IAEA Safety Series109/1/ to sheltering and then show an example involving sheltering. We emphasize that all the calculations for (GILs) followed the same methodology used to taking shelter and therefore are not showed in this paper.

5 CALCULATION TO SHELTERING

The publication /1/ in its page 76, developed a method to estimate optimized intervention levels for the movement of people and features the following equation to calculate the avoided dose that would justify a certain level of intervention:

$$\Delta\dot{E}(t) == \frac{a}{\alpha} \qquad (1)$$

Where:

$\Delta\dot{E}(t) =$ individual dose avoided per unit time
$a =$ the individual cost of maintenance actions per unit of time
$\alpha =$ unit cost of avoided collective dose

The publication/1/, are also the following considerations:

The measure of protection for sheltering consists, in general, to remain in indoors with doors and windows closed. No need to transport and food is available. The most important cost in this case is that caused by loss of productivity in the population involved. This cost can be estimated from the annual per capita gross domestic product divided by the number of days in the year:

$$(\text{US}\,\$20000 \div 365 \cong US\$55\,)$$

The value used for the publication was 20,000US$(Sv-person)$^{-1}$also value based on annual gross domestic product per capita. However, the publication on page74,explains that there are factors that could modify the alpha value, and that for this calculation was considered a factor of 2 uncertainty causing the alpha value ranges between10,000 and40,000US$(Sv-person)$^{-1}$.

Thus, the calculation has to sheltering as follows:

$$\Delta\dot{E}(t) = \frac{a}{\alpha} = \frac{\$55\,/\,day}{\$10000\ until\ \$40000\ \ /\,Sv} \cong 1{,}5\ to\ 6mSv\ \ /\,day \qquad (2)$$

As the approach for more than two days is considered detrimental to the population, the GIL was estimated for this time period. Then the dose value to justify avoided taking shelter is between 3 and 12mSv, it was then adopted the10 mSv value that is within this range.

6 SHELTERING EXAMPLE

For example we developed a calculation of the dose range of values that would justify taking shelter for two days for some countries of the European community based on the GDP per capita of 2013 the euro zone equivalent to US$ 34,060 as the International Monetary Fund (2013).

Table 3: Dose interval for Sheltering for 2 days for some countries of the European community based on the GDP per capita of 2013 in the euro zone

Country	PIB (US$)	a	alpha times two (Dose mSv)	alpha US$ 34,060 (Dose mSv)	alpha divided by two (Dose mSv)
Luxemburg	110423	303	9	18	36
Austria	48956	134	4	8	16
Netherlands	47633	131	4	8	15
Finland	47129	129	4	8	15
Ireland	45620	125	4	7	15
Belgium	45383	124	4	7	15
Germany	44999	123	4	7	14
France	42999	118	3	7	14
Italy	34714	95	3	6	11
Spain	29150	80	2	5	9
Cyprus	24761	68	2	4	8
Slovenia	22756	62	2	4	7
Greece	21857	60	2	4	7
Portugal	20727	57	2	3	7
Slovakia	17706	49	1	3	6
Lithuania	16003	44	1	3	5

Note that if the GDP of a country is high, higher shall be the dose value which justifies the sheltering, and if the GDP is low, lower doses justifies the sheltering. As we can see on table 3, the country that best fits the range found (3-12 mSv) in the SS-109[1] is Italy and this is because its GDP, US $ 34,714, is roughly equal to the GDP, US $ 34,060, of the European Community, which would reduce the expression (1) to the range below.

$$\Delta\dot{E}(t) = \frac{a}{\alpha} = \frac{\frac{\$}{365}}{\frac{\$}{2}\ until\ 2\$\ /\ Sv} = \frac{\frac{\$}{365}}{\frac{\$}{2}}\ until\ \frac{\frac{\$}{365}}{2\$} = \frac{2}{365}\ until\ \frac{1}{730} \cong 1{,}5\ to\ 6mSv\ /\ day \qquad (3)$$

7 CONCLUSION

The GSG-2/2 / deals with the issue with greater scope and in fact provides a common term for the value system that should be used as a basis for the application of the Protection Actions. In this publication the dose values equivalent and effective dose are presented (general criteria), which must not be exceeded and if it happen or will trigger the protection actions, or will be accepted provided it is proven its justification through an optimization process. Still according to this reference, a set of generic standards, expressed in terms of projected dose is compatible with reference levels within a range of 20-100 mSv. Protective actions to keep the doses compatible with this range, prevent the occurrence of any deterministic effects and reduce the risk of stochastic effects to acceptable levels. Given the above, the concept of Avoided Dose is only useful for evaluating the efficiency of the options available for protection actions and never as an end in it*s*elf.

The BSS nº109 [1] treats the dose avoided as justification to trigger protective actions such as taking shelter or evacuation, but avoided dose does not tell us what is the acceptable dose, did not exist, in this publication, a reference level that can be used as a guide. When we adopted, for example, 10 mSv of avoided dose for taking shelter, the question is: how much dose, not avoided, are accepted? What is acceptable in a nuclear or radiological accident? In terms of Radiological Protection, considering the methodology used in the calculation to justify taking shelter and observing the example of dose to sheltering developed in this work to countries of the European community, which means exactly avoid these 10 mSv? Suppose in an accident dose projected for a given individual is 11 mSv and that with taking shelter are avoided 10 mSv, it would be wise to keep this individual trapped inside your own home, which should remain with doors and windows closed for two days to it receives only 1mSv? This can be a very controversial issue, especially after comparison with the Reference Levels proposed in reference [2] which are among 20-100mSv, and according to the publication ensure that there will be no deterministic effect and the risk of stochastic effect is within acceptable limits.

8 REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Intervention Criteria in a Nuclear or Radiation Emergency, IAEA Safety Series No. 109. Vienna (1994).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency, IAEA General Safety Guide -2. Vienna (2011).

The reality of the low radiation dose on population in Fukushima Daiichi 20km zone

Jun Takada*

Physics Division, Sapporo Medical University, Sapporo 060-8556, Hokkaido, Japan

Abstract. For about four years since the 2011 earthquake and the Fukushima Daiichi Nuclear Power Plant (NPP) accident, the author's group continues the dosimetry survey focusing on the residents in the 20 km zone of the NPP. Including on-site condition, the results indicate external exposure at the level never inducing acute radiation injury. In particular, the effective dose to the people subjected to forced evacuation was lower than 10 mSv and their health effects would be negligible. The Fukushima incident demonstrated that the light water reactor accident would cause much lower dose condition than that of the graphite reactor accident of Chernobyl. The dose assessment for the residents in the 20 km zone determined by the former administration as uninhabitable was scientifically incorrect. Their assessment estimated the radiation exposure as high as over 50 mSv per year based on air dose measurement. Based on the individual dose measurements during the author's field survey for three days and two nights and the attenuation trend in the measurement results of surface cesium contamination density, annual dose in 2014, three years after the accident, has been estimated as 10 mSv or lower in many places and 0.5 mSv or lower in some areas. The exposure in the areas within the 20 km zone is at the levels allowing the people to return to their homes. From consideration to be compared with the historical events of the nuclear radiation disaster, Fukushima light water reactor events, was rated as level 6 a little less than in the international nuclear event scale.

KEYWORDS: Fukushima Daiichi 20km zone; Dosimetry studies; 2011~2014t.

1 LOW DOSE FOR FUKUSHIMA PEOPLE IN THE YEAR OF THE ACCIDENT

The author started the field survey of radiation hygiene for the Fukushima people in April 2011 by using a portable laboratory, the year of the earthquake disaster and has continued the survey for four years[1-7)]. The survey covers wide areas from the site boundary of the Fukushima Daiichi NPP to Futaba Town and Namie Town, Minami-soma, Iitate Village, Nihonmatsu City, Fukushima City, Koriyama City, Iwaki City, and Aizuwakamatsu City within the 20 km zone.

The fundamentals of the study are individual dose measurements of the Fukushima people including the author himself. External exposure dose and internal exposure dose due to Iodine-131 in thyroid and whole-body cesium have been measured. The measurement methodology is basically the same as that of the previous field survey in the world nuclear hazard zones[8, 9)].

Since there were no measured external dose data for the evacuees from the 20 km zone in emergency situation, the integrated dose for three days in March 2011 has been estimated based on the external individual dose of 0.05 mSv/day for two days measured by the author in April 2011 in the 20 km zone by taking account of decay rule (for every seven fold increase in time, total radioactivity decreases by a factor of ten, $A(t) = A_0t^{-1.2}$) [10)]. Table 1 shows the calculation procedure.

External dose to which the Fukushima people outside of the 20 km zone were exposed might be estimated as being much lower than the values in Table 1 based on the differences in the conditions of outdoor air dose rate and cesium contamination. Nevertheless, the doses are vague because no measured data have been published from the governmental emergency response headquarters.

A dosimetry expert in Koriyama City, Fukushima Prefecture measured and reported her individual dose [11)]. Based on her report, the author has analyzed emergency dose tracing back to March [10)]. As the result, annual external dose to a citizen in Koriyama City in 2011, the year of the disaster was estimated as 1.8 mSv. The value is lower than the estimated dose to the evacuees from the 20 km zone by a factor of three or larger.

[*]Presenting author, e-mail: juntakada@sapmed.ac.jp

Assuming that internal dose might be proportional to external dose, the total of internal and external effective dose to the Koriyama citizens in 2011, the year of the disaster can be estimated as 2.4 mSv/y[13)]. The value may be considered as significantly lower dose.

The basic survey of the health management survey for the Fukushima people by the Fukushima Prefecture has estimated external dose for four months in the early stage of the accident based on the investigation on activities of the people. According to the aggregated results until July 2013, among 435,788 people excluding radiation workers, 65.9% received dose lower than 1 mSv, 94.8% received dose lower than 2 mSv, 99.8% received dose lower than 5 mSv, and maximum was 25 mSv [12)].

Although these values do not represent directly measured individual dose, they are almost consistent with the author's measured values and measured values in Koriyama City. In summary, annual external dose to the Fukushima people in 2011, the year of the disaster, can be generally assessed as 5 mSv or lower.

Table 1: Estimation of external dose to the evacuees by the author's field study near Fukushima Daiichi in April.

Date	Elapsed days	Total decay ratio	Dose rate (mSv/day)
March 11, 2011			
March 12	1	1.000	2.84
March 13	2	0.435	1.24
March 14	3	0.268	0.76
			4.84
April 8	27	0.019	
April 9	28	0.018	0.05
April 10	29	0.018	

In addition, the author has summarized other exposure data[10, 13)]. Sources consist of the primary data on emergency workers reported by the Tokyo Electric Power Company (TEPCO), the dose data on the Japan Ground Self Defense Force (JGSDF) staff measured by the JGSDF medical unit, and the thyroid dose data on the Fukushima people reported form the various organizations including the Fukushima Prefecture, the National Institute of Radiological Sciences, and the group of the Hirosaki University[14-17, 13)]. Table 2 shows external, internal doses, and dose levels based on effective dose.

The study of the effects on fetuses in the survivors from nuclear disasters in Hiroshima and Nagasaki revealed that a threshold might be at about 0.10 Sv. In a case of instantaneous exposure, the exposure under the level can induce no health damage[18)]. Hence, no effect on fetus can be induced among the Fukushima people by radiation exposure due to the Fukushima Daiichi NPP accident.

The epidemiological data of late effects due to instantaneous organ dose for 80 thousand cases of survivors from Hiroshima nuclear disaster has showed elevated relative risk higher than 200 mSv[19)]. Therefor we can say no risk of such as thyroid cancer for the Fukushima people.

The following-up study of 78 survivors within the 500 m radius of the ground zero in Hiroshima indicated that although they were exposed to 1-6 Sv and the average was 2.8 Sv, their average age at death was 74 years old. Furthermore, although they had acute radiation damages, 13 people and three people lived a long life of up to the age over 80 and over 90 years old, respectively [9, 20)]. It has been proved by studies on Hiroshima nuclear survivors that the human body is resistant to ionizing radiation.

Table 2: Dose resulting from the Fukushima Daiich NPP accident

Group	Dose (Sievert)			Dose level [e]
	External exposure maximum	Internal exposure maximum		Effective dose
		Thyroid Iodine-131	Whole body Cesium	
NPP personnel [a]	0.2	12	0.05	D – C
JGSDF [b]	0.08	0.01 – 0.1 [c]	0.004	D – C+
Vicinity residents * [d]	0.005	0.04	0.001	D

a According to TEPCO, there were 20,103 emergency workers at the Fukushima Daiichi NPP until January 2012. Tissue weight factor for the thyroid was designated as 0.05 (ICRP60, 1990 Recommendation).
b Among the JGSDF staff dispatched in response to the nuclear emergency, 168 (including two women) were at levels over 5 mSv. JGSDF report.
c Thyroid dose of the JGSDF staff is based on the estimation reported in Takada, 2014.
d There has been no measured external dose data for residents within the 20 km zone provided by the government. For these residents, individual dose estimates are based on April 2011 measurements taken by Takada. A dosimetry expert in Koriyama City with a personal dosimeter indicated the levels were at 0.002 Sv. Thyroid dose for 1202 persons is cited from the 2012 NIRS sponsored expert meeting.
e Dose is classified by six levels. (see Figure 1).

Even in a case exposed to higher dose not reaching to lethal range, if overcoming acute effects, no significant life- shortening effect might be induced. Therefore, no life-shortening effect will be observable among the Fukushima people exposed to the effective dose lower than 10 mSv. True risk is superstitious fear on radiation and extended forced evacuation.

In case of Hiroshima nuclear explosion disaster, since the radiation source is rising rapidly in the sky, the exposure dose rate was very high of several Sievert in 1 minute. In contrast, the exposure of Fukushima radiation disaster is a low-dose rate of up to a few tens of days from a few days. This also means that radiation risk in Fukushima prefecture is in a negligible range.

2 THE FUKUSHIMA LWR EVENT TO BE CLASSIFIED AS LEVEL SIX

On April 12, 2011, the Nuclear and Industrial Safety Agency (NISA), the Japanese competent authority at the time announced the interim assessment of the fourth report that the Fukushima light-water reactor (LTR) event in 2011 was classified as level seven in the International Nuclear and Radiological Event Scale (INES). The basis of the assessment was the trial calculation of the total amount of radioactivity released from the nuclear reactors in the Fukushima Daiichi NPP into the atmosphere. The estimated total amount of radioactivity released into the environment would normally involve significant uncertainty. However, no report including such the scientifically sound argument was prepared by the NISA. Namely, no scientific basis was presented for the classification of level seven that the NISA concluded. In the Journals of the Atomic Energy Society of Japan, the classification of level seven for the Fukushima event that the NISA concluded has been objected and suggestions that "the event should be classified as level five based on core damage conditions and a proper classification might be level six at maximum" have been submitted [21)].

In this paper, we discuss the level assessment of the Fukushima LWR event based on the comparison with the dose of general public in the historical nuclear events [8, 9)]. The classification of level seven was assigned to the Chernobyl graphite reactor (GR) accident that caused acute deaths onsite (onsite dose level: B to A; offsite: C). A judgement that the Fukushima LWR event without acute death (onsite dose level: D to C; offsite: D) should be classified as the same would be apparently inconsistent.

In the Chernobyl accident that the reactor was being operated with shut-off power supply to the cooling system and a runaway of nuclear fission chain reaction in a reactor occurred, the reactor was destroyed instantaneously and then operating personnel and firemen received radiation dose of 4Sv or higher. There were thirty acute death. Just after the runaway, high temperature graphite fire occurred. This allowed large amount of radioactivity to be released into the environment and strontium and plutonium as well as radio-iodine and cesium contaminated the surrounding area significantly [9)].

On the other hand, the LWRs in the Fukushima Daiichi NPP terminated their nuclear fission chain reaction automatically with the detection of seismic wave and no reactor was destroyed. But, then tsunami of up to 15 m high caused their cooling function to be lost and the reactor cores to overheat. As a result, the molten reactor cores penetrated through the reactor pressure vessels to the containment vessels and eroded the bottom of the containment vessels by the depth of about up to 60 cm [2, 13)].

Resulted hydrogen gas leaked into the reactor buildings and hydrogen explosion occurred after 24 hours or later. Short-lived radioactivity having the half-life of minutes to hours decayed substantially. Thus, although volatile radioactive materials dispersed into the environment, majority of metallic nuclides were contained in the containment vessels. Hence, while Iodine-131, Cesium-134 and -137 contaminated the land area significantly, the contamination of Strontium-90 was insignificant and its level was a thousandth or lower as compared with cesium contamination.

Due to the difference between the reactor designs, the Chernobyl event and the Fukushima event became a higher dose accident and a lower dose accident, respectively. As a result, the latter caused no acute radiation injury and nobody died (see Table 3 and Fig. 1).

A container of high-level radioactive wastes at the Mayak plutonium production facility exploded in the former USSR in 1957 [22)]. So called Kyshtym accident led high level contaminations in a wide area of 120km with Sr-90 of 74 GB/km^2 (74 kBq/m^2) and caused the general public in the vicinity to receive 50 to 500 mSv (dose level: D+ to C). This disaster was classified as level six. The radioactive contaminations and doses of the Fukushima LWR event is much lower than that of Kyshtym disaster.

The dose in the Fukushima is significantly higher than that of the Three Mile Island LWR event classified as level five (onsite: D to D+ , thyroid: D, offsite: E) [23, 24)]. Therefore, the Fukushima LWR event should be classified as level six or little under level six [27)].

Figure 1: Dose classification by six levels

Table 3: International Nuclear Event Scale

Level	Evaluation example
7 Severe accident	Chernobyl accident Graphite reactor(1986, USSR)
6 Large-scale accident	Kyshtym accident Nuclear waste storage facility (1957, USSR) Fukushima accident Light water reactor (2011, Japan)
5 Accident involving a risk to the off-site	Three Mile island accident Light water reactor (1979, USA)
4 Accident not involving a significant risk to the off-site	JCO Criticality accident Uranium fuel processing (1999, Japan)

Maximum thyroid dose that the general public received in the Chernobyl accident was 50 Sv. About 4800 children in the vicinity suffered from thyroid cancer and were treated. Among them, 15 died.[25)] In contrast, maximum thyroid dose that the general public received in the Fukushima accident was 0.04 Sv and a thousandth or lower as compared with the Chernobyl. Even if we estimate the risk based on the LNT model, only one person per ten million people would suffer thyroid cancer. Hence, nobody would suffer from thyroid cancer in the Fukushima Prefecture having a population of about two million. Actually, a case-control study by the Fukushima Prefecture and the Japanese Government for thyroid using 10 MHz ultrasound echo could reveal no increase in thyroid disease being proportional to radiation dose among the Fukushima children[26)].

3 REMARKABLE REDUCTION OF ANNUAL DOSE IN FUKUSHIMA 20KM ZONE

In March 2012, one year after the accident, we stayed for three days and two nights in the Suenomori section of Namie town, 10km far from Fukushima Daiichi. A personal dosimeter attached to chest of the author indicated 0.051 mSv per 24 hours. Taking attenuation due to the physical half- life of two Cs isotopes (2 years and 30 years) into account, if a person was to take up residence on a ranch in the area for a year in 2012, his annual external dose could be assessed as 17 mSv [5, 6)].

Significantly, this value is less than the 20 mSv the government has recommended as the level at which residents should be allowed to return to their homes. Moreover, without conducting any personal dosimetry, the accident response headquarters of the government determined as 50 mSv or higher per year. If homes and pasture surface soil are decontaminated under governmental supervision, the dose in Suenomori will become 5 mSv or lower quickly. The current policy, however, has no scientific basis and most of the area in the 20 km zone is considered uninhabitable.

Using the values measured by a personal dosimeter for external exposure in March 2012 and the attenuation function of cesium concentration measured annually in the disaster year (2011) and subsequent years in pastures in Namie, the annual external dose from cesium gamma rays can be calculated. Figure 3 shows the results.

In 2012, one year after the accident, Takase had been in the condition allowing return. Namely, the dose was 1 mSv or lower. Also in Suenomori, since the dose was 20 mSv or lower in 2012, preparation for return could be started. In Komaru under more severe condition, the dose will be 20 mSv or lower in 2015 or later and hence the environment can be expected to become radiation condition allowing preparation of return. These external doses represent the values under the condition without the decontamination of cesium on surface soil.

The estimated annual external dose (mSv) in 2014, three years after the initial incident, based on the attenuation function of cesium radioactivity is 0.29 in Takase, 7.4 in Suenomori, and 27 in Komaru, three sites within the 20 km zone. In September 2014, a personal dosimetry trend in time series was measured in Namie and based on the results and the cesium surface concentration on pastureland, the annual external dose under the assumption that a person would live in the town could be estimated. Figure 4 shows the map within the 20 km zone summarizing the doses in 2014.

Scientifically assessed dose value is less than the 20 mSv the Government has recommended as the level at which residents should be allowed to return to their homes. If homes and pasture surface soils are decontaminated under governmental supervision, the dose in Suenomori will become 5 mSv or lower quickly. The current policy, however, has no scientific basis and most of the area in the 20 km zone is considered uninhabitable.

Figure 3: Trend of annual external dose (Cs) in Namie within the 20 km zone of the Fukushima Daiichi NPP. Cesium radioactivity on a ranch in the zone has already significantly decreased by the effective half-life of 1.7 years. Measured internal exposure by Cs of stockmen is less than 0.1 mSv in third year or later. Year 1 = 2011.

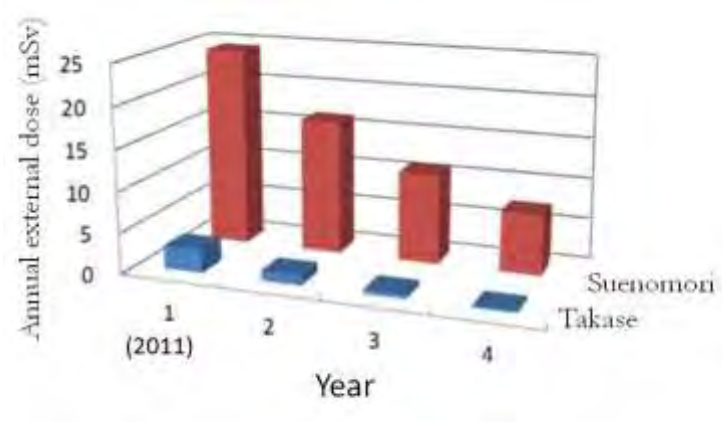


Figure 4: Sectioning of habitation-restricted zones in Namie Town and estimated annual dose rate in 2014, cesium radioactivity decays with an effective half-life of 1.7 years between 2012 and 2014.

Three years after the disaster, the annual dose rate assessed by a personal dosimetry for those residing within the 20 km zone is 10 mSv or lower in many places and 1.0 mSv or lower in some areas. Both the decontamination of the surface soil in agriculture/stock-raising lands and the rapid restoration of damaged social infrastructures to expedite return are the responsibility of the current administration.

4 CONCLUSION

The Fukushima LWR incident was a low radiation dose event quite different from the Chernobyl GR accident that induced higher radiation dose. No deaths occurred due to radiation exposure in the Fukushima Daiich NPP site. External dose to the general public in the vicinity was generally 5 mSv or lower in 2011, the year of the accident and thyroid dose was lower than 40 mSv. According to the knowledge from conventional radiological protection, these dose levels were within the range in which no health risk could be imposed. Fukushima LWR events, was rated as level 6 a little less than in the international nuclear event scale.

All other LWR accidents such as Three Mile Island induced lower radiation dose or no radiation dose. The results can be understood as the consequence from the fundamental mechanism of LWRs. It could be verified that the basic measures to protect the general public would be the sheltering indoors, the administration of stable iodine drugs and a short period control of food distributions for LWR emergency cases. Cesium radioactivity in ranch surface soil within the 20 km zone continues to decay by an effective half-life of 1.7 years between 2011 and 2014. In 2014, three years after the accident, based on the personal dose measurements during our field survey for three days and two nights and the attenuation trend in the measurement results of surface cesium contamination density, we could estimate annual dose in the 20 km zone. The annual dose rate was 10 mSv or lower in many places and 0.5 mSv or lower in some areas. The dose in many areas within the 20 km zone is lower than or equal to the annual dose limit of 20 mSv allowing the people to return to their homes specified by the accident response headquarters of the government. Hence, actually such the areas should be certified as the zone being prepared to return.

5 REFERENCES

[1] Jun Takada, Fukushima: Myth and Reality, Iryokagakusha, 2011. (in Japanese)

[2] Jun Takada, Fukushima: Myth and Reality, Iryokagakusha, 2012.

[3] Jun Takada, Path to reconstruction in Fukushima, as seen through fieldwork in the East Japan, The fourth "True Interpretations of Modern History" essay contest 2011. Best Seiji Fuji Award: 1 winner paper, APA Group, http://www.apa.co.jp/book_ronbun/vol4/2011japan.html. (in Japanese)

[4] Jun Takada, Path to reconstruction in Fukushima, as seen through fieldwork in the East Japan, The fourth "True Interpretations of Modern History" essay contest 2011. Best Seiji Fuji Award: 1 winner paper, APA Group, http://www.apa.co.jp/book_ronbun/vol4/2011english.html 2011.

[5] Jun Takada, The radiation hygiene survey for human and cattle in Fukushima: No health damage in low dose range, Sapporo Journal of Medicine, 81, 1-5, 2012. (in Japanese)

[6] Jun Takada, Low dose radiation and no health risk in Fukushima in contrast to Chernobyl, Genes and Environment, 35, 69-72, 2013.

[7] Jun Takada, Mohan Doss, and Sadao Hattori, Risk of no radiation, Overcoming the world crisis caused by the LNT model, Iryokagakusha, 2014. (in Japanese)

[8] Jun Takada, Worldwide studies on radiation hazards, Kodansha, 2002. (in Japanese)

[9] Jun Takada, Nuclear Hazards in the World, Springer and Kodansha, 2005.

[10] Jun Takada, Definitive results from Fukushima Dosimetry Survey. J. Rad. Prot. Dos. Med. 11, 1-10, 2015.

[11] Midori Iwasaki, Radiation dose I have been exposed while living in Koriyama City, proc. of 50th Annual Conference of Isotope and Radiation, Japan Radioisotope Association, 2013. (in Japanese)

[12] Implementation status of the Basic Survey, Health Management Survey for Fukushima People, People Health Management Section, Health and Welfare Division, Fukushima Prefecture, Homepage of Fukushima Prefecture, http://www.pref.fukushima.lg.jp. uploaded/attachment/50301.pdf. (in Japanese)

[13] Jun Takada, Definitive Studies on Fukushima Radiation Hygiene, 1-203, Iryokagakusha, Tokyo, 2015.

[14] Toshiyuki Hayashida, Akira Suzuki, Hideki Oda, Toshihiro Ueneo, and Masayoshi Goto (Tokyo Electric Power Company), Radiation dose assessment and current situation after Fukushima Daiichi Nuclear Power Plant Disaster, proc. of 47th Annual Conference, Japan Health Physics Society, 2014.

[15] Tokyo Electric Power Company: Latest radiological condition, http://www.tepco.co.jp./cc/press/2014/1236869_5851.html (in Japanese); Interim Report of the Accident of Fukushima Daiich Nuclear Power Station, June 20, 2012 (in Japanese), http://www.tepco.co.jp/nu/fukushima-np/interim/index-j.html (in Japanese); and On the information of exposure dose to workers engaged in the recovery activities for Fukushima Daiichi NPP accident submitted to the World Health Organization, http://www.tepco.co.jp/nu/fukushima-np/images/handouts_121206_01-j.pdf. (in Japanese)

[16] Tetsuo Yamamoto, Hidetaka Eguchi, and Yutaka Naoi, Nuclear emergency activity by Medical Unit, Japan Ground Self-Defense Force, Journal of Radiation Protection Medicine, No. 8, 10-16, 2012. (in Japanese)

[17] Shinji Tokonami et el: Thyroid equivalent doses due to radioiodine-131 intake for evacuees from Fukushima daiichi nuclear power plant accident, proc. of The 1st NIRS symposium on reconstruction of early internal dose in the TEPCO Fukushima Daiich Nuclear Power Station

accident, 51-58, 2012.
[18] Health effects of atomic bomb radiation, 1992, Hiroshima International Council for Health Care of the Radiation-exposed, Bunkoudo, Tokyo, 1992. (in Japanese)
[19] M. Ichimaru, T. Ishimaru, and J. L. Belsky, Incident of leukemia in atomic bomb survivors belonging to a fixed cohort in Hiroshima and Nagasaki, J. Rad. Res. 1978.
[20] Nanao Kamata et al., Comprehensive medical study of close range atomic bomb survivors, 25th report, Follow-up survey results for 25 years, Hiroshima Journal of Medicine, 51, 355-357, 1998. (in Japanese)
[21] N. Narabayashi and K. Sugiyama. J. Japan Atomic Energy Association, 387-400. 2011.
[22] M. M. Kossenko, M.O. Degteva et. al. Rad. Res. 148, 54-63.1997.
[23] Nuclear Regulatory Commission Conference Documents, May 3, USA, 1979.
[24] Nuclear Regulatory Commission Preliminary Notification, PNO-79-67 A-Z, April-June, 1979.
[25] World Health Organization: Health Effects of Chernobyl Accidents and Special Health Care Programs, 2006.
http://www.who.int/ionizing_radiation/chernobyl/who_chernobyl_report_2006.pdf
[26] Fukushima Prefecture: Thyroid examination,
http://www.pref.fukushima.lg.jp/sec/21045b/kenkocyosa-kojyosen.html. (in Japanese)
[27] Jun Takada, Worldwide studies on radiation hazards revised, Iryokagakusya, Tokyo, revised 2016. (in Japanese)

Application of virtual reality technology to minimize the dose to the example of staff emergency training at the center for radioactive waste management

K. Chizhov[a*], I. Kudrin[a], I Mazur[a], A. Tsovyanov[a], L. Bogdanova[a], M. Grachev[a], I. Tesnov[a], N.-K. Mark[b], I. Szöke[b], V. Kryuchkov[a]

[a]SRC-FMBC.
[b]Institute for Energy Technology.

Abstract. The paper presents the use of two complementary computer software that both are based on the Virtual Reality (VR) technology: Andreeva Planner (dynamic visualization of a radiation environment) and EasyRad (information-analytical system for radiation safety). The scenario of the accident: during the transportation of a container with radioactive waste from one dry storage unit to a second storage site an emergency situation occurs that leads to a fire and a subsequent release into the atmosphere of a mixture of radionuclides. Using data about the surface activity the ambient dose rate is calculated for the site. These data is then added to the 3D model of the site, which allows one to calculate the doses to the personnel at any trajectory on the territory of the facility. It is demonstrated how the application Andreeva Planner can be used for analysis of the consequences of the accident, for planning radiation surveys and additional inspections, for mapping the radiation situation, analysis of charts and maps to identify vulnerabilities. Andreeva Planner and EasyRad solve the following challenges during the elimination of the accident consequences: bypass of control points, decontamination of road surface, paving the main routes of personnel and planning major radiation-hazardous works. Several scenarios of each work are developed using Andreeva Planner for optimization of liquidator's doses. Creation of scenarios takes into account two principles: the principle of a major source and the principle of conquering space. The results of all the calculations are given in an interactive report that is available for viewing in the geographic information system.

KEYWORDS: radioactive waste management, research emergency exercise, accident, software, virtual reality technology.

1 INTRODUCTION

Training activities and exercises in the Northwest Russia are carried out to maintain the required level of preparedness of the FMBA's territorial units involved in the emergency response. During the Research Emergency Exercise (REE) that will take place in June 2016, the advanced computer methods will be used to simulate the radiation situation, assess the preparedness of the rescue emergency teams and medical brigades, including their psychological training in the mitigation of the consequences of a radiological accident. The results from previous exercises have been an important tool for improving emergency preparedness and working out standards for caretaking to the Federal Medical-Biological Agency of institutions (FMBA) of Russia.

The practical significance of this exercise is that it takes place at an enterprise with a modern infrastructure for the management of spent nuclear fuel and radioactive waste where preparations are taking place for the removal of radioactive materials for further processing.

The development of the scenario showing the radiological consequence of the conditional radiation accident was done by simulating interrelated values of radiation parameters and exposure pathways affected by using special software. This includes the modelling and assessment of radioactive release on the territory of the neighbouring countries based on the data network for hydro meteorological monitoring and surveillance.

[*]Presenting author, e-mail: kchizhov@fmbcfmba.ru

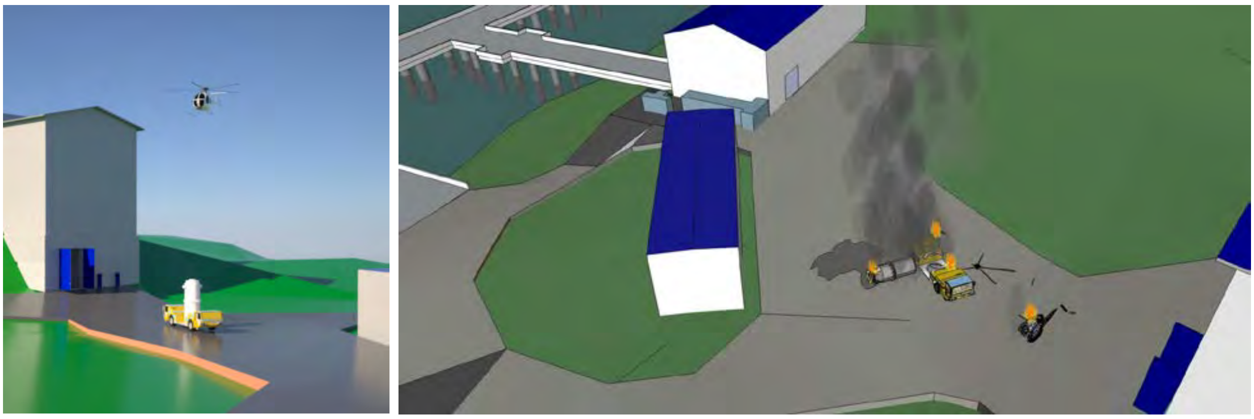
An important outcome was the testing and implementation in practice of software for the visualizing of the radiation situation and for the optimization of radiation-hazardous works. The purpose of this article is to show the ability to use virtual reality technology - as an example of a new approach - using specialized computer technology in emergency planning and in the response for a number of purposes: carrying out emergency exercises and training, modelling of the radiation situation, radiation survey planning, planning and optimization of radiation-hazardous work, optimization of doses for the participants in the liquidation of the consequences of the accident in addition to dose reconstruction for witnesses and victims of the accident.

2 INITIAL PARAMETERS OF REE

At the time just before the accident the staff is on the workplace at the site, where the work is going on in accordance with the plan for the removal of the spent fuel elements. The purpose with the work is to move the transport packing container (TPC) to the funded area. The reference levels of the radiation factors are not exceeded.

The initial event of the accident is the following: A light aircraft in the form of a helicopter falls on the vehicle loaded with the TPC (Fig. 1). The consequences are: a fire due to the helicopter crashing into the vehicle with the TPC, partial depressurization of the TPC with an increase in the dose rate of gamma radiation in the area of the accident and the release of radioactive pollution into the atmosphere from the heated air due to aerodynamic forces to a height of several meters, which results in the contamination of a location nearby. The radiation exposure level outside the technical area is normal. The fire is put out as a result of the actions of the facility workers and staff of the special fire brigade. The consequences of the radiation accident are then localized.

Figure 1: 3D-model showing the fall of the helicopter on the vehicle resulting in a fire.



3 MODELLING OF RADIATION ENVIRONMENT

Due to the chaos caused by the accident there are no useful data on the release of radioactive contamination just after the accident. Therefore an impact assessment and the initial steps have to be carried out on the basis of the results of measurements of the dose rate and the surface contamination. For the REE case we calculated these parameters. These numerical values are quantitatively related to each other on the basis of model calculations performed by IBRAE RAN. As an initial assumption the maximum dose rate was set to 100 μSv/h. The results of simulating the radiation conditions were as follows:

- The industrial site is found to be contaminated
- A field with a concentration of radioactive particles is found at a height of 1.5 meters above the ground
- The dose rate at time increases from 0 to 1200 seconds just after the accident happens

These calculations were performed using the specialized software tools designed to assess the radiation situation within the industrial site. The calculation takes into account the contribution of the buildings in addition to the uneven terrain and the atmospheric conditions (different stratification, wind power). In the simulation of the radiation situation the following initial data was chosen:

- Wind speed at 10 m above the surface
- Atmospheric stability category
- The value of dry deposition velocity
- Release time
- Wind direction

4 SOFTWARE INVOLVED IN THE REE

Two complimentary applications based on virtual reality technology will be used during the exercise: The Andreeva Planner and the Information-analytical system for radiation safety of workers (EasyRAD).

4.1 Andreeva Planner

Andreeva Planner is a real-time software tool for modelling and characterising nuclear environments, for planning a sequence of activities in the modelled environment, optimising protection against radiation, and producing job plan reports with dose estimates [1,2]. It offers the possibility to refine the radiological model to improve the accuracy of estimates and configure the dosimetric output provided. The software can also be used as an aid to producing post-work review reports, with real measurements included. Furthermore, the software provides support for presenting information to different types of users for briefing and decision-making, thus serving as an aid to communication between stakeholders in an intervention. To support the user in interpreting the results of calculations, the Andreeva Planner provides charts, graphs, and 3D radiation visualization, updated immediately to reflect any changes to the modelled radiological condition, such as changing shielding materials, and human activities over time. The software supports the activities in a typical radiological protection planning, optimization and monitoring workflow [3].

4.2 EasyRAD

EasyRAD is an information-analytical system for the radiation safety of workers [3]. This software can be used in both normal and emergency situations on the enterprise. It allows you to have a database with measurements of the radiation situation and enterprise schemes, calculate dose rate maps, and calculate doses of workers. EasyRAD also has a number of analytical functions designed for radiation safety services: delineation of territory at selected dose levels, search for the best ways to work around selected points and finding optimal routes of evacuation [4].

5 ACTIVITIES JUST AFTER THE ACCIDENT

After the accident, the virtual reality environment (in the form of a 3D model) is used to analyze the "evidence of the accident"[2], to plan radiation survey and inspection of damages, plot radiation situation specific maps, examine the maps and reveal vulnerabilities of these maps, schedule additional radiation survey and plot refined radiation situation specific maps.

[2] The evidence of the accident includes indications of automated radiation monitoring systems, data from stationary dosimeters (if available), and individual dosimeters from witnesses and victims of the accident.

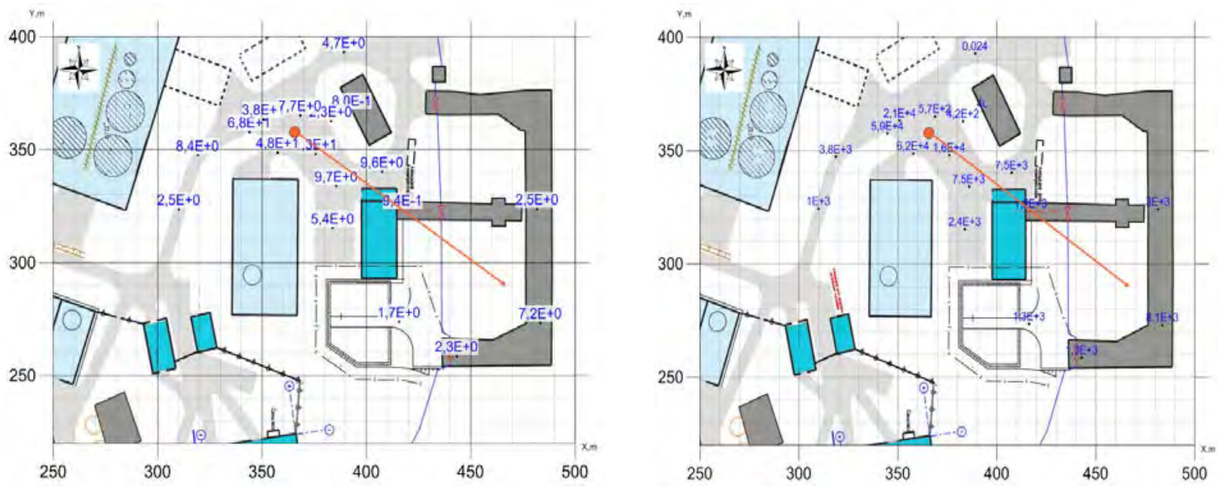
5.1 Analysis of "evidence of the accident"

Just after the accident, the radiation safety service gathers individual dosimeters from witnesses and victims of the accident, collects information about their routes from the moment of the accident and until they get into the safe area. These routes are used as input to EasyRAD for future comparison between the results of the instrumental dosimetry and the calculations. The second input is data from automated radiation monitoring systems, under the condition that these systems are still working, and also input from stationary dosimeters, if they are available at the moment.

5.2 Radiation survey planning

The primary planning of the radiation survey is done with significant uncertainty about the real situation. This planning procedure is based on information about the location of the accident, data on the wind direction at the moment of the accident and fragmentary observations from witnesses to the accident. The survey should start in the lee of the periphery of the proposed affected area with the greatest possible use of existing roads and pedestrian crossings in the direction perpendicular to the axis of the intended track. The groups of dosimetrists pass through specified routes making measurements in the set of reference points and reports the values of dose rates.

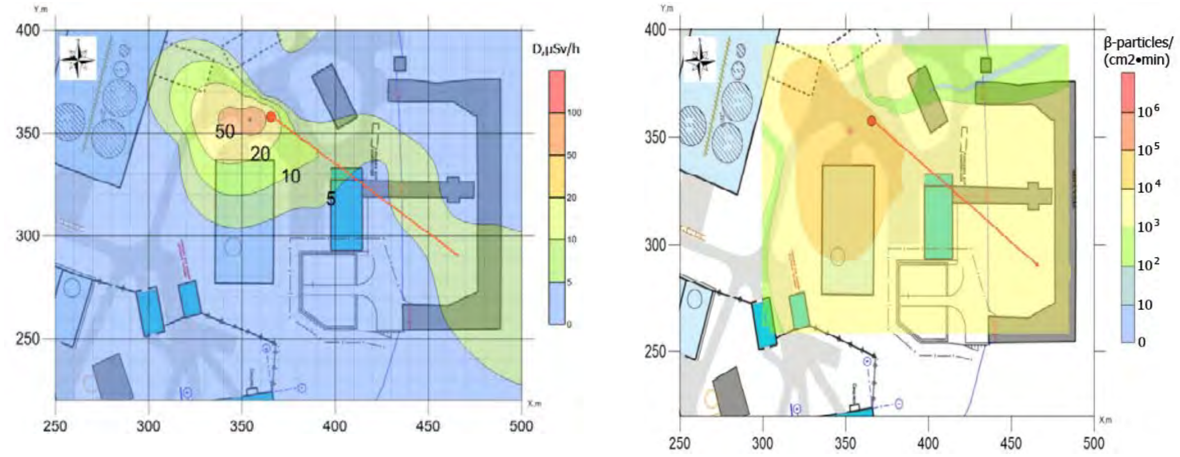
Figure 2: The results of the dose rate measurements in μSv/h (left figure) and the surface contamination in β-particles/(cm^2·min) (right figure). The orange circle is the crash site of the helicopter and the orange arrow is the wind direction.



5.3 Plotting the specific radiation situation map

The radiation survey results are put into EasyRAD. Then, the measured data are interpolated [5] to obtain radiation situation grids and plot the associated maps (fig. 3).

Figure 3: Left figure - example of the radiation environment map (μSv/h), right figure - surface contamination (β-particles/(cm^2·min)), plotted on the basis of results of the first (primary) radiation survey.



5.4 Analysis of the radiation situation specific maps and finding vulnerabilities of such maps

The radiation safety service analyses the radiation situation maps to reveal vulnerabilities. The vulnerability analysis is done using two algorithms, the first on based on the highest relative difference between the measured and interpolated dose rate values in the points of measurement and the second one by using the maximal dose rate gradient [4]. Each algorithm identifies 10 vulnerability points and the ranking of each point is shown by the size of the point. A set of vulnerability points from the first algorithm is compared with the points identified by the second one. The result of the comparison is the vulnerability points integral ranking and based on that the final set of points is selected in which one needs to do additional measurements. Further, the radiation survey can be repeated by carrying out new measurements in points selected in the previous step meaning that one will do a minimum number of measurements while getting the maximum understanding of the radiological situation.

6 AFTERMATH MITIGATION ACTIVITIES

At the time starting the aftermath mitigation activities, the radiation situation and the damage to buildings and equipment is well known[3].

6.1 Planning the inspection of reference points

The radiation survey will be a routine procedure that is repeated periodically. An optimal sequence of the reference points to be inspected should be determined (The reference points of the first survey plus those of the refined ones) for making measurements. For this purpose, the problem called the "traveling salesman"[4] of the graph theory should be solved. Moreover, one should try to determine the maximum number of tasks. That is solving the problem of choosing the optimal set of reference points which describe the radiation situation with an error rate not exceeding the specified one.

6.2 Planning of the decontamination pavement

One of the first tasks to be performed is cleaning the roads and driveways within the industrial site. In particular, the problem of optimal routes should be solved using graph theory methods.

[3] According to the exercise scenario, there is no damage to the buildings.
[4] The name of the classical problem of graph theory.

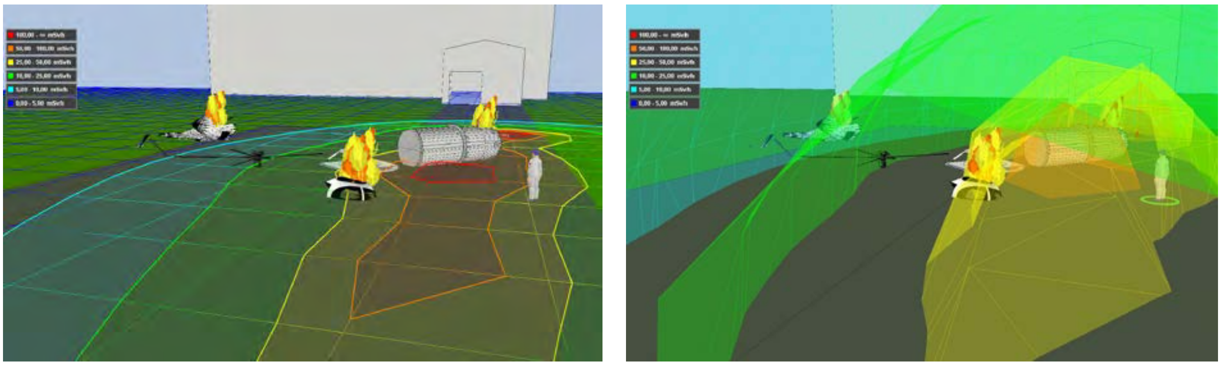
6.3 Driving the main routes

A list of locations for frequent staying of the workers during the aftermath work can be generated in advance. This includes the entry point to the controlled access area, sanitary pass, decontamination facility, etc. Using EasyRAD, the optimal routes between such points can be determined.

7 DEVELOPMENT OF SEVERAL SCENARIOS OF WORK IN THE SOFTWARE

Scenarios for the aftermath operations are developed using the Andreeva Planner software on the basis of the work plan. Each scenario is planned in the virtual reality environment and accompanied by calculating the time for work completion, the individual dose to each worker and the collective dose to all players. Then, based on the comparison of dose rates, the optimal scenario should be selected.
The Andreeva Planner offers an efficient way to model (simulate) work scenarios and estimate associated radiological risks in real-time (fig. 4). In addition, the tool allows the user to easily identify the source of the exposure to the participants in the modelled scenario. The user is able to quickly associate elevated dose to radiation sources or radionuclides present in the scene and to work steps within the scenario. This allows the user to quickly find alternative solutions that result in more optimal exposure.

Figure 4: Scenario in Andreeva Planner, 2D and 3D visualization of radiation environment.



The real-time capability of the Andreeva Planner allows the user (decision maker) to see how exposure changes dynamically with only the slightest modification of the work strategy. This allows the user to dynamically modify the strategy and experiment with alternative solutions until an optimal solution is found. The real-time capability also grants the user the ability to see how radiological conditions within the scene dynamically change as a result of changes to the scene resulting from actions of workers, e.g. due to adding or removing biological shielding. This capability makes it easy for the user to determine the optimal directions or updating a work scenario in the process of finding better alternatives. However, in some situations, very different multiple work strategies, involving different technology and different number of participants, need to be compared. The Andreeva Planner offers user-friendly functionalities for storing work scenarios and quickly comparing multiple scenarios in terms of multiple parameters, e.g. individual and collective dose and time requirements.

7.1 Planning local decontamination

When planning such operations, two principles must be followed: The major source principle and the principle of the space conquest [6].

7.1.1 The major source principle

The entire area is conventionally subdivided into several sites enclosing powerful radiation sources. The working site is selected, during the work, where the source is suppressed, which gives the maximum contribution to dose and dose rates at other sites. Within this site, contribution from “foreign” sources will be the lowest and much lower than that of the “native” source. Thus this site is determined as the first one, where local decontamination operations should be carried out. Therefore, “useless” exposure induced by sources, not suppressed during this operation, becomes minimal.

7.1.2 Principle of the space conquest

The optimal sequence to carry out some parts of the decontamination work (within the certain site) always corresponds to the beginning of work with the most radioactively contaminated side of the site accompanied by compulsory suppression (protection, removal) of all sources behind. Deviation from this rule (work beginning from the middle or from the less contaminated side, leaving unprotected sources “in rear”) leads to two-fold and more increase in dose rates.
The decontamination planning is reduced to preparation of the plan of decontamination operation in compliance with two above stated principles.

8 DOSE RECONSTRUCTION

At the final stage of the exercise, the resulted reconstructed doses to witnesses and victims of the accident are demonstrated. In reality, such a demonstration can take place two-three weeks after the accident, after investigation of all circumstances of the accident and completion of all necessary procedures to simulate discharge and following assessment of the surface contamination.
Doses from fallout to the ground and from the jet are reconstructed. Simulation of the discharge jet consists of two tasks: simulation of jet as a physical object and simulation of this jet as a gamma radiation source. The discharge jet as a physical body is simulated as a translucent object of complex shape that varies in time. The jet, as radiation source, is simulated by superposition of point radiation sources. All point sources are inside the jet as a physical object.

9 CONCLUSION

The software presented here show the results in an interactive 3D scenario, with the ability to view every action of the workers linked to the timeline. The special software also generate a standard report with drawings, diagrams, tables, containing a dose assessment of workers, the best (in terms of the minimal dose) routes on the territory of the enterprise and demarcation of areas at the site.
Virtual reality technologies can be used effectively in the procedures for emergency planning and response, which are processed in emergency exercises and training. Use of special software with virtual reality technology is designed to help decision makers. Computer simulation tools allow one to present the results of calculations in a convenient and understandable form.

10 ACKNOWLEDGEMENTS

Authors would like to thank NRPA, Institute for Energy Technology, EMRDC SRC-FMBC, Center for radioactive waste management - Andreeva Bay Northwestern Center on Radioactive Waste Management "SevRAO" - a branch of the Federal State Unitary Enterprise “Enterprise for Radioactive Waste Management “RosRAO”, IBRAE RAN.

11 REFERENCES

[1] Szőke I, Johnsen T 2013 Human-centred radiological software techniques supporting improved nuclear safety, Nucl. Safety Simul. 4, 219–225.

[2] Szőke I et al 2014 Real-time 3D radiation risk assessment supporting simulation of work in nuclear environments J. Radiol. Prot. 34, 389–416.

[3] Chizhov K et al 2014 3D simulation as a tool for improving the safety culture during remediation work at Andreeva Bay J. Radiol. Prot. 34,755–773.

[4] Chizhov K, Simakov A, Kryuchkov V 2013 Method to solve analytical tasks for the purpose of radiation protection of workers when planning aftermath operations based on radiation field interpolation, Equipment and news in ration measurements 2, 70-77.

[5] Kryuchkov V et al 2009 Radrue method for reconstruction of external photon doses for Chernobyl liquidators in epidemiological studies, Health Phys. 2009 Oct, 97(4), 275-298.

[6] Kryuchkov V et al. The accident at ChNPP: Doses to participants of aftermath activities: emergency monitoring, retrospective assessment, 2011, Moscow, Edition of SRC-FMBC, 280 pp.

Emergency Preparedness – a continuously improving Process

Marcel Lips*

Goesgen Nuclear Power Plant, P O Box, CH-4658 Daeniken, Switzerland.

Abstract. Emergency preparedness has passed through a continuous improvement process in the Swiss nuclear industry. Improvements are and will be going on as a result of societal demands, technological developments, operational experience and regulatory issues. The first reactors went into commercial operation in the late 1960s. At that time emergency preparedness was mostly a technical issue. Containment isolation, flood and seismic resistance have been the most influential design parameters. External emergency preparedness was not considered to be a necessity and core damage frequencies around 1E-3 per year were seen as adequate. Increasing knowledge from scientific research programs, lessons learned from incidents and accidents, probabilistic safety assessments and deterministic design reviews led to several retrofits making the plants considerably more resistant with a reduced risk of core damage frequencies and large activity releases. Long term consideration of safety functions and additional mobile emergency equipment has further increased technical plant safety. In parallel, procedures against beyond design accidents were developed and human factors have been taken into account. The current status in Switzerland is that the power plants comply with the IAEA requirements for new build. Despite these improvements, external emergency concepts have been implemented and continuously adapted. Operators, regulators, a national emergency operation centre and governmental bodies play an important role in protecting the general public in case of an emergency, depending on the expected doses. Strategies have been developed for emergency response teams in order to adequately comply with international and national requirements for Radiation Protection. Additional proposals on long term post accidental actions are currently under development.

KEYWORDS: Emergency Preparedness; Nuclear Industry; Continuous Improvement.

1 INTRODUCTION

When the first Swiss reactors went into commercial operation about 45 years ago, the common global understanding of safety and protection of the general public was significantly different compared to the current view. Cars for example were not equipped with seatbelts or headrests and poor attention was paid to industrial safety. Technological development was driven by economic growth. The belief in a more comfortable lifestyle overruled safety and protection issues to some extent. On the downside of the development an increasing environmental pollution became more and more an issue. At this time the Swiss Government decided to rely on nuclear power to satisfy the growing electricity demand, which could no longer be covered by hydro power only. Fossil power plants were not seen as an adequate solution. Some potential risks were already associated with nuclear power at that time, so enhanced safety demands were implemented compared to other industrial sectors. Nevertheless, emergency preparedness was mostly a technical issue. External emergency preparedness was not considered a necessity. In subsequent years an ongoing program to enhance plant safety and to increase protection of the public was launched in line with societal demands, technological developments, operational experience, human factors and regulatory issues. Today in Switzerland five reactors are in operation – see Table 1.

Table 1: Overview of the nuclear power plants in Switzerland.

Plant	Reactor type	Net Output	Year of commissioning
Beznau 1	PWR	365 MW	1969
Beznau 2	PWR	365 MW	1971
Muehleberg	BWR	373 MW	1972
Goesgen	PWR	1010 MW	1979
Leibstadt	BWR	1220 MW	1984

* Presenting author, e-mail: mlips@kkg.ch

2 IN THE BEGINNING …

The decision to build nuclear power plants in Switzerland was taken in the mid-1960s. This was widely accepted by the Swiss population – especially in the light of a contribution to a less environmental polluting power generation compared to fossil alternatives. Knowing that the safety demands to these plants was significantly above industrial standards, the general public did not question plant safety and their own protection against ionizing radiation. Therefore, planning, construction and commissioning was done within a few years and in December 1969 the first rector went into commercial operation.

From a technical point of view the ability to isolate the containment, to withstand a flood occurring once in 1000 years and to resist the strongest historically known earthquake north of the Alps have been the most influential design parameters. Containment isolation was assured by duplex block valves on each penetration, controlled by different safety systems. Sites were either selected at an elevation not being affected by the design high flood or heaped up accordingly. The 1356 earthquake, which occurred in the northern part of Switzerland, is estimated to have had a 6.5 magnitude on the Richter scale resulting in ground accelerations between 0.1 and 0.2 g (depending on site location) has been chosen as the design reference. Spatial separation of safety systems were not implemented consequently. Resulting core damage frequencies around 1E-3 per year were seen as sufficiently low and beyond design accidents resulting in significant activity releases were not further considered. As a consequence neither external emergency preparedness nor radiological protection issues for the general public were addressed.

In the context of reactor accidents, ICRP 9 [1] – the relevant Recommendations in these days – specified a whole body dose of 1 Gy always calling for an action. Generally unplanned exposures and the risks resulting from them were seen as possibly less hazardous than the risks of remedial measures. In this sense emergency preparedness was set at the right level at that time.

3 MAIN STEPS IN IMPROVING SAFETY

3.1 Initial developments

In the 1970s when the first probabilistic risk assessments were carried out (as for example the WASH-1400 study [2]), core melt down accident frequencies of power reactors were calculated at higher levels than assumed so far. Small LOCA (Loss Of Coolant Accidents) did contribute to these results. It was also recognized that a core melt could penetrate the containment leading to large radioactive releases. These, coupled with unfavourable weather conditions could give raise to large consequences. In 1970 in Switzerland an airplane crashed because of a terroristic attack just two kilometres from an operating power reactor (twin unit), generally challenging the resistance of power reactors against external events.

To enhance pant safety and the resistance against emergencies, spatial separation of safety systems, airplane crash resistant containment buildings and an (airplane crash resistant) bunkered, independent emergency system became part of basic design recommendations for new plants in Switzerland. For the already existing plants bunkered emergency systems became a part of a retrofit program.

Four independent emergency systems (powered either by an external 400 kV supply, 220 kV supply or an emergency diesel), two independent bunkered emergency systems (400 kV, 220 kV, diesel) – see Figure 1, two independent river water intakes (one of them equipped with two diesel engines) and two independent ground water intakes were available for the Goesgen NPP at the day of commissioning in 1979. Such a design was above international standard. Nevertheless, the responsible authorities decided to implement the first external emergency concept at that time – mainly an alarming system for neighbour municipalities.

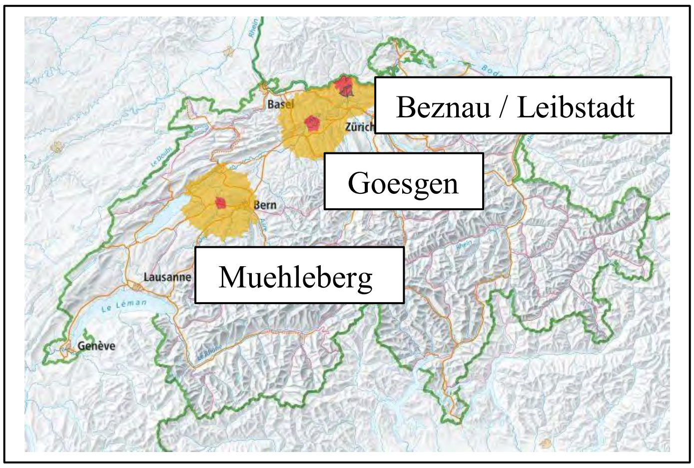
Human factors became an issue after the Three Mile Island accident leading to intense training programs for operators. Investigations about human performance and man machine interface issues are still ongoing and often used in experience utilization coming from internal or external events.

Figure 1: Bunkered emergency system building of the Goesgen NPP.

3.2 Tschernobyl

The Tschernobyl accident 30 years ago had a huge impact on the attitude and the general understanding in the matter of emergency preparedness. Despite the completely different plant design of light water reactors compared to the Tschernobyl type reactors, significant activity releases became part of enhanced preparations against nuclear emergencies. Two emergency planning zones were established around each unit – see Figure 2. An inner zone with a radius of about 5 km where immediate protective actions where seen necessary in case of a severe accident and an outer zone with a radius of about 20 km were protective actions were still seen as a necessity but less urgent. No protective actions were foreseen for areas outside the planning zones – except of harvest restrictions in wind direction. For the Beznau twin unit and the Leibstadt unit common emergency planning zones were established due to the proximity of the locations.

Figure 2: Emergency preparedness zones.



Protective actions were mainly based on radiological numbers like effective dose or thyroid dose and included in-house stay, sheltering and intake of iodine tablets [3]. Evacuation was not a primary issue due to the availability of shelter rooms for each citizen. Iodine tablets were distributed to each household inside the emergency planning zones and centrally preserved in the rest of the country. A dense network of remote controlled acoustic alarm sirens was installed inside the emergency planning zones and plant specific alerting criteria were included into the emergency procedures.

The emergency response tasks of affected bodies were specified in a concept [4] and a regime of emergency drills including all affected bodies – even across the boarders – was established.

All 5 units in Switzerland were recommended to install filtered containment venting systems to reduce the source term for iodine at least by a factor of 500 and for aerosols at least by a factor of 10'000 in case of a beyond design severe accident. In the mid-1990s the systems were fully operable and they allowed the Government to set the reference source term (basis for the emergency planning stage) to
3E18 Bq for noble gases, 1E14 Bq for iodine and 1E12 Bq for Aerosols, corresponding to a core melt accident with a filtered activity release. Model calculations showed that the defined emergency planning zones were adequate to such a source term – which was defined as an A2 source term.

3.3 Further developments

Lessons learned resulting from research programs as well as experience and knowledge exchange lead to considerations in long term behaviour of plant equipment after beyond design accidents. Reliability improvements for actuating drives, for different measuring instruments and for electric cables were identified and converted into a retrofit program lasting several years.

Together with considerations of beyond design accidents the whole complex of themes around hydrogen generation was intensely discussed. With the installation of hydrogen recombinators inside the containment of all units the risk of hydrogen explosions was significantly reduced in case of a severe accident.

At the same time the units started to develop procedures and strategies for the response to beyond design accidents which were not covered by the operational handbook so far. The already existing internal emergency preparedness organisations were adapted to the new tasks correspondingly and a regular collaboration with external organisations (fire and rescue, care giving, police, etc.) was institutionalized.

Fittings enabling core and spent fuel pool cooling with mobile equipment (tank fire-fighting vehicles, motor pumps, fire hoses) in case of a total station blackout were installed at different locations in the plants (see Figure 3).

Figure 3: Fire hose fitting for steam generator water supply.

3.4 PEGASOS

The PEGASOS study (Probabilistic Seismic Hazard Analysis for Swiss Nuclear Power Plant Sites)
was initiated in 1999 by the Swiss Federal Nuclear Safety Inspectorate in order to incorporate scientific advances into engineering practice – a process that is still ongoing after the formal completion of the original project [5].

The understanding obtained from the PEGASOS and follow-up refinement projects does not represent the indisputable truth, but serves to improve the quality of the hazard analyses step by step. New information and seismic data will continue to be available to the power plant operators and will contribute to the updating of the seismic hazard analysis in the future.

Switzerland has broken new ground in Europe with the PEGASOS project. The project and its subsequent refinement mean that the seismic hazard of the Swiss power plant sites has been determined according to the state-of-the-art knowledge and the newest data, with the ever-present uncertainties being considered conservatively.

3.5 New basic design recommendations

In 2005 a new legislation for the use of nuclear energy was implemented in Switzerland (Nuclear energy act [6] and nuclear energy ordinance [7]). Based on these, several subsequent ordinances were implemented as well. One of these governs the rules on provisional shutdown of nuclear power plants based on deterministic safety assessments [8]. For example the units had to provide evidence that they satisfy the dose limit for the general public (1 mSv) for any incident or combination of incidents with a probability of occurrence down to 1E-4 per year – irrespective of original plant designs.

As a consequence the design highflood switched from the 1E-3 per year event to the 1E-4 per year event, leading to additional safety precautions like a new safety wall towards the river at the Goesgen site. In combination with the determination of the new seismic hazard (see chapter 3.4) design ground accelerations increased from 0.1 – 0.2 g up to 0.6 – 0.8 g with huge impact. Programs to strengthen the seismic resistance of the plants had to be implemented in short time. Figure 4 shows an example strengthening seismic resistance.

Figure 4: Emergency diesel on new aseismic bumpers.

A big problem was the estimation of these 1E-4 per year events. Starting from an observation period of about one century with solid results the estimation contained huge uncertainties, which had to be incorporated in a conservative way.

3.6 Fukushima

After the accident at the Fukushima Daiichi nuclear power plant the Swiss Government initiated a general review on the preparedness against extreme emergency situations. In 2012 an action plan containing 56 individual actions was published [9]. Nearly all of them were related to the nuclear industry showing a big imbalance in the perception of the different risks.

One of the actions was the review of the reference source term defined in the 1990s. A systematic probabilistic approach using the newest computer technologies was chosen to show which source term would satisfy the large release frequency of less than 1E-5 per year according to the Basic Safety Principles INSAG-12 [10]. Taking into account the new seismic hazard (PEGASOS) the result of the review [11] showed that the current reference source term A2 does no longer satisfy the INSAG-12 recommendations. From a technical point of view an A3 source term (3E18 Bq for noble gases, 1E15
Bq for iodine and 1E14 Bq for Aerosols) corresponding to a core melt accident with an unfiltered activity release (containment filter bypass) would satisfy the IAEA recommendations.

Figure 5: Testing of mobile emergency power generators.

For purely political reasons it was decided to base future emergency response planning on an even worse so called A4 reference source term (1E19 Bq for noble gases, 1E16 Bq for iodine and 1E15 Bq for Aerosols) with the result that emergency planning actions became necessary outside the emergency planning zones. Precautionary evacuation of both emergency planning zones, subsequent evacuation of hot spots outside the planning zones, enhanced distribution radius (50 km) for iodine tablets and many other things became necessary due to that political decision.

Plant safety was reassessed in various ways, such as systematic design review, stress test, review of emergency procedures and emergency organisations, training, etc. Further actions strengthening seismic resistance were implemented and additional equipment enabling emergency response actions were installed (for example: two new emergency diesel generators in two new different buildings at the Goesgen site).

Figure 6: Air transportation of mobile equipment during an emergency drill exercise.

All units had to provide evidence that they are able to prevent a core damage accident after a total station black out without external support for 72 hours. The corresponding mobile pumps, power generators (see Figure 5), fire hoses and other equipment was procured and is now stored on site in seismic resistant buildings. Training in the use of this equipment was provided to almost all employees.

As a backup solution the Swiss utilities established a common external warehouse containing mobile emergency response equipment (water pumps, diesel generators, fire hoses, radiological protection material, etc.). The equipment is stored on racks and in containers ready to be air transported by helicopters or road transported by trucks. The Swiss Army is in charge of transportation in case of a severe emergency (see Figure 6).

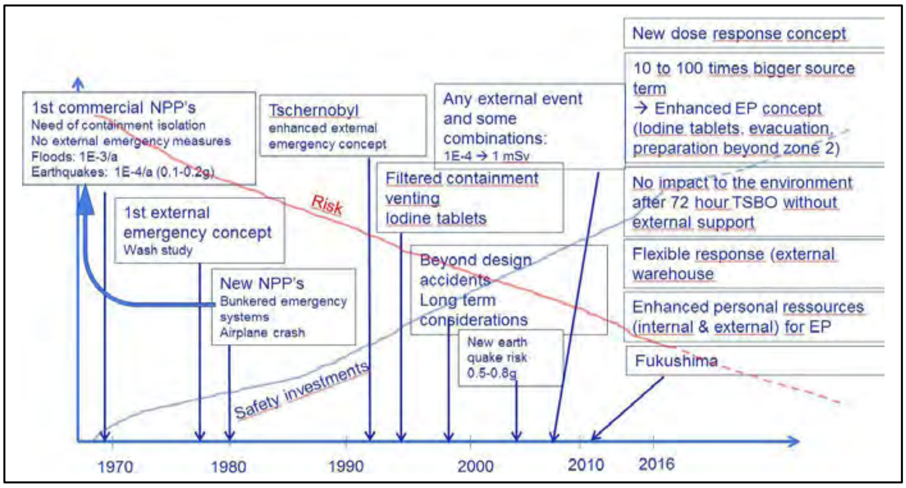
3.7 The future

Investments into plant safety will continue irrespective of the remaining plant lifetime. For the Goesgen plant for example a big retrofit program was recently initiated. The project has already started and will add additional onsite cooling water capacity and will extend the functionalities of the bunkered emergency system. The main goal of the project is to regain safety margins (which were mainly used up due to the new seismic hazard analysis) to be able to face any future requirements.

4 CONCLUSION

Emergency preparedness is not a pure technical and organisational issue. It is influenced by society demands as well as by incidents, operational experience, regulatory issues and state of the art developments. Nonetheless, emergency preparedness is a continuous process reducing the corresponding risk and, in case of the nuclear industry, the risk of potential radiation exposure (see Figure 7).

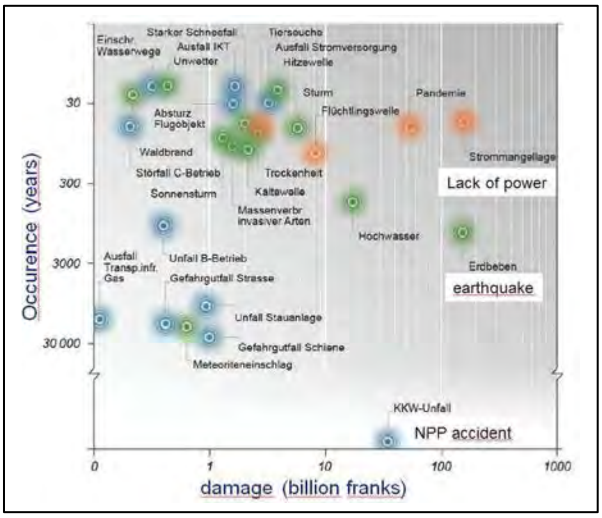
Figure 7: Schematic emergency preparedness development



In Switzerland nuclear power generation has been in use for almost 5 decades. During this time core damage frequencies have been lowered by at least two orders of magnitude and due to enhanced external emergency response planning, the risk of significant radiation exposures to the general public has been lowered even more.

A recent review of the different natural and civil risks to the Swiss population carried out by the Federal Office for Civil Protection [12] showed that a nuclear accident with an A4 source term would have a huge economic and societal impact. Due to the very low probability of occurrence this risk does not figure among the top 10 risks (see Figure 8). Earthquakes, severe pandemia or any long lasting lack of power are similar in consequences but much more frequent. Investments into civil protection should now move from the nuclear industry to these issues.

Figure 8: Natural and civil risk assessment



5 REFERENCES

[1] ICRP Publication 9, 1966. Recommendations of the International Commission. Pergamon Press, OXFORD.

[2] U.S. NUCLEAR REGULATORY COMMISSION, 1975. REACTOR SAFETY STUDY AN ASSESSMENT OF ACCIDENT RISKS IN U.S. OMMERCIAL NUCLEAR POWER PLANTS.

[3] EOR: operational organisation ordinance at elevated radiation levels (in German), 1991. From http://www.lexfind.ch/dtah/24024/2/732.32.de.pdf

[4] Federal Commission for NBC-Protection, 2006. Emergency preparedness concept in the surrounding of nuclear installations (in German). From https://www.baselland.ch/fileadmin/baselland/files/docs/jpd/amb/katast/kkw-zone2/2_konzept-notfall.pdf

[5] Swissnuclear, 2011. Evaluation of an intermediate seismic hazard for the existing Swiss nuclear power plants. From http://static.ensi.ch/1341581117/intermediate-hazard_may2011_27-06-2011_oeffentlicheversion.pdf

[6] KEG: Nuclear Energy Act, 2005. From https://www.admin.ch/opc/en/classified-compilation/20010233/200901010000/732.1.pdf

[7] KEV: Nuclear Energy Ordinance, 2005. From https://www.admin.ch/opc/en/classified-compilation/20042217/201205010000/732.11.pdf

[8] Ordinance on the methodology and boundary conditions for the criteria review on provisional shutdowns of nuclear power plants (in German), 2008. From https://www.admin.ch/opc/de/classified-compilation/20071450/200805010000/732.114.5.pdf

[9] Swiss Federal Department of Environment, Transport, Energy and Communications, DETEC (UVEK), Review of Emergency Preparedness Measures in Switzerland. 2012. From http://www.ensi.ch/en/wp-content/uploads/sites/5/2013/03/ida_nomex_e.pdf

[10] IAEA, 1999. INSAG-12 „Basic Safety Principles for Nuclear Power Plants“ 75-INSAG-3 Rev.1. Vienna.

[11] Federal Nuclear Safety Inspectorate, 2014. Review of Reference Scenarios for the Emergency Planning in the Surroundings of Nuclear Power Plants (in German). From http://www.ensi.ch/de/wp-content/uploads/sites/2/2014/06/ueberpruefung-der-referenzszenarien_ida_nomex_ensi-an-8293.pdf

[12] Federal Office for Civil Protection, 2016. Disaster and Emergency Situations Switzerland – Technical Risk Report 2015 (in German). From http://www.babs.admin.ch/

Medium and Long-Term Inhalation Dose Following a Major Radioactive Deposition Event

Mauro Magnoni

ARPA Piemonte – Dipartimento Tematico Radiazioni, Via Jervis, 30 – 10015 Ivrea (TO), Italy.

Abstract. The evaluation of the medium and long term inhalation dose following a major radioactive deposition event is one of the key issue in the management of the consequences of a nuclear accident. In fact, while the irradiation dose can be assessed quite easily by means of standard gamma rays measurements, the estimation of the inhalation dose is much more difficult and uncertain. In particular, the difficulty of a direct measurement of the airborne activity concentration of all the radionuclides species, especially true for the non-gamma emitters, limits in a substantial way the possibility of a simple and reliable evaluation of the activity introduced into the body. This lack of experimental data often leads to a different and indirect approach: the inhalation of radionuclides and the inhalation dose are therefore estimated from a much more easily available source, the deposition data. In fact, the airborne activity concentration can be estimated by means of the resuspention factor concept, defined as: $K=C/I$, where C is the airborne activity concentration (Bq/m^3) and I is the deposition value or inventory (Bq/m^2). Unfortunalely, one of the problems of this standard approach is that is a static one, not evolving with time, while the resuspention factor is no longer a constant parameter, but rather a quite strong function of the time, $K(t)$. Many empirical models have been proposed in order to tackle this problem; however, all of those models assume that, given an initial deposition value D_0, the resuspention into the atmosphere is the same for all the radionuclides. In this paper, a new model for the re-suspention factor is assumed and applied to a typical deposition event following a major nuclear accident: this model, in particular, allows a different resuspention behaviour for each radionuclide, being related to the different migration rates in soil: the medium and long-term inhalation dose is then calculated and compared with the results obtained using the standard available models.

KEYWORDS: re-ruspention factor; medium-long term inhalation effective dose.

1 INTRODUCTION

The medium and long-term inhalation effective dose is essentially driven by re-suspention mechanisms that re-inject into the atmpsphere the previously desposited radioactivity. The atmospheric re-suspention of radionuclides is usually modelled by means of the resuspention factor concept K, defined as the ratio of the volumetric air concentration C_a (Bq/m^3) to the initial soil deposition I_0 (Bq/m^2). This parameter is generally considered a time depending function. Therefore we can write:

$$K(t) = \frac{C_a(t)}{I_0} \quad (m^{-1}) \tag{1}$$

K values span several order of magnitude: typical values in the range 10^{-5} – 10^{-10} m^{-1} are reported in literature for different environmental conditions and the time elapsed since the deposition event (Sehmel, 1980, IAEA Tecnical Report Series n°472, 2010).

The resuspention factor concept is very useful in radioprotection in order to estimate the inhalation of radionuclides resuspended from contaminated surfaces when direct atmospheric measurements are lacking or difficult to perform. However, the choice of the proper values of K is usually not a simple task, being quite site specific and related to the meteorological, geomorphologic and environmental characteristics of the area to be studied. Moreover, several investigations showed clearly that the values of K are a decreasing function of time. The current available models for the resuspention factor are based on empirical formulas whose parameters are highly site dependent and cannot be easily related to some physical quantity.

m.magnoni@arpa.piemonte.it

Several models for the resuspention factor were proposed and tested in the last years using the Chernobyl accident data; in fact the widespread contamination of large areas of Europe allowed, especially in the first years after the accident, the measurements of airborne radionuclides concentrations (Plutonium and Cesium-137, in particular), in very different environmental conditions. These researches lead to the formulation and testing of some empirical models, commonly used for the prediction of time evolution of the volumetric air concentration C_a(t) in the years following a fallout event. A detailed discussion of these models can be found in the BIOMOVS II final report (Garger, E.K, 1999).

The more recent (2010) IAEA publication n°472 suggested the use of three models, corresponding to three different environmental conditions, such as: a) rural conditions; b) urban conditions; c) arid and desert condition.

For rural conditions the Garland model, who obtained the best scores in the IAEA BIOMOVS II intercomparison, is proposed. It consist of a simple time decreasing relationship defined as follows:

$$K(t) = \frac{K(0)}{t} \tag{2}$$

where *K(0)*=$1.2 \cdot 10^{-6}$ days$\cdot$m^{-1}, the time *t* is expressed in days and the formula is valid for *t* > 1 day.
For urban environment the Linsley model, consisting in an exponential decreasing function and a constant asymptotic value, is suggested:

$$K(t) = K(0) \cdot e^{-0.0\ \cdot t\ 1} + 1\ ^{-9}0 \tag{3}$$

where *K(0)*=10^{-6} m^{-1} and the time *t* is expressed in days.
For arid and desert conditions, where resuspention is generally more effective, the Anspaugh model is recommended:

$$K(t) = K(0) \cdot e^{-0.1\ \sqrt{t}} + 1\ ^{-9}0 \tag{4}$$

with *K(0)*=10^{-6} m^{-1}.
More complicated models, based on a modification and/or a combinations of the above mathematical expressions where also proposed and tested by other authors, in order to find a better agreement with local experimental data. However no significant improvements in the capability predictions was achieved. In general, the intercomparison exercises performed in the past years showed that the major source of uncertainty for all models is the choice of the initial conditions, i.e. the *K(0)* values, rather than the differences in the mathematical form of the expressions. The experimental available data for *K(0)* generally range between 10^{-5} and 10^{-6} m^{-1}, in the first days or months after the fallout event and are highly site specific.

All these empirical models provide quite good results for some specific situations but the lack of knowledge of the underlying physical processes limits the possibility of generalizing in an easy way the results obtained in a given site to others locations with different environmental characteristics. Moreover all these models predict an unique behaviour for all the radionuclides species, neglecting the big differences existing among different radioactive elements concerning their fate in the environment. For all those reasons, at the moment, none of these current models has broad prediction capabilities (both in time and space). These facts limit in a substantial way the capabilities of these models to be used as a reliable tool for the evaluation for the inhalation committed doses.
Therefore, in these work, we decided to use a little more sophisticated model, recently developed, that takes into account for the progressive downward migration of the radionuclides in soils: this model allows the calculation of a specific re-suspention rate for each radionuclide and therefore permits a

more realistic and reliable evaluation of the inhalation dose: the dose values calculated in this way are finally compared with those obtained using the standard models.

2 MATERIAL AND METHODS

2.1 The re-suspention scenario and the inhalation dosimetric model

The long-term inhalation committed dose is estimated from an hypothetical "Chernobyl-like" scenario, where a massive fallout occurred in a very large area. The simulation takes into account only for the doses delivered by the long-lived radioisotopes (i.e with $t_{1/2}$ > 1 year), thus neglecting all the short-lived radionuclides (iodines, lanthanides, rare earths, etc.). For the sake of simplicity we will consider only the following long-lived radionulclides, usually released into the environment in large quantities during major nuclear accidents, considered as representative for the medium-long term exposure:
^{137}Cs, ^{134}Cs, ^{90}Sr and ^{239}Pu.

All these radionuclides are supposed deposited on the ground at the time of the accident, t=0.
From that time, the inhalation exposure due to re-suspention starts. A breath rate of 1 m^3/hour is assumed. If the concentration in atmosphere of the generic radionuclide j is given by $C_{aj}(t)$ (Bq/m^3) and the corresponding inhalation dose coefficient is e_j, (Sv/Bq), the global committed effective dose delivered in a time τ, is calculated as follows:

$$E_\tau = \beta \cdot \sum_j e_j \cdot \int_0^\tau C_{aj}(t)dt \qquad (5)$$

It is also possible to evaluate a global effective dose rate:

$$\dot{E} = \beta \cdot \sum_j e_j \cdot C_{aj}(t) \qquad (6)$$

Both the equations (2) and (3) need the knowledge of the function $C_{aj}(t)$ for each radionuclide j resuspended from the soil. All these functions are in general unknown and therefore have to be calculated by means of a model. In this case we will assume that all the $C_{aj}(t)$ could be expressed by using the re-suspention factors $K_j(t)$, that are functions of the time and are in general different for each radionuclide j:

$$C_{aj}(t) = I_{0j} \cdot K_j(t) \qquad (7)$$

where I_{0j} is the deposition value of the radionuclide j at the time of the accident $t=0$.
The inhalation doses can thus be easily calculated, provided we know the initial deposition values and the $K_j(t)$ functions. As the deposition data are generally available, the problem to be solved is the proper choice of the form of the $K_j(t)$ functions. We will discuss this point in more detail in the next section.

2.2 The resuspention model

The re-suspention factor is modelled following the new theoretical approach described in a previous paper (Magnoni, 2012). The re-suspention factor can be expressed by the equation:

$$K(t) = \frac{\left[1 + \frac{v \cdot \Delta z}{2 \cdot D}\right] \cdot \Delta z \cdot K(0)}{2 \cdot \sqrt{\pi \cdot D \cdot t}} \cdot e^{-\frac{v^2 \cdot t}{4D}} \qquad (8)$$

where v and D are, respectively, the advection velocity and the coefficient of dispersion, i.e. the parameters characterising the downward migration of the radionuclides in soils described by the convention-diffusion equation equation (Konshin, O.V., 1992; Kirchner G., 1998; Bossew P. and Kirchner G., 2003). This model fits the available resuspention data much better than the most used models, especially in the medium-long term. It is therefore more appropriate for the evaluation of the medium-long term inhalation doses.

3 RESULTS AND DISCUSSION

The calculation of the inhalation effective dose rate and committed dose was performed, respectively, by means of equations (2) and (3), while the activity concentrations of the resuspended radionuclides $C_{aj}(t)$ were estimated using the formulas given by the equations (4) and (5).
In the following table the values of the migration parameters v and D used in the simulation, for all the considered radionuclides, are summarized. These values were taken interpolating the cesium soil profiles (gathered in Piemonte, North-West Italy), assuming that the downward migration could be described by the convection-diffusion equation. Very similar values were found by Kirchner (2003), in Austrian soils. The values for ^{90}Sr were properly scaled taking into account for the greater mobility of this radioisotope. For the ^{239}Pu values close to those of Cesium have been assumed, in order to follow a conservative approach.

Table 1: Values assumed for the parameters aof the convection- diffusion migration model

	^{137}Cs	^{134}Cs	^{90}Sr	^{239}Pu
v (cm/year)	0.112	0.112	0.248	0.110
D (cm^2/year)	0.037	0.037	0.075	0.035

In figures 1 and 2, the values of the activity concentration and the corresponding inhalation dose of all the considered radioisotopes are shown for the next 30 years after the deposition event. It can be seen that, while for the activity concentration the highest value is that of ^{137}Cs, the inhalation doses are largely dominated by ^{239}Pu.

Table 2: Values assumed for the initial deposition value I_0 for each radioisotope

	^{137}Cs	^{134}Cs	^{90}Sr	^{239}Pu
I_0 (kBq/m^2)	2000	1000	150	6

Figure 1: Activity concentration levels predicted by the model: equations (4) and (5).

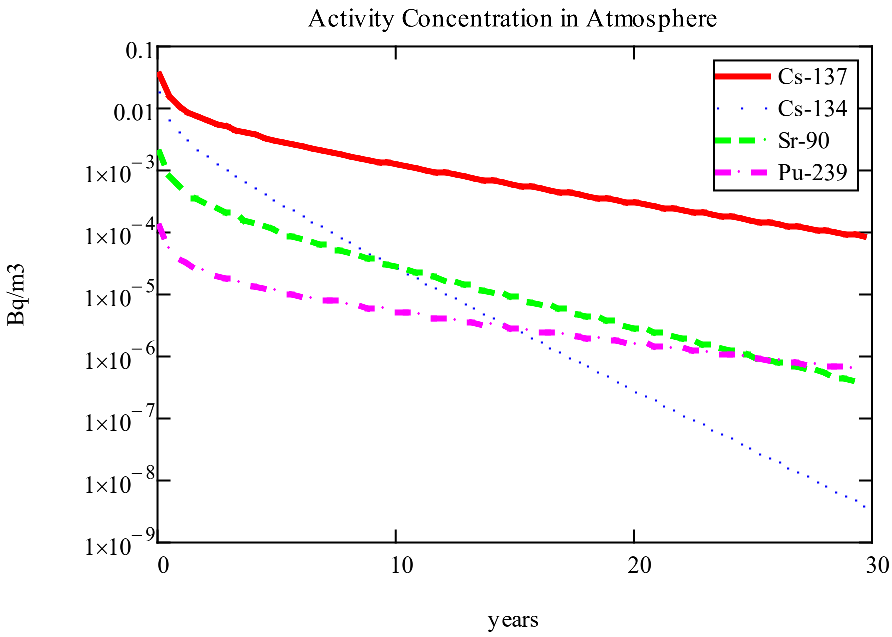
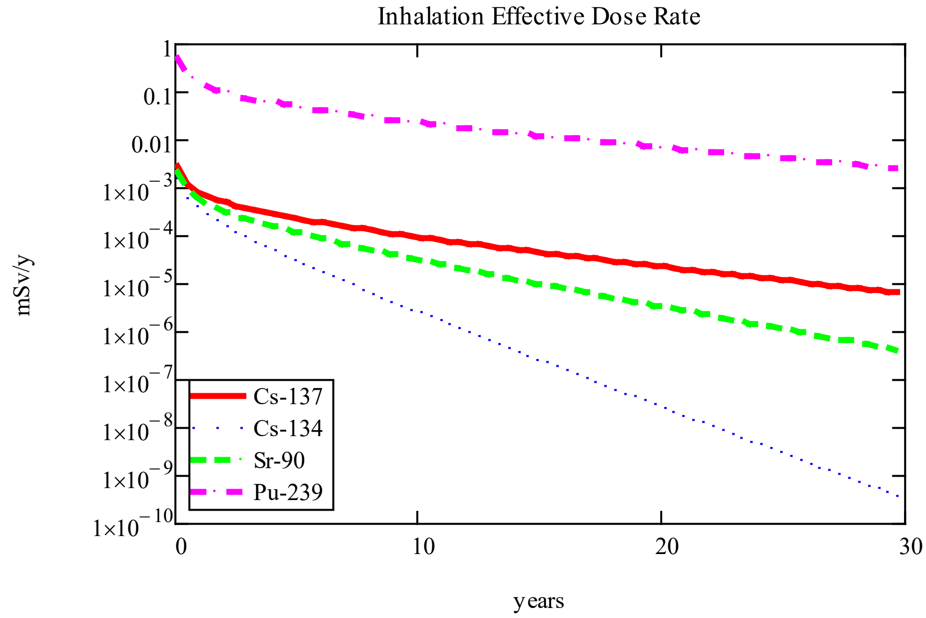
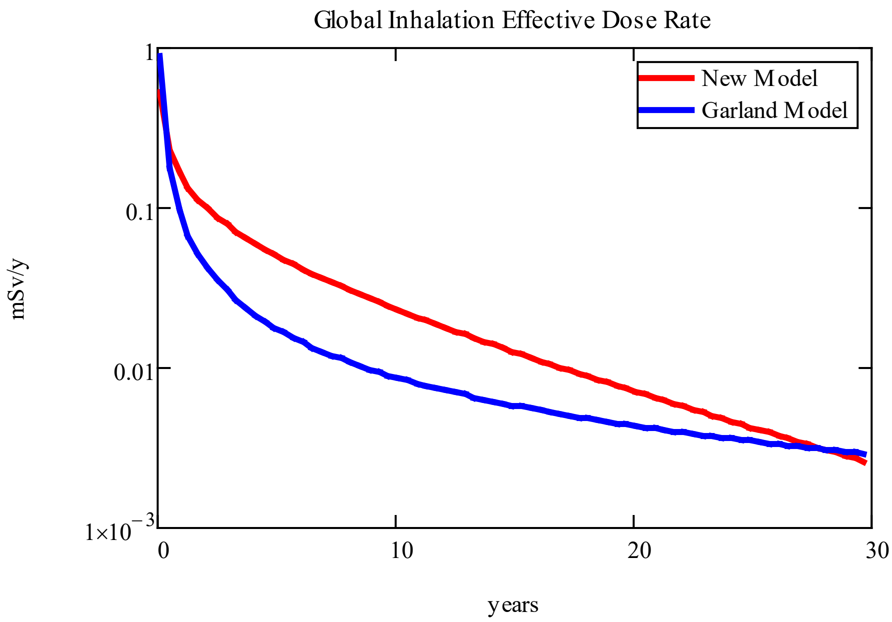


Figure 2: Inhalation doses for all the considered radioisotopes: the highest contribution is due to ^{239}Pu: a value close to 1 mSv/year is reached in the first years after the deposition event.



In figure 3 a comparison between the Garland model, given by a simple iperbolic function, $K(t)=K_0/t$, and the proposed model (5), is shown.

Figure 3: Comparison between the Garland and the new model: the Garland model underestimates in a substantial way the doses in the time range 0 – 30 years after the deposition event.



The Garland model seems to underestimate in a substantial way the doses in the time range 0 – 30 years after the deposition event. As a consequence, also the inhalation committed effective dose estimated using the Garland model (integration time =50) resulted in 0.55 mSv, a quite lower value compared to that evaluated with the new model, about 1 mSv.

4 CONCLUSIONS

The highest activity concentration levels in atmosphere due to the resuspention of the radionuclides deposited after a major nuclear accident are those of ^{137}Cs, while the dose contribution is largely dominated by ^{239}Pu. The dose evaluation, based on a new re-suspention factor theoretical model, considering a convection-diffusion equation mechanism for the migration of radionuclides in soils, gives higher and more reliable dose values than the standard empirical model.

5 REFERENCES

[1] Bossew P. and Kirchner G., “Modeling the vertical distribution of radionuclides in soils. Part 1: The convection-dispersion equation revisited”, *Journal of Envirnonmental Radioactivity,* (2003).

[2] Capra D., Facchini U., Gianelle V., Ravasini G., Ravera O., Pizzala A. and Bacci P., “The Chernobyl accident: soil deposition of radionuclides in Po Plain and Alpine Valleys”, *Acqua&Aria,* Vol. 3, 305-319, (1988).

[3] Facchinelli A., Magnoni M., “Radioattività ambientale e radiocontaminazione dei suoli piemontesi”, *Collana Ambiente n°14* – Regione Piemonte, (1998), in Italian.

[4] Facchinelli A., Magnoni M., Gallini L. and Bonifacio E., “Cs-137 contamination from Chernobyl of soils in Piemonte: spatial distribution and deposition model”, *Water Air, and Soil Pollution,* Vol. 134, 341-352, (2002).

[5] Garger E.K. et al., ”Test of existing mathematical models for atmospheric resuspension of radionuclides”, *Journal of Envirnonmental Radioactivity,* Vol. 42, 157-175, (1999).

[6] Hatano Y. and Hatano N., “Formula for the resuspention factor and estimation of the date of surface contamination”, *Atmospheric Environment,* Vol. 37, 3475-3480, (2003).

[7] IAEA, “Handbook of Parameter Values for the Prediction of Radionuclide Transfer in Terrestrial and Freshwater Environments”, *Tecnical Report Series n°472* (2010).

[8] Konshin O.V., “Applicability of the convection-diffusion mechanism for modelling migration of ^{137}Cs and ^{90}Sr in the soil”, *Health Physics,* Vol. 63, 291-300, (1992).

[9] Konshin O.V., “Mathematical model of ^{137}Cs migration in soil: analysis of observations following the Chernobyl accident”, *Health Physics,* Vol. 63, 301-306, (1992).

[10] Kirchner G., “Applicability of compartment models for simulating the transport in soils”, *Journal of Envirnonmental Radioactivity,* Vol.38, 339-352, (1998).

[11] Magnoni M., “Stima dell’evoluzione temporale della risospensione per mezzo di misure di fallout”, *Proceedings of AIRP National Congress,* Vasto, 1-3 october 2007 (in Italian).

[12] Magnoni M., Losana M.C., Bertino S., Bellotto B., Tripodi R., Ghione M., “La migrazione del Cs-137 nei suoli del Piemonte: dati sperimentali ed effetto della distribuzione verticale sui livelli di dose da irraggiamento ”, *Proceedings of the XXXIII Congresso Nazionale AIRP di Radioprotezione,* Torino, 20-23 september 2006, (in Italian).

[13] Magnoni M., “A theoretical approach to the re-suspention factor”, European Physical Journal, Web of Conferences, 05008 (2012).

[14] Sehmel G.A., “Particle resuspension: a review”, *Env. Int. 4,* 107-127, (1980).

Development and Current Status of a Carborne gamma-ray Survey System, KURAMA-II

Minoru Tanigaki[a*], Ryo Okumura[a], Koichi Takamiya[a], Nobuhiro Sato[a], Yasuhiro Kobayashi[a], Hirofumi Yoshino[a], Hisao Yoshinaga[a], Akihiro Uehara[a]

[a]Research Reactor Institute, Kyoto University, Kumatori, Osaka, 590-0494 Japan.

Abstract. A carborne gamma-ray survey system, named as KURAMA (Kyoto University RAdiation MApping system), was developed as a response to the nuclear accident at TEPCO Fukushima Daiichi Nuclear Power Plant in 2011. Now the system has evolved into KURAMA-II, characterized by its compactness, ruggedness, autonomous operation. KURAMA-II serves a continuous monitoring by city buses in residential areas in Fukushima prefecture and periodic carborne gamma-ray surveys in East Japan. The outline and current status of KURAMA-II including some examples of extending applications are introduced.

KEYWORDS: radiometry; mapping; γ-ray; carborne survey; air dose rate; Fukushima Daiichi nuclear power plant.

1 INTRODUCTION

The magnitude-9 earthquake in eastern Japan and the following massive tsunami caused a serious nuclear disaster of Fukushima Daiichi nuclear power plant. Serious contamination was caused by radioactive isotopes in Fukushima and surrounding prefectures. KURAMA [1] was developed to overcome the difficulties in radiation surveys and to establish air dose-rate maps during and after the present incident. KURAMA was designed based on consumer products, and enabled operations of a large number of in-vehicle units for large-scale surveys owing to its high flexibility in the configuration of data-processing hubs or monitoring cars. KURAMA has been successfully applied to various activities in the radiation measurements and the compilation of radiation maps in Fukushima and surrounding areas. As the situation becomes stabilized, the main interest in measurements moves to the tracking of the radioactive materials that have already been released into the environment surrounding the residential areas. Since most of carborne survey systems including KURAMA require a trained operator and a driver in each monitoring vehicle, surveillances in residential areas for several tens of years are almost impossible. Such monitoring can be realized efficiently if vehicles that periodically move around the residential areas, such as city buses, delivery vans or motorcycles for mail delivery, have compact and full-automated KURAMAs onboard. KURAMA-II [2] is designed for such a purpose, characterized by its compactness, autonomous operation, and additional functions such as the measurement of pulse height spectrum. In this paper, a system outline and the development of KURAMA-II as well as the results of continuous monitoring using KURAMA-II will be introduced.

* Presenting author, e-mail: tanigaki@rri.kyoto-u.ac.jp

2 KURAMA-II SYSTEM

2.1 System Outline of KURAMA-II

The system outline of KURAMA-II is shown in Fig. 1. As was the case in KURAMA, KURAMA-II is designed to realize large-scale surveys rather than pursuing the sensitivity or the precision of each unit. KURAMA-II basically stands on the architecture of KURAMA [1], but the in-vehicle part has been totally re-designed. In original KURAMA, a notebook PC was used in an in-vehicle unit, but KURAMA-II is based on the CompactRIO series of National Instruments [3][4] to obtain better toughness, stability and compactness.

The radiation detection part has been replaced from the conventional NaI survey meter to the C12137 detector series by Hamamatsu Photonics [5]. This CsI detector series is characterized by its compactness, high efficiency, direct ADC output and USB bus power operation. The direct ADC output enables one to obtain γ-ray pulse height spectra during operation. All of the components for the in-vehicle part are placed in a small tool box for a better handling (Fig. 2). KURAMA-II accepts multiple detector connections via USB port. At least three connections of C12137 units can be simultaneously used in the case of conventional CompactRIO units such as cRIO-9076. The air dose rate and pulse-height spectrum for each measurement point are collected by a simple file- transfer protocol based on RESTful in the current KURAMA-II scheme. The gateway server receives and combines these small files to the data files of air dose rate and pulse height spectrum, respectively. These data files are shared by remote servers using Dropbox, as was done in original KURAMA.

Figure 1: System outline of KURAMA II.

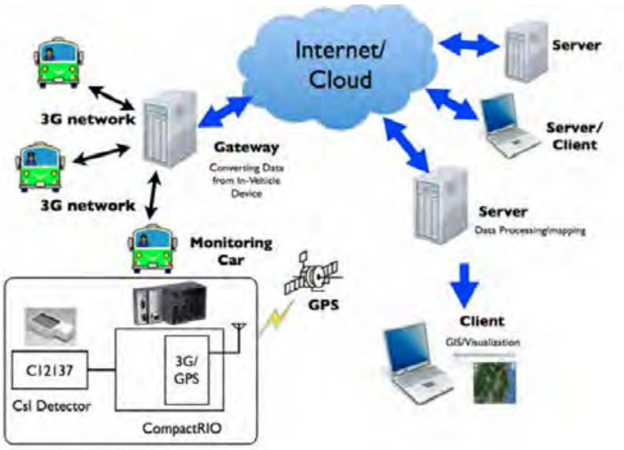


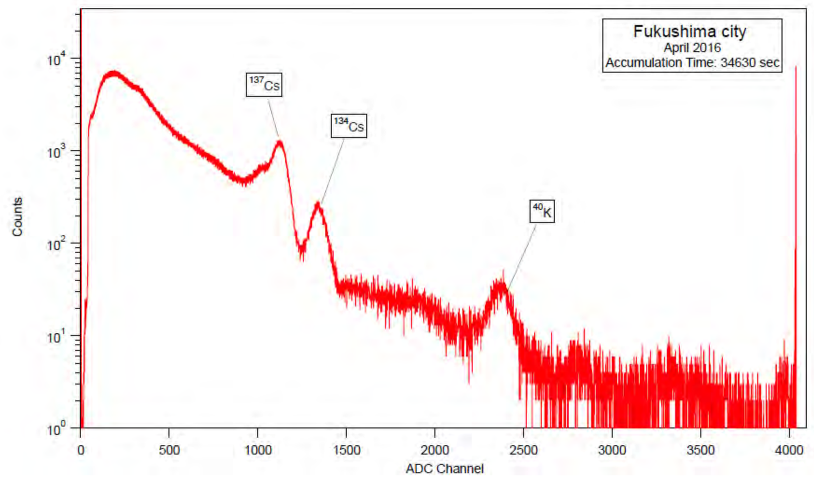
Figure 2: In-vehicle unit of KURAMA-II. All components are placed in a small tool box for a better handling. The in-vehicle unit of KURAMA-II is usually placed in the backside space of the rear seat row, corresponding to the center site of the road.

2.2 Evaluation of Air Dose Rate

In KURAMA-II, the air dose rate is obtained from the pulse-height spectrum by using spectrum-dose conversion operators, the so-called G(E) function method [6][7]. This time, G(E) functions for $H^{*}(10)$ were determined by the Japan Atomic Energy Agency (JAEA) group for C12137-00 and C12137-01. Details concerning the determination of these G(E) functions and the characteristics of KURAMA-II on radiation detection are available in ref. [8].

In G(E) function method, the energy calibration of detector is crucial. In KURAMA-II, the 796 keV peak of ^{134}Cs, which is still a well-isolated peak in pulse height spectra taken in Fukushima, is always monitored during its operation (Fig. 3). Up to now, the peak drift is at most 3% throughout the operation for one year. This corresponds to at most 5% of the drift in the air dose rate, one-third of the tolerance for typical portable survey meters used for the air dose rate measurements in Fukushima. The monitoring peak may be changed to the 662 keV of ^{137}Cs or 1460 keV of ^{40}K in future (in ten years or so) due to the short half-life of ^{134}Cs.

Figure 3: A typical pulse height spectrum in Fukushima city accumulated by KURAMA-II in April 2016. The 796 keV peak of ^{134}Cs is still a well isolated in a gamma-ray pulse height spectrum by CsI detector in Fukushima, thus suitable for the usage as the monitoring reference for gain shift.



3 APPLICATIONS OF KURAMA-II

3.1 Continuous Monitoring by City Buses

One of the major application of KURAMA-II is the continuous monitoring in residential areas by city buses. City buses are suitable for the continuous monitoring in residential areas because of their fixed routes in the center of residential areas, and their routine operations.
As of April 2016, fifty KURAMA-II units are continuously operated throughout Fukushima prefecture as an official project organized by Fukushima prefectural government under the collaboration of Kyoto University and JAEA. Each in-vehicle unit is fixed to a certain city bus and completes bus routes in its operation area over three to five days based on a transportation plan determined by its respective bus operator. The air dose rate at 1 m above on the road is determined by multiplying the shielding factor of bus body determined by the comparison with the results on the same route of periodical carborne surveys by Japanese government. The results from this measurement clarify the changes of radiations in residential areas (Fig. 4, 5), or the sudden increase of air dose rate caused by reasons other than the nuclear accident (Fig. 6).

Figure 4: Deviation of the air dose rate in 2013 from the reference period (from Dec. 20 to 31, 2012) in Fukushima monitored by KURAMA-II units on city buses. Heavy snow greatly reduced the air dose rate by shielding the radiation in January, February, and December.

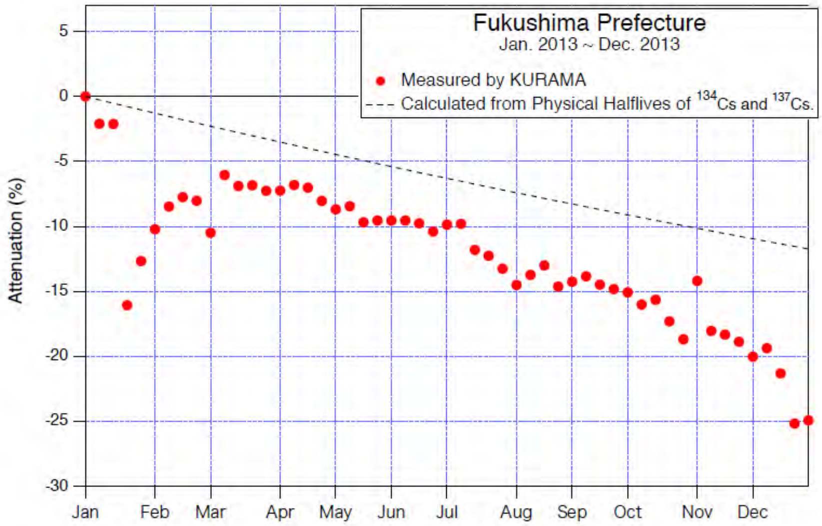


Figure 5: A typical example of the decontamination effect observed through monitoring by KURAMA-II on a local bus. The decontamination was performed at the grid pointed out in the figure in May, 2013 and a drastic reduction of air dose rate along with the decontamination was observed.

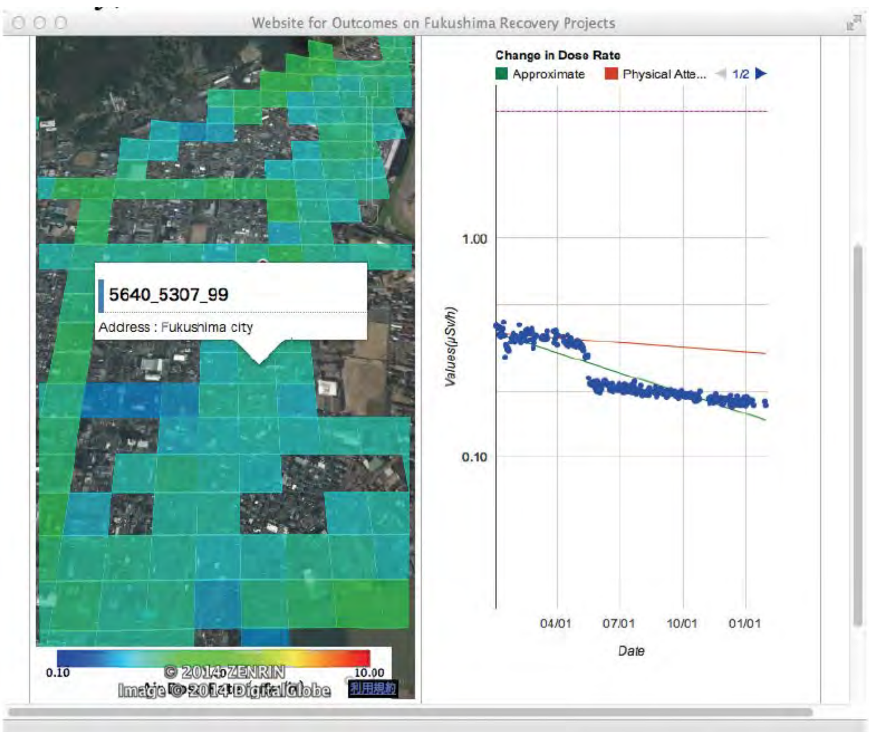


Figure 6: The pulse height spectra obtained by KURAMA-II on the bus during a patient after the treatment using radiopharmaceuticals on board (blue) and just before the patient on board (red). ^{99m}Tc was clearly detected in the pulse height spectrum.

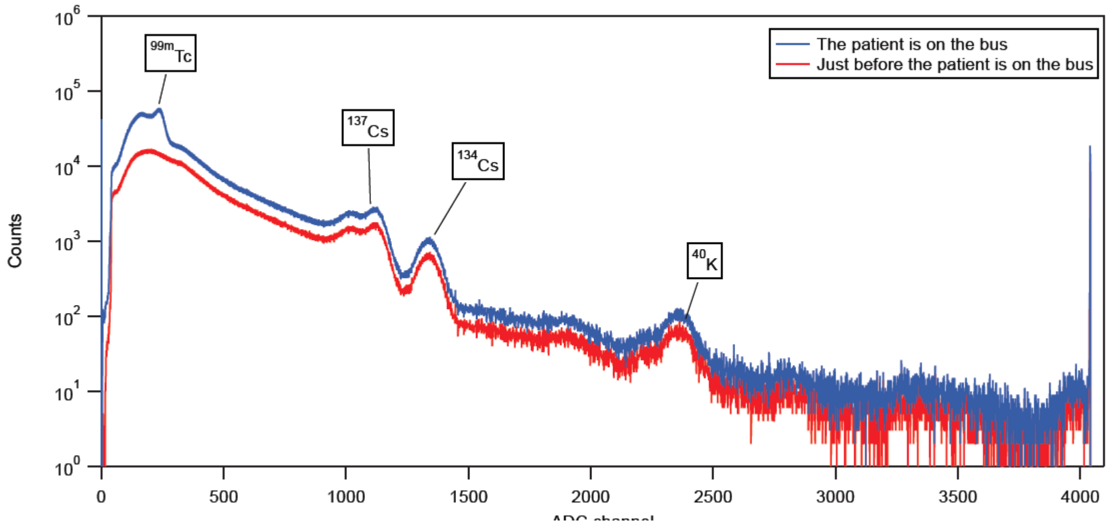
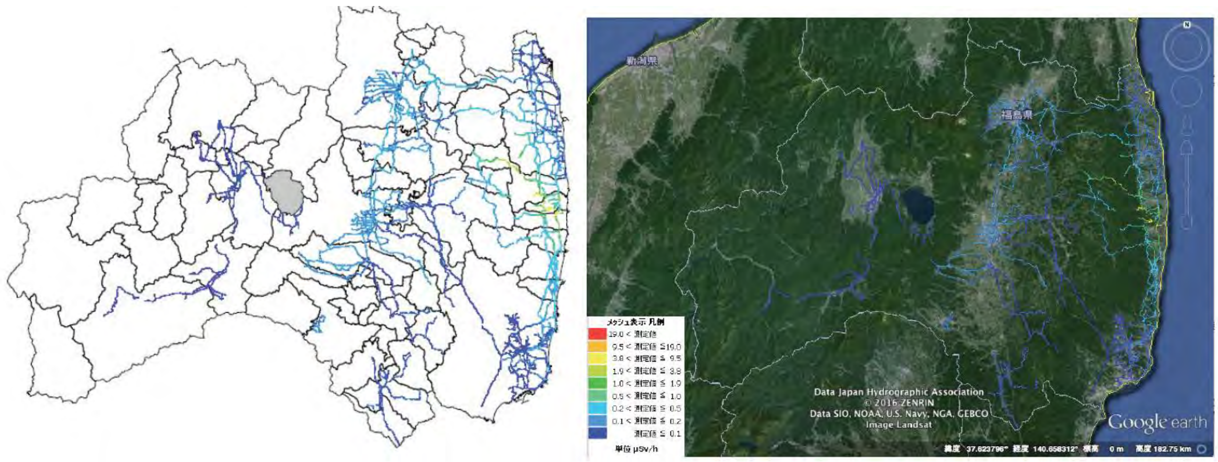


Figure 7: A typical result of continuous monitoring by city buses from Fukushima prefectural government as of 24~30 Jan. 2016. More than fifty KURAMA-II units are under operation. Summarized maps (left) and their KML data (right) for each week are released from the web site [9].



Real time data is released to the public through the display system at the public space of a building in Fukushima city, and the summarized results are available on a weekly basis on the web site of Fukushima prefectural government (Fig. 7) [9].

3.2 Periodical Surveys by Japanese Government

The Ministry of Education, Culture, Sports, Science and Technology of Japan (MEXT) had introduced KURAMA-II for the periodic carborne surveys in eastern Japan since March 2012 [10]. This project has been taken over to Nuclear Regulation Authority (NSR) because of the reorganization of atomic energy administration in Japan. In this project, around one hundred KURAMA-II are deployed to the local municipalities in eastern Japan twice a year. The staff members in each municipality just attach KURAMA-II into the backside of their conventional sedan cars, and drive around their own municipalities. The results of these periodic surveys are collected by MEXT or NSR in real time, then the temporary results are summarized and sent back to municipalities. The summarized data is released through the websites [11] [12]. Various analyses are performed based on the results of these periodical surveys including the evaluation of ecological half-lives of radioactive cesium or the long- term predictions of ambient dose equivalent rates in Fukushima [13].

3.3 Prompt Estimation of Cs Density in Soil

The development of a quick, easy, and non-destructive method for the estimation of soil contamination is crucial for the recovery of vast farmland in Fukushima. No explicit cautions are paid for the contribution from the contamination of surrounding structures such as houses or trees in conventional methods [14], and this could cause a large error in the estimation. We have proposed a simple method for the proper estimation of soil contamination. In our method, a pair of detectors, one a directional detector towards the ground with solid angle Ω and the other is uncollimated one, are used (Fig. 8). The radiation detected by the collimated detector Φ_C should be from the localized radioactive substances right below the detector Φ_L and the additional component of surrounding radioactivity Φ_F with the fraction of Ω . The radiation detected by the uncollimated detector Φ_U is approximately equal to Φ_F unless the localized radioactive substances dominates the air dose rate for that point. Therefore, Φ_L is given as:

$$\begin{aligned}\phi_L &= \phi_C - \Omega \times \phi_F \\ &\sim \phi_C - \Omega \times \phi_L. \end{aligned} \tag{1}$$

We have performed preliminary studies of this method in unplowed orchards in Fukushima with KURAMA-II with two C12137-01 detectors [15]. As shown in Fig. 9, a fairly good correlation was obtained. A series of studies on the correlations between Φ_L and the cesium density in solid for the various soil types are on the way.

3.4 Survey meters for primary responses in nuclear accident

One of the regrets of the nuclear accident in Fukushima is the lack of data for the short-lived nuclei in the environment. Generally, such systematic monitoring for such short-lived nuclei requires a large number of spectrometers and trained measurers, but neither of them were available at that time in Fukushima. Such information would have been automatically collected regardless of the situation if survey meters used under the emergency situations had been the capability to obtain pulse height spectrum along with its location data without requiring the special knowledge. In this point of view, we have developed a survey meter based on KURAMA-II scheme, named as KURAMA-mini [16]. We have just started a feasibility study of monitoring under the emergency situation.

Figure 8: Principle of estimation of soil contamination. The collimated detector gives the sum of the radiation from the radioactive substances on the soil Φ_L and that from the surroundings Φ_F. The contribution from the soil contamination is separated by using Φ_F determined by the uncollimated detector.

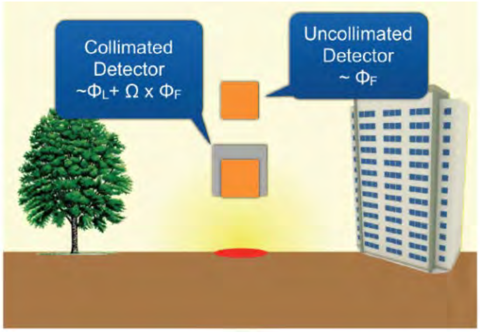
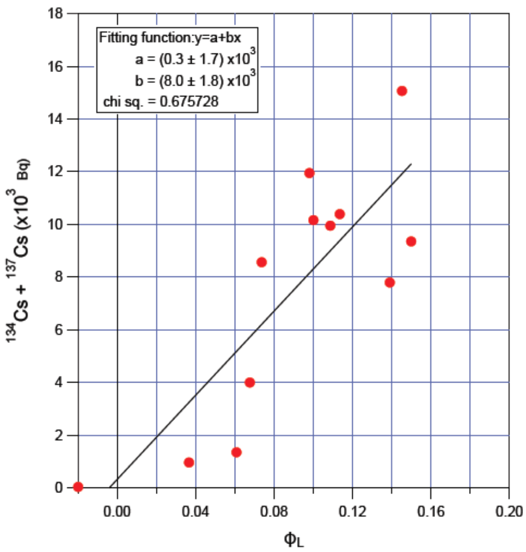


Figure 9: The correlation between Φ_L determined by KURAMA-II and the cesium contamination of soil samples determined by Ge detector for the unplowed land in Fukushima [16].



4 ACKNOWLEDGEMENTS

The authors are grateful to Dr. Mizuno, Mr. Kimura, Mr. Ito and Mr. Kojima of the Fukushima prefectural government for a continuous support to field tests of KURAMA-II and the establishment of a monitoring scheme based on KURAMA-II on city buses. The authors are indebted to Dr. Saito, Mr. Yoshida and Dr. Takemiya at JAEA for discussions concerning the operation of KURAMA, and to Dr. Tsuda at JAEA for evaluating the *G*(E) functions of C12137 series. Technical support regarding LabVIEW and CompactRIO was served by National Instruments Japan Corporation under a support program named as “Monotsukuri Fukkou Shien Project” by National Instruments Japan Corporation, aiming to help in the recovery from the Great East Japan Earthquake. The authors are grateful to Mr. Matsuura and Mr. Yasuoka at Matsuura Denkosha Corporation for discussions on the KURAMA-II source codes. Mr. Muto and Mr. Inomata at Fukushima Transportation Inc., Mr. Suzuki at Shin Joban Kotsu co., ltd., and Mr. Sugihara and Mr. Ishikawa at Aizu Bus co. ltd. for their cooperations in the field test of KURAMA-II on city buses. Finally, the authors would like to express their gratitude to Mr. and Mrs. Takahashi and the staff members at “Matsushimaya Ryokan”, an inn at Iizaka hot spring in Fukushima city, for their heart-warming hospitality and the offer of a foothold for our activities in Fukushima, regardless of the severe circumstances due to the earthquake and the following nuclear accident.

5 REFERENCES

[1] M. Tanigaki, R. Okumura, K. Takamiya, et al., Nuclear Instruments and Methods in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment **726** (2013) 162-168 .

[2] M. Tanigaki, R. Okumura, K. Takamiya, et al., Nuclear Instruments and Methods in Physics Research Section A: Accelerators, Spectrometers, Detectors and Associated Equipment **781** (2015) 57-64.

[3] National Instruments. The CompactRIO Platform, web page. Retrieved 2016-05-05 from: http://www.ni.com/compactrio/

[4] S.E.A. gmbh. Mobile Communications, web page. Retrieved 2016-05-05 from: http://www.sea- gmbh.com/en/products/compactrio-products/mobile-communications/

[5] Hamamatsu Photonics K.K. C12137 series, web page. Retrieved 2016-05-05 from: https://www.hamamatsu.com/jp/en/C12137.html

[6] S. Moriuchi and I. Miyanaga, Health Physics **12** (1966) 541.

[7] S. Moriuchi, "A Method for Dose Evaluation by Spectrum-Dose Conversion Operator and the Determination of the Operator", (JAERI, 1970).

[8] S. Tsuda, T. Yoshida, M. Tsutsumi et al., Journal of Environmental Radioactivity **139** (2015) 260-265.

[9] Fukushima Prefecture. Retrieved 2016-05-05 from: https://www.pref.fukushima.lg.jp/sec/16025d/soukou.html (in Japanese)

[10] MEXT (2012). Results of Continuous Measurement of Air Dose Rates Through a Vehicle-borne Survey (as of March 2012). Retrieved 2016-05-05 from: http://radioactivity.nsr.go.jp/en/contents/6000/5637/24/338_Suv_091218_e.pdf

[11] Japan Map Center. The Extension Site of the Distribution Map for Radiation Dose. Retrieved 2016-05-05 from: http://ramap.jmc.or.jp/map/eng/

[12] Japan Atomic Energy Agency. Database for Radioactive Substance Monitoring Data. Retrieved 2016-05-05 from: http://emdb.jaea.go.jp/emdb/en/

[13] S. Kinase, T. Takahashi, S. Sato et al., Proceedings of International Symposium on Radiological Issues for Fukushima's Revitalized Future, 30-31 May. 2015, Fukushima, Japan. pp. 40-43. Online. Retrieved 2016-05-05 from: http://www.rri.kyoto-u.ac.jp/anzen_kiban/outcome/Symposium'15_Proceedings_EN.pdf

[14] A.J. Cresswell, D.C.W. Sanderson, M. Harrold et al., Journal of Environmental Radioactivity **124** (2013) 22.

[15] M. Yuda, K. Abe, H. Shimura et al., Studies on the radioactive substances of farmlands in Fukushima prefecture (31), Proceedings of Annual meeting of Japanese Society of Soil Science and Plant Nutrition, 9-11 Sep. 2015. pp.169 (Japanese).

[16] Matsuura Denkosha Co.,Ltd. KURAMA series. Retrieved 2016-05-05 from; http://www.denkosha.net/product/kurama (Japanese).

Socially - psychophysiological Adaptation of the Patient, who has Suffered at Failure the Chernobyl Nuclear Power Station, transferred Acute Radiation Sickness of IV Heaviest Degree and Local Radiation Injuries I-IV of Severity Level.

N.A. Metlyaeva

State Research Center, Burnasyan Federal Medical Biophysical Center of Federal Medical Biological Agency, Russia, Moscow, Zhivopisnaya str., 46, 123182.

Abstract. *The purpose:* the clinical-psychophysiological estimation of feature of socially-psychophysiological adaptation of patient acute radiation sickness of the heaviest degree T.A.M., the victim in failure Chernobyl Accident (24 years of supervision), according to its individual mental adaptation and the periods stage its mental adaptation. *Material and methods:* T.A.M., 1962 year of birth, worked as the engineer - driver inspector on Chernobyl, formation technical, on April, 26th, 1986 fulfilled not in the change of a duty of the turner on duty. In April, 1986, at the age of 24 years, as a result of failure Chernobyl Accident, has transferred acute radiation sickness IV heaviest degree from rather uniform external a gamma-beta of an irradiation, a bone marrow syndrome of IV degree (a dose on a marrow of 9,8 Gr), local radiation injuries: artificial both eyes, plural angioectasia, chondralatrophic, cicatricial changes of a skin. The Asthenic syndrome. A chronic hepatitis of the mixed etiology (alimentary-toxic with HCV +) with moderate activity of process and inside hepatic cholestatic. Baselioma skin of the left hip, after operation 01.12.2008. Diffuse the changed thyreocele of II degree, thyroid balance. Abusing alcohol about 5 years, especially appreciable last 3 years. Psychological inspection T.A.M. with use of test MMPI. The test of Kettell, Raven, sensorimotor reactions it was spent in dynamics in 1999, 2000 and 2002. Clinical supervision over the patient was spent within 24 years (1986-2010). Psychophysiological inspection is spent with use of the automated programmno-methodical complex "Expert +" intended for research of personal properties of the person, memory and intellectual features of the person. Estimation of efficiency of mental adaptation under test MMPI spent taking into account height of an indicator of T-points. The height of indicators of T-points <70> 30 specifies in effective psychophysiological adaptation, between 70 and 80 T-points - on unstable mental adaptation, above 80-points – on an overstrain of mental adaptation. Deviations from average values MMPI downwards less prognosis are significant, than liftings. Unlike MMPI, in father-in-law Kettella prognosis the significant can be both liftings, and decrease in values of separate factors. Average value of any factor of the test of Kettella is 5,5 wall. The expressed deviations from averages are values of factors more than eight and less three. *Results:* the Average profile of multilateral research of the person (MMPI) T.A.M. and dynamics of indicators on years of supervision (1999, 2000, 2002) exceed borders of population statistical norm (<70> 30) on a scale of T-points and testify to an overstrain of mental adaptation T.A.M., transferred acute radiation sickness IV heaviest degree and plural local radiation injuries I-IV of degree. Dynamics of an emotional pressure and intensity of its mental adaptation (the test of Kettell) correspond to relative density of infringements of mental adaptation, characteristic for the period of adaptable exhaustion. The tendency to occurrence of personal lines at T.A.M is revealed in the form of apathetic depression (1999), with transition in disturbing depression (2000) with somatic alarms and to display of lines of the classical disturbing-depressive person (2002) with a low energy potential. For psychophysiological inspection rather high indicators of intelligence (factor B-7 of walls) and good is figurative-logic thinking - the test of Raven remained. Speed simple and difficult sensorimotor reactions and reaction to moving object in norm, but is absent accuracy of combination as its real functional reserves can't provide this accuracy because of the expressed uneasiness. Efficiency of psychophysiological adaptation depends not only on a dose of an irradiation and weight of the transferred disease, but in more to a measure from premorbid properties of the person of the victim, its genetic features with propensity to alcohol addiction which have the major value for maintenance not only the biological properties underlying physiological aspects of adaptation, but also features of socialization on which depends both actually mental, and socially-psychological adaptation and sociolabor installation. *Conclusions:* Specific features of psychophysiological adaptation T.A.M. and its expressed personal lines, have appreciably defined its behavior and death from an alcoholic intoxication of the patient almost restored after an irradiation in a dose of 9,8 Gr.

KEYWORDS: acute radiation sickness; ionizing radiation; Chernobyl accident; local radiation injuries, adaptation.

1 INTRODUCTION

The condition of physical and psychological health and success activity appreciably depends of efficiency of process of adaptation. It is known that, law mental and psychophysiological adaptations appears to be its stability. In the stage of mental adaptation there are three periods: primary, stable, adaptation and adaptation of fatigue. Periods of increased risk of change adaptation observed in the first 3 years of stay in unusual circumstances and in the period after 10 years, the minimum in the period of 4 to 10 years [1, 2, 11].It is seen that the survivors of emergency prone to alcoholism. Clinical model for the evaluation of the state of adaptive mechanisms can serve as alcoholism, which can be considered as sickness of disturbed adaptation. Moreover, due to the change adaptation include the acceleration of development of abstinence syndroms, and transformation of alcoholism in unusual circumstances [4]. Development of alcoholism, in most cases, is in critical periods of adaptation and during adaptation of fatigue. On the rate of formation of chronic alcoholism also vary critical periods stable adaptation. In the period of primary adaptation alcoholism develops within 1 year of alcohol abuse in almost 2/3 of the surveyed patients, in the period of adaptation of fatigue - 56%, in the period of stable adaptation - only in 1/3 of the patients [5]. A number of authors [6] distinguished syndrome disadaptation as the characteristic of the recovery period of acute radiation sickness. A.K. Guskova, recognizing the reality of a number of pathophysiological mechanisms of this syndrome, it does not see in it anything specific in comparison with the effects of other heavy somatic diseases and believes that a degree it is connected not only with the severity of the injuries, but more with the original background neuro-endocrine regulation and some situational factors, a kind of separate patients [7].

The aim of this study is to evaluate the characteristics of the socio-psycho-physiological adaptation of a patient ARS of extremely severe degree of T.A.M., suffered from the ChNPP accident (24 years of observation), according to his individual mental adaptation and periods of the stages of his psychological adaptation.

Tasks of clinical-social and psycho-physiological examinations are:
- assessment of the clinical and social data at the time of the psycho-physiological examinations;
- evaluation profile personality and current mental status (MMPI test);
- character assessment of the person (Kettell test);
- assessment of figurative logical thinking according to the Raven test;
- assessment of the operator's health according to sensomotoric reaction (PSMR, SSMR) and reaction to the moving object (RMO).

2 MATERIAL AND METHODS

T.A.M., 1962 year of birth, worked as a machinist-crawler on ChNPP, education is medium-technical, in 26 April 1986 was executing for not his shift responsibilities of the duty mechanic. In April 1986, at the age of 24 years old, due to the ChNPP accident, suffered ARS IV a very severe degree from relatively evenly external gamma-beta irradiation, bone marrow syndrome of IV stage (dose to the bone marrow 9,8 Gr), local radiation injuries: artificial of both eyes, multiple teleangioectasia, fibro-atrophic, cicatricial changes of skin. Asthenic syndrome. Chronic hepatitis of mixed etiology (alimentary toxic with HCV+) with moderate activity, process and inside hepatic cholestasis. Baselioma of skin of the left hip, after excision of 01.12.2008. Diffusive modified goiter of the II stage.

T.A.M. from 1986 to 2002 entered the clinic the Institute of Biophysics regularly, was moderately sociable, loved to play chess, passed willingly psycho-physiological examination. For the first 7-8 years after the transferred ARS he often recurred beam ulcers and conducted re-plastic operations, i.e. these 7-8 years primary adaptation went to treatment, on care of young children (two girls of age 2 years and 2 months, were born before the ChNPP accident), on the decision of housing for themselves and parents (got comfortable apartments in Moscow), wife employment, provision of childcare facilities). In 1989 he was involved in public activity in the organization «Union «Chernobyl» of Russia».

In the period of stable adaptation appeared the desire to work. In 1992, the entire group of ARS patients, left in Moscow to continue the qualified treatment, organized his own company and was involved in commercial activities. T.A.M. did the work of the founder. Worked actively within 3 years, then commercial activity were forced to stop because of the peculiarities of reorganization period in Russia. Since 1993 he joined the participants of the public organization «Our right», which is engaged in charitable activities, providing aid to liquidators of Chernobyl, victims in the accident at «Mayak», professional patients, participants of nuclear weapons testing at the Semipalatinsk test site, the New earth and the other victims of radiation accidents and incidents.

In the family helped his wife to raise children, helped around the house reluctantly. He was irritable and short-tempered. The children grew up, his concern was no longer in need, showed interest to grandson, interest quickly passed. Was closed, increased irritability, irascibility. Became interested by the construction of the unfinished house in the village and was addicted to drinking. Since 2003, less of the time was coming to the clinic the Institute of Biophysics, with the active calls from the inspection was refused, especially categorically from the psycho-physiological examinations. Gradually became badly to make contact, was are taciturn, sullen, and often sat in silence in the side, was isolated from friends. In August 2004 got in a car accident. At the Last admission to hospital 15 March 2010 complained of headaches, poor sleep, pain in the lumbosacral spine. In the analysis of blood from 15.03.2010.: HGB 126 g/l, RBC 41, PLT 136 10^9\ l, WBC 7000.b.0,2, e.2,1, s. 1, seg.68,4, lym. 17,3, m.11, ESR 45 mm/h. It can be noted thrombocytopenia and accelerated sedimentation rate, due, perhaps, chronic hepatitis with moderate activity process. From the words of his wife, he had abused alcohol during the last 3 years, «he was growing irritability, irascibility. He rushed, did not find a place to be, rapidly depleted. It would be better, she said, if he would perform some feasible work, but temper and irritability and fatigue hardly have suffered him to work successfully. Alcohol for him, rather, was the cure for stress. In connection with the fact that the family and the doctors continue to prevent alcoholism, he found the solution in dealing with distant relatives, living in the village and willingly drinking with his money. He left for his relatives in the village of Krupets Rylsk region. 18.10.2010 he entered the Rylsky Clinic in serious condition. According to the words of relatives, patient for a long time abused alcohol, during the last two weeks – in 1,5 liters per day. Condition rapidly deteriorated in the days before admission, when the patient was unable to leave his bed. Taken to the clinic by his own transport, hospitalized. The patient has spent in hospital for 16 hours. In spite on the started treatment, the status progressively worsened, was observed phenomena liver failure, hypotension. T.A.M. died in 19.10.2010. According to the autopsy study of a patient with multiple organ manifestations of chronic alcoholic intoxication syndrome of portal hypertension, developed bleeding from esophageal varices, along with the occurred liver failure caused by cirrhosis of the liver, caused to death.

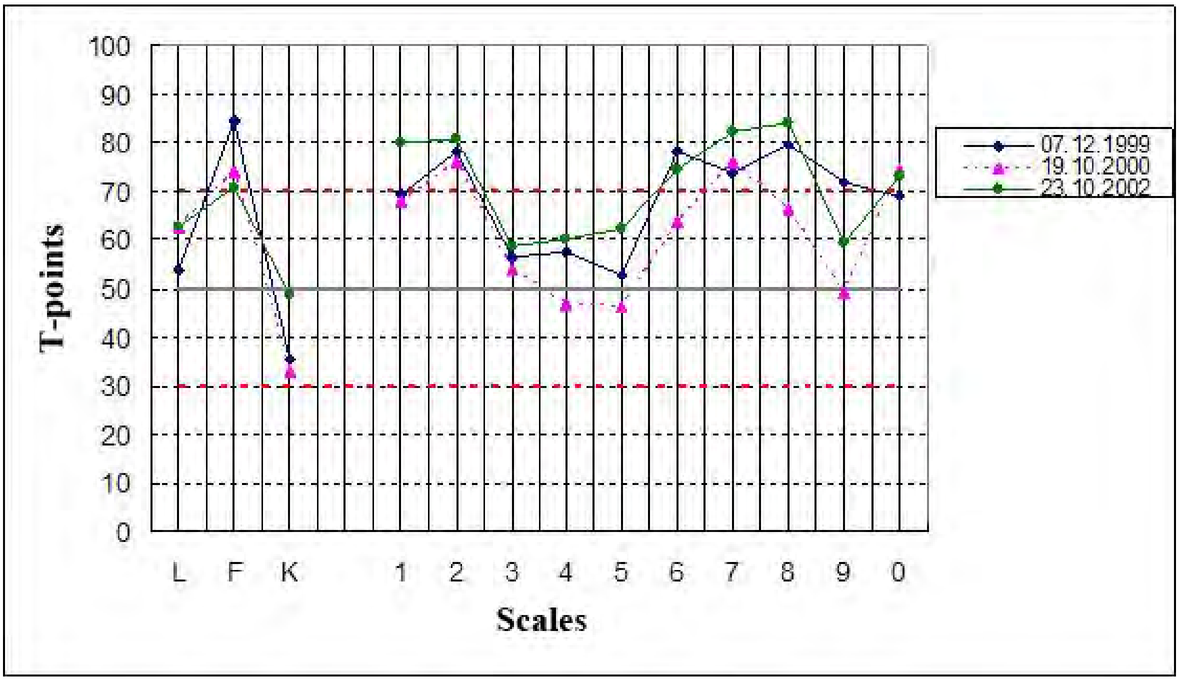
3 RESULTS AND DISCUSSION

Psychological examination of T.A.M. using MMPI, Kettell and Raven tests, of senso-motor reactions were carried out in dynamics in 1999, 2000 and 2002. Clinical observation of patient was held for 24 years (1986-2010). Psycho-physiological examination carried out using an automated program-methodical complex «Expert +», intended for research of personal property rights, cognitive and intellectual features of personality. Evaluation of effectiveness of mental adaptation MMPI test was performed with the height of the T-points index. Borders of population of statistical standards on a scale T-points are 70 points (M+2 sigma) and 30 T-points (M-2 sigma), with 95% probability limiting the target population.

Rises MMPI for 70 T-points with high (till 95%) reliability indicate an increased risk of mental and social adaptation. Height values between 70 and 80 T-points shows on unstable mental adaptation, above 80 points - to the surge of mental adaptation. Deviations from the average MMPI down less prognostically important, than ups. Unlike MMPI, in a Kettell test prognostically important are as lifts and lowers in the values of certain factors. The average value of any factor Kettell test is 5,5 wall. Significant deviations from the average values are factors more than eight and a minimum of three.

According to psychological MMPI test can be noted that the state of T.A.M. at the time of the survey in 1999 (fig.1) is characterized clinically apparent originality of thinking, lack of approach to the solution of vital problems, feeling of strangeness, singularity and incomprehensibility of what is happening around (scale 8), simultaneously he feels a sense of loneliness, abandonment, and its state is determined by the fact that the Clinic is called apathetic depression, i.e. he is indifferent to everything, that early had made him an encouraging experience. His depression is characterized by the same feeling of the strangeness, unusualness, unconventional of his status, and therefore, on the one hand, he feels the need to communicate, on the other hand, it limits contacts, and these contacts are quite formal and superficial character (scale 0). He is in this period is quite sociable, free of prone to neglect the social norms and rules of conduct (scale 4), not impulsive, self-esteem is lowered, and he in sort of way explains, why the ambient life became indifferent and uninteresting (scale K). Given that this man, who has been in a situation, where on earth visited units, not counting those, who in Japan was hit by an explosion, so he feels himself, like, those who were exposed to radiation, feel odd, unusual, his status, his social situation, his health status, he feels strange and unusual (scale 8).

Figure 1: Psychological profile of multilateral study of personality (MMPI) patient G.O.I. with distant effects ARS IV degree of severity, which had suffered from the ChNPP accident, for the period 1999, 2000, 2002.



The Scales of accuracy: L - lies, F - reliability, K - correction.
Main scales of MMPI: 1Hs - hypochondria, 2D - depression, 3Hy - hysteria, 4Pd - psychopathy, 5Mf - of masculinity, femininity, 6Pa - paranoia, 7Pt - psychasthenia, 8Sch - schizophrenia,
9Ma - mania, 0Si - introversion.

Eviction is working badly, as it should be in person, that has clinically expressed schizoid features. Inclined to exaggerate the difficulties, associated with his condition, tries to attract the attention to his problems, firstly, because he is sick, secondly, because his condition is absolutely non-standard compared to the other patients, he expressed quite enough features of self-criticism, he is not inclined to such a primitive way to exaggerate his dignity and his some primitive positive qualities. However, he is stubborn, persistent, purposeful, sensitive and irritable, and it sometimes seems that others downplayed, disliked (scale 6). He is not demonstrative, is not inclined to exaggerate his own advantages and external sides with her some positive qualities. Hypochondriac mechanisms (scale 1), which are used to optimize the mental state which is working, and he, of course, pay attention to his health, but to completely remove the anxiety that it is in presence, he cannot do with these mechanisms, and quite possible that his conversion mechanisms (scale 3) somehow worked, but they didn't work effectively, because he is not demonstrative, the vegetative system is performing poorly. But all of these neurotic mechanisms are not enough for, to remove the anxiety and tension, and then he attracts already psychotic mechanisms, actually allowing him to have sharply raised the scale 8. Struggling he is trying to explain his state of the uniqueness of what happened to him, but it's also did not explain everything, that’s why he has anxiety and depressive features (scale 2, 9) and psycho statistic features (scale 7) are existing quite obvious and this just is not enough in order to optimize the state completely.

When you re-run of psychological examination (fig.1) by the MMPI test (year 2000), we see that significantly decreased the originality of thinking (scale 8), significantly decreased the level of activity (scale 0). The combination of originality of thinking, anxiety depressive features, features of apathetic depression with high indices of activity, indicate the presence of him the features of emotional instability. The imbalance of the vegetative regulation is not inconsistent with his tensions and fears about his prospects in life, an understanding of what is happening to him, all of this that he has is hanging in the air and causes so poorly organized, in a sense, chaotic activity, which is implemented as necessary in connection with the originality of thinking, because of that, that the psycho statistic features combined with increased activity indicate a low level of organization and systematization of action and behavior. All of this has sharply decreased. He became much less active.

Although anxiety-depressive features of him have been preserved, and his status from a state of apathetic depression with increased excitability turned into a classic alarming depression - is 2, 7, 0, sharply ↓-9, low 4 - he is a typical anxiety-depressive personality. He felts a strong concern about unmet symbiotic needs, the needs in communication, however, he has closed, closed much more (scale 0). If in 1999, he tries to be open, trying somehow to establish contact, here he is completely closed. He has a small number of significant, considerable contacts, by the number a few contacts, contacts are very much limited. Here his indicators raised high over 70 t-points and a gap in scale 9 indicates that his activity is not enough to communicate. The peculiarity of him, is also sharply felt (scale 8). Instead of explicitly identity remained only the psycho statistic features (scale 7). He doubts in everything, unsure in everything, feels that he needs repeatedly to analyze the situation, but no efforts are sufficient to acquire the confidence and peace, although he has become quite social mobile, although he had always been a low rate in 1999 by the scale 4, the indicator of him is even lower, it is even more carefully observes social norms and rules of conduct, even more often thought about what he is doing something wrong. The Self-esteem of him has became even lower, in comparison with 1999, when he was relatively high self-esteem due to increased activity, now it is sharply dropped (scale 9), and he went into a classic depression. As for his the vegetative system and hypochondriac manifestations, there is actually nothing has changed, he is still concerned about the state of his health. Assessment scales practically nothing has changed, only increased the desire to look in the most favorable status, but it is not excluded, that this desire is real, not in order to present himself in a most favorable status, but he really struggling to comply with all, even less substantive norms and rules of conduct. Preemption is still weak, it is not enough to optimize the state and to minimize anxiety and the anxiety it is present in all its manifestations, as well as depressed features. It's all at least in the norm of clear accentuation. If in 1999, you can tell about clinical manifestations apathetic depression, now he is dominated by anxiety and depressive personality features.

In 2002 he is much more concerned about his health condition (fig.1). He appeared again on the background of hypochondria (scale 1) depressed features, maximum is in 2002 (scale 2, 9). Again, in comparison with the year 2000, it has dramatically increased the originality of thinking (scale 8) and even higher than in 1999, psycho statistic features in him were maximum in 2002 (scale 7). With comparison to 1999, in him in 2000 the features of affective rigidity (scale 6) were reduced, and in 2002 also peaked maximum, he became suspicious, irritable, more often began to come to his head, that people are unkind to him, and do not allow it to implement useful qualities, that attach to him a little importance. The Sociability he has not increased and activity raised up and most likely, it will implement it in any aspirations associated with proof of his own rights and of what he has suffered. It is quite possible that in this period became involved in the search for truth and proof that he is ill, suffered what he wanted to do more, to take care of. All this on a background of reduction of masculinity (scale 5). Probably it is connected with the fact that it had deteriorated endocrine status, it is possible he has early menopause, infertility and was broken hormonal background. At the same time in comparison with the year 2000, he became more impulsive, more irritable and short-tempered, the features of affective rigidity are intensified. Scales 4 and 6 of him are tall and slightly expressed, but, nevertheless, features that compared with previous years could be called an increased mood irritability, became stronger in him, 4 and 6 became the most high. He peculiarly is looking for the truth and, at the same time, implements his not very positive from the point of view of social encroachments in terms of what he wants something, what it is that somebody is trying to implement not quite the most ideal ways. Although 4 continues to remain at a high level, i.e. this scale is relatively increased. But it has increased the displacement, i.e. scale K is the highest in him and it is quite effective in significantly better displaces the signs of his trouble, but apparently this is not enough to shut down additional mechanisms. The mechanism of somatic reaction of anxiety is joined, more actively, though he certainly is combined with proper signs of violation of the vegetative regulation, which have become more prominent. In the profile height turns out that this period of 2002 for him is the most disadvantaged. The mechanism of somatic reaction of anxiety is joined and showed the features of typical anxiety-depressive personality.

According to the Kettell test (fig.2), intelligence that he has is above average (factor B = 7 walls). The results for Ravenna test (fig.3) indicate good figuratively-logical thinking (one mistake). At the same time, well-mannered forms of behavior is reduced (factor N), he is practical (factor M). Reduction of emotional control, integrative and shyness of behavior, subject to very prominent anxiety (factor F1). This concern is also highly expressed on the factor O and the factor Q4. He is very ambitious, his Q4 - 10, he is experiencing anxiety and tension and desire to achieve high social status. He is experiencing the anxiety, at the fact that his social status does not match his own ambitions. Very high level of pretensions, ambition, really can be explained and the tension in connection with the fact that he is in frustration. He is frustrated because he wanted more, than what is need for him. He is very closed, emotionally unstable, cool, relatively easily disarmed in critical situations, directly, with the spontaneous manifestations, anxiety helps him.

Figure 2: The Characteristic specialties of a person of Kettell test of the patient T.A.M. with long-term consequences of ARS IV degree of severity,
of the victim at ChNPP accident, for the year 2000.

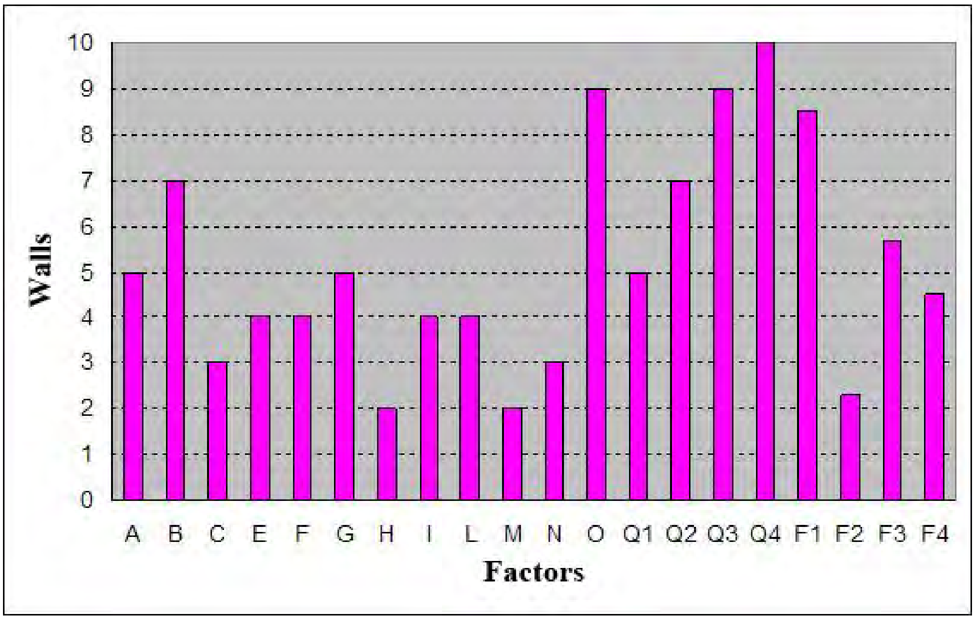
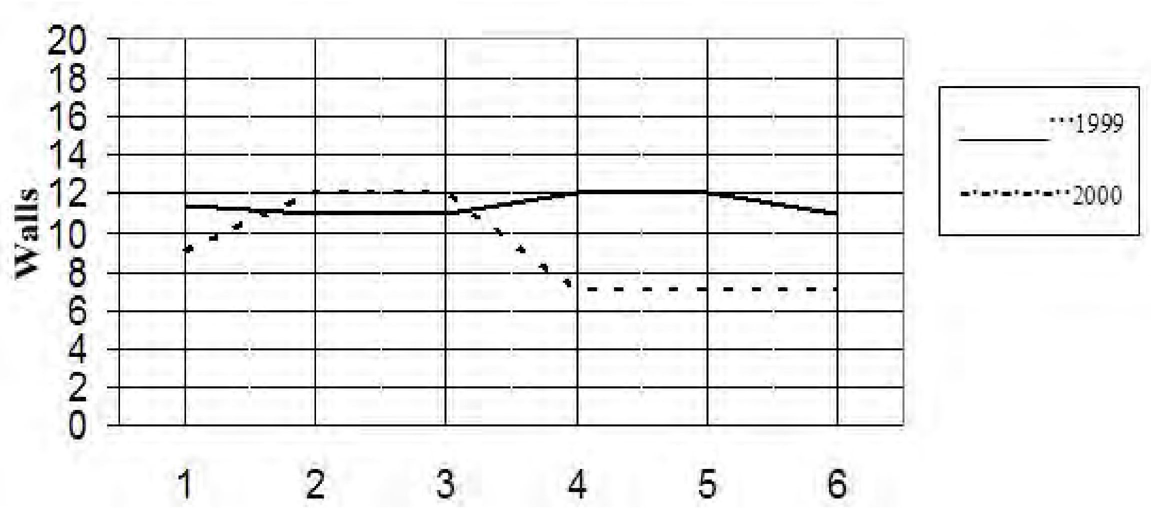


Figure 3: Parameters of the Raven test of T.A.M. patient with consequences of ARS IV degree of severity and LRI I-IV degree of severity.



Speed simple sensomotoric reaction (SSMR) good (229,3 ms), good mode (201 ms), high speed of reaction in the histogram. The speed sensomotoric good reaction (395.6 ms) within the norms admitted 4 mistakes.
Reaction to the moving object (RMO) not bad (1069.1 ms), but exactness of the combination is none (2%), he is anxious, he is trying to do as accurately as possible, and his actual functional reserves for this precision cannot be provided.

The first period of primary adaptation (the stage of anxiety) directly connected with the change of conditions in which lived and worked T.M.A. The Tension of adaptation mechanisms and the development of mental stress in this period can be explained by the fact that due to the ChNPP accident occurred eliminating the usual familiar environment (G. Pripyat, the work of mechanic-inspector at ChNPP) and the appearance of another environment, on one hand, it changes the structure of the situation, and in the another - reduces the effectiveness of adaptive behavior, based on acquired in other conditions skills (treatment in the clinic of Hematology research center, Institute of Biophysics - KB № 6 about the extremely heavy ARS and local radiation injuries of I-IV grades within 7-8 years, loss of job, housing, the new residence in Moscow, supervision and treatment in the dynamics, family with two small children).

The Tension of adaptation mechanisms in the period of primary adaptation can be placed on dependency of the mobilization of psychophysiological resources of the body, characteristic for adaptation voltage [11]. This mobilization provided T.A.M. a sufficient effectiveness of the adaptation process and the transition into a stable mental adaptation (resistance stage). Concerning adaptation value resulting voltage can be seen in the fact that along with the increase in the severity of characteristics that play a role in the formation of emotional stress and reflecting its intensity, for this period typically significant in adapting the increased level of motivation of achievement, which determines the desire for active mastering of the situation, to effective action, quickly ended. Along with the increase in the severity of characteristics that play a role in the formation of emotional stress and reflecting its intensity, a period of adaptation is increasing - stage of exhaustion (a frustrated tension, her attitude to the integration of conduct and the threshold of frustration, the level of realized lability, dissatisfaction of the situation and the position in it, anxiety). The role of profound personality changes as a result of the adaptation of exhaustion [9, 10, 11]. Along with this, noted the highest frequency of mental adaptation and the lowest special weight of the accented personalities, not detecting such violations. Excessive sharpening of accented features is getting difficult at mental adaptation and leading to decompensation of a mental condition. The deterioration of quality of mental adaptation in the period of adaptation of exhaustion reflects the integration (psychical and psychophysiological) on a new, less efficient level [4]. Such as the trend towards the emergence of a personality features, T.A.M. early as apathetic depression (1999), with the transition to the alarming depression (2000) with somatic reaction of anxiety and classic manifestation of anxiety-depressive personality (2002) with low energy potential. Persons with these characteristics (accentuated personalities by Leongard), in usual conditions normally would not find violations of mental adaptation. Expressed personal features, which largely determine their behavior, even contribute the mental adaptation in that case, if these features meet the requirements of the environment. However, if conditions change (alcoholism), this match disappears and long-term stress coping mechanisms leads to unwanted acute accented features, adaptive opportunities of an individual are violated, and accented features facilitate the emergence of intra-psychic and interpersonal conflicts, leading to decompensation of mental status [11, 12].

Thus, T.A.M. survived the difficult psychological situation, stress and mental trauma of the accident at the Chernobyl NPP and with extremely severe acute radiation sickness with local radiation lesions of the I-IV degree. The way out of this situation depended not only on the dose and severity of the disease, but mostly from premorbid personality features victim genetic features, including a possible predisposition to alcoholism. Given the use of T.A.M. at the time of the survey (1999, 2000, 2002) of high enough intelligence, good figuratively-logical thinking and good speed sensorimotor reactions, it can be concluded that alcoholism in T.A.M., probably, would be secondary to the growing depression (symptomatic alcoholism). This is confirmed by free and clinical interviews with him and observing his behavior, as well as information obtained from his wife (abused alcohol in last 3 years) and his friends (about 5 last years), i.e., apparently, somewhere beginning in 2006-2008.

Individual peculiarities of psycho-physiological adaptation of T.A.M. and his expressed personal features largely determined his behavior and death from alcohol intoxication of the patient, almost recovered after irradiation in the dose of 9,8 Gr.

4 CONCLUSIONS

1. The average profile of the multilateral study of personality (MMPI) T.A.M. and dynamics of indicators by the years of observation (1999, 2000, 2002) exceed the boundaries of population statistical norm (<70>30) on the scale of T-points and show the strain of mental adaptation of T.A.M., carried the ARS IV extremely severe degree and multiple local radiation injuries of I-IV degree.

2. Dynamics of emotional stress and tension to his mental adaptation (Kettell test) correspond to the specific weight of mental adaptation typical of the period of adaptation of fatigue.

3. Revealed the tendency to appearance of personality features of T.A.M. as apathetic depression (1999), with the transition to the alarming depression (2000) with somatization of anxiety and manifestation of features of classic of anxiety-depressive personality (2002) low-energy potential.

4. For the period of psycho-physiological inspection relatively high levels of intelligence (factor B-7 walls) and good figuratively-logical thinking – Raven test.

5. Speed simple and complex sensomotoric reaction and the reaction to a moving object in normal, but the lack of precision alignment due to the severe anxiety, indicates a failure of its real functional reserves, able to provide this level of accuracy.

6. The effectiveness of psychophysiological adaptation depends not only on the dose and severity of the disease, but mostly from premorbid personality features of the victim, genetic features, including a possible predisposition to alcoholism, which are crucial, not only on the biological properties of underlying physiological aspects of adaptation, but also peculiarities of socialization, refusal of employment, the upkeep of the family, from participation in the fate of their comrades who underwent the same severe disease, on which depends the actual mental, and social-psychological adaptation and social and labour installation.

7. Alcoholism of T.A.M., probably, would be secondary to the background of growing depression (symptomatic alcoholism). Individual peculiarities of psycho-physiological adaptation of T.A.M. and his expressed personal features, largely defined its behavior, and death from alcohol intoxication of the patient, almost recovered after irradiation in the dose of 9,8 Gr.

5 REFERENCES

[1] *Berezin F.B.*, *Miroshnikov B.P.*, *Sokolova E.D.* Methodology of multilateral study of personality / The structure, principles of interpretation, some areas of application. M., - 2011, p. 318.

[2] *Berezin F.B.*, *Sokolova E.D.* The human adaptation to the conditions of the North and mental health / Scientific-technical progress and sub polar medicine: Abstracts of the IV international Symposium on sub polar medicine. Novosibirsk, - 1978, 1, P. 252-253.

[3] *Korolenko C.P.* Psychophysiology of the person in extreme conditions. L., - 1978, p. 150.

[4] *Sokolova E.D.*, *Kalachev V.F.*, *Dolnikova A.A.* The Clinical aspects disorder of mental adaptation / Mental adaptation of the person in conditions of the North. Vladivostok, - 1980, P. 77-96.

[5] *Malkov U.M.* The influence of the adaptation period to the conditions of the Far North on the development and course of chronic alcoholism / Pathogenesis and clinical of alcoholic diseases. M, - 1970, P. 64-67.

[6] *Glazunov I.S.*, *Blagoveschenskya V.V.* Journal of neuropathology and psychiatry of S.S. Korsakova, - 1968, 8, p. 1129.

[7] *Guskova A.K.*, *Baisogolov G.D.* Radiation sickness of a person / Medicine, M., - 1971, p. 384.

[8] *Kaznacheev V.P.* Essays on the theory and practice of ecology of human. M., - 1983, p. 180.

[9] *Francuzskich J.R.P.*, *Jr.*, *Rodgers W.* Adjustment as person-environment fit /J. Soc. Issues, - 1974, № 18, P. 316-333.

[10] *McCrae R.R.* Situational determinants of coping responses: Loss, threat and challenge / J. Pers. Soc. Psychol, - 1984, 46, № 4, P. 919-928.
[11] *Berezin B.F.* Mental and psychophysiological adaptation of a person. - L: Science, - 1988, p.267.
[12] *Baevsky R.M.*, *Berseneva A.P.*, *Boyko E.R.* with co-auth. Medical and ecological research within the space experiment «Mars-500» / In collection: «Nosogeny. Problems of healthy lifestyle». Materials of the 5th International scientific Conf., SpB., - 2009, 17-18 December 17-18, P. 181-183.

Early measurements of members of the public after the Fukushima Daiichi NPP accident: Data made available to the EURADOS WG7 Survey

P. Fojtík[a*], M.A. Lopez[b], D. Franck[c], J. Ośko[d], U. Gerstmann[e], C. Scholl[f], A.L. Lebacq[g], B. Breustedt[h], L. del Risco Norrlid[i]

[a]SÚRO, Bartoškova 28, 140 00 Praha 4, Czech Republic.
[b]CIEMAT, Centro de Investigaciones Energéticas Medioambientales y Tecnologicas, Avda. Complutense 40, 28040 Madrid, Spain.
[c]IRSN, Institut de Radioprotection et de Sûreté Nucléaire, LEDI. Fontenay aux Roses, France.
[d]NCBJ, National Centre for Nuclear Research, A. Sołtana 705-400 Otwock, Poland.
[e]BfS, Bundesamt für Strahlenschutz. Ingolstädter Landstraße 1, 85764 Neuherberg, Germany.
[f]LIA.NRW, Ulenbergstr. 127-131, 40225 Düsseldorf, Germany.
[g]SCK-CEN, Belgian Nuclear Research Centre, Boeretang 200, 2400 Mol, Belgium.
[h]KIT, Karlsruhe Institute of Technology, Hermann-von-Helmholtz-Platz 1; 76344 Eggentein-Leopoldshafen; Germany.
[i]SSM, Strålsäkerhetsmyndigheten, Stockholm, Sweden.

Abstract. Results of in-vivo and in-vitro measurements and dose assessments of foreign citizens returning from Japan after Fukushima Daiichi NPP accident were made available to EURADOS Working Group 7 "Internal dosimetry" by bioassay laboratories in Europe and Canada. Due to the lack of early measurements of the internal contamination in the members of public in Japan, further analysis and processing of the data from the EURADOS survey can be considered as a complementary source of information contributing to clarify the situation. In this paper, the degree of correspondence between the measurement results and values estimated from the radionuclide air concentrations data is shown for certain locations and radionuclides. Case study approach is applied for determining intakes of radionuclides by monitored individuals, taking results of their in vivo counting in to calculation. Data to support assumptions taken to dose estimates based on atmospheric dispersion models are given. Also, scarce early data on the ratio of radioiodine and radiocaesium intakes not available elsewhere are presented, as well as contributions of short-lived radionuclides to the early dose.

KEYWORDS: Fukushima Daiichi NPP accident; internal contamination; in vivo measurement.

1 INTRODUCTION

Failures caused in the Japanese Fukushima Daiichi NPP in March 2011 as a consequence of a natural disaster resulted in large radionuclide releases to the environment [1].
Measurements, on voluntary basis, of persons who left Japan for their foreign countries soon after the disaster were performed by local dosimetry services in the countries. Some results were made available to the EURADOS Internal dosimetry working group (WG 7) for publication [2, 3]. In the absence of early information outside Japan on the radionuclide amount released to the atmosphere, the level of internal contamination of persons returned from Japan could have been an indicator of radionuclide air concentrations in Japan. Due to the lack of direct human measurements in Japan early after the radionuclide releases, atmospheric dispersion simulations [4-6], ground deposition data [7], and radionuclide concentrations measured in the air [8] are the basis for radionuclide intake and dose estimations made for members of the Japanese public. Questions arise about the extent to which the inhalation intake expected based on those indirect data correspond with the intakes calculated for the radionuclide retention measured in people exposed to the contaminated air. In this paper, the inhalation intakes calculated from in-vivo measurements are compared to intakes expected on the basis on available radionuclide air concentration data.
Intake ratio of ^{131}I to ^{137}Cs is a key parameter for dose reconstruction algorithm that is based on ^{137}Cs whole body measurements performed among the Japanese population later after the radioiodine release had ceased [9].

*Presenting author, e-mail: pavel.fojtik@suro.cz

Contribution of very early (~ 12-15 March 2011) inhalation of short-lived radioiodines to thyroid doses is estimated as great as 30–40% of the ^{131}I dose for members of the public in Fukushima and neighbouring prefectures [10].
However, no results of in vivo measurements have been published so far to supplement that estimation. The results of in vivo gamma-ray spectrometry measurements of persons who left Japan in March and April 2011 were processed in the frame of the "EURADOS Survey on in vivo monitoring data and internal dose assessment" [3]. The present analysis of homecomings individual monitoring data aims to contribute to clarify the above mentioned open questions.

2 MATERIALS AND METHODS

2.1 Cases of exposed persons

Only those cases of persons that show positive retention values for iodine, caesium and tellurium radioisotopes are considered in the present paper. Cases with radionuclide retention values lower than detection limit (DL) cannot contribute to the intention of this paper. An exemption is made for lower than DL values of very early inhalations.

2.2 Data and parameters

2.2.1 Radionuclide air concentrations

For Japanese sampling sites, concentrations (in $Bq \cdot m^{-3}$) of radionuclides in the surface air in March 2011 are available in documents published either in printed or electronic (web) media.

Table 1: References to radionuclide air concentration data used for intake calculations

Sampling site	Reference	Sampling site	Reference
Tōkai	JAEA [11]	Yokota Air Base	OTR [16]
Ōarai	JAEA [12]	Tokyo	TMITRI [17]
Chiba pref.	JCAC [13]	Takasaki (Gunma pref.)	OTR [16]
Tsukuba (Ibaraki pref.)	NIES/KEK [14]	Sendai (Miyagi pref.)	OTR [16]
Wakō (Saitama pref.)	RIKEN [15]	Ishinomaki (Miyagi pref.)	OTR [16]

Radionuclide air concentrations for sampling sites in Kantō region are summarized by Tsuruta *et al.* [18]. In their paper, time pattern for air concentration ratios of ^{131}I to ^{137}Cs and of elemental to particulate form of ^{131}I is presented. The latter ratio is used in the present paper for the calculation of ^{131}I intake using radioiodine air concentration data.

2.2.2 Intake calculations

For the calculation of intakes by individual person cases, the ratio of 60:40 is taken for the proportion of the elemental and particulate form of ^{131}I in the surface air [10] Kim [19]. The following parameters are applied for the intake calculations: AMAD of 1 μm for the particulate form of all radionuclides, lung absorption type F for the particulate form of radioiodine, class SR-1 and type F for the elemental form of radioiodine, absorption type F for radiocaesium, type M for tellurium. Thyroid or whole body retention function values for different times of intake are calculated by IMBA® Professional Plus software [20] for both ^{131}I forms, their 60:40 mixture, and for the particulate form of radiocaesium. Retention function values from German regulation [21] are used for tellurium retention in the whole body.

2.2.3 Exercised activity and ventilation rates

The pollution of air masses was caused by several events occurring in the Fukushima DII NPP. Thus, temporal variations of radionuclide air concentrations at a particular air sampling site can be described by periods and peaking times [18, 22]. Where possible, radionuclide air concentrations were therefore estimated in the Survey for specific time intervals during a day, namely for 0-6 o'clock (sleep), 6-8 (sitting), 8-18 (light exercise), 18-22 (sitting), 22-24 (sleep).

All cases are adult persons. Ventilation rates as recommended in ICRP 71 [23] for the adult member of public were used for the exercise level corresponding to each of the timeslots of the day. In combination, the exercised activity budgets and ventilation rates result in the daily inhaled volume of 22 m^3.

3 RESULTS AND DISCUSSION

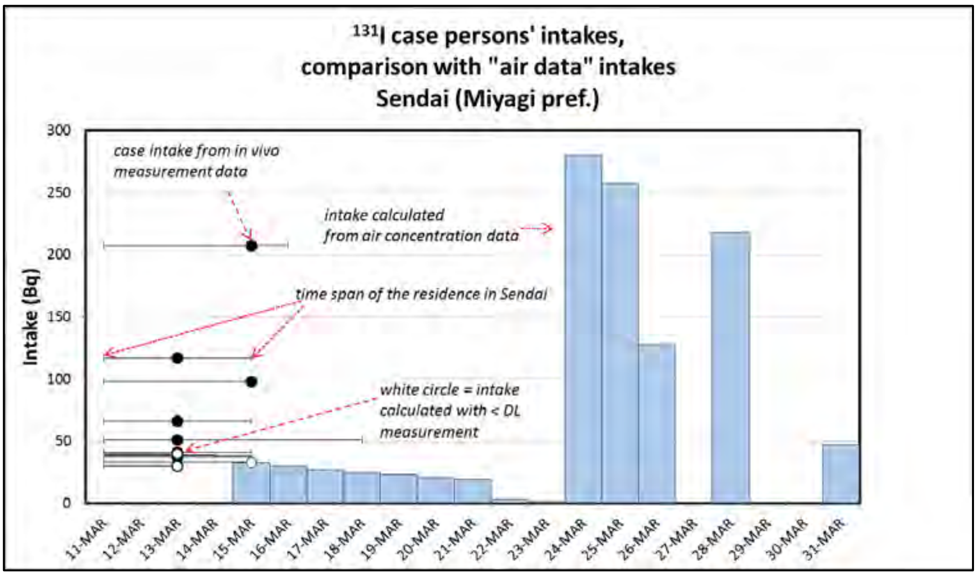
3.1 Comparison of inhalation intakes

Intakes calculated for person cases with the use of in vivo measurements results and those expected based on radionuclide air concentrations are compared for individual cases in Figures 1 to 6.
The intake calculated for a particular person case is indicated as a circle in Figures 1 to 6 and is assigned to a certain day spent by the person in a particular place. The day in which polluted air masses were passing the place of residence is chosen for intake assignment. A time span of person's residence in that particular place is also indicated in the charts. The expected inhalation intakes calculated based on the radionuclide air concentration for the place of residence are indicated as bars in the chart. No allowance is made in the calculations for any sheltering or reducing factors such as staying indoors or for duration of the stay as they are not known.

3.1.1 Cases in Sendai (Miyagi pref.)

In Figure 1, the comparison of ^{131}I intakes is shown for persons that stayed either in Sendai (Miyagi pref.) only or partly in Sendai and in a region not affected by radionuclide releases.
Neither radionuclide air concentration nor external dose rate data are available for Sendai before 15 March. Internal contamination of cases in Figure 1 confirms the transport of contaminated air masses by southerly wind to Sendai after the NPP reactor Unit 1 wet venting and the hydrogen explosion occurred at Unit 1 on 12 March and the subsequent radionuclide dry deposition to the surface air in Sendai before 15 March. No positive human in vivo measurement data is available to the Survey for radiocaesium and Sendai residents.

Figure 1: ^{131}I inhalation intake; comparison for Sendai (Miyagi prefecture).



3.1.2 Cases in Tokyo metropolitan area and Chiba

In Figure 2, 3, and 4 the comparison is made for ^{131}I, ^{137}Cs, and ^{132}Te intakes in the Tokyo metropolitan area and in Chiba prefecture. The intakes shown as bar charts are calculated based on radionuclide air concentrations reported for three sampling sites, namely for the Tokyo Metropolitan Industrial Technology Research Institute sampling site located at the Tokyo Bay, for the sampling site in Wakō city (north-west border of Tokyo), for the sampling site at JCAC, Chiba city, and for data from Yokota Air Base (some 40 km westwards from the Tokyo Bay).
All ^{131}I intakes calculated for the case persons based on their in vivo measurements are lower than one half of the corresponding intake estimated from ^{131}I air concentration data. On the other hand, ^{137}Cs intakes calculations agree better with each other.
In Figure 4 intakes are compared for ^{132}Te. The intakes calculated for person cases can be explained by inhalation.

Figure 2: ^{131}I inhalation intake; comparison for Tokyo metropolitan area and Chiba.

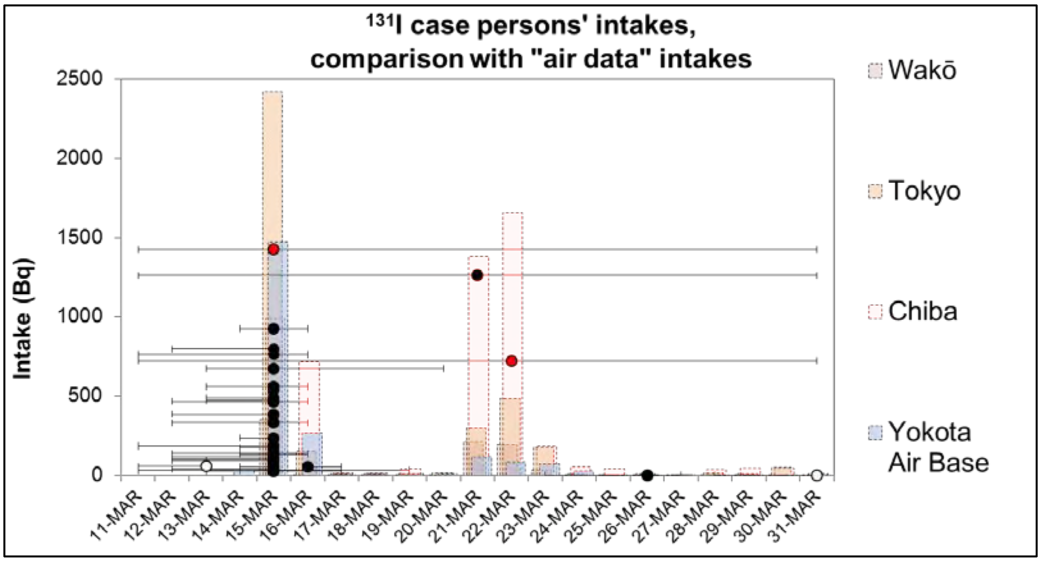


Figure 3: ^{137}Cs inhalation intake; comparison for Tokyo metropolitan area and Chiba.

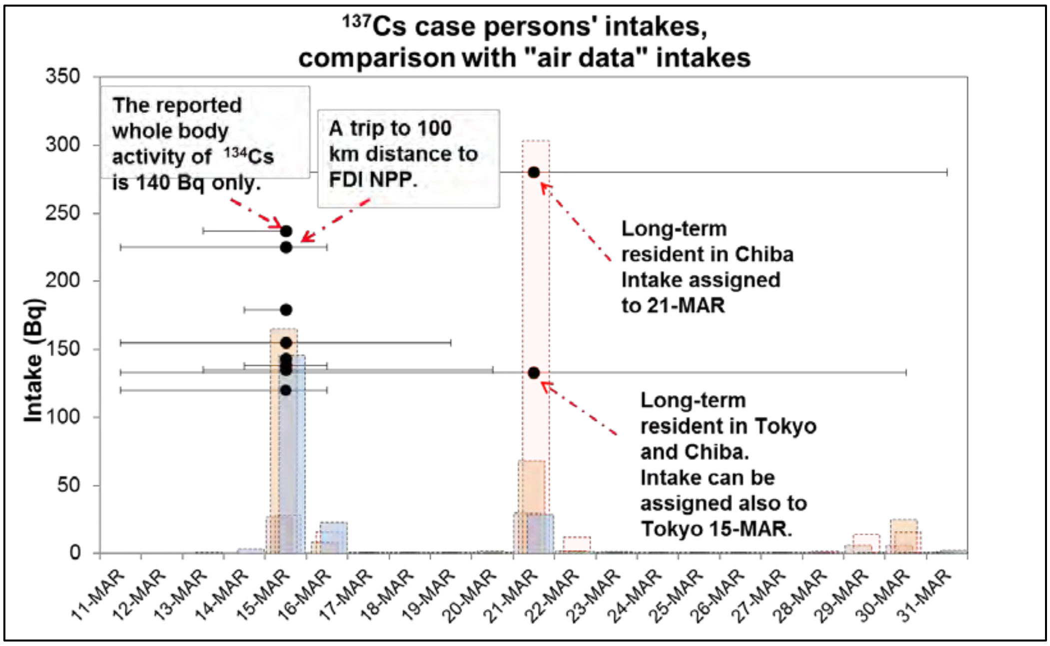
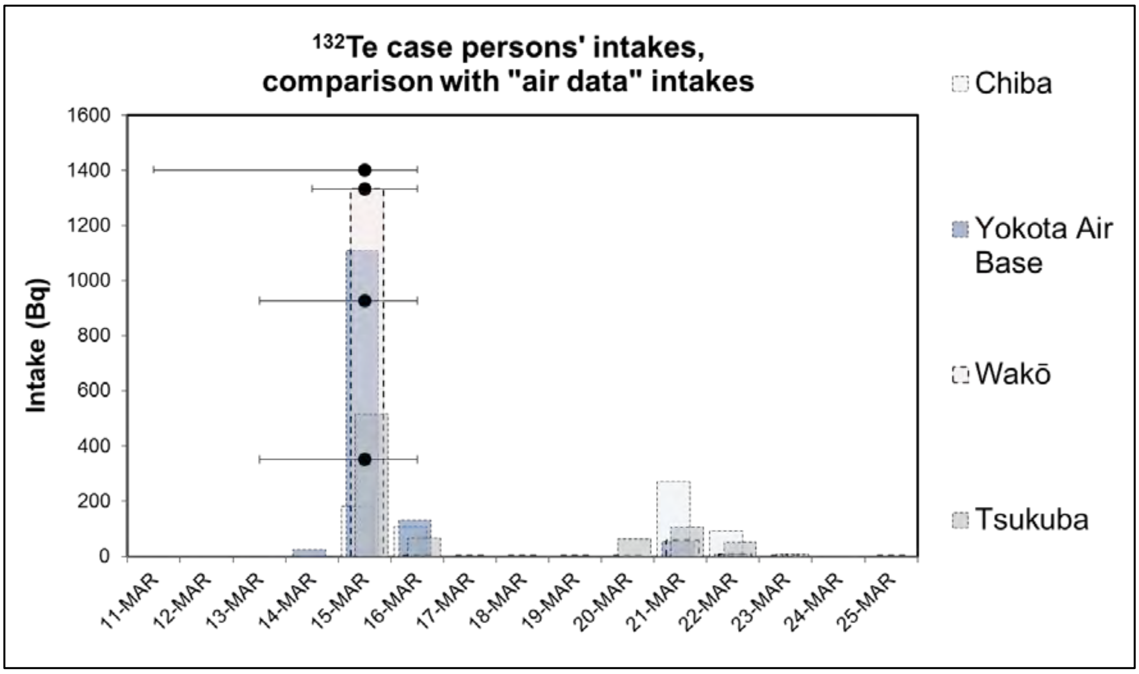


Figure 4: ^{132}Te inhalation intake; comparison for Tokyo metropolitan area and Chiba.



3.1.3 Cases in Ibaraki prefecture

In Figure 5 and 6, the comparison is made for ^{131}I and ^{137}Cs intakes in Ibaraki prefecture based on data from two sampling sites of JAEA [11, 12]. A source of intake for two cases of persons with ^{131}I intakes around 5 kBq assigned by them to the 13March has not been explained. No dose rate elevation is reported for their place of residence before 15 March. Those persons returned to Europe on 15 March probably *via* Tokyo airport and could have had an intake in Tokyo metropolitan area as well. No whole body radiocaesium measurement is reported.
Again, estimated and calculated intakes of ^{137}Cs agree better with each other than for ^{131}I.
Among persons showing measurable ^{132}Te whole body retention in the Survey, only one person can be supposed to have an intake in Ibaraki prefecture.

Figure 5: ^{131}I inhalation intake; comparison for Ibaraki prefecture.

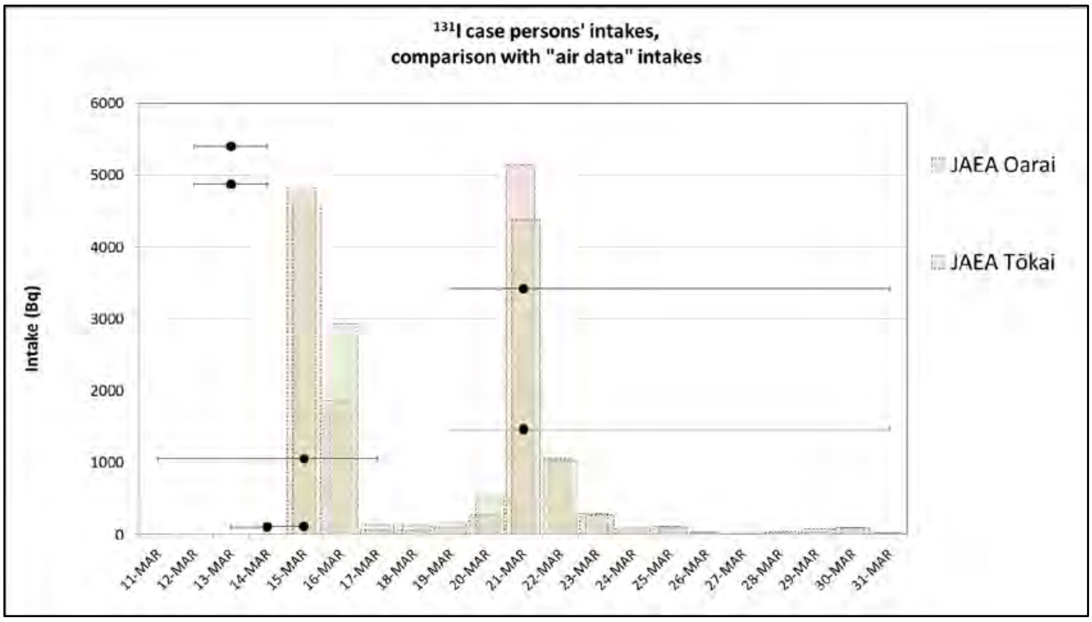
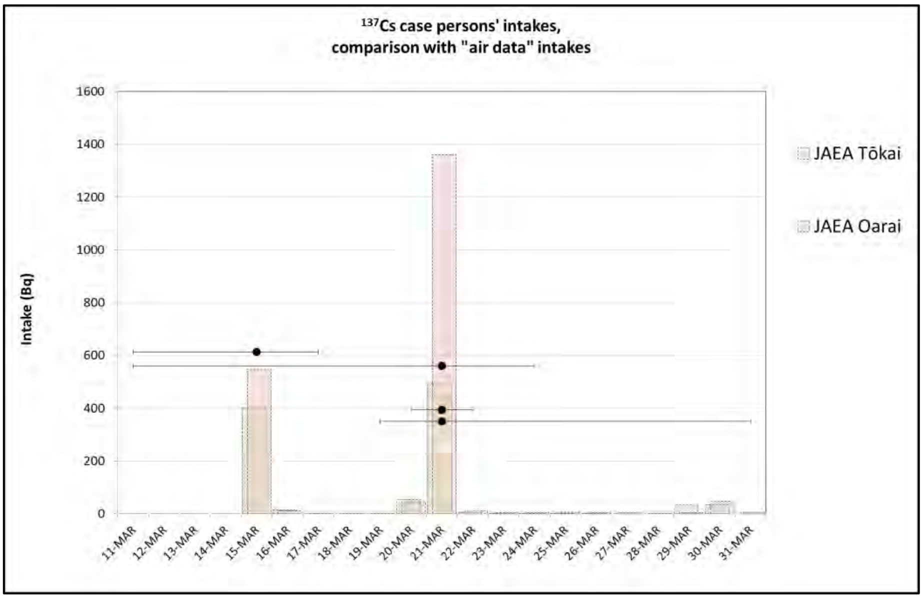


Figure 6: ^{137}Cs inhalation intake; comparison for Ibaraki prefecture.

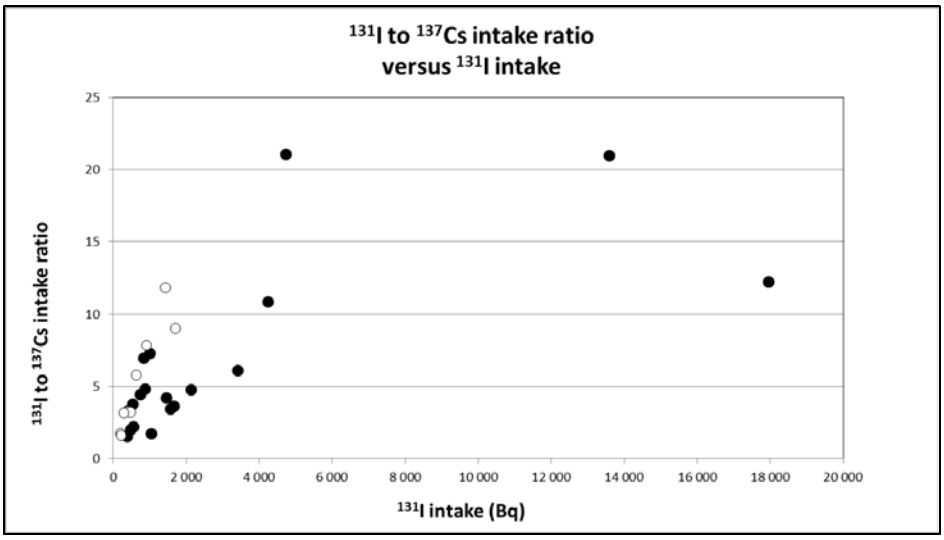


3.2 Intake ratios

3.2.1 ^{131}I to ^{137}Cs intake ratio

For 37 cases in the Survey, positive retention results for both ^{131}I and ^{137}Cs are available. Those cases are therefore eligible for the intake ratio calculation. Time of intake and the ratio of particulate and gaseous forms of ^{131}I were set as described above (sections 2.2 and 3.1).
^{131}I to ^{137}Cs intake ratios were found ranging from 0.2 to 24.8. Certain low values were associated with radioiodine retentions on the DL level of the respective laboratory and were excluded from the data set. Certain values were also excluded the intake of which could not be estimated for a single intake time.
Median of the intake ratio values for the adjusted data set was found to be equal 4.7 (GSD = 2.1). Higher ratio values are associated with higher ^{131}I intakes. No trend can be observed for the ratio dependency on the ^{137}Cs intake.

Figure 7: ^{131}I to ^{137}Cs intake ratio vs. ^{131}I intake (White circles = ^{137}Cs measurement was on the level of detection limit).



Kim *et al*. [19] discuss the ratio values found by some Japanese authors. It should be noted to this issue that not always "intake ratio" is reported by the Japanese authors and comparisons should be made carefully. The retention ratio can be confused sometimes with the intake ratio.
^{131}I to ^{137}Cs intake ratio found in the EURADOS Survey for mostly Tokyo residents is somewhat higher than the findings by Kim *et al*. for Fukushima residents and is broadly similar to the air activity ratios.
Two persons who travelled together and had their inhalation intake in the period of 20-22 March in Ibaraki prefecture only show the intake ratio 4.2 and 6.1 respectively.

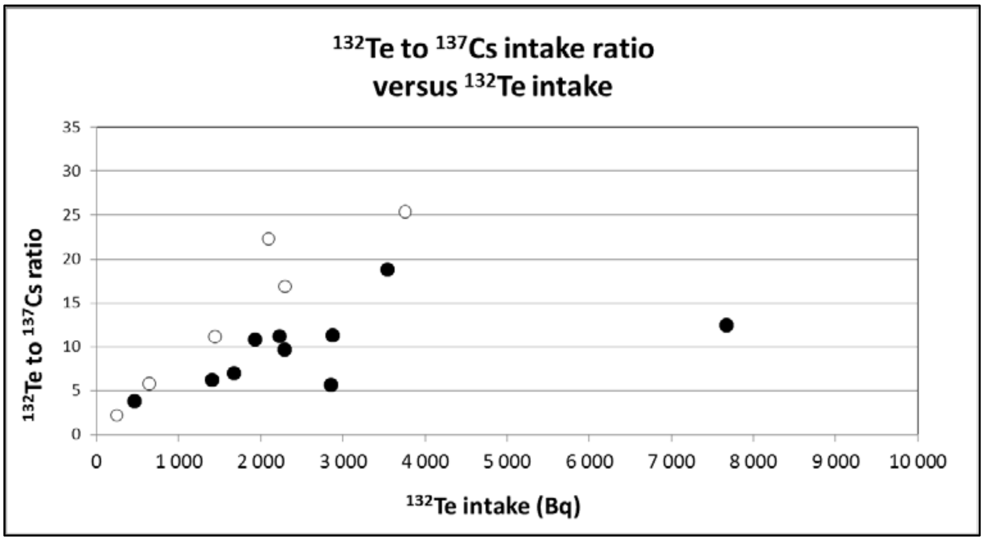
3.2.2 ^{134}Cs to ^{137}Cs whole body retention ratio

Retention ratios of ^{137}Cs to ^{134}Cs for 34 case persons were plotted against the day of measurement and no statistically significant difference from the expected early value 1:1 was found for the retention ratio.

3.2.3 ^{132}I to ^{132}Te retention ratio and ^{133}I

The last positive ^{132}Te counting in humans is reported in the EURADOS Survey on 30 March for a case person who stayed in Fukushima and Tokyo. However, the residence time and place were not specified precisely.
For 23 case persons, positive retention values in the whole body are recorded for the "radionuclide couple" ^{132}Te and ^{132}I. Activity ratios of ^{132}I to ^{132}Te found for the whole body measurements are ranging between 0.4 and 1.4. The ratio of 1 is expected between ^{132}I and ^{132}Te activity in the whole body. The differences can be accounted to the effect of the measurement geometry and the distribution of radiotellurium and -iodine within the body (tellurium in the lungs and iodine also in thyroid).
No specific measurement was made for the ^{132}I activity in thyroid and ^{132}Te in the lungs. The activities ratio 0.2 ± 0.03 found by Balonov *et al* [24] for Chernobyl evacuees could not be therefore reviewed for the case persons in the EURADOS Survey.
The ratio found by Balonov *et al.* remains the only guidance for the estimation of the ^{132}I thyroid dose due to the intake of ^{132}Te.
For the purpose of this estimation, ratios of ^{132}Te and ^{137}Cs are plotted in the Figure 8. The intake ratios broadly correspond with the air activity ratios found for 15 March at sampling sites in Tokyo area (measured in Chiba and Wakō and plotted *e.g.* by Katata [25]).

Figure 8: ^{132}Te to ^{137}Cs intake ratio (White circles = ^{137}Cs measurement was on the level of detection limit).



No in vivo measurement of ^{133}I was reported to the EURADOS Survey.

4 CONCLUSION

Case study approach was applied to the available measurement results of foreigners who left Japan in March 2011 shortly after the Fukushima Daiichi NPP accident and underwent thyroid and whole body counting in their respective countries. Case persons' inhalation intakes calculated for ^{131}I, ^{137}Cs, and ^{132}Te generally do not exceed the intakes estimated based on radionuclide air concentrations reported for the persons' time and area of residence in Japan. Intake ratio of ^{131}I and of ^{132}Te to ^{137}Cs is calculated for case persons whose in vivo measurements provided positive and reliable data, ranging from 0.2 to 24.8. ^{131}I to ^{137}Cs intake ratio found in the EURADOS Survey for Tokyo residents is higher than the findings by Kim et al. for Fukushima residents and is broadly similar to the air activity ratios. The ratios are the guidance to assess the thyroid dose from short lived radioiodines when thyroid measurements of ^{132}I and ^{133}I are not reported.

5 ACKNOWLEDGEMENTS

Ch. Li, Health Canada; I. Malátová, SÚRO, Czech Republic; S. Holm. Righhospitalet, Denmark; J. Huikari and M. Muikku, STUK, Finland; V. Kamenopoulou and K. Potiriadis, GAEC, Greece; I. Balashazy and P. Zagyvai, MTA EK, Hungary; S. Busone, Florence Hosp., Italy; S. De Crescenzo, Hosp Niguarda, Italy; F. Rossi, Florence Hosp., Italy; B. Lind, NRPA; Norway; R. Kierepko and J.W. Mietelski, IFJ PAN, Poland; T. Pliszczynski, NCBJ, Poland; J.F. Navarro, T. Navarro, and B. Perez, CIEMAT, Spain; H. Pettersson, Linköping University Hospital, Sweden, G. Etherington and J.E. Scott, PHE, UK; V. Vasylenko, NRCMR, Ukraine.

6 REFERENCES

[1] Tanaka, S.: Accident at the Fukushima Dai-ichi Nuclear Power Stations of TEPCO —Outline & lessons learned—. Proc. Jpn. Acad., Ser. B 88 (2012).

[2] Lopez, M. A. *et al*.: EURADOS network on internal dosimetry. *In*: Proceedings of the 13th International Congress of the International Radiation Protection Association, Glasgow, Scotland, 13–18 May (2012).

[3] Lopez, M.A., *et al*.: Lessons Learned from the EURADOS Survey on Individual Monitoring Data and Internal Dose Assessments of Foreigners Exposed in Japan Following the Fukushima Daiichi NPP Accident. Radiat. Prot. Dosim. (DOI:10.1093/rpd/ncv510).

[4] UNSCEAR 2013 Report, Attachement C-12, Methodology for the assessment of dose from external exposure and inhalation of radioactive material. UN 2014

[5] Tokonami, S. *et al*.: Thyroid doses for evacuees from the Fukushima nuclear accident. Sci. Rep. 2, 507; DOI:10.1038/srep00507 (2012).

[6] Kim, E. *et al*.: Estimation of Early Internal Doses to Fukushima Residents after the Nuclear Disaster Based on the Atmospheric Dispersion Simulation. Radiat. Prot. Dosim. (DOI:10.1093/rpd/ncv385).

[7] WHO 2012: Preliminary dose estimation from the nuclear accident after the 2011 Great East Japan Earthquake and Tsunami. World Health Organization 2012

[8] Priest, N.D.: Radiation doses received by adult Japanese populations living outside Fukushima Prefecture during March 2011, following the Fukushima 1 nuclear power plant failures. J. Environ. Radioactivity 114 (2012) 162-170

[9] Kurihara, O.: NIRS's Project for the Reconstruction of Early Internal Dose to Inhabitants in Fukushima After the Nuclear Disaster. *In*: Radiation Monitoring and Dose Estimation of the Fukushima Nuclear Accident. (Sentaro Takahashi Ed.). Springer Open (2014). (DOI 10.1007/978-4-431-54583-5)

[10] Shinkarev,S.M. *et al*.: Estimation of the Contribution of Short-Lived Radioiodines to the Thyroid Dose For the Public in Case of Inhalation Intake Following the Fukushima Accident. Radiat. Prot. Dosim. 164(1–2) 2015, 51–56.

[11] Ohkura, T. *et al*.: Emergency Monitoring of Environmental Radiation and Atmospheric Radionuclides at Nuclear Science Research Institute, JAEA Following the Accident of Fukushima Daiichi Nuclear Power Plant. JAEA-Data/Code 2012-010.

[12] Furuta, S. *et al*.: Results of the Environmental Radiation Monitoring Following the Accident at the Fukushima Daiichi Nuclear Power Plant -Interim Report (Ambient Radiation Dose Rate, Radioactivity Concentration in the Air and Radioactivity Concentration in the Fallout), JAEA-Review 2011-035

[13] Amano, H., *et al*.: Radiation measurements in the Chiba Metropolitan Area and radiological aspects of fallout from the Fukushima Dai-ichi Nuclear Power Plants accident. J. Environ Radioactivity 111 (2012) 42-52.

[14] Doi, T. *et al*.: Anthropogenic radionuclides in the atmosphere observed at Tsukuba: characteristics of the radionuclides derived from Fukushima. J. Environ. Radioactivity 122 (2013) 55-62.

[15] Haba, H. *et al*.: One-year monitoring of airborne radionuclides in Wako, Japan, after the Fukushima Dai-ichi nuclear power plant accident in 2011. Geochemical Journal, 46 (4), 2012, 271-278, (including Supplement Material)

[16] Operation Tomodachi Registry. https://registry.csd.disa.mil/registryWeb/docs/registry/optom/DTRA-TR-13-044.pdf. Last access 30.4.2016

[17] Tokyo Metropolitan Institute of Public Health. Measurement results of radiation and radioactivity levels in Tokyo. http://monitoring.tokyo-eiken.go.jp/en/index.html. Last access 30.4.2016

[18] Tsuruta H., Takigawa, M., Nakajima, T.: Sumary of Atmospheric Measurements and Transport Pathways of Radioactive Materials Released by the Fukushima Daiichi Nuclear Power Plant Accident. *In*: Proceedings of 1st NIRS Symposium on Reconstruction of Early Internal Dose in the TEPCO Fukushima Daiichi Nuclear Power Station Accident. (Ed: Kurihara O. *et al*.) NIRS-M-252, Chiba, Japan 2012

[19] Kim, E. *et al*.: Intake Ratio of ^{131}I to ^{137}Cs Derived from Thyroid and Whole-Body Doses to Fukushima Residents. Radiat. Prot. Dosim. 168 (3) (2015), 408–418.

[20] IMBA® Professional Plus: Integrated Modules for Bioassay Analysis (www.phe-protectionservices.org.uk/imba).

[21] Richtlinie für die physikalische Strahlenschutzkontrolle zur Ermittlung der Körperdosen. Anhang 7: Retentions- und Ausscheidungsdaten, Dosiskoeffizienten und SEE-Werte. Bundesamt für Strahlenschutz, Germany.

[22] Tsuruta, H. *et al*.: First retrieval of hourly atmospheric radionuclides just after the Fukushima accident by analyzing filter-tapes of operational air pollution monitoring stations. Sci. Rep. 4, 6717; DOI:10.1038/srep06717 (2014).

[23] ICRP, 1995. Age-dependent Doses to Members of the Public from Intake of Radionuclides - Part 4 Inhalation Dose Coefficients. ICRP Publication 71. Ann. ICRP 25 (3-4).

[24] Balonov, M. *et al*.: Contributions of Short-Lived Radioiodines to Thyroid Doses Received by Evacuees from the Chernobyl Area Estimated Using Early In Vivo Activity Measurements. Radiat. Prot. Dosim. 105 (1–4) 2003, 593–599.

[25] Katata G., *et al*.: Detailed source term estimation of the atmospheric release for the Fukushima Daiichi Nuclear Power Station accident by coupling simulations of an atmospheric dispersion model with an improved deposition scheme and oceanic dispersion model. Atmos. Chem. Phys., 15(2015), 1029–1070.

Large scale monitoring of radioiodine in thyroid: equipment and preparedness in the Czech Republic

P. Fojtík[a*], S. Kavan[b], L. Novotná[b], T. Svobodová[a], E. Čermáková[a], J. Baloun[a], J. Helebrant[a]

[a]SÚRO, Bartoškova 28, 140 00 Praha 4, Czech Republic.
[b]HZS Jč kraje, Pražská 52b, 370 04 České Budějovice, Czech Republic.

Abstract. In the case of an accidental release of 131I, a system for large-scale monitoring of the population for the radionuclide intake is needed, which has been recognized by various national and international initiatives. In the Czech Republic, the monitoring is ensured by SÚRO (National Radiation Protection Institute) in the frame of the Radiation Monitoring Network. Arrangements for the involvement of third parties, namely nuclear medicine departments, are also made. For a proper assessment of the thyroid dose in population, a monitoring system is required to be capable of measuring adult as well as child subjects across a wide range of ages. Such system has been developed by the Institute and the company Envinet, a.s. NUVIA Group. Six gamma-ray spectrometry scintillation units equipped with collimators are connected to at least two subject registration sites. Barcode and RFID readers built in the measurement units facilitate matching the subject and measurement result in such multi-unit detection system. In order to achieve up to thousand subjects to be counted within one day, two- or tree-minute measurement time is sufficient for the detection of 100 Bq of ^{131}I in thyroid in non-elevated ambient background. In the paper, general considerations, monitoring system and its performance are presented, together with a scenario of an exercise for the evaluation of measurement capacity.

KEYWORDS: thyroid measurement; radioiodine; preparedness.

1 INTRODUCTION

In the case of an accidental release of ^{131}I, a system for large-scale monitoring of the population for the radionuclide intake is needed, which has been recognized by various national and international initiatives. In the Czech Republic, the monitoring is ensured by SÚRO (National Radiation Protection Institute) in the frame of the Radiation Monitoring Network. Arrangements for the involvement of third parties, namely nuclear medicine departments, are also made. For a proper assessment of the thyroid dose in population, a monitoring system is required to be capable of measuring adult as well as child subjects across a wide range of ages. Training and capability testing are essential components of the preparedness for radiological emergency [1]. Collaboration between specialized radiological emergency response units and other emergency services within the frame of the Integrated Rescue System can provides a platform for training activities and education of the public for emergency situations as well. In the Czech Republic, the Fire Rescue Service coordinates the rescue works in emergency situations. SÚRO was invited by the Fire Rescue Service of South Bohemia to take advantage of their activities developed for fire brigade training as well as for education of the public.

2 MATERIALS AND METHODS

2.1 Large scale monitoring of radioiodine in thyroid

2.1.1 Goals of the monitoring

The monitoring is governed by the following goals:

- Estimation/determination of thyroid dose for individuals;
- Identification individuals for medical follow-up;
- Alleviation of people's health concerns;
- Planning medical services;
- Verification of and making corrections to dispersion model predictions;

* Presenting author, e-mail: pavel.fojtik@suro.cz

- Assessment of the efficiency of protective measures taken;
- Making prognosis of health consequences.

2.1.2 Monitoring strategy

Limited time is available for the thyroid monitoring in the case of an accidental release of ^{131}I.
Within a maximum of four weeks after the passage of the polluted air masses, the monitoring should be carried out of people in the affected areas. Later measurements do not provide reliable results due to the radioiodine decay. Also limited number of trained radiation protection specialists can be expected to be allocated for human monitoring tasks. The involvement of hospitals performing treatment of thyroid disease with radioiodine requires protocol to be established where all measurement details and hospital's instrument settings should be recorded for later evaluation of the measurement.

2.2 Equipment

The system intended in SÚRO for the large scale monitoring of radioiodine in thyroid consists in a transportable set of six collimated scintillation gamma-ray spectrometers linked as a portable instruments with personal computer-based software. The hardware and software ensure

- registration of people for measurement;
- measurement semi-control;
- gamma-ray spectra evaluation; and
- result recording and further processing.

Three trained persons are required for operation the system during the monitoring process. With two minute counting time the detection level of 100 Bq can be expected for ^{131}I in non-elevated ambient background.
The thyroid monitoring system is shown in the Figure 1.

Figure 1: Thyroid monitoring system in SÚRO

2.3 Field exercise

Field exercise was conducted to train the deployment of the system and to test its capacity in terms of number of persons measured per hour. Non-professional figurants, mainly students, played the role of monitored persons. To avoid possible ethical issues, the desired presence of child figurants was ensured by the SÚRO and Fire Rescue Service employees, who came with their children to the exercise site, see the Figure 2. An adverse situation was simulated by an intentionally unannounced power cut-off in the course of measurement.

Figure 2: Child measurement during the field exercise with the monitoring technique

3 RESULTS AND DISCUSSION

The field exercise provided required information on

- real time of the deployment of the system for measurement
- throughput in number persons that can be measured per time unit
- reaction of the system to power cut-off
- effort demanded on the operators, etc.

The presence and behaviour of non-professional figurants has revealed variety unexpected issues and has extended the number of situations that should be considered and trained in advance. Under uncomplicated conditions, around 100 persons can be counted for thyroid activity per hour. The necessity to inform the measured person about the measurement result may require that another operator should be in place and communicate with measured persons about the results.

4 CONCLUSION

Large scale monitoring of the thyroid in case of an accidental release of ^{131}I should be based on a strategy developed on the goals to be achieved by the monitoring. Limited time available and the lack of trained human resources in any such adverse situation require an effective monitoring instrument to be utilized. Field exercise of the emergency response team with full deployment of the monitoring systems should follow after development of the monitoring instruments and operational procedures. Evaluation of the exercise is the basis for further modifications to strategy and development and upgrading the monitoring technique.

5 ACKNOWLEDGEMENTS

The work is funded by the project of the Ministry of the Interior of the Czech Republic VF20162016050. The authors would like to thank all the adult and child figurants for they cooperation in the exercise.

6 REFERENCES

[1] EPR-EXERCISE (2005). Preparation, Conduct and Evaluation of Exercises to Test Preparedness for a Nuclear or Radiological Emergency. IAEA, Vienna, 2005.

Investigation of the radiological performances of commercial Active Personal Dosemeters for the use of the PSI fire brigade radiation protection squad

R. Philipp, B. Hofstetter-Boillat, E. Hohmann, S. Mayer

Paul Scherrer Institute (PSI); 5232 Villigen PSI; Switzerland

Abstract. The radiation protection squad of the PSI fire brigade needed new Active Personal Dosemeter (APD). Based on special criteria for fire fighters, the radiation protection squad shortlisted two types of APD. The radiological performances of these two types of APD were tested in photon reference radiation fields available at the verification and calibration laboratory for radiation protection instruments of PSI. The comparison of dependence of energy, angle and dependence of doserate are discussed in the presented studies.

KEYWORDS: comparison; active personal dosimeters; fire fighters; radiation protection squad.

1 INTRODUCTION

The Paul Scherrer Institute (PSI) maintains a fire brigade with a radiation protection squad included. This special unit of the fire brigade is trained and equipped for operations in environments with increased levels of ionising radiation and emergency interventions. Essential part of the equipment is an APD raising an alarm in case of increased dose or dose-rate. The current APD used by the radiation protection squad of the fire brigade are not any longer compliant with requirements defined by Swiss national standard SN EN 61526 and actual needs [1] - in addition they are old and worn out. As response to these needs, the radiological properties of preselected commercially available APD fulfilling special criteria for the fire brigade use were investigated in reference radiation fields available at the verification and calibration lab for radiation protection instruments of PSI.

2 REQUIREMENTS

Non radiological requirements for the preselection process of the fire brigade have mainly been audibility of the acoustic alarm with wearing protection gear, a case easy to decontaminate and a shock and dust proof design. According to these criteria, two APD, namely the Mirion DMC 3000 and the Tracerco PED meeting the constraints were further investigated (Figure 1) [2, 3].

Figure 1: Selected APD for further investigation: Mirion DMC 3000 (left) and Tracerco PED (right).

The requirements for the radiological performance of the APD a are defined by national standards and their correct indication of high dose and dose-rate - Swiss radiation protection ordinance accepts up to 250 mSv per person for lifesaving emergency operations [1, 4].

According to the manufacturer, the DMC 3000 is suited to measure photons with energies from 15 keV up to 7 MeV, dose-rates from 10 µSv/h up to 10 Sv/h and doses starting from 1 µSv up to 10 Sv [2].

The PED is designed to be able to measure photons with energies from 33 keV up to 1.25 MeV, dose-rates from 0 µSv/h up to 0.1 Sv/h and doses from 0 µSv up to 10 Sv [3]. These specifications are summarized in Table 1.

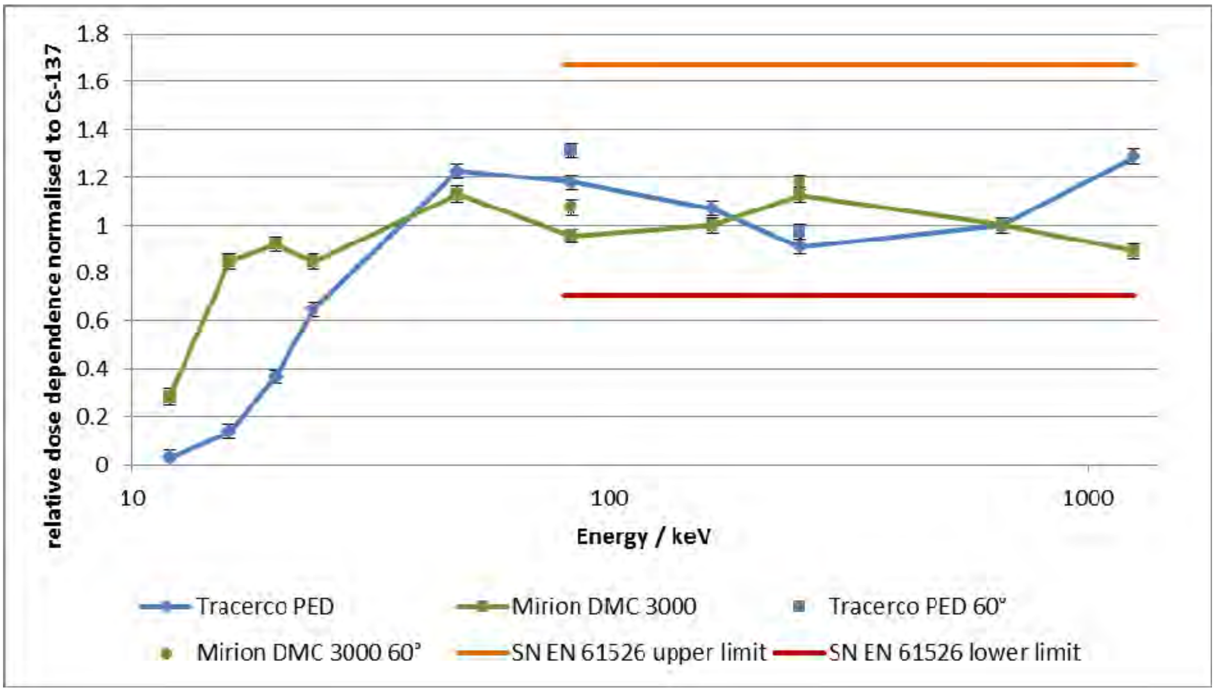
Table 1: Technical data of investigated APD according to the manufacturers' specifications.

Quantity	Dosimeter					
	Mirion DMC 3000			Tracerco PED		
Energy	15 keV	-	7 MeV	33 keV	-	1.25 MeV
Dose-rate	10 µSv/h	-	10 Sv/h	0 µSv/h	-	0.10 Sv/h
Dose	1 µSv	-	10 Sv	0 µSv	-	10.00 Sv

3 RESULTS

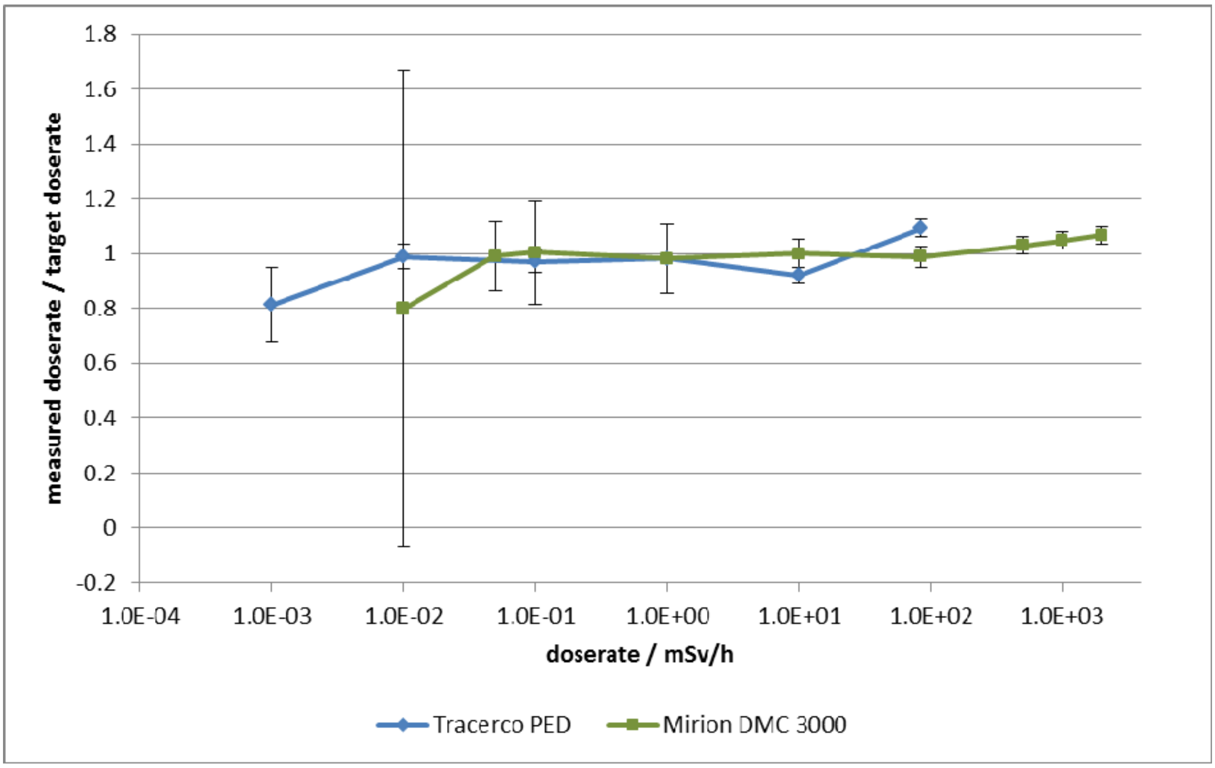
Figure 2 shows the energy and angular dependency resulting from dose measurements of both investigated APD. Both APD underestimate the dose for photons with energies less than 24 keV. However this effect is less pronounced by the DMC 3000. Both APD also comply with the limits of the Swiss national standard SN EN 61526 applicable for energies from 80 keV up to 1250 keV [1].

Figure 2: Measured energy and angular dependency resulting from dose measurements of both investigated APD arising from irradiation with ^{137}Cs, ^{60}Co and x-rays.



The PED was found to have an upper dose-rate limit of 89 mSv/h compared to 100 mSv/h as stated by the manufacturer (Table 1). The dose indicated of the DMC 3000 at lowest indicatable dose-rate of 10 µSv/h varies with the discrete values 0 µSv/h and 10 µSv/h resulting in an increased measurement uncertainty. The results are summarized in Figure 3.

Figure 3: Measured dose rate dependency of both investigated APD arising from irradiation with ^{137}Cs.



4 CONCLUSIONS

The radiological investigations described above showed, that both APD comply with the specifications stated by the manufacturer as summarized in Table 1 and requirements according to the Swiss national standard SN EN 61526 [1]. However, the performed tests were focused on properties according to requirements of the PSI fire brigade radiation protection squad and not for type testing.

5 ACKNOWLEDGEMENTS

The presented work was partly funded by the Swiss Nuclear Safety Inspectorate under contract No. 100979.

6 REFERENCES

[1] Electrosuisse, Radiation protection instrumentation - Measurement of personal dose equivalents Hp(10) and Hp(0,07) for X, gamma, neutron and beta radiations - Direct reading personal dose equivalent meters and monitors; SN 61526:2007 (2007).

[2] Mirion Technologies. DMC 3000 – Personal Electronic Dosemeter, 1151199EN-G. https://mirion.app.box.com/s/4zz8p1chhd716qd2cvck

[3] Tracerco. The Intrinsically Safe TRACERCO™ Personal Electronic Dosimeter (PED) T404. http://www.tracerco.com/pdfs-medialibrary/Tracerco%20T404%20PED.pdf

[4] The Swiss Federal Council, 1994 (Status as of 2014). Radiological Protection Ordinance. https://www.admin.ch/opc/en/classified-compilation/19940157/index.html

The Analytical Platform of the PREPARE project

W. Raskob, S. Möhrle, S. Bai, T. Müller

Karlsruher Institut für Technologie (KIT), Eggenstein-Leopoldshafen, Hermann-von-Helmholtz-Platz 1, Germany, wolfgang.raskob@kit.edu, Tel.: +4972160822480.

Abstract. The European project PREPARE (Innovative integrated tools and platforms for radiological emergency preparedness and post-accident response in Europe) aims at closing gaps that have been identified in nuclear and radiological preparedness following the first evaluation of the Fukushima disaster. Among others, a work package was established to develop a so-called Analytical Platform exploring the scientific and operational means to improve information collection, information exchange and the evaluation of such types of disasters. The Analytical Platform contains several modules supporting the work of experts in analysing an ongoing event and in communicating with the public.

KEYWORDS: Analytical platform, decision support, event analysis, communication, PREPARE.

1 INTRODUCTION

The PREPARE project (http://www.prepare-eu.org/index.php) has been created to close gaps that have been identified in nuclear and radiological preparedness following the first evaluation of the Fukushima disaster. As the management of the Fukushima event in Europe was far from optimal, a so called Analytical Platform has been developed exploring the scientific and operational means to improve information collection, information exchange and the evaluation of such types of disasters.

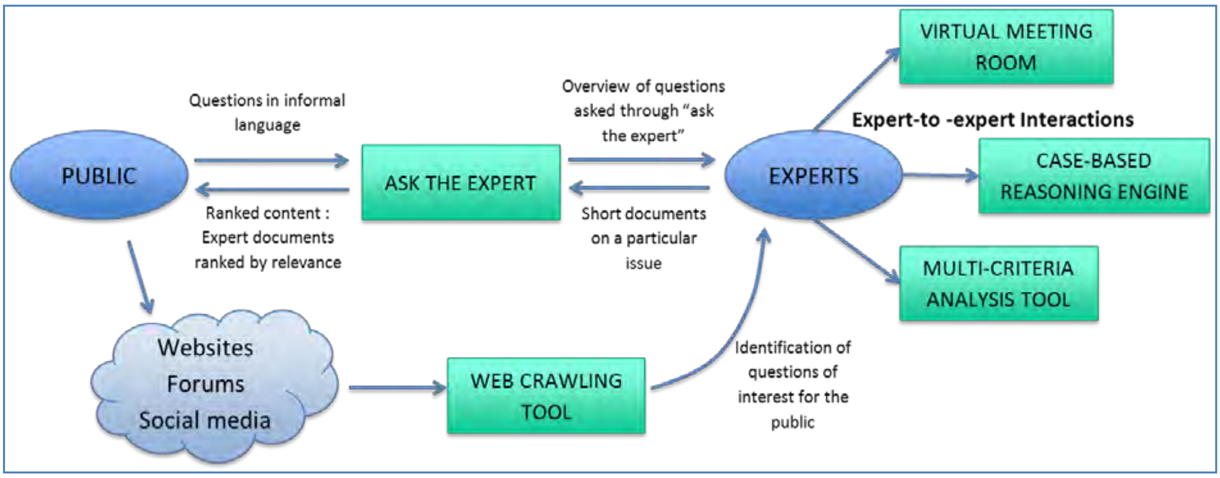
This paper shortly summarises the key functions of the Analytical Platform.

2 FUNCTIONS OF THE ANALYTICAL PLATFORM (AP)

2.1 Overview

The general idea of the AP is to provide an easy to access platform for information exchange in times of a nuclear or radiological crisis, allowing discussions between experts and to disseminate information to the public community. If an alert is issued based on a protocol to be developed as part of the operational procedures, members of different organizations would be attached to the platform and start working. The following flowchart represents the key components and the work flow of the AP. In the following chapters, key components of the AP will be described.

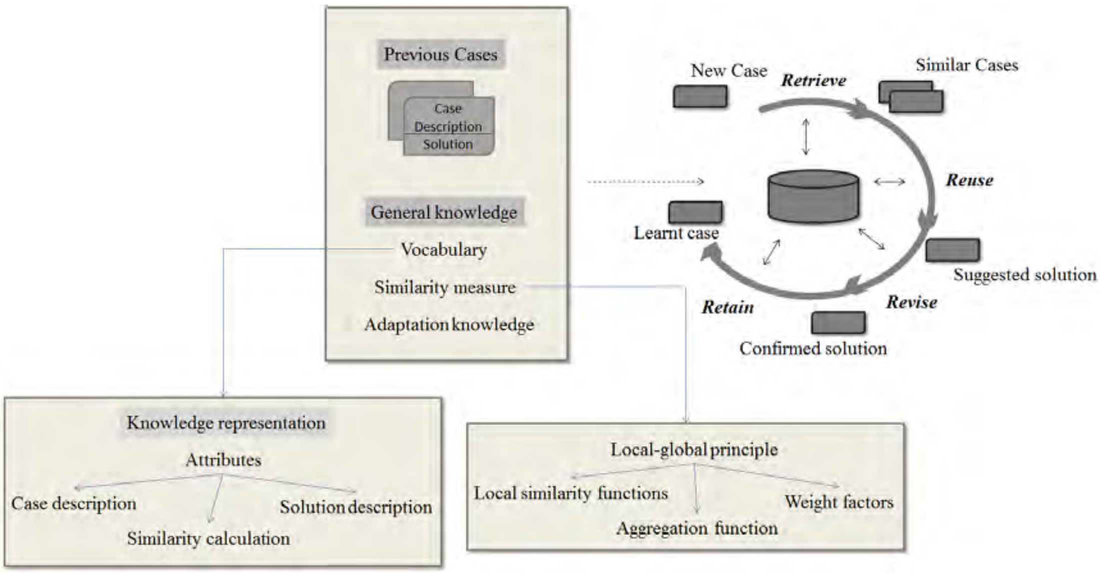
Figure 1: Analytical Platform with its components and work flow.



2.2 Knowledge database and case based reasoning

The core of the AP is the knowledge database with historic events and scenarios and its Case Based Reasoning (CBR) algorithm to compare them with the on-going event. The CBR methodology is a process to identify that scenario which fits best to the one under investigation. To facilitate the retrieval mechanism, appropriate similarity functions had to be developed to compare the on-going event and the one from the knowledge data base (Figure 3). Uncertainty handling is important for any complex natural or man-made disaster. Therefore also a concept for dealing with uncertain information was defined.

Figure 2: CBR cycle and knowledge included in the system



16 historic cases from Windscale fire, Chernobyl and Fukushima accidents and the poisoning of Alexander Litvinenko have been integrated into the data base. These cases are shown in Figure 3. As the number of historic cases was small, additional 96 scenarios have been added to the database. In addition, the new HERCA-WENRA scheme was adopted for the pre-release phase. It proposes a methodology for a common European approach allowing to recommend urgent protective actions as well as a minimum common level of preparation for these actions:

- evacuation should be prepared up to 5 km around nuclear power plants, and sheltering and Iodine Thyroid Blocking (ITB) up to 20 km
- a general strategy should be defined in order to be able to extend evacuation up to 20 km and sheltering and ITB up to 100 km

The parameters affecting the result are the following and are not used in the AP:

- risk of a core melt;
- integrity of the containment;
- wind direction known;
- evacuation can be performed in time or not.

For the release phase, 4 key parameters were identified to define 96 scenarios:

- INES scale – sourceterm (5, 6, 7);
- Season of the year (spring, summer, autumn, winter);
- Weather (with rain, dry, low wind speed, medium wind speed);
- Population distribution (urban, rural).

Figure 3: Number of historic cases implemented into the knowledge data base

Phase / Event	Chernobyl nuclear power plant accident	Fukushima Daiichi nuclear power plant accident	Windscale fire	Poisoning of Alexander Litvinenko
Release	3	1	1	1
Transition	1	-	1	1
Long-term post-accident	4	1	1	1

Affected areas

- Pripyat
- Narodichskiy district of Zhitomirskaya oblast
- Bryansk region
- Novo Bobovichi
- United Kingdom

Milk ban area

Population who were given urine tests (includes both UK and overseas testing)

In the longer term, only one weather condition and the population distribution were taken into account for the scenario construction. The values were normalised to a certain contamination level (100 mSv in the first year) and the area of one square kilometre. This allows for scaling to other conditions. Having used the scenarios as defined for the early phase, the dose distribution would be too inhomogeneous to define measures in a proper way. So far 8 scenarios have been implemented – more will be defined within the European project HARMONE.

2.3 "Ask the Expert" service and the "web crawling" facilities

Further important tools of the AP are the "*Ask the Expert*" service and the "web crawling" facilities. Both tools are based on the *NOMAD Web Crawling Platform* developed by NCSR-D in the context of FP7-ICT NOMAD.[1]. The following components were developed:

- The *NOMAD Web Crawling Platform* provides the underlying crawling functionality. The NOMAD Platform persists crawling results in a MySQL database. The Nomad Platform is a Tomcat Web Service controlled via a REST API;
- The PREPARE *Web Content Discovery Service* acquires and sifts through public on-line forums in order to categorize content according to its relevance to several sub-categories of the overall nuclear and radiological events theme. This service executes on a Liferay portal and provides a user interface for controlling the Crawling Platform as well as the functionality for using Cron to periodically repeat crawls along with the functionality for extracting and displaying statistics about the terms that occur in the crawled documents;
- The *PREPARE Ask the Expert Service* provides functionality for managing and retrieving documents, including functionality for recommending documents based on their relevance to user query terms.

The Bing™ search engine, the Google+™ social network and the Twitter™ social network can be scanned. The user can define particular search queries and can run them for a short or longer term. In particular for longer term queries, a statistics will be provided that shows the importance of the expression in the longer term development. This allows to react on an emerging topic which is at present not realised as relevant.

[1] For more details, please cf. http://www.nomad-project.eu

The *Ask the Expert Service* comprises a document management. The user can upload documents, annotate them with terms, or update documents with new versions. The documents in the collection can be searched using a full-text index of all the terms appearing in them or for the user-provided annotation terms. This panel is based on the standard Liferay Documents and Media Library as part of the AP.

2.4 Situation awareness

2.4.1 Incident Manager

The incident manager provides means to organize an incident as a timeline clearly displaying the sequence of events, to collect measurements in a list, to calculate a simple dispersion model, and to display this information on a map. These tools are accessible by tabbed panes.

2.4.2 Time line

The timeline provides an overview of events during an incident that are displayed as points in time or intervals with start and end point. These events are displayed on a floating time line. Events may contain additional meta information like geographic coordinates or internet addresses. A set of events can be grouped. As some events may lie in the future or are fictional, each event can be marked as to be actual, real or not. The timeline requires manual interactions from the expert respectively admin users to be up to date.

2.4.3 Measurements

The measurements tab provides a series of input fields at the top, followed by a button bar and a table view in the center. Each line in the table represents a measurement. A measurement features a label, a coordinate determined by latitude and longitude, a measured value, and a checkmark whether the measured value was confirmed or not. Measurements can be selected respectively unselected by clicking onto the according line in the table. The values can then be edited using the input fields. Changes are stored the moment the field loses focus. Invalid changes are ignored. To add a new measurement you have to first add it to the list, then select it afterwards and change the default values. The buttons provide means to add and remove measurements, to delete, import, or export the whole list as either XML or CSV, and to focus the map on a selected measurement respectively on all measurements.

2.4.4 Dispersion

The dispersion tab provides means to create simple dispersion calculations and overlay them on the map. It displays a list of active calculations on the right and on the left an overview of the parameter fields required to define and compute a simple dispersion. Bear in mind that the simple model is only meant to quickly provide an estimate which areas are likely to be threatened by the release. The dispersion model is based on a Gaussian point release of a continuous stream of material.

2.5 Virtual meeting room

The virtual meeting room (VMR) provides a communication and messaging environment for experts. It allows to exchange documents and messages for a well-defined group of experts brought together to solve a particular issue. The communication is not accessible by others. It is based on the M/KSID system and runs as a separate application embedded in the portal as a web page. Accessing the page will try to automatically log into the subsystem using the current users email as login name and “prepare” as password. If the user is not yet defined the default login mask will appear. For new users of the PREPARE portal the administrator can register them in M/KSID by their emails and mentioned password.

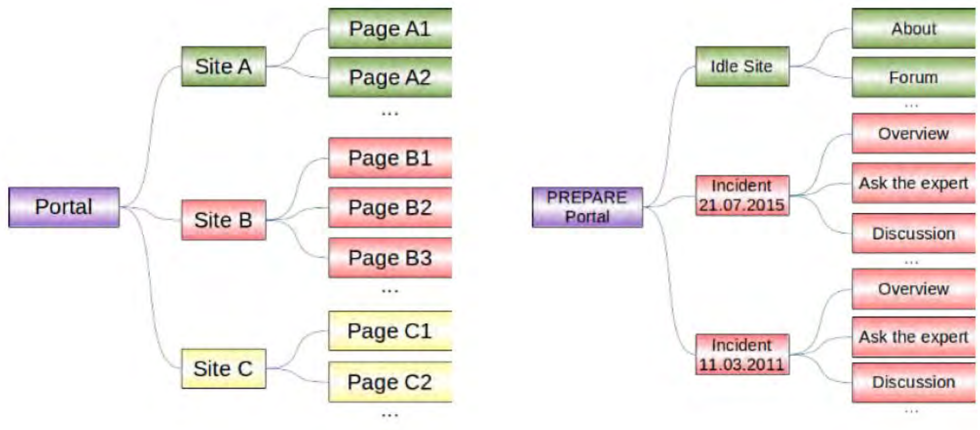
2.6 Decision Support

The decision support provides a simple means to analyze a situation, where an alternative has to be chosen from a set of alternatives based on several criteria. The actual alternatives and criteria have to be defined manually or by an external application and provided as XML. The actual values of the criteria can be adapted in the web interfaces. The decision support is intended to be prepared long term by experts defining the criteria and alternatives but applied in short term by adapting the actual values to the current situation.

2.7 Realisation

The AP was realized as a portal and based on Liferay as content management system. Liferay organizes the content in Sites, Site templates, Pages, and Page templates (Figure 4).

Figure 4: The left side shows the Liferay portal structure in general, containing different sites, each site providing different pages. On the right side the actual organization of sites and pages in the PREPARE Portal is shown.



3 TRAINING AND DISSEMINATION

The first draft of the AP has been tested in several countries and finally implemented for a training course in September 2015 in Trnava. The final version was demonstrated in Bratislava at the final dissemination meeting January 2016.

The training course was developed providing the necessary information on the Analytical Platform, the scientific methods and tools developed for collecting information, analysing any nuclear or radiological event and providing information about the consequences and its future development. A particular attention was given to the conditions and means for pertinent, reliable and trustworthy information to be made available to the public in due time and according to its needs in the course of nuclear emergency and post-emergency context. The targeted audience of the workshop were researchers, organisations and actors in emergency and recovery preparedness and management from all levels and sectors.

The theoretical and practical aspects of Analytical Platform management and use have been addressed through:

- Half a day exercise in a form of table-top exercise and facilitated workshops split in two parts: the first part focused on information gathering and distributing, whereas the second part was devoted to a nuclear event using the INEX 5 exercise scenario;
- Feedback from the exercise;
- Round table and discussion about the AP: use of scientific means, operational procedures, technical platform, information structuring and collection, virtual meeting room.

Each course day was closed with a round table and discussion on the topic(s) of day or tools presented/demonstrated and trained. The overall resonse was very positive. However, it was also pointed out, that further training sessions and demonstrations are necessary before the platform is widely used.

4 CONCLUSION

The AP is a unique system that supports experts from all areas in analysing an on-going nuclear or radiological emergency. It serves as a focal point for collecting information, providing means to interpret them and finally to disseminate them to the public.

At the end of the PREPARE project, the AP has been released in its first operational version. The next steps will be to explore their application. In particular the question of the users has to be discussed. So far, the AP will be managed by the NERIS platform. The NERIS platform represents more than 50 organisations from the operational community, research and NGOs, interested in emergency management and long term rehabilitation. A working group on information, participation and communication has been established aiming to explore the usage of the AP by institutional and non-institutional experts. Also the connection of official European systems such as EURDEP and ECURIE should be explored.

5 ACKNOWLEDGEMENTS

The research leading to these results has received funding from the European Atomic Energy Community Seventh Framework Programme FP7/2012-2013 under grant agreement 323287.

6 REFERENCES

[1] Liferay, http://www.liferay.com/
[2] NERIS-Platform, http://www.eu-neris.net/

Determination of Radionuclides Surface Concentration and Radiation Level In Fukushima Prefecture, Japan: November 2014

Godwin Ekong, Isa Sambo, Saiyadi Sulaiman

Nigerian Nuclear Regulatory Authority, Plot 564/565 Airport Road, PMB 559, CBD, Abuja. Nigeria

Abstract. The determination of radionuclides and radiation levels in some of the most affected area of Fukushima, Japan (Lat. 37046N Long. 140028E) after the 2011 Tsunami employing the use of Radiagem 2000/Rados (RDS – 31) dose rate meter for radiation survey measurement and hand held Hyper pure Germanium (HpGe) detector for in-situ Surface Concentration (SC) measurements was the aim of this study. Six (6) different locations were measured and analysed, Insitu surface concentration measurements prominently of ^{134}Cs at energies 569, 605 & 796 keV and ^{137}Cs at energy 662 keV were established with little radiation from NORM alongside with dose rate measurement at the same locations. The ability of the detector to measure a Minimum Detectable Activity (MDA) as low as (0.10±0.02) Bqm^{-2} from a background area of about 400 m^2 established 95% of confidence limit on detection before commencing the Insitu measurements. The ranges of Insitu SC measurements for ^{134}Cs and ^{137}Cs for the Locations were found to be (162.11±0.46 – 59,649.39±6.61) Bqm^{-2} and Dose Rate measurements for the same locations were found to be (0.87±0.03 - 22.84±1.26) µSvh^{-1}respectively. The presence of ^{134}Cs and ^{137}Cs in these locations of Fukushima Prefecture, Japan showed an evidence of the nuclear accident as a result of the Tsunami disaster. The radiation levels recorded during the monitoring exercise was above the value of 0.23 µSvh^{-1} which Japanese Government hopes to achieve at the end of it ongoing environmental remediation before re-habitation of it evacuated citizens.

1 INTRODUCTION

March 11, 2011 a 15 metre tsunami disable the power supply and cooling of three Fukushima Daiichi reactors as a result of foremost earthquake which led to largely melting of all the three cores within few days. This accident led to the discharged of several amounts of radioactive materials / fission products in the fuel to the environment causing the radiation level to rise above levels expected for human habitation that resulted in the evacuation of over 100,000 residents of the area within 20-30 km from the Nuclear Power Plant (NPP) with a criterion of 20 mSv/yr [1]. The radioactive materials / fission products released to the environment were majorly of Iodine – 131 (131I) with half-life of 8 days, Caesium – 134 (134Cs) with half-life of 2.5 years, Caesium – 137 (137Cs) with half-life of 30.07 years etc. which were easily carried from the plume resulting in land contamination [I]. Therefore out of these fission products, Caesium presently remains the major concern in the environment because of its yield and longer half- life.

Caesium is a metal that may be nonradioactive or radioactive, found in the alkali group of the periodic table. Caesium undergoes beta decay (β−) and is also a strong emitter of gamma radiation. . It can easily be taken up by the body without concentrating in any particular organ with a biological half-life of 70 days. Dust particles contaminated by Cs may become air-borne and inhalation of contaminated dust can result in an internal exposure. Radiation from the radionuclides damages the body's soft tissues upon ingestion or inhalation as it is distributed evenly in the body [1, 2]. Slightly higher concentrations of Cs are found in muscle, while slightly lower concentrations are found in bone and fat.

The most common radioactive form of Cs is ^{137}Cs which is more significant as an environmental contaminant than Caesium- 134 with its lower yield and shorter half-life. The primary source of ^{137}Cs in the biosphere is from atmospheric nuclear weapons testing from the 1940s to the 1960s where about 90% of ^{137}Cs was produced by atmospheric testing, 6% was created by the Chernobyl accident and 4% by nuclear fuel reprocessing facilities [3], which was dispersed and deposited all over the world and the accident at Chernobyl Although natural radionuclides result in a relatively small amount of radiation dose.

Several radiation monitoring programmes has been instituted by nuclear Experts all over the globe with the host country playing the leading role as well as radiation monitoring points/stations set up since March 11, 2011 till date to ensure that accurate radiation level is constantly reported before and up till the end of remediation action before Return of Evacuees. A monitory exercise was conducted on 17-21 November 2014 for determination of radionuclides activity / surface concentration and radiation level of some locations of Fukushima Prefecture, Japan.

2 SURVEY METHOD

The survey/sampling methods employed may be categorised into probability and non-probability sampling. However, Non probability was employed during this radiation monitoring exercise because the survey points were picked in such a way that it does not have equal chances of other points being selected.

This was majorly Judgemental - visual, historical site use or pre-identifies areas having potential of higher radioactivity concentrations. *According to IAEA-TECDOC-1415.*
"However, although judgmental samples are not necessarily representative samples, in the sense that they represent the presence and concentration levels of the target analytes of the whole site, they are still intended to be representative of that sub-portion of the site being investigated".[4]

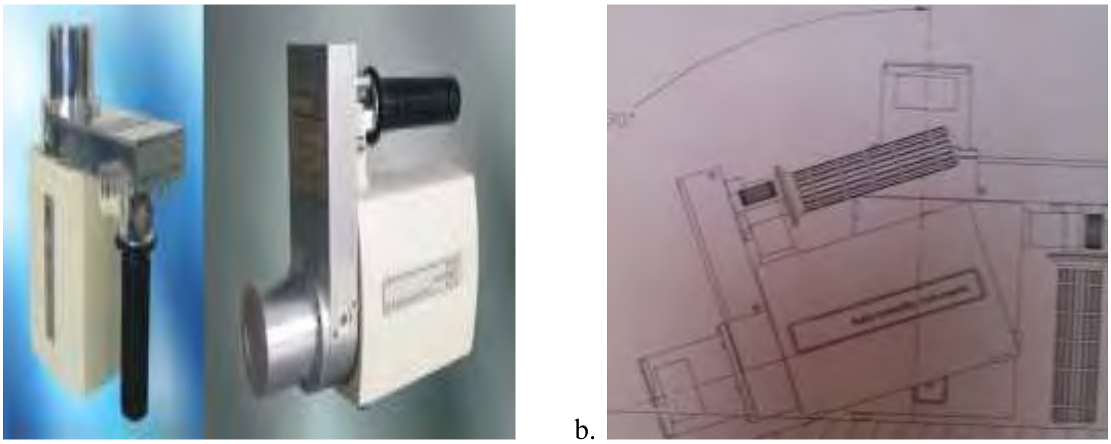
Six major locations were chosen where the monitoring exercise took place. Their results and discussions with respective coordinates are presented in Para 5 and 6 respectively.

3 EQUIPMENT SECIFICATION/ THEORY

Spectrometer – HpGe Handy GPD 25300

i. Detector properties, including:
- Type - HPGe p-type. All data analysis of was done using WINSPEC software
- Cooling mechanism and Dewar - LN2 . MCA Battery operation time – 20 hrs
- Resolution (FWHM) at normal operation at 122 KeV – 0.58 KeV

Figure 1: (a) Showing Picture of HpGe Handy GPD 25300 system setup and (b) Schematic with 20 degrees of orientation without tripod



a. b.

i. Calibration procedures and parameters were obtained using TECDOC 1092 and stepwise given below [5]:

- Certified reference point sources with a reasonably long half-life, having gamma line(s) in the medium energy range (58 keV line of Am-241, 122 keV line of Co-57, 662 keV line of Cs-137 and 1173 & 1332keV of Co-60 were considered as good choices).
- The angular correction factor (R_f/R_o) for germanium detectors with comparable diameter and length was taken to be close to 1.
- The geometrical factor (θ/As) was determined for the energy of interest from Figure D2.

- Response factor (R_0/Ø) was determined for this lines by following steps 3.1 to 3.5 in Procedure Dla uncollided flux (Ø) was determined from step 3 of D1a and measured corrected count rate (R_0) for 120 seconds from the counter, A plot of response factor (R_0/Ø) as a function of Energy (keV) gave the efficiency calibration which corresponds to that obtained in figure D3
- Parameters listed above [Geometrical factor (θ/As), Response factor (R_0/Ø) and Angular correction factor (R_f/R_o)] were therefore used to determined the Calibration Factor (C_f) given below:

$$\mathrm{Cf} = (\mathrm{R0}/\emptyset) * (\theta/\mathrm{As}) * (\mathrm{Rf}/\mathrm{Ro}) \quad ....(1)$$

A plot of C_f as a function of energy (E) determined the efficiency ($\mathcal{E}ff_{ab}$) of the counter as shown in Figure 2 below which is designated in Equation 2 as G.

Please note that the efficiency (*G)* of energy (*E)* which lies between two measured energies of E_1 and E_2 with efficiencies G_1 and G_2 could be then calculated where required using linear interpolation as given below [6].

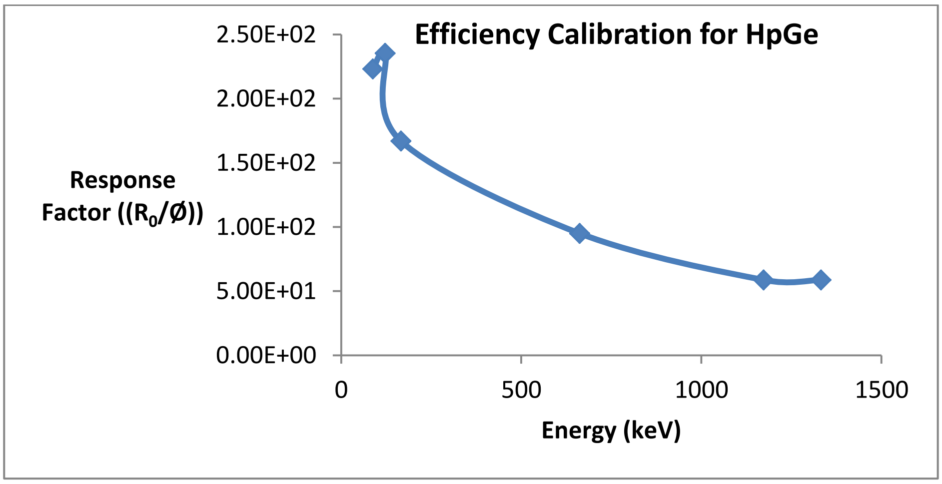
$$G = G_1 + (G_2 - G_1)\frac{(E-E_1)}{(E_2-E_1)} \quad ....(2)$$

It is relevant to note that when $E = E_1$ this correctly gives the efficiency G_1 and that when $E = E_2$ it correctly gives the efficiency G_2. This is more precise than simply averaging G_1 and G_2 because averaging only applies to mid-point of interval of E_1 and E_2.

- Calibration full energy efficiency:
    - Efficiencies and (preferably) uncertainties for energies

      Efficiency of 4.27±1.17 for Energy 88 keV, Efficiency of 4.15 ±0.64 for 122 keV, Efficiency of 3.20±0.53 for Energy 166 keV Efficiency of 1.82±0.22 for Enery 662 keV, Efficiency 1.12 ±0.10 of for Energy 1173 keV and Efficiency of 1.12 ±0.10 for 1332 keV

Figure 2: A plot of C_f as a function of energy (E) determined the efficiency of the counter.



The value of efficiency ($\mathcal{E}ff_{ab}$) rose steadily where the photoelectric effect is considered to be dominant, requiring a few millimetres depth of the detector window for photon interactions and This efficiency curve conformed to that obtained from other studies [7,8].

- Description of calibration efficiency geometry;
  Crystal shape of detector sensitive area is 500mm^2 while the sensitive region thickness – 14 mm and thickness of input window Al 0.5mm, therefore approximately 3cm distance between

the detector cap and source/sample were observed for all dimensions measurements. This was taken in consistence with same distance applied in plume survey in Procedure A1.

- Energy calibration:
  Channel 147 for Energy 88 keV, Channel 204 for 122 keV, Channel 276 for 166 keV, Channel 1337 for Energy 662 keV, Channel 1933 for Energy 1173 keV and Channel 2194 for 1332 keV as given in the system equation below:

$$E = 0.609\ keV/Ch * N\text{-}3.091\ keV + 0.000E + 0\ keV/Ch^2 * N^2 \ldots\ldots\ldots(3)$$

- FWHM versus energy:
  FWHM of 1.6 keV for 88 keV, FWHM of 1.32 keV for 122 keV, FWHM of 1.12 keV for 662 keV, FWHM of 1.30 keV for 1173 keV, FWHM of 1.50 keV for 1332 keV
- From equation 1, S_C can be calculated thus [5]:

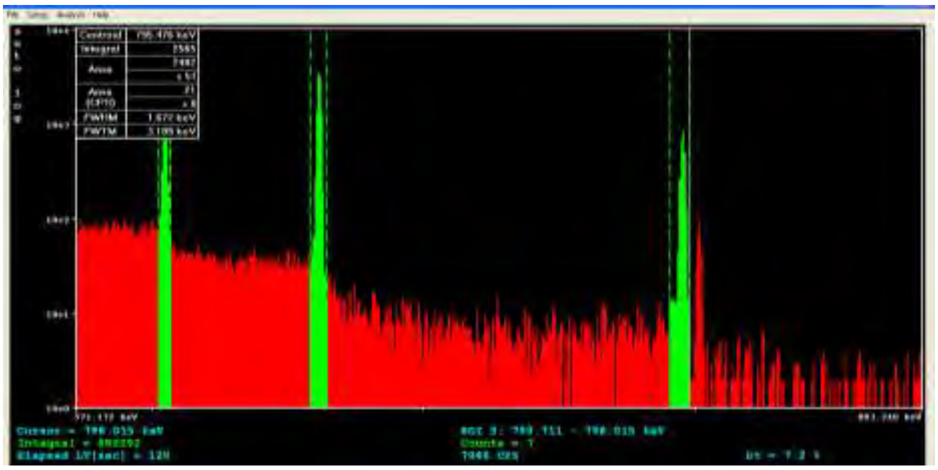
$$S_c = \frac{10\ X\ N_{count}}{C_f \times I\gamma \times t} \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots(4)$$

Where N is net count, C_f is the calibration factor earlier determined, $I\gamma$ is absolute gamma decay intensity of the specific energy peak [IX] (also known as the emission probability per transformation for a specific energy photo peak) and t is the time of counts. Its unit of activity concentration is Bqm^{-2}.

4 RESULTS

Data acquired from the spectrum during the Insitu measurement was done using WINSPEC software. The HpGe was held with maximum orientation of 0^0 - 20^0 (see Para. 4, Fig. 1b) and at a 3cm because of its thick window which could not measure at 1m from the ground. The time of spectra acquisition was kept at 120 seconds throughout the entire monitoring exercise considering the high level of radioactivity in the environment. The S_C was calculated using equation 4 above taking into account all the prominent multiple energy lines of each ^{134}Cs at 605 keV, 795 keV etc and ^{137}Cs at 662 keV as shown in Figure 3 below .

Figure 3: Acquired Spectrums indicating energy lines of Cs – 134 at 605 keV and 795keV, Cs – 137 at 662 keV obtained during the insitu monitoring



The results of the S_c evaluated from acquired spectrum for the Insitu and dose rate measurement are presented in Table 1 and 2.

5 DISCUSSION

The presence of ^{134}Cs and ^{137}Cs and high radiation dose rates in the six (6) locations of Fukushima, Japan showed the an evident of nuclear accident resulting from Tsunami disaster where these radionuclides were discharge to the environment (Figure 5). The level is higher than that of exemption of bulk amount and clearance of solid material without further consideration for ^{137}Cs is 0.1Bg/g which poses appreciable risk to the environment [9]. Figures 4 below affirmed that stated in Para 2 that the most common radioactive form of Cs is ^{137}Cs which is much more significant as an environmental contaminant than the fairly common ^{134}Cs and can also be evident in their percentage contribution where ^{137}Cs contributed 68.7% of environmental contamination as against 31.3% contribution of ^{134}Cs.

Figure 4: Showing Percentage Contr. Of Cs-134 & Cs-137 contamination of the environment

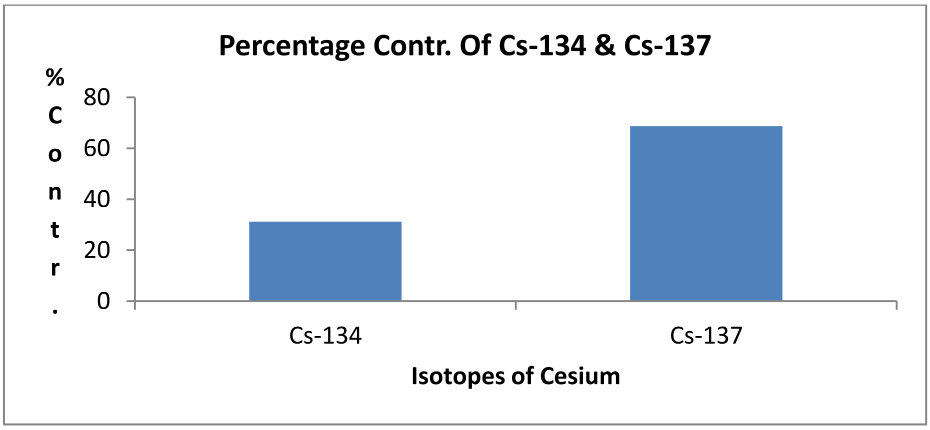
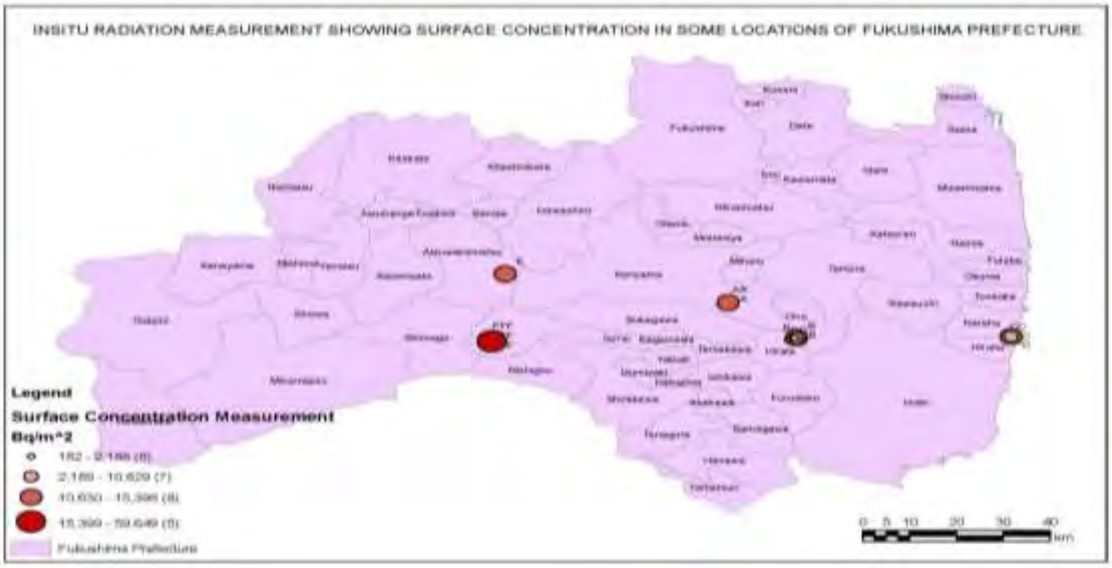


Figure 5 shows Dose Rate (μSvh^{-1}) measurement for six (6) monitored locations. The ranges of Dose Rate measured values were observed to be (0.87±0.03 - 22.84±1.26) μSvh^{-1} , lower Dose Rate values indicted areas where environmental remediation had commenced while higher Dose Rate values as shown in the legend indicated areas observed to be the direction of the plume and where environmental remediation are yet to commenced or the primary spot of the disaster where the NPP is locate.

Figure 5: Ambient Dose Rate (**µ**Svh^{-1}) measurement for six (6) locations

Figure 6 shows Insitu S_C (Bqm^{-2}) measurement for six (6) monitored locations. The ranges of measured Insitu S_C values were observed to be (162.11±0.46 – 59,649.39±6.61) Bqm^{-2}. As noted earlier in Dose Rate measurements, lower values indicted areas where environmental remediation had commenced while higher values as shown in the legend indicated areas observed to be the direction of the plume and where environmental remediation are yet to commenced or the primary spot of the disaster where the NPP is locate.

Figure 6: Insitu Surface Concentration (Bqm^{-2}) measurement for six (6) locations



The radiation levels recorded during the monitoring exercise exceeds recommended level as contained in ICRP publication 103 of 2007 where there is likelihood of resulting consequences of increased radiation exposures of both humans and other biota through various pathways if not properly monitored and controlled. It is therefore pertinent to note the evacuation in these areas of Fukushima has been effected and Japanese Government hopes to achieve at the end of it ongoing environmental remediation a dose rate value of $0.25\mu Svh^{-1}$ before resettlement of evacuated citizens.

6 RECOMMENDATION

It is hereby recommended that periodic environmental monitoring be maintained to ensure that recommended dose rate is achieved before re-habitation.

7 CONCLUSION

Environmental monitoring exercise was carried out in the six (6) locations of Fukushima, Japan to determine current dose rate level as well as in-situ Surface Concentration major radionuclide contaminant in the area. The result showed high level of radiation especially in major hotspot area with the presence of ^{134}Cs and ^{137}Cs appearing as major contaminants with minute traces of NORMs which were not of interest to the exercise. ^{137}Cs was seen to contribute majorly to the contaminant of the environment than it other isotope, ^{134}Cs. Therefore, intensive cleanup and periodic environmental monitoring exercises is required until the area become habitable again in future.

8 ACKNOWLEDGEMENT

Special gratitude goes to Incidence and Emergency Centre, International Atomic Energy Agency (IEC, IAEA) for putting up the Response and Assistance Network (RANET) workshop November 2014 and providing the funding to enable effective participation for the workshop.

Also, many thanks to the Japanese Government for permitting entry as well as playing a good host to the participant throughout the period of the exercise. Finally, the Nigerian Nuclear Regulatory Authority (NNRA) – our employer, granting approval and affecting our prompt release is deeply appreciated.

9 REFERENCES

[1] Fukushima Accident – World Nuclear Association – www.world-nuclear.org/.../ Fukushima Accident. Last accessed January 9, 2015

[2] National Council on Radiation Protection and Measurements (NCRP), *Cesium-137 in the environment: radioecology and approaches to assessment and management: recommendations of*

the National Council on Radiation Protection and Measurements, November 21, 2006, NCRP Report no.154, Bethesda, Md., 2007, ISBN: 9780929600918.
[3] US Environmental Protection Agency. (2012) '*Caesium*'. Available at: http://epa.gov/rpdweb00/radionuclides/cesium.html. Accessed: 22nd December 2014.
[4] IAEA-TECDOC-1415
[5] Generic procedures for monitoring in a nuclear or radiological emergency, IAEA - TECDOC – 1092
[6] Thorne etal 2012, Thorne, M., Balding D., Egan, R., Paulley, A. *LLWR Radiological Handbook; LLWR Environmental Safety Case Project*, 2012, LLWR/ESC/R(10)10033 Issue 1.4
[7] Knoll, G. F., *Radiation detection and measurement*, 4thed,New York: John Wiley & Sons, 2010
[8] Al-Sulaiti, H., *Determination of Natural Radioactivity Levels of the State of Qatar Using High-Resolution Gamma-ray Spectrometry*. MPhil to PhD Transfer Report, 2009 Department of Physics (University of Surrey).
[9] *Update of X Ray and Gamma Ray Decay Data Standards for Detector Calibration and other Applications* , International Atomic Energy Agency, Vienna, Austria. Volume 1, ISBN 92-0-113606-4.
[10] IAEA General Safety Requirement Part 3 Pp.126

Selective Bone Marrow Shielding as an Approach to Protecting Emergency Personnel

Oren Milstein[a], Gideon Waterman[a], Kenneth Kase[b*], Itzhak Orion[c]

a* Department of Research & Development, StemRad, Ltd., 6 Raoul Wallenburg St. 1st Floor Tel Aviv 6971905, Israel

b 955 N California Ave, Palo Alto, CA. 94303-3407, USA

c Department of Nuclear Engineering, Ben-Gurion University of the Negev. P.O.B.653, Beer-Sheva 84105, Israel

Abstract. The importance and current feasibility of protecting emergency responders through bone marrow selective shielding is highlighted in the recent OECD/NEA report on severe accident management (NEA/CRPPH/R(2014)5). Until recently, there was no effective personal protection from externally penetrating gamma radiation. In Chernobyl, first-responders wore makeshift lead sheeting for protection. In Fukushima, emergency personnel undertook disaster mitigating activities without any protection from gamma radiation. Older personal shielding solutions used thin layers of shielding over large body surfaces which are ineffective for energetic gamma radiation. Acute exposures may result in Acute Radiation Syndrome where the survival-limiting factor up to 10 Gy is irreversible bone marrow damage. Protracted exposures may result in malignancies of which bone marrow is especially susceptible. This is compounded by leukemia's short latency time. This highlights the importance of shielding bone marrow for preventing deterministic and stochastic effects. Due to the extraordinary regenerative potential of hematopoietic stem cells, to effectively protect from deterministic effects of bone marrow exposure, it is unnecessary to protect all or even most of the bone marrow. This is exemplified in transplantation, where <5% of the donor's marrow is sufficient to rescue a lethally irradiated recipient. This biological principle allows for a new class of personal protection equipment providing unprecedented attenuation of external radiation to select marrow-rich bodily regions, deferring Acute Radiation Syndrome to much higher doses (>7 Sv vs. as low as 1 Sv today). As approximately 48% of the of the body's active bone marrow is contained within the pelvis, shielding this region holds great promise for preventing the deterministic effects of bone marrow exposure and concomitantly reducing stochastic effects. Here we show the efficacy of a device providing selective shielding to the pelvic region in mitigating Acute Radiation and reducing cumulative doses to bone marrow and other radiosensitive organs in the abdominal area.

KEYWORDS: gamma radiation shielding; bone marrow; personal protective equipment; acute radiation syndrome; hematopoietic stem cells; selective shielding.

1 INTRODUCTION

The biological effects of ionizing radiation exposure are categorized as stochastic or deterministic. Stochastic effects describe the probability of radiation induced cancer and/or hereditary defects where increased exposure corresponds to a higher probability of the effects with some latent onset time. Deterministic effects occur beyond a certain dose threshold and increase in severity with increasing dose. Acute exposure can lead to Acute Radiation Syndrome (ARS). Usually, cells are able to repair the damage in cases where low doses are received. At higher levels of radiation, apoptosis results and cells that are lost as part of normal tissue turnover are not replaced because of damage to the stem-cell compartment, leading to tissue failure [1]. Many casualties of the Hiroshima and Nagasaki atomic bombs, and many of the firefighters who first responded to the Chernobyl nuclear power plant accident, became ill with ARS [2]. The probability of survival of those inflicted with ARS decreases with escalating radiation dose. Most of the people who do not recover from ARS will die within a few weeks to a few months after exposure, with the primary cause of death being the destruction of the body's bone marrow (BM) [2].

BM is comprised of hematopoietic stem cells (HSCs) which are responsible for the constant renewal of blood cells [3]. Due to their high rate of proliferation, HSCs are especially vulnerable to ionizing radiation,

* Presenting author, e-mail: kr.kase@stanfordalumni.org

but are endowed with remarkable regenerative potential [4-8]. Owing to their central role in blood production, lethal irradiation of HSCs leads to death from severe anemia, infection and internal bleeding. This relationship between high doses of radiation and HSC apoptosis has led to the use of HSC transplantation as a life-saving intervention in cases of acute exposure [9-13]. This is demonstrable in medical practice, where thousands of individuals have undergone supra-lethal Total Body Irradiation (TBI) for purposes of cancer therapy, and were rescued by BM transplantation [9, 14]. Indeed, life-threatening damage may be reversed by BM transplantation in individuals receiving doses of radiation of up to 10 Gy [2]. At doses > 10 Gy, damage to gastrointestinal (GI) tissue may become a limiting factor in survival [2, 15]. Remarkably, in the common procedure of BM transplantation, the number of HSCs extracted from a single active BM site containing less than 5% of the donor's BM tissue is sufficient to support the complete reconstitution of the HSC compartment in a lethally irradiated recipient [16, 17]. This capacity of BM to expand and replenish is due to the high regenerative potential of the HSCs it harbors.

In contrast to mature blood cells, which are dispersed throughout the body, HSCs are confined to the bones, allowing for effective targeted shielding. This has been confirmed in several animal models, where sparing the HSC-rich area of a subject receiving otherwise full-body irradiation is sufficient to support hematopoietic functions and allow survival [18-21]. Approximately 48% of the human body's active BM is contained within the lumbar vertebrae and the pelvic region [22]. This high concentration of BM makes the pelvic region an ideal area to shield for preventing the deterministic effects of radiation exposure.

Stem-cell rich tissues like active BM are also more sensitive to the stochastic effects (i.e. cancer) of radiation exposure in addition to the deterministic. The high amount of BM in the pelvic region, combined with the presence of the sensitive gastrointestinal system and female gonads in this same area of the body allows for the possibility of significantly reducing stochastic effects of radiation by preventing radiation exposure to this vital area. Thus, shielding this region holds great promise for both acute and protracted exposures.

Shielding the human body from gamma rays requires large amounts of material, in stark contrast to alpha particles that can be blocked by paper or skin, and beta particles that can be blocked by foil. Gamma rays are best blocked using high-Z materials (high density). For this reason, lead is a favorable material for radiation attenuation [23].

Here we present a specialized radiation shield for the protection of active BM concentrations, which to the best of our knowledge is the first to be developed for humans. This belt-like shield focuses on the protection of BM that is present in the hip bones. The device protects the medullary volume from which BM is commonly extracted for transplantation (i.e. the iliac bones) [16, 17] while allowing relatively unhindered movement of the wearer. In order to optimize the use of shielding materials towards the protection of active BM, the shielding uniquely brings into account the natural shielding properties of human tissue, with thickness being inversely related to the thickness and radiodensity of the underlying tissue at each point surrounding the area being protected. Here we present the findings resulting from both experimental and simulation testing of this device

2 MATERIALS AND METHODS

2.1 Engineering of the Active Bone Marrow Shield

First we sought to identify the ideal target for protection. HSCs are present in several BM locations in the human body, the foremost being the hip, sternum, ribs, vertebrae and skull. In adults the iliac bones of the hip are the most attractive targets for protection due to a high content of active BM (225 g), the iliac bones' relatively small surface area to volume ratio, and the positioning of the iliac bones in the pelvic girdle at the body's center of gravity - an ideal anatomical location for weight-bearing purposes [24]. The iliac bones serving as the source of HSCs in BM transplantation validates this choice [16,17].

Since, depending on recipient weight, between 23 and 58 grams of net active BM is harvested and transplanted on average (after deduction of plasma and blood cell infiltrates) [25, 26], the device was developed so that this volume of BM will remain viable at least up to a radiation dose where tissues other than BM sustain major damage. The 2nd most radiosensitive tissue is the gut [2], and the gut sustains irreversible damage at about 11 Gy [2], so our shield was engineered to protect this critical volume of BM at doses as high as 11 Gy (radionuclide energy dependent), a level which covers most nuclear catastrophes.

In nuclear disasters, radioactive materials are presented in the form of nuclear fallout in a cloud geometry. This dictated that our shield be able to attenuate radiation emanating from all directions, so it was engineered with a circumferential arrangement. The device is designed to closely wrap around the area of the body containing the BM selected for protection. Additionally, the device covers body surfaces, which are adjacent to the protected BM in order to sufficiently attenuate any radiation approaching the BM through the body of the wearer. To engineer a shield of a weight and design that does not limit mobility, we sought to minimize shield mass without compromising protection. Selectively shielding the iliac bones provided an effective means of dramatically reducing shield weight compared to non-selective strategies, but the optimal shield would incorporate into its design the attenuation of the underlying tissue. Also, different tissues (bone>muscle>adipose) have different radiodensity. This shield accounts for the natural shielding properties of human tissue by being of differential thickness inversely related to the thickness and radiodensity of the underlying tissue at each point surrounding the target for protection. Thus, at any given point on the device the radiation attenuation factor is such that it accommodates the variation in tissue thickness and radio-density in the circumference of the protected BM, reducing shield mass substantially without compromising protection.

Desired Total Attenuation (A_D) was defined such that the surviving volume of viable active BM is sufficient to allow for hematopoietic reconstitution after exposure. Knowing the radiosensitivity of human HSCs and progenitors and the volume of active BM that is protected by the radiation protection device, formula 1 is used to deduce A_D [27].

$$A_D \geq \frac{D_U}{D_V} \tag{1}$$

$$P_R = \frac{V_N}{V_P} \times 100 \tag{2}$$

Where A_D = desired total attenuation; D_U = unprotected radiation dose and D_V = dose at which the percent viability of the BM cell is equal to percent viability of active BM necessary for hematopoietic reconstitution (P_R). P_R = percent viability of active BM necessary for reconstitution; V_N = volume necessary for reconstitution (23 to 58 cm^3, size dependent) and V_P = volume of protected active BM.

The Visible Human Data set was employed to calculate Tissue Attenuation (A_T). The Visible Human Project is the creation of complete, anatomically detailed, 3D representations of the human body [28-31]. The data set includes complete transverse and reconstructed longitudinal cryosection images of representative male and female cadavers. This tool, which has been used for the construction of accurate digital phantoms in several radio-dosimetry studies, allowed us to measure the thickness and determine the overall radiodensity of tissues surrounding active BM sites [32, 33].Using cross sectional data of the Visible Human pelvic area, we created an anatomically accurate digital phantom (Fig. 1C, right) allowing us to map the tissue type and thickness present between the selected BM centers and radiation entry points for hundreds of points around the waist area. By inputting this data into the following formula we arrive at the true tissue attenuation at a given point:

$$A_T(x, y, z) = b * e^{-\mu x} \tag{3}$$

A_T=tissue attenuation; b=build-up factor for one energy at tissue thickness x; μ=linear attenuation coefficient in cm^{-1}; x = tissue thickness between BM and body surface point (x,y,z) in cm. A_T and A_D then allowed the calculation of the Required Attenuation (A_R) of the shielding at any given point (equation 4) and subsequently to the shielding thickness at any point (equation 5):

$$A_R(x,y,z) = \frac{A_D}{A_T} \quad (4)$$

$$Thickness\ (x,y,z) = \frac{\ln(b*A_R)}{\mu} \quad (5)$$

This led to a belt-like radiation protection device with variable thickness, using only the minimal amount of shielding material needed (Fig. 1). This device in combination with the body's tissue is configured to provide a substantially uniform 4-fold total radiation attenuation (Cs-137) to 300 cm^3 of active BM in the posterior pelvis and lesser degrees of protection to an additional ~300 cm^3 of active BM. The shielding material of the device is provided in the form of multiple uniquely shaped 1mm sheets of virgin lead, which are layered upon each other forming a shielding device of a topography inversely related to the thickness and density of the tissue present between the device and the protected BM in the iliac crest.

Figure 1: Shielding Device **A**. The outer shell of the device is designed for maximal comfort. **B**. The shielding component of the device is comprised of uniquely shaped 1 mm sheets of virgin lead with friction reducing dividers placed between them. **C**. The topography of the shielding reflects the anatomy of the underlying BM such that it is of a thickness inversely related to the thickness and density of the tissue between the device and the BM, thereby minimizing weight without compromising protection.

2.2 Experimental Setup and Configuration

Irradiations were performed at Pacific Northwest National Laboratory (PNNL) of a male phantom that would result in an approximate simulation of a radiation exposure of an individual to a cloud-like source of Cs-137. This source-phantom irradiation geometry could also simulate the radiation dose to an individual walking and turning numerous time in an enclosed environment that contains multiple sources at various heights relative to the individual. These irradiations were conducted with the phantom both shielded and unshielded with embedded dosimetry to measure changes in absorbed doses at internal points of interest (active BM concentrations and abdominal organs) resulting from the device.

2.2.1 RANDO Phantom Geometry

The male RANDO® phantom used for the test irradiations at PNNL, Richland, WA, USA was manufactured by Alderson Corporation. The RANDO® man represents a 175 cm tall and 73.5 kg male figure without limbs. RANDO® is constructed with a real human skeleton that is cast inside soft tissue-simulating material. The phantom is constructed of horizontal slices of 2.54 cm thickness to allow access to the thermoluminescent dosimeter (TLD) cavities. Each slice contains approximately 40 of these cavities, each 4.8 mm diameter in a 3.5 cm grid pattern. Forty additional TLD cavities were added to the RANDO® in order to measure the absorbed dose to the active BM tissue in the lower spine and pelvis. These 40 additional cavities were 33 cavities distributed equally in the pelvic BM and 7 representing equal volumes of lumbar vertebrae BM. Another 52 TLD cavity locations were used to measure doses to the colon, small intestine, and ovaries. Absorbed doses were measured at 92 distinct cavities in the presence vs. in the absence of the device. Three LiF TLDs were positioned within each of these cavities.

2.2.2 Structure and Positioning

To ensure precise distances between the source, floor, and RANDO®, a reference point was defined near the middle of the torso, at the geometric center on the top of slice 29. The z-axis of rotation was relative to this reference point. The device was fit to RANDO® with the posterior of the device spanning between slices 24.5 and slice 33.5. This fit was consistent with the proper donning of the device for ergonomic and shielding of the active BM purposes. Fig. 2 shows the device on RANDO® although the irradiation was done without clothing. The posterior side of the shielding measures 18 cm in height and the vertical center of this posterior shielding was positioned to be lined up with the reference points in both the RANDO® phantom and the MCNP simulations.

2.2.3 Source Geometry

We used an encapsulated Cs-137 source consisting of a total of 0.078” stainless steel and 0.125” aluminum. Encasing results in the elimination of the beta particle part of the spectrum associated with the nuclide, and only the gamma spectrum is seen (peaks at 662 keV). To create a realistic fallout setting, the source was shifted in relation to the phantoms to 5 discreet positions (0°, +/- 22.5° and +/- 45°) along the Z-axis relative to the reference point while the phantom itself was rotated at a rate of 1 RPM along the X-Y plane. In the -45˚ source position, the amount of ground scattering was measured to be approximately 4.5% and was taken into consideration. The total exposure at the reference point was measured using a Capintec Model PR-18 ionization chamber to be 4.518 R.

2.3 Monte Carlo n-Particle (MCNP) Code Simulations

MCNP reproduced the experimental conditions and evaluated the ability of the device to reduce the dose absorbed in the iliac and vertebral BM using point sources of Cs-137. A version of the computerized ORNL-MIRD phantom with a modified pelvic and lumbar spine geometry to provide higher resolution of absorbed dose to the active BM was used [34]. The phantom was positioned standing on a concrete slab (30 cm high) to allow for ground scatter. Twenty point sources were used in a geometry similar to that used in the PNNL irradiation; 5 different heights at 0°, +/- 22.5° and +/- 45° on the z-axis and rotated in 4 positions in the x-y plane every 90° relative to the origin (same as ref. point used at PNNL) which is the center of torso cell at the height corresponding to where the L-5 vertebra and sacrum meet.

3 RESULTS AND ANALYSIS

3.1.1 Experimental Results

According to PNNL’s formal report for the irradiation testing, the mean relative dose of the shielded phantom’s BM volumes divided by the mean relative dose of the unprotected phantom’s BM results in a mean belt on/belt off ratio of 0.59 [35]. One minus this ratio, 0.41, provides the mean percent reduction in dose to the pelvic BM attributed to the device (41%). While the shielding provided by the device was significant at all pelvic area points studied, it was especially evident in the posterior iliac crest which showed a 58% reduction in absorbed dose to the BM from the device. In the experiment, the absorbed dose reduction to the total BM tissue in the body was 19%; the ovaries had a 35% dose reduction, and the large intestine had a 27% dose reduction [35].

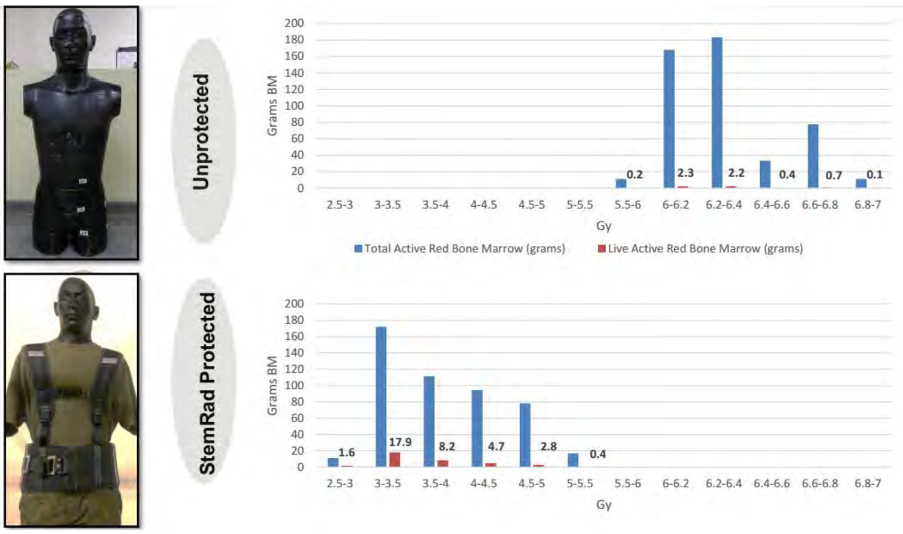
3.1.2 Simulation Results

The energy deposition (MeV/g) reported by F6 tally results for the active BM regions record absorbed doses for active BM regions which were compared in the shielded and unshielded geometries. The maximum reduction in absorbed dose was in a region of the sacrum with a 57% reduction compared to the same region in the unshielded phantom, and the mass weighted average of absorbed dose reduction for the active BM sampled in the simulation was 42%. In the simulation, the absorbed dose reduction to the total BM tissue in the body was 17%; ovaries - 29% dose reduction; intestines - 30% dose reduction, and the stomach had a 19% dose reduction.

3.1.3 Analysis

Using the attenuation conferred by the device at each point in the pelvic marrow, we determined the absorbed dose to active BM assuming an ambient dose of 9 Gy (Cs-137). 9 Gy was chosen as a high dose point of interest because it is near the maximum dose where the hematopoietic sub-syndrome of ARS is the primary cause of death [2]. To this end, we assumed that the total amount of active marrow of our phantom was identical to that of the Computerized Anatomical Man (CAM) age corrected for a 25 year-old (1266 g) [24]. The distribution of the active BM in the body was used to determine how much resided in the lumbar spine and pelvic regions [43]. Since the TLD cavities were evenly distributed through the pelvic volume, we correlated the mean absorbed dose of the 3 TLDs in each of the 33 pelvic cavities to a precise mass of active BM. The vertebral TLD cavities' mean absorbed doses were also applied to the specific masses of active BM which they represented. Figure 2 shows the absorbed dose to the active BM histogram based on this analysis; a dramatic shift in absorbed dose to active BM is evident in the presence of the device. Based on the human BM radiosensitivity curve from J.S. Senn and E.A. McCulloch, the precise amount of live active BM that would remain following a 9 Gy whole-body exposure in the absence vs. in the presence of the device (Fig. 2 - red bars) was determined [27]. Adding together the amounts of live active BM gave a total of 6 grams for an individual exposed to 9 Gy without protection and 36 grams for an individual equipped with the device. This process of dose extrapolation to 9 Gy was repeated for the MCNP analysis using the shift of absorbed dose to the active BM to determine the amount of viable active BM for the shielded phantom and unshielded cases. The results were similar to that of the experiment with 33 g of viable active BM in the shielded case and 4 g for the unshielded.

Figure 2: Dose to active BM at 9 Gy and resulting live BM quantities (shown in red). The distribution of absorbed dose to active BM in 50 and 20 cGy dose bins for unprotected and protected individuals.



4 DISCUSSION

The significant difference in the remaining viable active BM at 9 Gy with the tested device (36 g shielded vs. 6 g unshielded) should have enormous implications for the survival of the exposed individual, as the quantity of active marrow necessary for the reconstitution of a lethally irradiated average sized adult is approximately 2.5% or 32 grams [26]. In the absence of the device, an individual exposed to such a high dose would have virtually no chances of recovery from pancytopenia due to such a critically low mass of viable active BM. Contrarily, the amount of 36 grams of live active marrow preserved in an individual equipped with the device is sufficient to allow for complete reconstitution of the hematopoietic system leading to dramatically increased survival. In addition to protecting a critical amount of active BM to mitigate ARS, the reduction in absorbed doses to the total BM tissue in the body, stomach, colon and ovaries was significant (19%, 19%, 27%, and 35% respectively) and would lead to a reduction in radiation-induced cancer incidence of these high tissue-weighted organs.

5 REFERENCES

[1] Stone, H.B., et al., *Effects of radiation on normal tissue: consequences and mechanisms.* Lancet Oncol, 2003. **4**(9): p. 529-36.

[2] CDC. *Acute Radiation Syndrome: Fact Sheet for Physicians*. Accessed March 23, 2012; Available from: http://www.bt.cdc.gov/radiation/ars.asp.

[3] Wang, L.D. and A.J. Wagers, *Dynamic niches in the origination and differentiation of haematopoietic stem cells.* Nat Rev Mol Cell Biol, 2011. **12**(10): p. 643-55.

[4] Greenberger, J.S. and M. Epperly, *Bone marrow-derived stem cells and radiation response.* Semin Radiat Oncol, 2009. **19**(2): p. 133-9.

[5] Pearce, M.L., M.A. Greenfield, and W.N. Valentine, *Studies on the radiosensitivity of bone marrow. III. The dynamics of erythrocyte regeneration after hemorrhage and irradiation.* Blood, 1952. **7**(2): p. 207-13.

[6] Valentine, W.N. and M.L. Pearce, *Studies on the radiosensitivity of bone marrow. I. The relative sensitivity of erythroid and myeloid elements.* Blood, 1952. **7**(1): p. 1-13.

[7] Valentine, W.N., M.L. Pearce, and J.S. Lawrence, *Studies on the radiosensitivity of bone marrow. II. The effect of large, repeated whole body irradiation exposure on hematopoiesis.* Blood, 1952. **7**(1): p. 14-9.

[8] Dainiak, N., *Hematologic consequences of exposure to ionizing radiation.* Exp Hematol, 2002. **30**(6): p. 513-28.

[9] Thomas, E.D., et al., *Marrow transplantation for acute nonlymphoblastic leukemic in first remission using fractionated or single-dose irradiation.* Int J Radiat Oncol Biol Phys, 1982. **8**(5): p. 817-2.

[10] Chertkov, J.L., et al., *Autologous bone marrow transplantation in irradiated baboons.* Radiat Res, 1971. **46**(1): p. 129-43.

[11] Alpen, E.L. and S.J. Baum, *Modification of x-radiation lethality by autologous marrow infusion in dogs.* Blood, 1958. **13**(12): p. 1168-75.

[12] Arkin, R., et al., *NATIONAL PLANNING SCENARIOS*. 2006. p. 1-16.

[13] MacVittie, T.J., *Therapy of radiation injury.* Stem Cells, 1997. **15 Suppl 2**: p. 263-8.

[14] Goldman, J.M., et al., *Bone marrow transplantation for patients with chronic myeloid leukemia.* N Engl J Med, 1986. **314**(4): p. 202-7.

[15] Barnett, D.J., et al., *Understanding radiologic and nuclear terrorism as public health threats: preparedness and response perspectives.* J Nucl Med, 2006. **47**(10): p. 1653-61.

[16] Thomas, E.D., et al., *Bone-marrow transplantation (second of two parts).* N Engl J Med, 1975. **292**(17): p. 895-902.

[17] Thomas, E., et al., *Bone-marrow transplantation (first of two parts).* N Engl J Med, 1975. **292**(16): p. 832-43.

[18] Cole, L.J., H.M. Haire, and E.L. Alpen, *Partial shielding of dogs: effectiveness of small external epicondylar lead cuffs against lethal x-radiation.* Radiat Res, 1967. **32**(1): p. 54-63.

[19] Stearner, S.P., E.J. Christian, and A.M. Brues, *Modification of the radiation syndrome in the chick by partial body shielding.* Radiat Res, 1954. **1**(3): p. 270-81.

[20] Jacobson, L.O., et al., *Recovery from radiation injury.* Science, 1951. **113**(2940): p. 510-11.

[21] van Bekkum, D.W. and E. Schotman, *Protection from haemopoietic death by shielding versus grafting of bone-marrow.* Int J Radiat Biol Relat Stud Phys Chem Med, 1974. **25**(4): p. 361-72.

[22] ICRP, 2002. Basic Anatomical and Physiological Data for Use in Radiological Protection Reference Values. ICRP Publication 89. Ann. ICRP 32 (3-4).

[23] Jaeger, R.G. and International Atomic Energy Agency., *Engineering compendium on radiation shielding, prepared by numerous specialists*. 1968, Berlin, New York etc.: Springer-Verlag.

[24] Watchman, C.J., D. Hasenauer, and W.E. Bolch, *Derivation of site-specific skeletal masses within the current ICRP age series.* Phys Med Biol, 2007. **52**(11): p. 3133-50.

[25] Pichardo, J.C., et al., *Method for estimating skeletal spongiosa volume and active marrow mass in the adult male and adult female.* J Nucl Med, 2007. **48**(11): p. 1880-8.

[26] NMDP Standards Committee (1993) "Standards of the national marrow donor program", Transfusion 33, 172-180.

[27] Senn, J.S. and E.A. McCulloch, *Radiation sensitivity of human bone marrow cells measured by a cell culture method.* Blood, 1970. **35**(1): p. 56-60.

[28] NIH. *The Visible Human Project*. Last Accessed 11 August 2012; Available from: http://www.nlm.nih.gov/research/visible/visible_human.html.

[29] Spitzer, V.M. and D.G. Whitlock, *The Visible Human Dataset: the anatomical platform for human simulation.* Anat Rec, 1998. **253**(2): p. 49-57.

[30] Spitzer, V., et al., *The visible human male: a technical report.* J Am Med Inform Assoc, 1996. **3**(2): p. 118-30.

[31] Ackerman, M.J., et al., *The Visible Human data set: an image resource for anatomical visualization.* Medinfo, 1995. **8 Pt 2**: p. 1195-8.

[32] Xu, X.G., T.C. Chao, and A. Bozkurt, *VIP-Man: an image-based whole-body adult male model constructed from color photographs of the Visible Human Project for multi-particle Monte Carlo calculations.* Health Phys, 2000. **78**(5): p. 476-86.

[33] Caon, M., *Voxel-based computational models of real human anatomy: a review.* Radiat Environ Biophys, 2004. 42(4): p. 229-35.

[34] Eckerman K.F., Cristy M., Ryman J.C. (1996) The ORNL mathematical phantom series, informal paper, OakRidge, TN:Oak Ridge National Laboratory. available at http://homer.hsr.ornl.gov/VLab/mird2.pdf

[35] Murphy, M.K. et al., *StemRad 360Gamma Radiation Shielding Belt Testing - Final Report. Pacific Northwest National Laboratory formal test report[1], PNNL-24346, June 2015.*

RENEB – The European Network for Emergency Preparedness and Scientific Research

Ulrike Kulka[a], Elisabeth Ainsbury[b], Lleonard Barrios[d], Octavia Gil[e], Eric Gregoire[c], Alicja Jaworska[f], Simone Mörtl[g], Ursula Oestreicher[a], Gabriel Pantelias[h], Laurence Roy[c], Laure Sabatier[i], Sylwester Sommer[j], Georgia Terzoudi[h], Francois Trompier[c], Pedro Vaz[e], Anne Vral[k], Clemens Woda[g], Andrzej Wojcik[l]

[a]Bundesamt fuer Strahlenschutz, Department Radiation Protection and Health, Neuherberg, Germany

[b]Public Health England, CRCE, Didcot, UK

[c]Institut de Radioprotection et de Sûreté Nucléaire, Fontenay-aux-Roses, France

[d]Universitat Autonoma de Barcelona, Barcelona, Spain

[e]Campus Tecnológico e Nuclear, Instituto Superior Técnico, Universidade de Lisboa, Lisboa, Portugal

[f]Norwegian Radiation Protection Authority, Aas, Norway

[g]Helmholtz Zentrum München, Neuherberg, Germany

[h]National Centre for Scientific Research Demokritos, Athens, Greece

[i]Commissariat à l´Énergie Atomique, Fontenay-aux-Roses, France

[j]Instytut Chemii i Techniki Jadrowej, Warszawa, Poland

[k]Faculty of Medicine and Health Sciences, Universiteit Gent, Gent, Belgium

[l]Stockholm University, Institute Molecular Biosciences, Stockholm, Sweden

Abstract. The European Biodosimetry Network "RENEB" has been established to perform large-scale rapid biodosimetric radiation dose estimation. Specialized to handle a large number of samples, RENEB has ability to contribute to radiological emergency preparedness and large scale research projects. The network infrastructure is based on reliable assays and techniques combined with high performance standards. To enhance the effectiveness of the network, RENEB is linked to global emergency preparedness and response systems as well as to the European radiation research area. The network was initiated in January 2012 with 23 partners from 16 European countries with the support of the EC (EURATOM FP7, GA 295513). At this time the focus was on emergency preparedness with the aim to significantly increase individual dose reconstruction capacities in case of large-scale radiological scenarios. Individual dose estimation based on biological samples and/or inert personalized devices has been optimized to support the rapid clinically relevant categorization of many victims according to the received dose. Communication and cross-border collaboration was standardized and cooperation with national and international emergency and preparedness organizations such as IAEA and WHO were initiated. The value of RENEB to support topics also outside emergency preparedness is now evident. With established strategies to guarantee consistent performance between the partner laboratories, the network has the ability and capacity to contribute to large-scale research projects by analyzing radiation exposure biomarkers. This includes studies on the effects of low doses, group related radiation sensitivity, contribution to non-cancer diseases, and epidemiological studies where sampling and handling of biological samples is required. RENEB also drives the development and evaluation of new exposure markers with special view to their applicability for addressing acute or protracted exposures as well as exposures dating back years or decades. Last but not least, RENEB provides inter-comparisons, specialized courses and seminars open also to laboratories outside the network, thus being of relevance for E&T and Quality Assurance in biodosimetry.

Absorbed Dose Measurements using Ordinary Salt on Anthropomorphic Phantoms – Novel Conversion Coefficients to Effective Dose for Various Exposure Geometries

Christian Bernhardsson, Therese Geber-Bergstrand, Jonas Jarneborn, Maria Christiansson

Medical Radiation Physics, Department of Translational Medicine, Lund University, Malmö, Sweden

Abstract. There are several methods for retrospective dosimetry, but there is a lack of information on how to convert the measured response of the dosemeters to human risk related quantities i.e. effective dose. In a first attempt of establish such relationship we present the first conversion coefficients for the optically stimulated luminescence signal in ordinary salt (NaCl) to effective dose after exposure to ionising radiation. Two types of anthropomorphic dosimetry phantoms (Alderson, RANDO; ATOM, CIRS) representing adult males, were exposed in five different geometries (AP, PA, ROT and, with the radiation source in the trouser pocket and in a back- pack) using Co-60 sources. LiF thermoluminescent (TL) detectors were used inside the phantoms for estimation of organ doses and effective dose. Special personal dosemeters consisting of a combination of TLDs with LiF and household salt (NaCl) read-out with optically stimulated luminescence (OSLD), were placed onto the phantom surface in the chest area. In addition, ordinary salt portion packages, commonly found in e.g. fast food restaurants, were positioned on the front side of the leg of the phantom in height of the trouser pocket. The TLD absorbed dose to the effective dose was 1.56, 1.12, 1.19 mGy mSv^{-1}, for AP, PA and ROT exposure geometries, respectively. The corresponding conversion coefficients for the OSLDs were lower by about 50% and there were no significant difference between the doses to the OSLDs and to the salt in the portion packages for the AP, PA and ROT exposure geometries. However, in the special geometries (pocket and back-pack) it is important to consider were the salt is kept on the body. In the pocket geometry the difference in absorbed dose reached a factor of 14 between the OSLD and the salt portion package. The here presented conversion coefficients for NaCl provide the first physical measurements on the relation between the absorbed dose in NaCl and effective dose. It is indicated that there is no difference if the salt is kept in special dosemeter holders with PMMA build up or in their original packages when exposed to Co-60 photons. It is also indicated that the position of the salt on the body is not influencing the conversion factors when the photon beam is incident parallel to the body surface.

Monitoring and Dose Assessment for Children for Internal Radiation Contamination Following a Radiological or Nuclear Emergency

Chunsheng Li[a], Florence Ménétrier[b], Armin Ansari[c], George Etherington[d], Wi-Ho Ha[e], Jean- Rene Jourdain[f], Boris Kukhta[g], Osamu Kurihara[h], Maria Lopez[i], Arlene Alves dos Reis[j], Ana Rojo[k], Stephen Solomon[l], Jianfeng Zhang[m], Zhanat Carr[n]

[a]Health Canada, Ottawa, Ontario, Canada

[b]Commissariat à l'énergie atomique et aux énergies alternatives, Fontenay-aux-Roses, France

[c]Centers for Disease Control and Prevention, Atlanta, USA

[d]Public Health England, Chilton, UK

[e]Korea Institute of Radiological and Medical Sciences, Seoul, Republic of Korea

[f]Institut de Radioprotection et de Sûreté Nucléaire, Fontenay-aux-Roses, France

[g]Burnasyan Federal Medical Biophysical Center of Federal Medical Biological Agency, Moscow, Russia

[h]National Institute of Radiological Science, Chiba, Japan

[i]Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas, Madrid, Spain

[j]Institute of Radiation Protection and Dosimetry, Rio de Janeiro, Brazil

[k]Autoridad Regulatoria Nuclear, Buenos Aires, Argentina

[l]Australian Radiation Protection and Nuclear Safety Agency, Melburne, Australia

[m]Chinese Center for Disease Control and Prevention, Beijing, China

[n]World Health Organization, Geneva, Switzerland

Abstract. Following a radiological or nuclear emergency, affected populations may become internally contaminated with the radioactive material(s) involved, as demonstrated during the Goiânia, Chernobyl and Fukushima accidents. Children, as a special population, have been extensively monitored, assessed, or treated for internal contamination following these accidents. However, managing a large number of children in an emergency situation is still very challenging; technological or technical gaps remain outstanding. This paper presents a new project, under the coordination of the WHO Radiation Emergency Medical Preparedness and Assistance Network (REMPAN), to address such gaps, including thyroid monitoring, whole body monitoring, in vitro monitoring, triage for medical intervention, and communicating risks to children and their parents.

The Radiation Situation in the Area of Radioactive Trace Resulted From the Accident in a Nuclear Submarine at the Chazhma Bay

Dmitry Isaev, Alexey Titov, Sergey Kiselev, Vladimir Shlygin, Natalia Novikova, Renata Starinskaya

SRC - Burnasyan Federal Medical Biophysical Center, Moscow, Russia

Abstract. Over the whole operating period of nuclear submarines in the Soviet Union and Russia, there was the only radiation accident, accompanied by the release of radionuclides with a significant radiation impact on the environment. We mean here the accident of 1985 at the Chazhma Bay (Prymorie) nearby the pier of the naval shipyard. Due to violations of the nuclear safety requirements and technology, a spontaneous chain reaction has occurred during the reactor overloading. Following the heat explosion and accompanied fire, combustion products along with fission and activation products as well as particles of the fuel composition were released into atmosphere. The radioactive cloud of gaseous radioactive materials resulted from the accident, has crossed the Danube Peninsula and reached the sea in the area of the Ussuri Bay. The paper includes the results of the radiation survey of the above mentioned area 30 years after the accident. The radiation situation depends on Co-60 and Cs-137 radionuclides presented. Within this area, the trace area limited by 0.15 µSv/h gamma dose rate isoline is 0.31 km^2. At the local part of the trace, at 1300 m distance from the location of the accident, gamma dose rate reaches 0.5 µSv/h. At this area, the radionuclide specific activities in the surface soil layer (as to October 2014) reach 640 Bq/kg by Co-60 and 55 Bq/kg by Cs-137.

Mass media communication of emergency issues and countermeasures in a nuclear accident. Fukushima reporting in European newspapers

Eduardo Gallego[a], Marie Claire Cantone[b], Deborah. H. Oughton[c], Tanja Perko[d], Iztok Prezelj[e], Yevgeniya Tomkiv[c]

[a]Energy Engineering Department, E.T.S. Ingenieros Industriales, Universidad Politécnica de Madrid, José Gutiérrez Abascal, 2, 28006-Madrid, Spain

[b]University of Milan, Italy

[c]Norwegian University of Life Sciences, Norway

[d]Institute for environment health and safety, Belgian nuclear research Centre SCK- CEN

[e]University of Ljubljana, Faculty of Social Sciences, Slovenia

Abstract. Public communication is one of the most followed aspects of a nuclear emergency management. This paper presents the results of a large study of 1340 articles published by two major newspapers from each of six countries (Belgium, Italy, Norway, Slovenia, Spain and Russia) in the first two months after the Fukushima nuclear disaster. The focus of the analysis is on the application and overall impact of protective actions, both during the emergency phase and later, how the newspapers describe those actions, which differences were apparent between countries and what recommendations can be extracted in order to improve general communication about these issues. A clear lesson is that, even under uncertainty and recognising the limitations, responsible authorities need to provide transparent, clear, and understandable information to the public and the mass media right from the beginning of the early phase of any nuclear emergency. Media could be interested in evacuation since it can be presented as an event. Evacuation has to be communicated intensively not only to evacuees but also to a global public worldwide. Other early countermeasures affecting people, like iodine prophylaxis, or the control of water consumption and farming products are also topics of interest for the media already during an early stage. Clear, concise messages should be given. Mass media could play a key role in reassuring the public if the countermeasures are clearly explained.

KEYWORDS: ***emergency management; emergency communication; media communication; countermeasures.***

Radiation Protection Dosimetry (2017), Vol. 173, No. 1-3, pp. 163–169

doi:10.1093/rpd/ncw334

The Development of Concept of "Virtual Cytogenetic Biodosimetry Laboratory" for Radiation Emergencies in Occupational Field

Franz Fehringer[a], N. Maznyk[b], Chr. Johannes[c], T. Sipko[b], N. Pshenichna[b]

[a]Institute for Radiation Protection of the BG ETEM, Cologne, Germany

[b]Institute for Medical Radiology NAMSU, Kharkiv, Ukraine

[c]University Duisburg-Essen, Essen, Germany

Abstract. In radiation accidents countermeasure management it is always crucial to get reliable physical and biological dosimetry information soon after the accident. As there is a lack of cytogenetic biodosimetry laboratories around the world the idea to establish and run a small international network that includes biological dosimetry seem to be a good choice for radiation protection institution. We have organized such a network named "Virtual Cytogenetic Biodosimetry Laboratory" (VCBL). That allowed us bringing together the expertise from radiation protection, radiation biology and cytogenetic biodosimetry fields. Several studies were carried out within the scope of VCBL running in order to determine the best ways of procedures comprised sample collection, lymphocyte cultivation, slide processing and image and microscopy analysis. These studies included in vitro experiments and comparisons with blood probes from workers of different industrial fields. They demonstrated good correlation between microscopy and image analysis data and highlighted further issues of VCBL concept to be taken into account. The results will be discussed. The next studies were devoted to investigating the influence of image capturing system software parameters on the data output and inter-scorers variability. They comprised in vitro experiments and three cases of suspected radiation overexposure. The results and lessons learned will be discussed. The main issues of VCBL concept such as quality assurance procedures, handling the blood samples, validation of image and microscopy analysis, harmonization of scoring criteria, data security, data protection, communication as well as outlook to further work will be discussed. The implementation of VCBL concept will help to create additional capacities of cytogenetic biodosimetry especially for radiation protection applications in the industrial fields.

Radiological Situation at the Chernobyl Shelter Site Thirty Years after the Accident

Gunter Pretzsch[a], Viktor Krasnov[b]

[a]Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gmbH, Berlin, Germany

[b]Institute for Safety Problems of Nuclear Power Plants (ISPNPP), Chornobyl, Ukraine

Abstract. Thirty years after the accident the Chernobyl Shelter still remains a dangerous nuclear facility. Inside the Shelter remained about 96 % of the irradiated nuclear fuel inventory of the reactor before the accident, i.e. 180 t of Uranium of total radioactivity $7x10^{17}$ Bq. The radioactive releases to the environment were estimated to amount 4 %. Because of the radiation exposure the spent fuel inside the Shelter and the radioactive soil and groundwater contaminations at the site have an essential impact on all human activities which are presently under progress with the erection of the New Safe Confinement (NSC) in the framework of the Shelter Implementation Plan. The spread fuel in particular causes radiations fields with high dose rates. Airborne radioactive dust particles contribute to inhalation doses when raised inside the Shelter and released by air ventilation towards the environment during accidental situations, e.g. after roof collapse, or its dismantling after the NSC will be completed. The radiological situation and the potential hazards caused by the amount, modifications and local distribution of spent nuclear fuel inside the Shelter and by the radioactive contamination at the surrounding industrial site as well as the drying out cooling pond were studied in a long lasting cooperation project over more than 20 years between GRS Germany and ISPNPP Ukraine. The paper gives an overview of the main results.

Assessment of Measurement Capabilities in Nuclear Accident

Helena Janzekovic

Slovenian Nuclear Safety Administration (SNSA), Ljubljana, Slovenia

Abstract. The Fukushima accident as one of the largest nuclear accident opens challenging issues to nuclear as well as other countries. Namely, countries where nuclear facilities are sited as well as those without such facilities but which can be affected by accidents originating in other countries study their preparedness to cope with a severe accident and its consequences. They also estimate long term off site damages caused by severe radiological and nuclear accidents. Development in emergency preparedness based on the lessons learned from the accident mentioned has been reported by many countries. In 2013 the IAEA published the Actions to Protect the Public and an Emergency due to Severe Conditions at a Light Water Reactor (EPR-2013) which reflects such lessons. While initial measures in an emergency are based on information given by an operator, later, i.e. in a due course of an accident, measurements of contamination and dose fields are essential parts of regulatory activities. Namely, numerous measures, such as decontamination of schools and forests, are actually based on results of measurements.. So one of the basic regulatory issues is the preparedness for conducting such measurements in the state in due time when the very first phase of an accident is over. A systematic and quantitative approach to the assessment of this specific preparedness is proposed in the article. Namely, the author propose initial assessment of: measurement equipment available taking into account OILs from EPR-2013, and organisational aspects of such measurements in the state, e.g. availability of teams to conduct particular measurements of dose fields. The assessment is based on quantitative criteria, i.e. on the physical parameters from the EPR-2013 related to specific measures. As physical parameters, e.g. parameters related to area contamination, should be determined in a specific timeframe, e.g. a day, in order to effectively implement a specific measure, the same timeframe is also used when assessing the capabilities. The article is a first attempt to quantitatively assess preparedness to conduct necessary measurements in a state a function of a particular accident scenario.

Study on the Derived Response Level in Case of a Radiological Accident

Hyeong-Ki Shin[a,b], Juyoul Park[a,b]

[a]Korea Institute of Nuclear Safety, Daejeon, Chungnam, Republic of Korea

[b]FNC Technology, Seoul, Republic of Korea

Abstract. Introduction: In case of a radiological accident or terror, as well as radiological effect, a severe social unrest resulting from the public's potential psychological fear of the radiation may occur. Considering the huge sum of the cost needed for the decontamination of radiological materials and for the site restoration, it is necessary to establish an effective national emergency preparedness system against the radiological accident or terror. A radiological emergency response analysis tool is essential for the prompt evacuation of the public and the estimation of the radiological environmental consequence. However, an exclusive program for the consequence analysis of a malicious radiological act and an eventual radiation accident is not yet developed, so that a prompt first response cannot be performed successfully to minimize the accident consequences. Objectives and Methodology: In this study, after a scenario for a hypothetical radiological terror was selected by estimating risk and was analyzed by Hotspot code, its results were applied to derive a response level using RESRAD-RDD. Firstly, the domestic and foreign emergency operational guidelines on the radiological accident/terrorism are reviewed. Secondly, former data on radiation sources incidents are reviewed as well as IAEA Illicit Trafficking Database System (ITDB) to obtain a suitable radiation source. Next, a hypothetical scenario of radiological malicious terror with radiation dispersion device was selected by taking the reliable references and the reasonable assumptions. Through the Hotspot program, the selected scenario was analyzed to estimate the total effective dose equivalent (TEDE) and the ground deposition as function of downwind distance. Finally, the operational guidelines incorporated in RESRAD-RDD program are applied to derive a response level using estimated results of the hypothetical scenario, as well as the specific staying time during the early phase, etc. Results: In case of a radiological accident or terror, whether the prompt response is effective or not can largely affect the accident consequences. In this study, a systematic analysis procedure to estimate the radiological accident or terror was tried through estimating terror risk for the scenario, Hotspot code analysis for the radiation level and RESRAD-RDD code for the derived response level. The results show that this analysis procedure can be well applied to estimate a radiological accident or terror, such as for the judgement of staying time in the early phase and the radiological concentration for the access to business, and for the derived response level in the various environments, etc. Conclusion: From this study, it is expected that the radiological accident/terror estimation procedure serves well as an essential technical support tool to carry out the protective actions, the environmental monitoring and to derive guidelines in various cases. This support tool can also be used to train emergency staff to improve their capability for the radiological emergency preparedness and response. Finally, the social unrest by accidents could be alleviated and the social cost to recover from the radiological contamination could be reduced.

Training of RPEs for Emergency Response

Heleen van Elsäcker-Degenaar, Folkert Draaisma

Nuclear Research and Consultancy Group, Petten, The Netherlands

Abstract. In our nuclear installations we need an adequate and trained emergency organisation. Experience learns that a lot of attention is focused on the risk of radioactive releases and the potential consequences. Therefore, the Radiation Protection Expert (RPE) plays an important role in the emergency organisation. We developed a training program specifically for 24/7 available RPEs in order to give proper advice. These advices are derived from pre-defined scenarios and the potential exposure to members of the public. Topics addressed are e.g.: national nuclear emergency response plan, pre-defined scenarios (15) of our installations with source term release values, programs to calculate release and exposure in the environment and parameters influencing the effect of the release (mainly the weather conditions). Evaluation shows that the training is adequate and that practical refresher course are needed to improve the performance of the RPEs during table-top exercises. In the presentation attention will be paid to the development of the training program and the content of this program.

Research Priorities in Emergency and Recovery Preparedness and Response: the NERIS Strategic Research Agenda

Johan Camps[a], Thierry Schneider[b], Wolfgang Raskob[c]

[a]Belgian Nuclear Research Centre (SCK•CEN), Mol, Belgium

[b]Centre d'étude sur l'Evaluation de la Protection dans le domaine Nucléaire (CEPN), Paris, France

[c]Karlsruhe Institute of Technology (KIT), Karlsruhe, Germany

Abstract. NERIS is a European Platform on preparedness for nuclear and radiological emergency response and recovery, founded in June 2010. Its mission is to establish a forum for dialogue and methodological development between all European organisations taking part in decision making of protective actions in this field. 55 institutions are currently member of the NERIS platform from which 28 supporting organizations. NERIS aims at identifying gaps and needs for further research and developments and addressing new and emerging challenges. These needs are documented in the NERIS Strategic Research Agenda. Following the general presentation of the NERIS Platform, this paper will present the research priorities identified by the NERIS R&D Committee and discussed during the 2015 NERIS workshop and consultations of NERIS members. These research priorities concern both further developments of models and researches focused on the decision-making processes in emergency and recovery situations. For modelling, it is expected to make more reliable forecasts of atmospheric dispersion, including data assimilation and improved inverse modelling in different environments (e.g. urban areas) and/or at different spatial scales. It is also considered to develop local radio- ecological models for their integration in decision support systems. To improve the decision-making processes, it is expected to further investigate: 1) data uncertainties and how they can be communicated; 2) the structuration of the decision processes with the help of decision aid tools, and on the basis of feedback from stakeholder processes; 3) the development of sustainable preparedness strategy at Local, National and European level, to enable optimized remediation and contribute to the elaboration of robust recovery strategies; 4) the optimised use of monitoring resources by integrating new monitoring technologies and developing processes and tools for combining the monitoring results from experts and lay people.

Mapping of Radiation Fields in Areas with Complex 3D Source Geometries Using a Shielded Two-Detector Configuration and Data Processing

Jonas Jarneborn, Marcus Persson, Christopher Rääf, Robert Finck

Lund University, Malmoe, Sweden

Abstract. After a release of radioactive material from a nuclear accident or from an uncontrolled dispersion of a radioactive source, the radiation field in the environment can be quite complex because the radioactive substances may be in various physico-chemical forms and may deposit on different surfaces and structures in the environment. When trying to locate the radioactive material, the complex radiation field will complicate the determination of the variability and distribution of the activity because stronger and more widespread sources at distance can provide significant contributions over weaker and more closely located sources. This also hampers the possibility during decontamination to determine whether a particular area has become clean enough or not, because radiation from more remote areas provides a background contribution that disturbs the measurements. The measuring situation can be improved by the use of detectors that are sensitive to the radiation field in different directions. Such a detector constellation can be achieved by shielding the detectors in an appropriate manner to identify the direction of the radiation. Partial shielding of one or more detectors will increase the sensitivity and specificity to detect and locate weaker sources nearby. The present study describes a method using a two-detector system consisting of a collimated HPGe detector and a shielded NaI(Tl) detector. The detector system was tested in a radioactive waste treatment facility in order to locate sources on the ground. The method has shown to decrease the detection limit of Cs-137 sources on the ground from approximately 2.5 MBq to 100 kBq in a surrounding background dose rate of 45 μSv/h from Cs-137. The results indicate that the method may be useful in the clearance of areas in the nuclear industry, or for determining the efficiency of decontamination of radiocaesium on the ground after an uncontrolled release. It could also be suitable for locating radioactive debris or radioactive hot spots in the event of a malevolent act with an RDD.

Dose Re-estimation Method in Emergency Radiation Exposure Situation

Jungil Lee[a], Insu Chang[a], Jang Lyul Kim[a], Kisoo Chung[b]

[a]Korea Atomic Research Institute, Daejeon, Republic of Korea

[b]Gyeong Sang National Univesity, Jinju, Republic of Korea

Abstract. Radiation exposure doses of occupational workers are generally monitored by thermoluminescent dosimeter (TLD), optically stimulated luminescent dosimeter (OSLD) or glass dosimeter. In case of emergency radiation exposure situation (without artificial dosimeters), optically stimulated luminescence (OSL) from electronic components of personal electronic devices such as mobile phones could be used to estimate the doses exposed, however it is sometimes not very reliable because of the very high fading rate of the OSL signal from the samples. In this presentation, we are going to present a method to re-estimate of the exposed doses using thermoluminesence (TL) which is residual signal after the process of dose estimation of the exposed dose using OSL technique. In alumina rich components such as resistors and inductors, we have found that a "residual TL signal", after OSL measure measurement, is still remained in the samples. We have tested the dosimetric properties such as dose linearity and fading rate of the residual TL signals and concluded that the residual TL signal could be applied to verify the estimated exposed doses in emergency situation. Dose response of the signal was linear and the fading rate was much slower than that of the OSL technique. Therefore, we suggest that the residual TL signal method could be method for dose re- estimation in emergency situations.

Introduction of Uncertainty of Atmospheric Dispersion Calculation and Improvements of Urban Countermeasure Modelling in an Operational Decision Support System

Steen Hoe[a], Kasper Andersson[b], Jens Havskov Sørensen[c], Carsten Israelson[a]

[a]Danish Emergency Management Agency, Birkerod, Denmark

[b]Technical University of Denmark, Roskilde, Denmark

[c]Danish Meteorological Institute, Copenhagen, Denmark

Abstract. The ARGOS Decision Support System (DSS) has been operational worldwide for more than 15 years and was successfully used during the Fukushima accident – but only far away from Japan. The accident revealed a need for introduction of uncertainties in the early phase, as well as improvement of fast dose calculation and countermeasure simulation in urban areas. Today, most current DSSs do not consider uncertainties in atmospheric plume calculation, but merely allow for presentation of a single deterministic plume dispersion pattern. However, recent developments in numerical weather prediction modelling include probabilistic forecasting techniques addressing the inherent uncertainties, and this approach may be taken over by atmospheric dispersion modelling. Recently, such methodology has been developed and implemented operationally for long-range dispersion modelling in Denmark. On request by the Danish ARGOS installation, the Danish Meteorological Institute (DMI) automatically delivers statistical dispersion model output based on 25 ensemble members to ARGOS for presentation and dose modelling. As monitoring data become available, they provide important assistance for the decision making, and therefore ARGOS includes a comprehensive monitoring database. However, there is a need for advanced modelling for evaluation of the potential future doses corresponding to different countermeasures. The ERMIN model (European Model for INhabited areas) is an integral part of the ARGOS system. ERMIN was developed to allow system users to examine doses and dose reduction effects of different countermeasures in different types of inhabited areas as well as providing assistance in developing and refining an appropriate emergency management strategy. The integration of ERMIN includes interfaces to atmospheric dispersion, including uncertainty, for the very early phase, as well as an interface to monitoring data as they become available. The introduction of uncertainties of plume prediction, and faster urban dose simulation, has given the experts new important tools for expert guidance of the decision makers.

Protection Actions Decision Making Support during Nuclear Emergency in China

Jiangang Zhang, Yapeng Yang, Zongyang Feng, Linsheng Jia, Shutang Sun

China Institute for Radiation Protection, Taiyuan, Shanxi, China

Abstract. In order to avoid or reduce the exposure during emergency, effective protection measures could be taken according to the regulations and standards. After Fukushima accident, EPZs in the relevant standards are studying to change to adapt new requirement. At any stage, decision-makers will necessarily consider: the future impact the effectiveness of protective measures and the concerns. The information decision-makers need includes facility design data, geography data and real-time data such as operating condition, radiation monitoring in facilities, environment monitoring, meteorology, and disaster. During different accident stages, the protection actions decision-making is based on different methods, such as Initial Conditions (ICs) and Emergency Action Levels (EALs), consequence assessment, and Operation Intervention Levels.

Capabilities of the New Mobile Laboratory of the Nuclear Forensic Laboratory

Károly Bodor

CER, HAS, Budapest, Hungary

Abstract. The main task of the Mobile laboratory is to monitor the environment of the KFKI Campus and also extended with new task to explore and identify nuclear materials and orphan radioactive, nuclear sources. The capabilities of the Mobile Laboratory are: on route gammaspectrometry; on route dose rate measurements, dose map production; exploration of orphan, hidden, smuggled radioactive sources & nuclear materials; source identification; control of different events with significant participation of spectators in order to detect hidden nuclear or other radioactive materials; education, presentation & organization of national and international trainings in the field of nuclear forensic and nuclear security (Global Initiative to Combat Nuclear Terrorism (GICINT), Nuclear Forensics International Working Group (ITWG)); continuous air sampling & activity concentration assessment; soil, plant sampling, on site measurement & evaluation; rapid thyroid gland measurement (I-125, I-131) & internal dose assessment. The Mobile Laboratory is recently equipped with very large scintillation and neutron detectors, (1.85 m long, 22.5 cm diameter each), the system is very sensitive. The MEST team made an assessment of the natural background of Hungary in several places and made a survey near the Hungarian Coal power plants to measure the TENORM activities. With a new method the team are able to assess the internal dose from the thyroid gland by using the so called Quadratic Compression Conversation method, by this technique the measuring time can be reduced for a few seconds, the measurements are fast but reliable, this allows faster decision for Iodine profilaxis at an emergency preparedness.

Ongoing Work to Enhance Post-accident Radiation Protection at Swedish Nuclear Power Plants

Karin Fritioff[a], Staffan Hennigor[b], Marie Carlsson[c], Michael Pettersson[d], Ingela Svensson[e]

[a]Vattenfall AB, Stockholm, Sweden

[b]Forsmarks Kraftgrupp AB, Östhammar, Sweden

[c]Ringhals AB, Väröbacka, Sweden

[d]Oskarshamns Kraftgrupp AB, Oskarshamn, Sweden

[e]Swedish Nuclear Fuel and Waste Management Company (SKB AB), Stockholm, Sweden

Abstract. The Nuclear power plants in Sweden have established a working group with the objective to identify areas for improvement and develop action plans for the radiation protection capabilities during severe accidents. An overarching goal of the group is to harmonize, when needed and when possible, the strategies and procedures within post-accident radiation protection between the nuclear facilities. A common approach will facilitate the assistance from the others and the communication and interaction with relevant authorities. An assistance agreement, both regarding RP personnel and RP equipment has been established and a strategy document with recommendation on response functions and actions in the early phase on an emergency is developed. The WG will now move to the practical implementation of the strategies and the assistance agreement. The task is to identify procedure that could and should be adopted at the facilities. In addition the WG will give recommendations on Operational Intervention Levels that will support the RPM function in the emergency centres. With the dimensioning scenarios for the emergency preparedness and the guidance from the Nordic authorities regarding dose criteria, intervention levels for specific actions such as distribution of iodine thyroid block tablets, relocation of people, evacuation of an area and sending people to the medical treatment will be discussed and agreed upon. The contribution, oral or a poster, will present the results of the WG and in addition elaborate upon the rationale for the suggested OIL.

A Spectrometry Acquisition System for UAS based on Raspberry Pi

Magnus Gårdestig, August Ernstsson, Håkan Pettersson

Department of Medical Radiation Physics, Department of Medical and Health Sciences, Linköping University, Linköping, Sweden

Abstract. Unmanned aircraft systems, UAS, have developed to the point of readiness which enables their exploitation in radiation measurement applications. Any measurement scenario where unmanned aircraft can be used offer operation from safety distance in respect to irradiation as well as personal safety and forensic considerations. The envisaged main mission for the rotary wing system is localization and identification of individual radioactive sources. With a fixed wing system, with longer endurance, mapping of ground area of fallout is possible. The aerial platforms set limitations in payload. During recent years, commercial detectors have been developed towards rugged, lightweight and affordable detectors. CZT spectrometers lighter than 100 g are readily available. Light CsI(Tl) spectrometers are available and open source PMT-base multichannel analysers are available for NaI or LaBr scintillators. Feasibility studies with simple ad-hoc detector systems have been presented during the last years. As a part of a complete unmanned mobile radiometry system, a suite of programs has been developed to acquire, transmit, present and analyse gamma spectrometry data from detectors carried by unmanned vehicles. The acquisition system is based on the Raspberry Pi, a single- board computer, affordable and small. The detectors supported so far are the Kromek GR1, a CZT-based spectrometer, only 60 g and scintillators operated by the Bridgeport Instruments oemBase, a MCA and high voltage generator for PM tubes. Data is stored according to ANSI/IEEE N42.42 locally on the Raspberry Pi and in tandem sent to the ground control station by digital radio link. The spectrometry data is merged with the GPS data given by the autopilot through MAV link, Micro Air Vehicle Communication Protocol. This makes it possible to enhance the GPS accuracy with e.g. RTK technology keeping the MAV link protocol. The spectrometry data is presented as a waterfall diagram, energy spectrum, full spectrum or user specified ROIs, SDI graph and colour coded dose rates overlaid on a map. The system is intended for quadcopter or fixed wing aerial surveys, but could be used in a variety of measurement setups.

Comprehensive Approach to Assess Radiation Induced Individual Health Injuries and Prognostic Clinical Evaluation using Integrative "Dosimetry" Strategies

Matthias Port, Christina Beinke, Michael Abend

Institute of Radiobiology of the German Armed Forces, Munich, Germany

Abstract. Radiological events like deployment of nuclear weapons, improvised nuclear devices, dirty bombs, radiological or nuclear accidents require rapid and precise medical classification ("triage") of a large number of patients. Estimates on the absorbed dose and in particular predictions of the radiation induced health effects of soldiers is mandatory for optimized allocation of limited medical resources and initiation of patient centered treatment. Among the German Armed Forces Medical Services the Bundeswehr Institute of Radiobiology offers a wide range of tools to cope with different scenarios. The forward deployable mobile Medical Task Force has access to state of the art methodologies summarized into approaches such as physical dosimetry, clinical "dosimetry" (H-Modul) and different means of biological dosimetry (e.g. dicentrics, high throughput gene expression techniques, gamma-H2AX). The integration of these different approaches enables trained physicians of the Medical Task Force assessing individual health injuries as well as prognostic evaluation, considering modern treatment options. To enhance the capacity of single institutions, networking has been recognized as an important emergency response strategy. The capabilities of physical, biological and clinical "dosimetry" approaches spanning from low up to high radiation exposures will be discussed and an alternative to classic biodosimetry will be demonstrated..

NEA Framework for the Post-Accident Management of Contaminated Food

Michael Siemann, Matt Cooper, Ted Lazo

OECD Nuclear Energy Agency, Paris, France

Abstract. After the 1986 Chernobyl accident the international community, established, through the Codex Alimentarius Commission, criteria for the importation of contaminated food from areas affected by radiological contamination. The European Commission also issued a directive establishing such criteria for accepting food containing post-accident radioactivity within the European Union. However, the Fukushima Daiichi NPP accident demonstrated that the post- Chernobyl approach to such criteria was not universal. The importation of food goods from Japan following the Fukushima Daiichi NPP accident caused numerous national and international challenges that existing national and international criteria could not address. The NEA began study of post-accident contaminated food management in 2013, and in 2015 issued a report outlining a framework addressing food management in a more holistic fashion. The NEA framework addresses protection of those living in contaminated areas, addresses protection of those living in the accident country but in areas not affected by radiological contamination, addresses the export of food from the accident country, and addresses the import of food from the accident country. Further, the NEA developed a framework study of the impact that the modification of Japanese export criteria had on the importation of food goods from Japan. This paper will describe the results of both of these NEA reports.

Industrial Radiography Accident at the Rio Turbio Power Plant: Causes of the Event

Maria Teresa Alonso Jimenez, Irene Raquel Pagni, Eleazar Martin Ameal

Nuclear Regulatory Authority, Buenos Aires, Argentina

Abstract. On September 8th 2015 a call was received at the Nuclear Regulatory Authority (ARN from its Spanish initials) from an Industrial Radiography operator, stating that during the early hours on august 27th he was over exposed to an Ir-192 radioactive source of 1628 GBq (44 Ci). The operations were being done in a confined space inside a boiler at the Rio Turbio Power Plant, which was under construction. After realizing that the source had been exposed without his awareness, he put it back in its shielding position and carried on with the remaining operations. Days after, when an injury appeared on his right index finger he got scared and called ARN. Over the reconstruction of events, once again it became clear that the human factor had the main role. The manipulation of the projector was made by auxiliary personnel who, in spite of having received theoretical training by approving a specific course, he didn't have operational experience. The operator, who happened to be the Responsible for Radiological Safety of the company, didn´t comply with specific safety procedures of the practice, such as the use of the totality of radiological safety equipment, allegedly for being under contractual pressure. The objective of this work is to analyze the initiating event, the causes that lead to the accident and the string of deviations from the applicable standards. Currently this operator and his assistant are under medical surveillance. It is not the purpose of this work to present a dosimetric evaluation of the event.

Off-site Emergency Planning at UK Nuclear Licensed Sites

Paul Leonard[a], Gareth Thomas[b]

[a]CRA Risk Analysis, Leatherhead, Surrey KT22 7AH, United Kingdom.

[b]Paper produced with the support of: Nuclear Safety Inspector, Office for Nuclear Regulation, EP&R Team, Cheltenham, United Kingdom.

Abstract. Nuclear emergency planning arrangements in the United Kingdom are continually kept under review. This work proposes to outline how experience from nuclear exercises and undertaking emergency response duties can be based on radiological knowledge of specific sites and utilised in the future. In 2015, the UK regulator, the Office for Nuclear Regulation (ONR) revised their principles for the determination of off-site emergency planning areas around nuclear sites where pre-determined countermeasures and other protection measures are applied to protect those people who may be affected by a radiation emergency. The revised principles also enhanced communication from the nuclear site operators and Local Authorities to the public. This updated ONR's application of the UK Radiation (Emergency Preparedness and Public Information) Regulations 2001 (REPPIR) *http://www.hse.gov.uk/radiation/ionising/reppir.htm* , which includes details of minimising potential doses to the public, as well as assessment and reassurance, linked to other concurrent risks such as flooding. ONR undertakes site specific assessments of each operators' hazard identification and risk evaluation which includes consideration of whether the public might receive a significant radiation dose in the year following the emergency (excluding countermeasures in the first 24 hours). In defining the areas for off-site emergency planning, practical and strategic factors are then considered, which includes other local non-nuclear emergency planning arrangements and experience, and whether local geographic and demographic aspects could aid public credibility and confidence.

Radiation Protection Dosimetry (2017), Vol. 173, No. 1-3, pp. 157–162
doi:10.1093/rpd/ncw315

Preliminarily Integrated Simulation for Severe Accident of CLEAR-I based on Virtual Reactor Virtual4DS

Tao He, Jinbo Zhao, Zihui Yang, Pengcheng Long, Liqin Hu, Team FDS

Key Laboratory of Neutronics and Radiation Safety, Institute of Nuclear Energy Safety Technology, Chinese Academy of Sciences, Hefei, Anhui, China

Abstract. With the rapid development of Information Technology, such as High Performance Computing Technology, Numerical Simulation Technology, Visualization Technology and so on, Virtual Reactor is playing an increasingly significant role in enhancing the safety reliability and economy of nuclear power. “Virtual4DS”, virtual reactor in digital social environment, was initiated by Institute of Nuclear Energy Safety Technology, CAS•FDS Team. It is based on the deep integration of advanced information technologies such as cloud platform, high-performance computing, virtual reality, visualization, big data, collaboration technology and internet of things. It aims to realize a full- range and full-cycle of the reactor behaviours and the properties of high-fidelity multi-physics integrated simulation. Virtual4DS is considering radiation, neutron transport, burn-up, thermal- hydraulic, structural mechanics, material behaviours, fuel properties and many other aspects of the integrated simulation. The seamless integration of different physical process data, high sense of reality and immersive experience, virtual assembly and design verification, maintenance plan and virtual training, occupational exposure dose assessment and optimization can also be realized in the Virtual4DS. Virtual4DS has been used for the nuclear design and safety assessment of ADS reactor system, which is supported by “Strategic Priority Research Program” of Chinese Academy of Sciences. Typical transients of CLEAR-I as a reference reactor were simulated with developed CFD-system coupled code, which are unprotected loss of heat sink and unprotected loss transient of overpower for both critical and subcritical modes, for which, a sub-critical point kinetic neutronics model has been developed in Virtual4DS. The calculation results were compared with the results from the system code only, that shows the differences brought by the three dimensional flow in the pool type reactor between the new coupled code and the original system code prediction. Based Virtual4DS, researchers have successfully completed the first full-process integrated simulation of nuclear accident simulation for CLEAR-I, which will improve the safety of CLEAR-I.

Review on Spraying Water Soluble Resin to fix the Radioactive Material after Fukushima Accident

Qiong Zhang, Bo Wang, Ruiping Guo

Nuclear and Radiation Safety Center, Beijing, China

Abstract. After Fukushima nuclear accident, the method of spraying water soluble resin was used by Tokyo electric power company. The method was expected to restrain the spread of the radioactive dust, by forming consolidation layer in pollution area surface. This paper briefly introduced the motivation of spraying water soluble resin, spraying range and implementation process. From the relevant report on Fukushima nuclear accident, the effect of spraying water soluble resin for reducing pollution was analyzed, the mechanism of reducing pollution for water soluble resin and the application prospect were discussed. The spraying water soluble resin for fixing radioactive dust had reasonable effect for reducing pollution. It is worth to use as reference and study in other countries.

Training Emergency Plan in a Nuclear Installation

Ruy Ferraz, Ricardo Bitelli

Marinha do Brasil, Sorocaba, São Paulo, Brazil

Abstract. All kinds of plants must have an emergency response plan. A nuclear installation is no different. These nuclear facilities must have a response plan very specific emergency, detailed and trained several times. The aim of this paper is to present the details of the emergency response plan of a nuclear facility of the nuclear fuel cycle. The training strategy begins in the definition and study of the real risks that have the nuclear installation. After you completed this step, the strategy is to start training activities. This training must have some steps, including: i) theoretical exercises of emergency response; ii) training with the teams involved in the nuclear installation; iii) actuation the teams remotely (at a distance); local training; and finally the full exercise of the emergency response plan. The completed paper will detail all these steps presenting a new methodology for measuring and evaluation of training efficiency. This methodology based on concepts of the International Atomic Energy Agency (IAEA).

Suitability of portable radionuclide identifiers for emergency incorporation monitoring

Roman Galeev, Gernot Butterweck, Markus Boschung, Bénédicte Hofstetter-Boillat, Eike Hohmann, Sabine Mayer

Paul Scherrer Institute, 5232 Villigen PSI, Switzerland

Abstract. The suitability of portable nuclide inspectors for incorporation measurements were tested with three probes (LaBr, NaI and HPGe) differing in sensitive volume and energy resolution. The efficiencies for the measurement of whole-body and lung radionuclide burden were calibrated using a whole-body block phantom with traceable radionuclide sources of ^{60}Co, ^{133}Ba, ^{137}Cs, ^{152}Eu and ^{40}K. The phantom was placed in a standing position with contact to the detector. The standing geometry was chosen as it allows rapid positioning of persons for the measurements. Decision and detection limits were determined for the unshielded detector in a normal laboratory radiation environment according to ISO 11929 for ^{134}Cs, ^{137}Cs and ^{60}Co, which were considered as the relevant radionuclides for most emergency scenarios. The detection limits of all three probes were significantly higher than for a well-shielded dedicated whole-body monitor. Nevertheless, lung and whole-body burdens derived from dose constraints for routine and emergency conditions could be measured with all three probes with a counting time of one minute even for routine monitoring purposes.

KEYWORDS: incorporation monitoring; gamma-spectrometry; radiological emergency; radionuclide identifier.

Radiation Protection Dosimetry (2017), Vol. 173, No. 1-3, pp. 145–150

doi:10.1093/rpd/ncw330

Nuclear and radiological emergencies - making early health protection decisions under uncertainty

Stephanie Haywood[a], Simon French[b], Matthew Hort[c]

[a]Public Health England, Oxfordshire, UK

[b]University of Warwick, Coventry, UK

[c]UK Met Office, Devon, UK

Abstract. Dose estimates early in an emergency help determine whether actions (eg evacuation, sheltering, administration of stable iodine) are needed rapidly to reduce serious health effects. Understanding of an emergency situation and the estimate of likely doses will be very uncertain in the first few hours, but decisions on protective actions must be taken in spite of lack of knowledge. It is especially important to counterbalance the uncertainty associated with early dose estimates against the health risks associated with the early emergency countermeasures themselves, in particular evacuation. Rapid and comprehensive data for all significant radionuclides in all significant mediums cannot be achieved by measurements alone; modelling fills gaps in measurement data and also extrapolates to predict future impact. Assessments should consider not only available data but also what significant information is not yet known. Poorly understood aspects may include what has been released (amounts, radionuclides, particle sizes, release energy), the time distribution of the release and how this may continue, what influence the weather in the area will have (eg wind direction, precipitation). Combining the possible spreads in these factors leads to a range of different predictions of dose. The presentation of these alternatives, with the confidence associated with the different outcomes, will form the basis for decisions. In the UK, work is in progress to explore with decision makers the best way of presenting this information. Priority areas for future work are presented: the need for assessment tools to rapidly integrate monitoring and modelling results, with the use of real-time modelling of dispersion and deposition processes based on fine resolution meteorological information.; improved techniques for estimating source terms based on (eg) plant conditions; the development of systems which can reflect and visualise uncertainty in key areas.

Development of Prediction Models for Ambient Dose Equivalent Rates in Inhabited Areas after the Fukushima Daiichi Nuclear Power Plant Accident

Sakae Kinase[a], Tomoyuki Takahashi[a,b], Hideaki Yamamoto[a], Kimiaki Saito[a]

[a]Japan Atomic Energy Agency, Naka-gun, Ibaraki, Japan

[b]Kyoto University, Kumatori-cho, Osaka, Japan

Abstract. Radiocaesium originating from the Fukushima Daiichi nuclear power plant accident was deposited in the eastern region of Japan. For the implementation of adequate protection strategies, prediction models have been needed for ambient dose equivalent rates from the radiocaesium in inhabited areas located within a 80-km radius around the Fukushima Daiichi nuclear power plant. At the Japan Atomic Energy Agency, prediction models have been developed to provide the time variation of ambient dose equivalent rates in inhabited areas. The prediction models described using a double exponential form with ecological half-lives for land use, enable affected population to receive information on the level of ambient dose equivalent rates and its space and time distribution for the next 30 years after the accident. In the present study, ecological half-lives for land use were evaluated using the changes in ambient dose equivalent rates through vehicle-borne surveys. To validate the prediction models, comparisons were made between predicted and measured values for ambient dose equivalent rates in inhabited areas. In addition, distribution maps of ambient dose equivalent rates for the next 30 years after the accident were created by the prediction models. It was found that ecological half-lives in deciduous and evergreen forest areas are different from those in other areas. The prediction models were found to give information on ambient dose equivalent rates for the next 5-30 years after the accident, which would be predicted within a factor of approximately 2. The distribution maps of ambient dose equivalent rates would be useful for a better understanding of the radiological situation.

^{210}Po Version of the Yasser Arafat's Death. Results of the Russian investigations: Part 2. Medical research

Yulia Kvacheva[a], Vladimir Uiba[b], Leonid Ilyin[a], Alexandr Samoilov[a], Yuriy Abramov[a], Irina Galstyan[a], Boris Kukhta[a], Natalia Nadezhina[a], Vladimir Stebelkov[c], Alexandr Tsovyanov[a], Sergey Shinkarev[a], Vladimir Yatsenko[a]

[a]State Research Center - Burnasyan Federal Medical Biophysical Center, Moscow, Russia

[b]Federal Medical Biological Agency, Moscow, Russia

[c]Noncommercial Partnership "Laboratory of analysis of micro-particles", Moscow, Russia

Abstract. The objective of medical research was to have a detailed analysis of the Arafat's medical records to compare with the materials of our own clinical observations of accidental ^{210}Po injuries. In the very hypothetical case of ^{210}Po fatal intoxication, suggesting a very large intake of radioactivity (~10^8 Bq), it would be possible to expect the typical clinical signs of acute radiation syndrome (ARS). The main of them is a severe hematopoietic failure in the form of three-lineage hemo- and myelodepression, primarily causing the infectious complications. Y.Arafat, who died on the 30th day of the disease onset, showed no symptoms of hematopoietic depression. In addition, the patient didn't have fever or any signs of focal infection. Gastrointestinal symptoms of the Arafat's disease presented with intermittent vomiting, nausea and watery diarrhea were non-specific and similar to those of much common GI infections, as well as digestive pathologies. As evidenced by the Russian experience, in contrast to ARS in external radiation exposure, the clinical picture of emergency cases of ^{210}Po intoxication is characterized by a number of features. Of these hemorrhagic symptoms (spontaneous hemorrhages, intensive bleeding) are known to be associated with extremely rapid and prominent starting, even though in cases with relatively high platelet count. These ones were not documented in Arafat's case in spite of thrombocytopenia. There were no ^{210}Po characteristic features of hepatic and renal toxicity as well. The Arafat's signs of mild hepatopathy coincided with the patient following a strict religious fast, and his renal function remained intact until the latest stage of terminal multiorgan deficiency. A further possibility of ^{210}Pb associated health impairment was recognized insignificant in relation to both common lead toxicity and radiotoxic effects. Thus, summing up the comprehensive physical research to determine ^{210}Po and ^{210}Pb contents in the biological specimens and given that available medical records do not provide reliable data on whether Arafat had any symptoms of ARS, a direct causal relationship between the elevated activity of the above mentioned radionuclides in the remains of the deceased and his death should be excluded.

^{210}Po Version of the Yasser Arafat's death. Results of the Russian Investigations: Part 1. Physical Research

Sergey Shinkarev[a], Vladimir Uyba[b], Alexander Samoylov[a], Leonid Ilyin[a], Yulia Kvacheva[a], Yury Abramov[a], Irina Galstyan[a], Angelina Guskova[a], Boris Kukhta[a], Natalia Nadezhina[a], Vladimir Stebelkov[c], Alexander Tsovyanov[a], Vladimir Iatsenko[a]

[a]State Research Center - Burnasyan Federal Medical Biophysical Center of Federal Medical Biological Agency, Moscow, Russia

[b]Federal Medical Biological Agency, Moscow, Russia

[c]Laboratory for Microparticle analysis, Moscow, Russia

Abstract. Yasser Arafat, the founder and the first head of the Palestinian National Authority, died due to unidentified disease in 2004. At a later time a version on polonium (^{210}Po) poisoning has been suggested. In 2012 the exhumation of Y. Arafat's remains took place. Specialists from France, Switzerland and Russia participated in the exhumation and were given the equal number of samples recovered during the exhumation for further thorough analysis. The purpose of this paper is to present the results of physical research conducted by the Russian specialists to answer to the questions: (1) is elevated amount of ^{210}Po available in the remains of Y. Arafat and (2) was the Y. Arafat's death due to ^{210}Po poisoning. Materials used in the research were more than 20 biological and other samples. All possible methods to determine ^{210}Po and ^{210}Pb using radiochemical separation of ^{210}Po and measurements of both radionuclides with use of alpha-, gamma-, mass- spectrometric systems have been conducted. The ^{210}Po activity measured in the biological samples was definitely much higher (in the range from ten to one hundred times) than its background levels in the samples of internal organs and bones from various parts of the skeleton. Taking into account the uncertainties, it is important to stress that ^{210}Po is fully supported by ^{210}Pb in all measured samples, i.e. ^{210}Po and ^{210}Pb are in a state of radioactive equilibrium. Because the measurements of ^{210}Po and ^{210}Pb in the biological samples were carried out about 8.5 years after the Y. Arafat's death, the hypothetical lethal amount of ^{210}Po entered the body would have decreased by a factor of about 7 000 000 and would have been reduced to the values of natural background. Thus, elevated activity of ^{210}Po measured in the biological samples was supported by the current activity of its precursor ^{210}Pb rather than hypothetical lethal amount of ^{210}Po entered the Y. Arafat's body in 2004. However, it is impossible to unambiguously establish the fact whether the ^{210}Po radiation injury had taken place or not. Only complex physical and medical research conducted simultaneously is able to address that question. Thorough consideration of the medical research of the Y. Arafat's death is presented in a companion paper.

Health Risk Assessment of Emergency Personnel Regarding Radiation Exposures during the Aftermath of the Crash of Malaysia Airlines flight MH-17

Tjerk Kuipers

Ministry of Defence, Doorn, Utrecht, The Netherlands

Abstract. On the 17th of July 2014 298 persons lost their lives when Malaysia Airlines flight MH17 crashed near Hrabove, Ukrain. Four days later the UN unanimously adopted Resolution 2166 and demanded an independent airline investigation. Because of the unstable Ukrainian eastern region the investigation of wreckage and identification of the victims needed to be performed within a secure and stable environment. The Netherlands Ministry of Defence directed the multiple phases of the investigation among other parties, therefore taking responsibility for the health and safety of MOD personnel during all phases: three recovery missions, wreckage transport to The Netherlands and offering a safe and secure identification location. Radiation protection during the recovering phase was performed by the NLD MOD CBRN Response Team. Since the cause of the crash was initially unknown and the confirmed presence of radioactive components in the plane wreckage, radiation safety needed to be an issue. On the identification site multiple non-invasive techniques like mobile CT-scanners, mobile X-ray baggage scanners and hand held dental X-ray scanners were used by 120 international forensics to confirm the identity of the victims. Radiation safety during this identification process was supervised by the Health Physics Department of the NLD MOD. At the start of the identification process external radiation measurements were performed on the casket to exclude any radioactive contamination. The second phase was a twofold CT-scan of the casket. During the third phase the casket was X-rayed by a baggage scanner. The fourth identification phase covered dental identification techniques using five hand-held dental X-ray scanners. Radiation protection of personnel on the identification site was achieved by CT- scanner and baggage scanner perimeter surveys, lead shielding screens around and between the dental identification areas, lead aprons, lead gloves, lead thyroid collars and active and passive personal dosimetry. The results of radioactive contamination measurements at the crash site and identification site and radiation perimeter surveys at the identification site showed no significant elevated levels. The results of active and passive personal dosimetry are all well below legal dose limits.

Surveillance of Radioactivity in the Atmosphere by the Deutscher Wetterdienst (DWD) – Monitoring and Prognosis

Thomas Steinkopff, Joachim Barth, Axel Dalheimer, Jochen Förstner, Hubert Glaab, Michael Mirsch

Deutscher Wetterdienst, Offenbach, Germany

Abstract. Since 1955 in Germany the DWD is responsible for the surveillance of radioactivity in the atmosphere performed by measurements as well as by analyzing the meteorological situation to provide information about the transport of possibly contaminated air masses. The basic task of DWD within the German "Integrated Measuring and Information System for the Monitoring of Radioactivity in the Environment" is presented including the measuring program, the network of 48 measuring sites, the organization of aircraft measurements, the radiochemical procedures for the measurement of Sr-90, isotopes of americium, uranium and plutonium, tritium in precipitation, noble gases Kr-85 and Xe-133 as well as dispersion calculations for the transport of radionuclides after the release. The total strategy is demonstrated by examples of measurements with only very low activity concentrations due to emissions of different sources. The participation of DWD in regular emergency exercises and the operational routines guarantee a continuous verification of the procedures. A current challenge is the fast provision of information for the air traffic about the radiological situation in different flight levels in case of the release of radioactive material.

'SUDOQU': A New Methodology for Deriving Criteria for Radiological Surface Contamination

Teun van Dillen

National Institute for Public Health and the Environment (RIVM), Bilthoven, The Netherlands

Abstract. The National Institute for Public Health and the Environment (RIVM) has developed a new methodology to assess the annual effective dose resulting from a radiological surface contamination. This methodology, entitled 'SUDOQU' (SUrface DOse QUantification), was developed in the aftermath of the Fukushima nuclear accident (Japan, 2011). It revealed the absence of and the need for harmonized criteria for surface contamination of imported non- food (consumer) goods, containers and conveyances from the affected area. SUDOQU takes account of mechanisms related to the removability of contamination in a mass-balance framework. Based on the time-dependent contamination levels and on exposure scenarios for both (non-radiological) workers and members of the public (consumers), it calculates the annual effective dose per unit of surface contamination (microSv/year per Bq/cm^2). It comprises contributions of the relevant exposure pathways: external radiation, inhalation, ingestion and skin contamination. These results may then serve as input for the derivation of criteria for surface contamination, such as limiting values and operational screening levels for the ambient dose rate. We will present the concepts of the methodology by means of some practical examples. We follow a generic approach by presenting a contour plot of the annual effective dose as a function of contaminated area and duration of exposure. We also present preliminary results of dose calculations for dock workers handling contaminated freight containers. Results are discussed in context of the Regulations for the Safe Transport of Radioactive Material as established by the International Atomic Energy Agency (IAEA). The results show the versatility of the SUDOQU methodology for surface contamination, which could serve as a useful tool for policymakers and radiation-protection scientists.

Post-Fukushima Dai-ichi Review of Radioactive Materials Users and Panoramic Irradiators

Vincent Holahan, William Lee, Gordon Bjorkman

U.S. Nuclear Regulatory Commission, Washington, DC, USA

Abstract. Lessons learned after the accident at the Fukushima Dai-ichi nuclear power plant in Japan prompted the U.S. Nuclear Regulatory staff began a review of the events and assessed the possible implications for the safety of sealed radioactive sources after extreme natural events such as earthquakes, tornadoes, hurricanes, flooding or wildfires. Many of these events have the potential to cause the loss of licensee control of radioactive material. There are over 80,000 category 1 and 2 sources licensed for use in the USA. The largest industrial sources are panoramic irradiators. There are currently fifty-five panoramic, wet-source-storage irradiators and one underwater pool irradiator each containing a minimum of 37 PBq (1 MCi) of colbalt-60 licensed for use in the USA. Nine of these irradiators are located in seismic areas (i.e., eight in California and one in Arkansas). Staff evaluation of panoramic irradiator licensees were performed through document review of license requirements and regulatory framework, incident reports contained in the nuclear material events database and the daily event notification report, and internal meetings with license review experts. Facility design and available equipment must supply sufficient engineering controls and barriers to protect the health and safety of the public and licensee employees, keep exposures to radiation and radioactive materials ALARA, and minimize the danger to life and property from the uses of the types and quantities of radioactive materials possessed by the licensee. The assessment reviewed various external events to determine if the failure of a sealed source could reasonably be expected to result from an event that would be more severe than previously evaluated. NRC staff concluded that panoramic irradiators are licensed appropriately and have sufficient engineering controls to protect the health and safety of workers and members of the public. Based on the best available information, it is unlikely that extreme natural phenomena will result in a loss of control of radioactive material from an industrial irradiator that would have an adverse effect on public health and safety or the environment. As such, the NRC staff concluded that radioactive materials, panoramic irradiators in particular, are licensed appropriately for the given scope and potential hazard.

Possible Mechanism of Realization of High Doses from Beta-Particles Exposures to the Atomic-Bomb Survivors in the Area of Wet Fallout

Victor Kryuchkov[a], Sergey Shinkarev[a], Boris Kukhta[a], Evgeniya Granovskaya[a], Konstantin Chizhov[a], Masaharu Hoshi[b]

[a]SRC-FMBC, Moscow, Russia

[b]Hiroshima University, Hiroshima, Japan

Abstract. Recently numerous research studies have been initiated in order to explain the reasons of some biological effects ascertained among the atomic-bomb survivors at Hiroshima and Nagasaki in the area of wet fallout that were not explained by the existing DS02. The purpose of this paper is to present a description of a new possible scenario of residual radiation exposure where high doses to subsurface human organs and tissues due to external or contact beta-particle exposures might have been realized. After those nuclear explosions a "steel mist" and "loose" aerosols have been created. A "steel mist" was micron-sized steel spheres containing composition of radionuclides which behavior was similar to that of hot particles following the Chernobyl accident. In contrast to a "steel mist" each radionuclide included in the "loose" aerosols can be considered as an independent exposure factor, because intermolecular bonds in the "loose" aerosols are rather weak and can be easily broken. Under analysis of "loose" aerosols it is reasonable to choose those radionuclides that are capable to provide strong overexposure of subsurface human organs and tissues. The analysis shows there is such radionuclide - ^{139}Ba. One hour after the explosion in Hiroshima the activity of this radionuclide was 7% of the total activity of the nuclear debris. The ratio of β- to γ-dose to the open skin for this radionuclide is estimated to be about 180. It is supposed that the water vapor from the upper part of the storm front interacted with radioactive particles from "loose" aerosols in the radioactive cloud. Each particle from "loose" aerosols was an ice crystallization center Ih. The main competing reactions with Ba were considered and the equilibrium concentrations of various oxides of Ba were estimated. After that syngonies and parameters of crystals of these oxides were compared with the parameters of the ice crystals Ih in order to estimate effectiveness of these "loose" aerosols fragments as ice crystallization centers Ih. We suppose that a selective wet deposition of ^{139}Ba took place in the black rain area, because only crystals of oxides of this radionuclide were effective ice crystallization centers Ih under the considered conditions. According to our calculations in case of realization of such scenario, the estimated doses to the subsurface organs and tissues were high enough to cause significant health effects, while the increase of whole body dose was substantially smaller.

Regulatory Emergency Control Centre Improvement Initiatives

Vanessa Maree

National Nuclear Regulator, Cape Town, South Africa

Abstract. The National Nuclear Regulator (NNR) is established under the National Nuclear Regulator Act, 1999 (Act No. 47 of 1999) (NNRA). Its main object is to provide for the protection of persons, property and the environment against nuclear damage through the establishment of safety standards and regulatory practices. Regulatory practices include the evaluation and monitoring of the effectiveness of the implementation of Emergency Plans around Nuclear Installations such as Koeberg Nuclear Power Station and Necsa Research and isotopes manufacturing facilities. In order for the NNR to fulfil its role in case of nuclear and radiological accidents the NNR has established a Regulatory Emergency Response Centre (RERC). This centre provides a centralized location where key NNR staff members can receive notification from authorisation holders and other stakeholders, monitor the evolution of accident conditions, perform verification analyses and provide advice to off-site authorities regarding decisions that are taken to protect people and the environment. Currently the NNR is dependent on the on and off-site radiological information supplied directly by the licensees. One of the considerations of Emergency Preparedness and Response for a State Embarking on a Nuclear Power Programme is to implement an on-line radiation monitoring system. The NNR is in the process of implementing a system Plant Data Transfer and On-Line Radiation Monitoring that will allow the NNR to extract plant operational data and off-site environmental radiation information and transmit it to the RERC. The project team had to face various challenges and unexpected issues during the implementation of the project, i.e. exceeding initial budget estimates, lack of qualified equipment suppliers, short timeframes etc. Despite the challenges and difficulties, success of the entire project hinges around not compromising on functionality and quality of the establish high-tech RERC.

Application of Backpack Radiation Detection Systems for Evaluation of External Exposure after the Chernobyl Accident

Valery Ramzaev[a], Anatoly Barkovsky[a], Ivan Romanovich[a], Jonas Jarneborn[b], Sören Mattsson[b], Christian Bernhardsson[b]

[a]Saint-Petersburg Research Institute of Radiation Hygiene, Saint Petersburg, Russia

[b]Medical Radiation Physics, Department of Translational Medicine, Lund University, Malmö, Sweden

Abstract. Applicability of backpack gamma-radiation detection systems for evaluation of the external doses to humans was tested at the Gävle region (Sweden) and the Bryansk region (Russia) in 2013–2015. Both regions had been severely contaminated by ^{137}Cs due to the Chernobyl nuclear accident in 1986 with ground deposition densities (GDD) of about 100 kBq m^{-2} (Sweden) and up to 4300 kBq m^{-2} (Russia). Commercially available models of portable gamma-ray spectrometers (NaI(Tl); La_2Br_3) were employed in the investigation. Application of the spectrometric techniques allowed fractionating anthropomorphic and natural components of the exposure. The measurements were performed inside and outside settlements at typical outdoor locations (street, yard, arable land, undisturbed grassland, forest) and decontaminated ground areas. The ambient dose equivalent rate (ADER) in the pack-back geometry for a subject with the body mass of 70 kg and a height of 172 cm was by 12% (the geometry conversion factor of 0.88) lower than the ADER registered in the standard measurement geometry with a detector placed on a tripod at a height of 1 m above the ground. The effective dose rate for a human in a location was calculated by using the experimentally determined geometry conversion factor and the overall reduction coefficient for conversion from ambient dose equivalent to effective dose (0.52 Sv/Sv for adults), previously deduced in a parallel study. For the Novozybkov city of the Bryansk region of Russia (^{137}Cs GDD = 700 kBq m^{-2} in 1986), the current anthropogenic effective dose rate for adults was estimated as 0.1 mSv y^{-1} (selected paved areas in the city center), 0.6 mSv y^{-1} (a decontaminated ground area in the city center), 2.1 mSv y^{-1} (an undisturbed grassland in the city suburb) and 2.0 mSv y^{-1} (a forested area in the city suburb). Further calibration and application of the radiation detection back-pack systems for indoor measurements is planned to be performed during a survey in 2016.

Nuclear and radiological preparedness: The achievements of the European Research Project PREPARE

Wolfgang Raskob[a], Thierry Schneider[b], Florian Gering[c], Sylvie Charron[d], Mark Zhelezniak[e], Spyros Andronopoulos[f], Gilles Heriard-Dubreuil[g], Johan Camps[h]

[a]Karlsruher Institut für Technologie (KIT), Eggenstein-Leopoldshafen, Hermann-von-Helmholtz-Platz 1, Germany

[b]Nuclear Protection Evaluation Centre (CEPN), Fontenay-aux-Roses, France

[c]Federal Office for Radiation Protection (BfS), Neuherberg, Germany

[d]Nuclear Radiation Protection and Safety Institute (IRSN), Paris, France

[e]Ukrainian Center of Environmental and Water Projects (UCEWP), Kiev, Ukraine

[f]National Center For Scientific Research "DEMOKRITOS", Aghia Paraskevi, Greece

[g]MUTADIS, Paris, France

[h]Belgian Nuclear Research Centre, SCK•CEN, Mol, Belgium.

Abstract. The PREPARE project has the objective to close gaps that have been identified in nuclear and radiological preparedness in Europe following the first evaluation of the Fukushima disaster. With 46 partners from Europe and Japan, it collected the key players in the area of emergency management and rehabilitation preparedness. Starting February 2013, the project has ended January 2016. Among others, the project addressed the review of existing operational procedures for dealing with long lasting releases, cross border problems in radiation monitoring and food safety and further developed missing functionalities in decision support systems ranging from improved source term estimation and dispersion modelling to the inclusion of hydrological pathways for European water bodies. In addition, a so called Analytical Platform has been developed exploring the scientific and operational means to improve information collection, information exchange and the evaluation of such types of disasters. The tools developed within the project will be partly integrated into the two decision support systems ARGOS and JRODOS. This paper presents the final achievements in terms of methodological, mathematical and operational improvements in the area of emergency management and rehabilitation preparedness.

KEYWORDS: nuclear emergency preparedness; response; decision support; regulations; stakeholders; ARGOS; JRODOS.

Radiation Protection Dosimetry (2017), Vol. 173, No. 1-3, pp. 151–156

doi:10.1093/rpd/ncw318

UAV Carried Emergency Radiation Detection System for Severe Nuclear Accident

Yang Liu, Zhiping Luo, Luzhen Guo, Guowen Zheng, Hongchao Pang, Qin Chen

China Institute of Atomic Energy, Beijing, China

Abstract. The severe nuclear accident mostly combined with high radiation level, extreme environment conditions, and roads block. The radiation information around the accident area are the key parameter for the accident state judgment. But the normal radiation detectors nearly can be used or send to this area to get the information. For solving this problem, we newly developed a radiation detection system, including radiation detectors, unmanned aerial vehicle (UAV) system and emergency data management platform. Detectors' Special structure designed to protect the probe, GPS and data transmission module provides the tolerance of strong impact, high temperature and jetting water. We modified a DJI S1000 octocopter for carrying and throwing the detectors from 30mD100m high in the air to the accident area. The octocopter can carry two detectors and control in 5Km. Measurement data were transited to an emergency data management platform which is constructed on cloud server. The platform provides a service of exact orientation of the detectors and radiation data combine with Geographic Information System (GIS) map.

KEYWORDS: severe nuclear accident; UAV; radiation detection; emergency platform; GIS.

The Proceedings of the 14th International Congress of the International Radiation Protection Association

Volume 5 of 5

Area 11: Decommissioning, Waste Management and Remediation

Analysis of radioactive inventory for radionuclide contained in liquid effluents, resulting from the decommissioning of a nuclear research reactor

Carmen Tuca*, Ana Stochioiu, Mitica Dragusin, Daniela Gurau and Felicia Mihai

Horia Hulubei National Institute for Physics and Nuclear Engineering, P.O.B. MG-6, RO-077125 Magurele, Romania

Abstract. The paper presents the radionuclide inventory contained in liquid effluents which result from decommissioning of a nuclear research reactor and also from a nuclear spent fuel storage facility. Gamma-ray spectrometry analysis for liquid and residue samples was done in several notified laboratories of our institute in order to determine the activity concentration of the radionuclides. Gross-beta activity is measured only for the residue samples. The total amount of the activity is 7.29E+07Bq for liquid effluents resulted from reactor decommissioning activities (buffer tank) and 3.29E+08Bq for those arising from decommissioning of the nuclear spent fuel ponds. The highest concentration of activity is found for those radoionuclides identified in water samples from buffer tank and third pond. Comparing with the previously assessment, the radionuclide inventory is qualitatively increased due to ^{54}Mn and ^{241}Am radionuclides and quantitatively due to the evolution of the decommissioning process. The gross beta activity for solid residues is measured and expressed in activity equivalent to that of ^{90}Sr (fission product). The ^{90}Sr activity it is comparable with the ^{137}Cs activity for sludge samples resulted from liquid effluents originated from the spent fuel disposal facility, and is much higher for those resulted from reactor decommissioning due to the very different content of the radionuclides. The results are used for proper radiological characterization of liquid effluents and for radiological risk assessment of workers involved in those transfer to the radioactive waste treatment plant. In addition, comparing of gross-beta and gamma-ray spectrometry analysis is a proper way to improve liquid effluents monitoring in decommissioning process.

KEYWORDS: liquid effluents; concentration of activity; decommissioning.

1 INTRODUCTION

The paper aims to assess the radionuclides inventory of liquid effluents arising from the decommissioning of the nuclear research reactor and also from Nuclear Spent Fuel Storage (DCNU) ponds. Both facilities are owned by the Horia Hulubei National Institute for R&D in Physics and Nuclear Engineering (IFIN-HH) Bucharest-Magurele, Romania. A short description of reactor operation history is presented in details in [1] by Tuca et.al. The reactor is in the last decommissioning phase which consists of dismantling and demolition of the internal parts of reactor block, cooling pond for nuclear spent fuel and de-aerator; hot cells decontamination and removing of radioactive drainages. The liquid effluents resulted from reactor decommissioning are transferred by special drainage system in the underground buffer tank having 30 m^3 (source 1) for temporary storage. Then they are put in special containers and transferred to Radioactive Waste Treatment Plant (RWTP) (in a tank having 300 m^3 capacity) in order to be treated. Treated water is unconditionally released in the local river (Ciorogarla) respecting the derived emission limits approved by regulatory body (National Commission for Nuclear Activities Control (CNCAN)).

The nuclear spent fuel assemblies resulted from reactor operation were stored for short-term (one year) in the cooling pond which contains deionised water (8 m^3), inside of reactor building. Then these are transferred away from reactor into the ponds (7.21 m^3 each one) of the Nuclear Spent Fuel Storage (DCNU) [2], for long term storage. The ponds are concrete tanks lined inside with AlMg3 and outside with bitumen, against corrosion. Each pool has an aluminium rack (60 kg) - a cells network for assembly storage. Pond 1 was designed for emergency purpose and the others for normal operation. The spent fuel was repatriated into the Russia Federation in 2009 (S-36) and (EK-10) in 2012 [2].

* Presenting author. Carmen Tuca. Address: 30 Reactorului Street, POB MG-6, Magurele, 077125, Romania. Phone: +4021.404.23.00. Fax: +4021.457.44.40. E-mail address: tuca@nipne.ro

Now the ponds contain contaminated water. The emptying of DCNU pools (the second source) is necessary for interim storage of aluminium and graphite wastes resulted from research reactor decommissioning [3].

2 EQUIPMENT

The radiological characterization of both sources involves analysis of water and sludge probes for identification and determination of individual activity concentrations of constituent radionuclides. The measurements - gamma-ray spectrometry analysis and gross beta measurements - were carried out in our institute by the notified laboratories based on specific work procedures [4; 5; 6; 12] with proper equipments (see Table 1).

Table 1: Laboratories and equipments

Laboratory	Equipment
Radiologic Characterization Laboratory (LCR) from Reactor Decommissioning Department (DDR)	Gamma-ray spectrometry system consists of a GEM60P4-95 high-purity germanium coaxial detector (HPGe) with: relative efficiency 60%, resolution (FWHM) 1.95 keV, peak-to-Compton ratio 70:1, peak shape (FWFM/FWHM) 3.0, all evaluated at 1.33 MeV peak of ^{60}Co; a DSPEC jr.2.0 digital signal processing and a low-background shield. Am Mastero-32 and GammaVision-32 software are used for spectra acquisition and analysis.
Spectrometric Analysis Laboratory (LAS) from Radioactive Waste Management Department (DMDR)	Gamma-ray spectrometry system consists of a HPGe detector with relative efficiency 30%; a multichannel analyzer model DSA 1000 with a specially designed shield consisting in 10 cm lead, 1 mm cadmium and 2 mm copper; and the spectral analysis system, operation and analysis software.
Spectrometric Analysis Laboratory Gamma Spec (LGS) from Nuclear Physics Department (DFN)	Gamma-ray spectrometer consists of a HPGe detector, with measurement geometry Marinelli type; relative efficiency 30%, resolution 2.1 keV to 1332 keV energy of ^{60}Co, placed in a Pb shield (10 cm thickness). Measurements can be performed for: solid samples (mass of ~30-200 g); liquid samples (100 g - 1 kg).
Personal Dosimetry and Environment Laboratory (LDPM) from Life and Environmental Physics Department (DFVM)	9300-PG, GFR, PROTEAN EQUIPMENT for the gross alpha and beta activity measurements with flow gas, window less proportional counter. Installation for alpha-beta-gamma activity gross measurement, with automatic exchange of samples, Canberra USA, S5XLB-G type, with measuring range of 0.01÷10.000 Bq for alpha channel, 0.06÷10.000 Bq for beta channel and up to 10.000 Bq for gamma channel. Both equipments were calibrated with a ^{90}(Sr+Y) standard source. The uncertainties were less than 10%. Equipment for primary processing of samples: evaporation, drying and calcinations. Analytical balance Model WAX 220, PARTNER, 10^{-4} measurement accuracy.

3 METHODS

3.1 Gamma Ray Spectrometry

Water samples (0.5 l each) are collected in 2013 and 2014 from buffer tank for radiological characterization of liquid effluents and theirs transfer at the Radioactive Waste Treatment Plant (RWTP). Similar samples are taken from DCNU ponds for activity content monitoring at that moment and measured by LAS and LCR laboratories. Using of DCNU as interim storage facility for aluminium and graphite wastes require the liquid effluents transfer to RWTP, aluminium racks removal and pond's cleaning. For this purpose in 2015 are collected water samples from bottom of pond close to the rack and from 1 m below the waterline. Also, are collected mixture samples (water and sludge) from three different points of each rack, due to the special physic-chemical behaviour of some radionuclides. To obtain solid residues the water samples are left to settle for several days, the water is pumped out and the sludge is evaporate to dryness. The residue is collected in Sarpagan containers. The water and residue samples are set directly upon the end cap of the detector in the lead castle and separately analyzed by gamma-ray spectrometry method [8]. The energy calibration of spectrometric systems is done with standard sources made by Radionuclide Metrology Laboratory (RML) of IFIN-HH. The sources are located on the symmetry axis of detector and they cover the energy domain of interest (50÷2000 keV) [9; 10]. The spectrometers calibration was done by the same laboratory, RML [11].

3.2 Gross Alpha-Beta Activity

The radionuclides identified by gamma-ray spectrometry analysis of both sources such as: ^{60}Co, ^{134}Cs, ^{137}Cs, ^{154}Eu, ^{235}U, ^{238}U and ^{241}Am that are highly-energetic alpha and beta emitters. The radionuclides such as ^{89}Sr and ^{90}Sr are pure beta emitters and they cannot be measured by gamma-ray spectrometry. The standard method consists of chemical separation from the sample and the measurement of the pure $^{90}(Sr+Y)$ sample in equilibrium conditions. Our laboratory does not perform chemical separation so an indirect method for gross beta measurements is used. The method consists of gross beta measurements of solid samples with relatively low surface density (mg/cm^2). ^{90}Sr content in the mixture (sludge) is determined and results are expressed in equivalent activity of this radionuclide for which was done the measurement system calibration. Gross alpha measurements are not performed because all alpha emitters are measured by gamma-ray spectrometry. The samples conditioned for gamma-ray spectrometry measurements taken from DCNU ponds and buffer tank are used. The samples were gravimetrically prepared, dried and weighted before measurement according to procedure [12]. The residue is uniformly distributed on a stainless steel plate (50±0.2 mm diameter), fixed with alcoholised water in order to avoid its spreading in the proportional volume of counter and measured into P10 gas atmosphere.

3.2.1 Working procedure

Gross beta determinations are performed for estimation ^{90}Sr content of liquid effluents. The counting rates are recorded and corrected for background contribution. The equivalent activity of ^{90}Sr is calculated using eq.1.

$$N_\beta = \varepsilon \cdot s \cdot f \cdot A_{echiv}$$

with

$$N_\beta = \varepsilon_{etalon} \cdot s \cdot f \cdot A_{^{90}Sr} \tag{1}$$

where:

- N_β - counting rate for sample and standard source;
- A_{echiv} - sample activity;
- ε - detection efficiency (the same for standard source and sample);
- s - beta radiation emission intensity (the same for standard source and sample);
- f - self-absorption factor of radiations in thick samples.

The total beta activity consists of beta-gamma emitters known activity and ^{90}Sr activity that is to be determined. The detection efficiency ε is 0.425 s^{-1}/Bq. The eq.1 allows calculating the equivalent activity, starting from N_β which is determined according to eq.2:

$$N_\beta = 0.425 \cdot \sum_i^n f_i A_i + 0.425 \cdot 2 \cdot f \cdot A_{^{90}Sr} \tag{2}$$

where:

- f_i - specific self-absorption factor of radionuclides from beta-gamma emitters mixture
- f - self-absorption factor for $^{90}(Sr+Y)$ mixture at the equilibrium.

For thin samples $f_i = f = 1$ and for thick samples the self-absorption factor *f* is calculated using eq.3 and eq. 4:

$$f = e^{-\frac{\rho_x}{R_{\max}}} \tag{3}$$

where:

- ρ_x - sample thickness;

m_{sample} - mass sample;

S_{plate} - plate surface;

$R_{\max}$ - maximum range of beta radiation [13].

$$\rho_x = \frac{m_{sample}}{S_{plate}} \tag{4}$$

4 RESULTS

4.1 Gamma Ray Spectrometry Measurements

The conservative values of the activity for liquid effluents are calculated as product between concentration of activity (Bq/l) and effluent volume of each pond (see Table 2). The total activity of a pond is a sum of all constituent radionuclides activity corresponding to the liquid volume (34.8 m^3 for buffer tank and 7.21 m^3 for each DCNU pond). The measurement uncertainties are between 3.00÷16.5 % ranges.

Table 2: The conservative values of the activity for liquid effluents

Radio-nuclide	Activity (Bq)				
	Source 1	Source 2			
	Buffer tank	1st pond	2nd pond	3rd pond	4th pond
^{60}Co	2.76E+07	4.59E+04	4.61E+04	2.80E+06	8.45E+05
^{134}Cs	8.71E+05	5.05E+03	7.21E+03	2.67E+04	8.65E+03
^{137}Cs	4.26E+07	1.67E+05	6.09E+05	3.16E+08	4.70E+06
^{108m}Ag	1.11E+05	2.43E+04	8.57E+03	1.57E+06	4.74E+04
^{152}Eu	-	-	6.26E+03	2.75E+05	3.99E+04
^{154}Eu	-	-	1.47E+04	1.36E+05	2.05E+04
^{54}Mn	8.07E+04	-	-	-	-
^{235}U	2.39E+05	-	-	-	-
^{238}U	2.94E+06	-	-	-	-
^{241}Am	-	-	-	-	2.81E+05
Total	7.29E+07	2.42E+05	6.92E+05	3.21E+08	5.94E+06

The liquid effluents resulted from reactor decommissioning contain fission products and activation products and also actinides. The activity inventory of source 1 (see table 2) was comparatively analyzed with the similar one presented by Tuca et.al. [1]. It can seen that the liquid effluents contain ^{54}Mn in addition and it is a significant increasing of activity concentration at about 3.52 times for ^{60}Co, 1.19 times for ^{134}Cs and 3.38 times for ^{137}Cs as a result of reactor decommissioning progress. These effluents do not meet the criteria for unconditional release into the environment due to activity concentrations higher than derived emission limits (DEL). The radionuclides such as: ^{108m}Ag, ^{235}U and ^{238}U arising from primary and secondary circuit of reactor and ^{54}Mn occurs in nuclear reactor by $^{54}Fe(n,p)^{54}Mn$ reaction [14]. The ^{108m}Ag which is an activation product of ^{107}Ag [15] is mainly found in pipes alloy membranes (Ag-In-Cd) and primary coolant circuit of reactor. The half-life high value involves analysing for radioactive waste storage purpose. ^{60}Co occurs through activation of cobalt in the metallic structure of nuclear reactor.

In ponds no 2, 3 and 4 of DCNU (source 2) was found ^{152}Eu and ^{154}Eu and ^{241}Am in addition in the 4th one. Europium is accumulated in the sludge due to the high adhesion of its salt at this component and this is the explanation of his presence in sludge samples collected from the pond bottom. The occurrence of ^{241}Am in pond 4, a prevalent isotope in the nuclear waste, is explained by the possibility of a fuel failure unidentified during the operation period. A significantly higher concentration of activity in the 3rd pond is a consequence of the leakages from the fuel assemblies which were detected and confirmed, since 1994, by the presence of ^{137}Cs radionuclide. A maximum total activity of 4.9E+07 Bq was registered in 2007. The activity of ^{137}Cs in sludge samples is 3.16E+08 Bq, in 2015, and is comparable with that of water samples.

4.2 Gross Beta Measurements

In order to estimate the ^{90}Sr content of liquid effluents, there were performed gross beta activity determinations. The ^{90}Sr activity uncertainty can be evaluated in the range of 30÷60% due to the counting statistics, gamma emitters' inventory uncertainties and approximations for relatively thick samples measurement. ^{90}Sr activity calculation is done according Eq.2 and values are presented in Table 3.

Table 3: Gross beta activity values of ^{90}Sr equivalent and comparison with ^{137}Cs

Location of measurement	ρ_x (mg/cm^2)	N_β (s^{-1}/l)	$A_{^{90}Sr}$ (Bq/l)	$\frac{A_{^{90}Sr}}{A_{^{137}Cs}}$
Pond 1	17.70	1.66E+01	2.68E+00	0.12
Pond 2	3.42	2.52E+02	2.48E+02	2.94
Pond 3	2.14	3.71E+04	2.13E+04	0.49
Pond 4	18.75	3.24E+02	1.32E+02	0.24
Buffer tank	1.78	1.10E+04	1.19E+04	9.69

One can remark two groups of values: one of them refer to the four ponds (1÷4) of DCNU where ratios between ^{90}Sr and ^{137}Cs activity are rather consistent as the radionuclide originate from the spent fuel. The lowest values of ^{90}Sr and ^{137}Cs activity ratio are for the thick sludge samples. It can be considered that the self-absorption effect was underestimated, so the method can be further improved. The other group (buffer tank) has significant higher values and reflect the very different content of radionuclides in liquid effluents arising from reactor decommissioning process (e.g. primary and secondary circuit, cooling pond for nuclear spent fuel, biological protection, wet cutting of reactor component, decontamination of tools, equipment's and workers).

The evaluation of ^{90}Sr activity becomes important as it is known that ^{90}Sr occurs with ^{137}Cs in the burned fuel and has a high radiotoxicity for internal exposure, at least two times higher than that of ^{137}Cs [16]. The comparison between the gross beta and gamma-ray spectrometry analysis is a proper way to improve the method used in decommissioning process for the liquid effluents monitoring.

4.3 The Future Use of Results

The comparative analysis of activity concentrations of each radionuclide (see Table 2 and Table 3) with the derived emission limits in force, calculated for nuclear reactor decommissioning by Tuca et.al.[1] shows the radioactive liquid effluents do not meet the criteria for unconditioned release into the environment. Therefore, it is necessary to perform radiological characterization of effluents and worker risk assessment during the liquid waste transfer operation to RWTP for treatment.

5 CONCLUSION

There was developed the methodology for radioactive inventory of liquid effluents arising from the decommissioning process of reactor consisting of radionuclide activities determination by gamma-ray spectrometry analysis and gross beta measurements. The radioactive inventory is qualitatively higher than previously, by ^{54}Mn and ^{241}Am radionuclide detection in water samples and also quantitatively as a result of decommissioning process progress. The sludge samples analysis emphasise that ^{108m}Ag, ^{152}Eu and ^{154}Eu are concentrated in sludge and it is necessary to recalculate the DELs for liquid effluents. The ^{90}Sr activity was evaluated for the first time through gross beta measurements and it is comparable with the ^{137}Cs activity for sludge samples resulted from source 2 (DCNU) where the radionuclides originate from the spent fuel, and is much higher for the first source (buffer tank-reactor decommissioning) which has a very different content of radionuclides. The comparison between the gross beta and gamma-ray spectrometry analysis is a proper way to improve the method used in decommissioning process for the liquid effluents monitoring. The analysis of radioactive inventory of liquid effluents is important in order to perform a safely transfer at the radioactive waste treatment plant and for radiological risk assessment of workers during this process.

6 ACKNOWLEDGMENTS

The authors offer many thanks to Dr. M. Sahagia for her suggestion and feedback when clarifications of issues were required.

7 REFERENCES

[1] C. Tuca, A. Stochioiu, M. Sahagia, D. Gurau, M. Dragusin, Assessment of derived emission limits for radioactive effluents resulted from the decommissioning activities of the VVR-S nuclear research reactor, *J. Environ. Rad.* **148**, 130-136 (2015).
[2] D. Stanga, E Ionescu, D. Gurau, Radiological characterization report of Nuclear Spent Fuel Storage (2013).
[3] C. Tuca, A. Zorliu, A., Conceptual plan for decommissioning of nuclear spent fuel storage, code: *PCD-03*, Rev. 0 (2013).
[4] D. Gurau, Gamma-ray spectrometry analysis with HPGe detector, model GEM60P4-95, *PC-DEZ-408* control procedure (2011).
[5] A. Pantelica, Method for measurements of gamma emitting radionuclide activity contained in environmental samples and materials, control procedure (2013).
[6] L. Done, Method for measurements of gamma emitting radionuclide activity contained in environmental samples and materials, *AC-PL-LAS-05* work procedure (2013).
[7] A. Stochioiu, A. Luca, M. Sahagia, R.M. Margineanu, I. Tudor, Quality assurance for measurements of the radioactivity in the area of the "Horia Hulubei" National Institute for Physics and Nuclear Engineering, IFIN-HH, *J. Environ. Rad.* **112**, 4-7 (2012).
[8] D. Radu, D. Stanga, O. Sima, ETNA software used for efficiency transfer from a point source to other geometries, *Appl. Rad. Isot.* **67 (9)**, 1686-1690 (2009).
[9] M. Sahagia, L. Grigorescu, Water equivalent standard sources, *Nucl. Instr. and Meth in Phys. Res. A* **312**, 236-239 (1992).
[10] M. Sahagia, A.C. Razdolescu, A. Luca, R. Macrin, R., Volume standard sources in soil matrix, *Rom. J. Phys.* **42 (9-10)**, 659-663 (1997).
[11] A. Luca, B. Neacsu, A. Antohe, M. Sahagia, Calibration of the high and low resolution Gamma-ray spectrometers, *Rom. Rep. Phys.* **64 (4)**, 968-972 (2012).
[12] A. Stochioiu, Gross alpha, beta and gamma activity monitoring for soil and sediment samples, PL-UMRM-03 work procedure (2012).
[13] J.E. Martin, „*Physics for Radiation Protection"*, A Handbook, second Edition, Completely Revised and Enlarged, Ed. Wiley-WCH Verlag Gmbh &Co. KGaA, ISBN 3-527-40611-5 (2008).
[14] M. Iqbal, M. Ejaz, Production of manganese-54 in a nuclear reactor, *The International Journal of Applied Radiation and Isotopes* **27(4)**, 258-260 (1976).
[15] L. Ferreux , M.C. Lepy, M.M. Be, H. Isnard, Photon emission intensities in the decay of ^{108m}Ag and ^{110m}Ag, *Appl. Rad. Isot.* **87**, 101-106 (2014).
[16] CNCAN, Fundamentals Norms for Radiological Safety, NSR-01, Approved through the 14/24.01.2000 CNCAN President Order, published in the 404/29.08.2000 Official Bulletin (part I) (2000).

Discussion on practical application of radiation protection system for radioactive waste management in existing exposure situations

Daisuke Sugiyama*, Takatoshi Hattori

Radiation Safety Research Center, Nuclear Technology Research Laboratory, Central Research Institute of Electric Power Industry (CRIEPI), 2-11-1 Iwado-Kita, Komae, Tokyo 201-8511, Japan.

Abstract. For environmental remediation after nuclear accidents, it is important to establish a radiation protection system for waste management in existing exposure situations involving contamination resulting from the release of radionuclides because the target dose level in existing exposure situations can be reduced by using a reference level selected from a band of more than 1 mSv/y order. This paper provides key aspects in establishing a reasonable radiation protection framework for waste management in existing exposure situations: universal recognition of the definition of 'exposure from waste management containing a radiation source specified in an existing exposure situation' for the adoption of intermediate reference levels, the selection of stepwise reference levels according to the progress of the remediation process and translation of the concept of the existing exposure situation into regulations.

KEYWORDS: radiation protection; radioactive waste management; existing exposure situation; intermediate target dose.

1 INTRODUCTION

Remediation processes are being carried out in the contaminated areas resulting from the accident at the Fukushima Daiichi nuclear power plant on 11th March 2011. Although there has been a significant reduction in the ambient radiation dose in the contaminated area as a result of decontamination and the natural decay of radionuclides, some areas should be recognized in the existing exposure situation as having residual contamination and a higher dose level than the normal background. There is a desire to reduce exposure to levels that are similar to those considered as normal in most existing exposure situations [1]. It should also be recognized that large amounts of contaminated wastes and substances have been accumulated as a result of environmental remediation that have to be appropriately managed under the concept of radiation protection. A framework of radiation protection in which intermediate reference levels for waste management are adopted gradually according to the progress in reducing the existing ambient dose has been proposed [2]; however, it is being realized that there are some aspects that should be discussed in the practical application of the framework on the basis of the experience after the accident at the Fukushima Daiichi nuclear power station.

In this paper, key aspects are discussed, namely, the designation of the existing exposure situation, exposure from waste management containing radiation sources in the existing exposure situation and the practical adoption of stepwise target dose levels, following a review of the proposed framework of radiation protection for waste management in existing exposure situations.

2 FRAMEWORK OF RADIATION PROTECTION FOR MANAGEMENT OF RADIOACTIVE WASTES AND SUBSTANCES IN EXISTING EXPOSURE SITUATIONS

In areas contaminated with radioactive materials under existing exposure situations, such as in the aftermath of a nuclear accident, remediation activities are carried out to reduce the individual dose. In decontamination, substances contaminated with radionuclides are removed and collected, and regarded as radioactive wastes. The generation of radioactive wastes is inevitable in decontamination, which means that the management of radioactive contaminated substances (hereafter radioactive wastes) must be regarded as an integral part of the remediation process, and can be justified when the

* Presenting author, e-mail: daisukes@criepi.denken.or.jp

remediation activity progresses and the individual dose in the existing exposure situation is reduced by the management of radioactive wastes. The existing ambient dose averted by decontamination should be optimized by taking the projected exposure from the accompanying management of radioactive wastes into account.

A radiation protection framework for the management of radioactive wastes in existing exposure situations has been proposed by referring to some of the related ICRP recommendations as follows [2]:

(1) The reference level for the management of radioactive wastes as a source-related restriction is selected to be below the reference level selected for the existing annual ambient dose in the environment.
(2) Intermediate reference levels are adopted gradually according to the progress in reducing the existing ambient dose in the environment by taking into account the practicability of the management of radioactive wastes and remediation including the participation of stakeholders.
(3) In the planning and execution of the management of radioactive wastes, it is ensured that the estimated dose from the management of radioactive wastes does not exceed the selected reference level.
(4) If the existing annual ambient dose and the reference level for the management of radioactive wastes are reduced to a certain level corresponding to the normal background level as the remediation proceeds, intervention can be exempted.
(5) A reference level of 1 mSv/y may be used as a target value in the decision of the final closure of the repository.

Figure 1 shows the relationship between the estimated exposure resulting from the management of radioactive wastes and the existing ambient dose averted by remediation that produces wastes under the proposed framework. In the remediation, the existing dose will be reduced progressively if the estimated dose resulting from the management of radioactive wastes is less than the dose averted by selecting a reference level for the waste management below the reference level for the existing ambient dose.

Figure 1: Relationship between estimated dose resulting from radioactive waste management and averted existing ambient dose in remediation using reference levels [2].

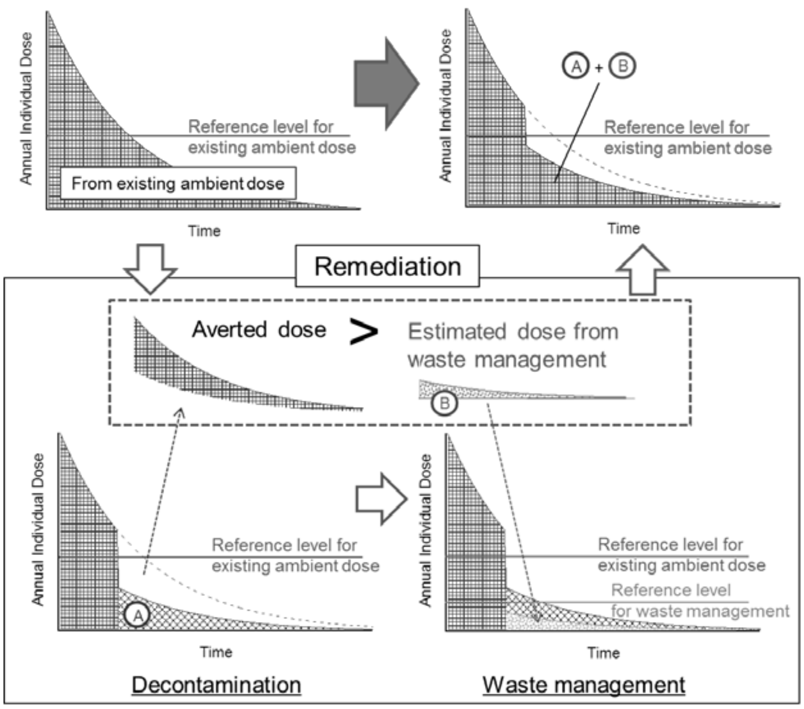
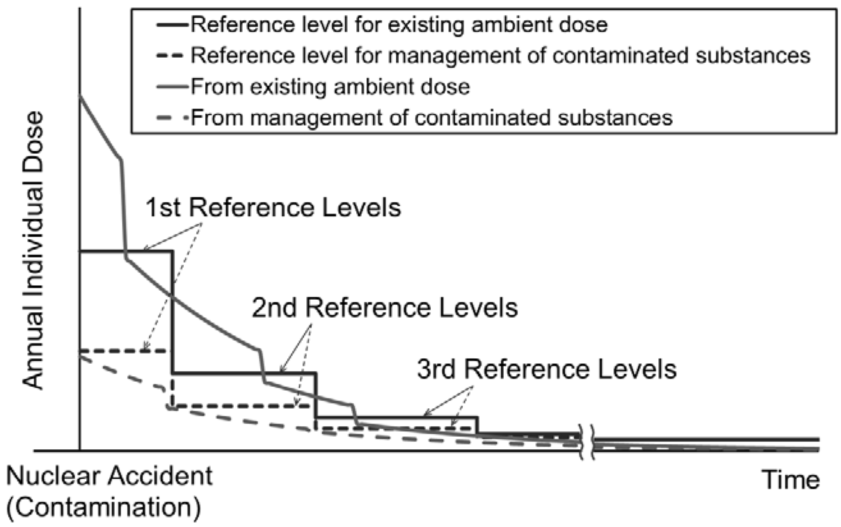


Figure 2 illustrates the concept of the reduction in the annual individual dose in environmental remediation including the management of radioactive substances in existing exposure situations under the proposed framework. The first reference level for the existing annual ambient dose is selected and the first reference level for the management of radioactive wastes is selected to be below the reference level for the ambient dose. When the existing annual ambient dose is reduced to below the first reference level by remediation activities including decontamination, the second reference levels for the existing annual ambient dose and for the management of radioactive wastes are selected to be lower than the first values. These procedures are repeated until the existing annual ambient dose is reduced to the normal dose level.

Figure 2: Conceptual diagram of reduction in individual dose in existing exposure situations (modified from Sugiyama and Hattori [2]).



3 DISCUSSION ON KEY ASPECTS

3.1 Exposure from waste management containing radiation sources specified in existing exposure situations

In existing exposure situations, remedial actions are carried out to reduce the annual individual dose, and the management of radioactive wastes (i.e., temporary and long-term storage, reprocessing and disposal) must be regarded as an integral part of the strategy for reducing exposure in remediation. This means that the radiation protection against exposure from the management of radioactive wastes under environmental remediation activities should be optimized under the framework of radiation protection in an existing exposure situation, in which the reference level is used as a tool in the optimization of protection to ensure that all exposures are kept as low as reasonably achievable taking into account societal and economic factors [1]. However, there has only been a radiation protection system for planned exposure situations with a normal background radiation dose level involving the planned introduction and operation of sources, as provided in ICRP Publication 77 [3] and Publication 81 [4], since exposure from waste management has usually been recognized as a planned exposure. Adoption of the currently available radiation protection system for waste management in planned exposure situations for remediation in contaminated areas in an existing exposure situation, where related radiation sources exist, would not be effective or practical if the removal and collection of the radioactive contaminated substances are restricted owing to limitations on the management of wastes under unreasonable criteria.

The above discussion suggests that the definition of ‘exposure from waste management containing radiation sources specified in an existing exposure situation’ should be universally recognized so that the framework of radiation protection in an existing exposure situation adopting intermediate reference levels selected from an annual dose band of 1-20 mSv can be appropriately implemented. The proposed framework [2] states that the reference level for the management of radioactive wastes as a source-related restriction should be selected to be below the reference level chosen for the existing annual ambient dose in the environment because radioactive waste management is carried out to reduce the existing annual individual dose. In other words, the exposure resulting from radioactive waste management should be optimized as an integral part of environmental remedial actions.

3.2 Adoption of stepwise target dose levels

During the decontamination after the accident at the Fukushima Daiichi nuclear power plant, a huge volume of removed soils and wastes containing radionuclides has been generated[1]. In the IAEA report [5], it is regarded that the application of a low reference level has the effect of increasing the quantity of contaminated materials generated and the cost of remediation activities.

A guideline published by the government has provided a value of 100,000 Bq/kg for the radionuclide concentration of cesium, which was calculated on the basis of an approach that the radiation exposure of residents living in the vicinity of the waste disposal facilities should be restricted to below 1 mSv/y in the operational phase and to below 10 μSv/y after the termination of institutional control [6]. The guideline was based on the radiation protection system for planned exposure situations with a normal background radiation dose level (e.g., in ICRP Publication 77 [3] and Publication 81 [4]), under which compliance with the target annual dose of 1 mSv for public exposure has to be demonstrated. However, if a single target value based on the radiation protection standard in planned exposure situations is applied to all the processes in the management of contaminated substances in existing exposure situations, there is some concern regarding the environmental remediation that the reduction in the existing ambient dose would not progress satisfactorily or would be delayed owing to limitations on the generation of waste and the removal of contaminated soil, and plans for the management of contaminated substances may be made economically and technically impractical. In view of the principle of optimization, i.e., individual doses should be kept as low as reasonably achievable taking economic and social factors into account, it is reasonable to recommend that an appropriate guideline harmonizing with the stepwise approach of radiation protection in existing exposure situations is established so that intermediate reference levels can be selected progressively according to the progress of the remediation process in the operational phase of waste management (including disposal) facilities.

Meanwhile, it is also recognized that the adoption of several intermediate reference levels for waste management is not necessarily reasonable in the practical design and management of facilities for waste disposal as stepwise refinements to the facility, such as additional barriers, would increase the burden on both regulatory bodies and operators. There would also be difficulties in public consultation. One possible way to overcome this would be the adoption of a generic criterion for waste disposal, which is the final step in waste management, so that no further active control is necessary after the closure of the repository. Therefore, regarding the final target for waste management, the proposed framework recommends that a reference level for the source-related dose attributable to radioactive waste disposal is set at an equivalent level to the generic intervention exemption level on the order of 1 mSv/y recommended in ICRP Publication 82 [7].

[1] The volume of contaminated soil and waste to be managed at the planned interim storage facility is estimated to be more than 10 million m^3 [6].

It is essential to establish environmental remediation plans including those for waste management together with relevant stakeholders living in existing exposure situations because the target dose levels have to be selected according to the progress in reducing the individual dose in the integrated plan of environmental remediation. Under an existing exposure situation, the related inhabitants have the right to participate in decision making in the selection of targets in actual environmental remediation.

3.3 Designation of existing exposure situation

Areas contaminated by the accident at the Fukushima Daiichi nuclear power station are categorized into the "Special Decontamination Area" and the "Intensive Contamination Survey Area" under the "Act on Special Measures Concerning the Handling of Radioactive Pollution" [8]. The Special Decontamination Area is constituted of the former Restricted Area (within 20 km from the NPS) and the former Deliberate Evacuation Area (with a risk of the annual cumulative dose exceeding 20 mSv). The Intensive Contamination Survey Area comprises areas in which an additional annual exposure dose of over 1 mSv was observed. Decontamination has been in progress in both areas [9]. The long-term goal of decontamination has been stated as a reduction of the exposure dose to 1 mSv/y or less, with stepwise and rapid reductions of the dose in areas with additional exposure exceeding 20 mSv/y. However, the current regulatory framework does not explicitly contain the concept of the existing exposure situation, although the basic concept has been provided by ICRP recommendations [1] and IAEA safety standards [10].

To apply the stepwise approach discussed above, the concept of the existing exposure situation will have to be appropriately translated into regulations. National authorities should play an important role in designating areas requiring remedial actions in the existing exposure situation because the remedial actions in the existing exposure situation would also influence the situation outside the affected areas. It has been recommended that a judgment by a regulatory authority is required to decide whether or not a component of an existing exposure is amenable to control, taking the controllability of the source or exposure, as well as economic, social and cultural factors into account [1]. Thus, it is suggested that another aspect in addition to the level of the radiation dose, i.e., the origin of the radiation source, should be considered to recognize existing exposure situations. This is because it is desirable to return to the normal situation and remedial actions are justified if they are likely to be beneficial to individuals and also to society in the case of exposure from contamination due to the release of a radiation source from an accident. The situation accompanying contamination due to accidents can be recognized as an existing exposure situation even with a very low additional annual individual dose. This consideration makes it possible to apply the proposed radiation protection framework with a stepwise approach to existing exposure situations having relatively high background radiation due to naturally occurring radioactive materials.

In general, it has been difficult to employ a higher and less stringent value than those in the radiation protection criteria established for a normal situation, even in a contaminated area with an elevated background radiation level, and modification of the criteria is not easy. Nevertheless, it is strongly recommended that a flexible approach is introduced into the environmental remediation strategy in existing exposure situations according to the progress in reducing the existing ambient dose. The dissemination of a practical radiation protection framework introducing stepwise intermediate reference levels in existing exposure situations should be a key element of the strategy.

4 CONCLUSION

Experience after the Fukushima Daiichi nuclear accident has led us to recommend that a stepwise approach is established for the management of radioactive contaminated wastes in the existing exposure situation. Important aspects that should be embodied in the regulatory system are definition of the exposure resulting from waste management involving radiation sources specified in the existing exposure situation, designation of the existing exposure situation and the adoption of stepwise reference levels. A reference level of 1 mSv/y may be used as a target value for the final closure of the repository.

5 ACKNOWLEDGEMENT

This study was carried out under the Radiation Safety Research Project of the Central Research Institute of Electric Power Industry (CRIEPI).

6 REFERENCES

[1] ICRP, 2007. The 2007 Recommendations of the International Commission on Radiological Protection, ICRP Publication 103, Ann. ICRP 37 (2-4).

[2] Sugiyama, D., Hattori, T., 2013. Radiological Protection from Radioactive Waste Management in Existing Exposure Situations Resulting from a Nuclear Accident, Radiation Protection Dosimetry, 153, 64-73.

[3] ICRP, 1997. Radiological Protection Policy for the Disposal of Radioactive Waste, ICRP Publication 77, Ann. ICRP 27 (S).

[4] ICRP, 1998. Radiation Protection Recommendations as Applied to the Disposal of Long-lived Solid Radioactive Waste, ICRP Publication 81, Ann. ICRP 28 (4).

[5] IAEA, 2015. The Fukushima Daiichi Accident Report by the Director General.

[6] Ministry of the Environment, 2012. Management of off-site Waste Contaminated with Radioactive Materials due to the Accident at Fukushima Nuclear Power Stations. http://josen.env.go.jp/en/documents/pdf/documents_04.pdf?130111.

[7] ICRP, 1999. Protection of the Public in Situations of Prolonged Radiation Exposure, ICRP Publication 82, Ann. ICRP 29 (1-2).

[8] Ministry of the Environment, Measures for Decontamination of Radioactive Materials Discharged by TEPCO'S Fukushima Daiichi NPS Accident. https://josen.env.go.jp/en/#top02.

[9] Ministry of the Environment, 2016. Progress on Off-site Cleanup and Interim Storage in Japan. https://josen.env.go.jp/en/pdf/progressseet_progress_on_cleanup_efforts.pdf?150918.

[10] IAEA, 2014. Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, GSR Part 3.

The Role of Radiation Protection Considerations in Decommissioning
A brief look at safety culture in Iraq and Ukraine

Eric K. Howell[a*], Rodolfo Avila[b], John Rowat[c], David Kenagy[d], Ronald Chesser[e]

[a]Facilia Projects GmbH, Vienna, Austria.
[b]Facilia AB, Stockholm, Sweden.
[c]International Atomic Energy Agency, Vienna, Austria.
[d]United States Department of State, Washington DC, USA.
[e]Texas Tech University, Lubbock, Texas, USA.

Abstract. Like any other operation resulting in potential exposure to radiological (and other) hazards, radiation protection measures are integral to the safety of decommissioning activities. The International Atomic Energy Agency, International Commission on Radiological Protection, and other institutions provide clarity on such radiation protection principles and requirements through the issuing of international standards of safety. The International Atomic Energy Agency, under its mandate, cooperates closely with international experts and organizations in development of capacities in radiation protection specifically during decommissioning works. This paper will review the radiation protection principles and requirements for decommissioning as defined by international standards of safety as well as explore case studies of application of such principles through International Atomic Energy Agency and European Commission programmes. The Iraq Decommissioning Project involved cooperation of over 20 countries in re-establishing radiation protection capacities in Iraq following the removal of the former regime in 2003. Iraq was faced with the challenge of decommissioning its destroyed former nuclear programme with a work force, which had been isolated from the international community for decades. The European Commission is presently assisting Ukraine with the decontamination, decommissioning, and remediation of the Pridneprovskiy Chemical Plant in the eastern portion of the country. This site was once home to significant uranium processing activities, which have left a legacy requiring clean up. Given the recent instability in Ukraine, the responsible organizations have been faced with unique challenges in this task. The aforementioned case studies will be explored, highlighting challenges, successes, and lessons learned from implementation of radiation protection principles during the unique decommissioning activities. The importance of establishment and maintenance of a sound safety culture will be examined in the planning and implementation of future unique decommissioning tasks in developing countries.

KEYWORDS: decommissioning; remediation; Iraq; Ukraine; safety standards; safety culture.

1 INTRODUCTION

Radiation protection considerations are essential to the safe conduct of decommissioning activities. The International Atomic Energy Agency (IAEA), European Union (EU), International Commission on Radiological Protection (ICRP), Western European Nuclear Regulators Association (WENRA), World Bank Group, the national legislations of each country, and many other institutions have each contributed to the international standards and best practices for radiation protection [1-3]. These standards focus on nearly all types of facilities and activities, which could be undertaken in the field of decommissioning and cover the range of topics from safety management to strategies and planning of decommissioning activities to the required safety verifications to the actual conduct of the decommissioning work. The international community has also published many case studies where the safety standards have been applied resulting in successful protection of people and the environment from radiological hazards [4-6]. Extreme environments and situations require special attention when planning and implementing such activities. This paper will explore the challenges, successes, and lessons learned from two of these case studies: The Iraq Decommissioning Project (IDP) and the Pridneprovskiy Chemical Plant (PChP) in Ukraine.

* Presenting author, e-mail: eric.howell@facilia.se

2 IRAQ DECOMMISSIONING PROJECT

2.1 Background

The decommissioning and remediation of Iraq's nuclear facilities and sites posed many unique and difficult challenges. Many of the facilities had been damaged or destroyed during warfare and looted due to a breakdown of security. Moreover, beginning in 2004, there was an insurgency that made civilian operations extremely difficult throughout the course of the project. Governmental organization was restructured, experienced management had been displaced, technical capability had been diminished, and many technical records had been lost. All of this work had to take place against the background of a breakdown in national infrastructure (e.g., electricity, water, communication, transportation). Further, Iraq had been isolated from the international nuclear community for over two decades. Some of the legal infrastructure was inconsistent with international standards and there was no effective regulatory oversight of Iraq's nuclear facilities [4]. Fig. 1 presents before and after photographs of Iraq's IRT-5000 research reactor. Fig. 2 indicates the geographical locations of Iraq's former nuclear facilities.

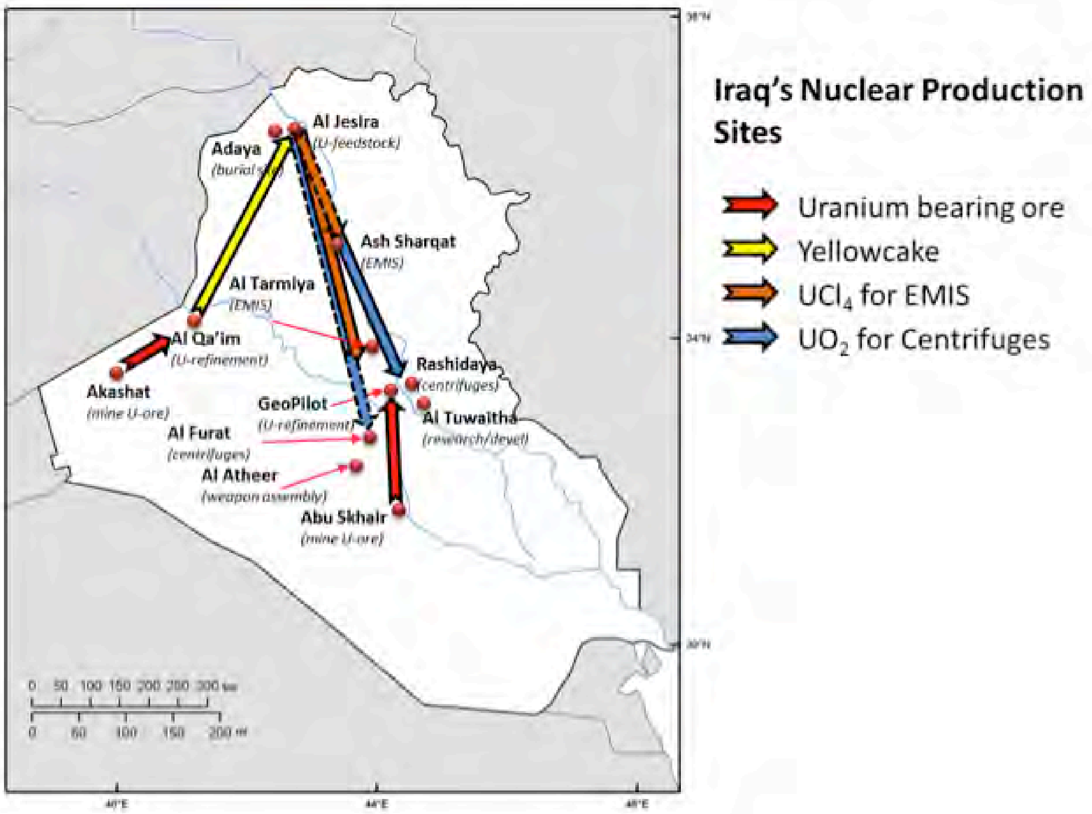
Figure 1: Iraq's IRT-5000 research reactor during operation (left) and after being bombed in 1991 (right).

The Iraq Decommissioning Project was initiated in 2006 after a request from the Interim Government of Iraq (GoI) to the IAEA in December 2004. The project was supported by U.S. extra-budgetary funding for IAEA, and included a significant bilateral programme between the U.S. and Iraq that contributed to the aims of the project. The programme continued from 2006 to 2013 with 20 countries and the European Commission (EC) providing financial and in-kind support through governmental and commercial organizations.

2.2 IDP Objectives

The principal objective of the project was to reduce the overall radiological risk to the public and the environment through decommissioning the former Iraq nuclear complex, waste management, and remediation of contaminated areas and disposal sites. The approach taken was to build capacity and expertise within the country. This was considered to be the most appropriate pathway to sustainability [4].

Figure 2: The geographical locations of Iraq's former nuclear facilities.



2.3 IDP Accomplishments

The extra-budgetary funding provided by the US Government ended in 2013, resulting in the closing of the IDP. Although work has since continued through alternative funding mechanisms, the principal achievements of the IDP included [4]:

- The national infrastructure for decommissioning, remediation, radioactive waste management, laboratory analysis, and quality management improved greatly since the inception of the project.
- There was a marked increase in Iraq's ability to independently implement the multitude of objectives of the project.
- Scientists, engineers and technicians in Iraq successfully and independently decommissioned four of the former nuclear facilities in and around Baghdad (Geo-Pilot Plant, Active Metallurgy Laboratory, Italian Radioisotope Production Facility, and Rashdiya).
- These decommissioning efforts were preceded by the establishment of regulatory oversight for licensing and inspection, the development of detailed decommissioning and waste management plans, and the improvement of technical capacity in radiation protection and the use of special equipment through various training exercises.
- The IDP also assisted Iraq in developing decommissioning and remediation plans for five of the higher-risk facilities (Radiochemistry Laboratory, Tammuz-2 Reactor, IRT-5000 Reactor, Fuel Fabrication Facility, and the Adaya Burial Site). These plans were approved by the relevant regulatory body, the Ministry of Environment (MoEN) in Iraq, and implementation of decommissioning for some of these facilities was initiated.

- The IDP contributed to the strengthening of regulatory infrastructure in Iraq. A law establishing a national atomic energy commission was enacted by the Iraqi Parliament and a law establishing a unified and competent regulatory authority was submitted to the Council of Ministers.
- Through the IDP, Iraq also established regulations for radiation protection and licensing, decommissioning and remediation, radioactive waste management and disposal, transport of radioactive materials, and security of radioactive sources and associated facilities.
- A marked increase in regulatory competence was evidenced by the review and assessment, licensing and inspection activities that led to release of decommissioned nuclear sites.
- Work was initiated with the Government of Iraq to implement a quality management system (QMS) within the Iraq Ministry of Science (MoST). The QMS served as a requisite for the international accreditation (ISO 17025) of the newly built Al-Tuwaitha Radioanalytical Laboratory (ATRL).
- The IDP was also integral to the equipping and staffing of the ATRL with state-of-the art instrumentation and trained personnel.

2.4 IDP Conclusions

The IDP demonstrated:

- A unique and successful approach for international post-conflict cooperation that provided reconstruction and capacity building over a broad spectrum of Iraqi governmental organizations while achieving positive impacts on public health and environmental concerns.
- That IAEA provides an effective mechanism for international cooperation.
- The effective use of extra-budgetary IAEA funds in initiating a project, which was then fully supported by the international community, effectively multiplying the overall resources applied to the project.

3 PRIDNEPROVSKIY CHEMICAL PLANT

3.1 Background

The Pridneprovskiy Chemical Plant was established in 1948 and was a multi-processing plant, which carried out the following industrial activities [6]:

- Processing of uranium-containing blast furnace slag, uranium ores and concentrates (1948-1991);
- Ion-exchange resins (1965-present);
- Apatite ore processing to produce rare-earth elements and mineral fertilizers (1974-1990);
- Processing of raw materials containing zirconium to obtain pure compounds of zirconium and hafnium (1982-2012); and
- Processing of North African phosphorus-containing raw materials to obtain fertilizers (1991-present).

In 1992, the number of plant personnel was 7,200 people including 1,000 staff working at uranium processing facilities. Uranium ore processing activities at the plant site ceased after disintegration of the Soviet Union in 1991 [6].

The PChP was restructured in 1996, providing access to private enterprises for various, especially chemical, activities (total of about 1,000 non-radiological workers). These private enterprises use many of the buildings, which housed the previous uranium processing activities. These buildings never underwent controlled and licensed decontamination or decommissioning. The site also consists of several tailings facilities, some of which have degraded or non-existent covers. Fig. 3 shows the geographical distribution of hazardous objects at the Pridneprovskiy Chemical Plant site. Fig. 4 indicates the overall gamma distribution levels at the Pridneprovskiy Chemical Plant industrial sector.

Figure 3: The geographical distribution of hazardous objects at the Pridneprovskiy Chemical Plant site including contaminated buildings and areas in addition to multiple tailings facilities.

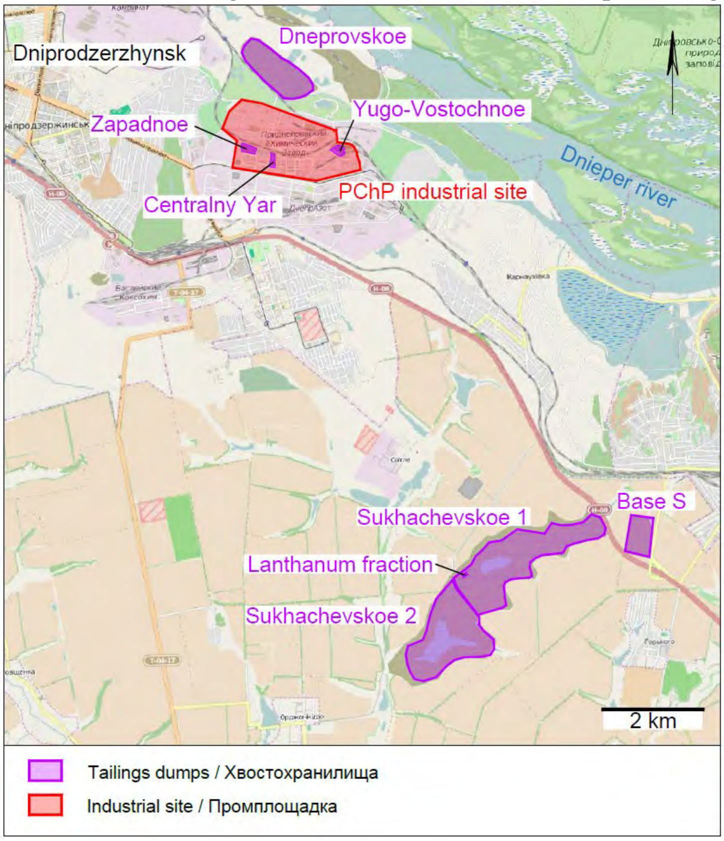
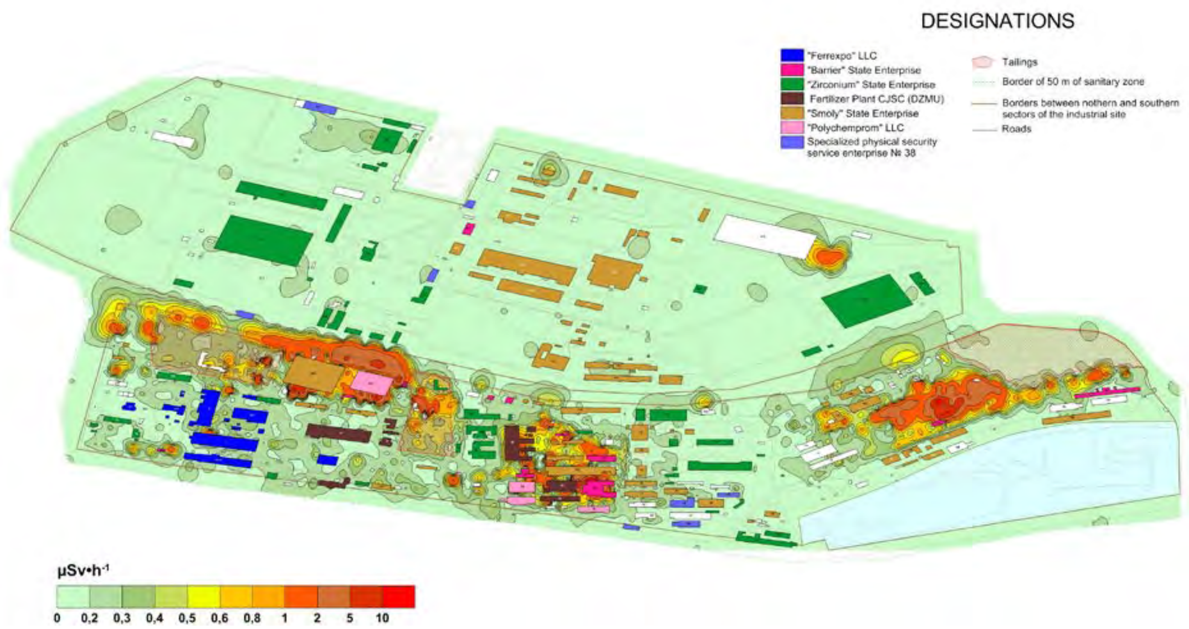


Figure 4: The overall gamma distribution levels at the Pridneprovskiy Chemical Plant industrial sector.



3.2 PChP Project Objectives

During the last decade, national PChP site remedial programmes have been supported by a number of international projects including technical assistance projects funded by the Swedish government, IAEA, and European Commission. The overall objective of the current EC project is to implement modern, effective methodologies and tools for planning remediation activities and to develop the method (a strategy and technology(ies)) for the remediation activities at the PChP based on international best practice. The specific objectives are as follows [5]:

- Compilation of an up-to-date inventory of tailings, buildings and site data describing radiological and chemical composition, physical condition (e.g. volume, area), etc.;
- Development of recommendations regarding the application of best international practice to the planning and implementation of remediation activities at the former Uranium facilities at the PChP site while, at the same time, considering the Ukrainian regulatory regime and readily available technologies;
- Assessment of the radiological risk for non-radiological workers on site and the population around the site;
- Development of a method (a strategy and technology (ies)) for the remediation activities at the PChP site; and
- Development of the decommissioning and remediation plans for selected facilities at the PChP site.

3.3 PChP Project Accomplishments

Although decommissioning and remediation work has not yet commenced at the PChP site, the site has been extensively characterized, an overall decommissioning and remediation strategy has been developed, and preliminary remediation designs have been prepared for the highest priority objects. Future support provided by the international community will prepare the site for decommissioning and remediation, evaluate potential radioactive waste management sites, provide support to revision and updating of the legal and regulatory framework, develop final decommissioning and remediation plans, and provide for at least initial implementation of these plans.

3.4 PChP Project Conclusions

Compared to similar projects, the international and national stakeholders are very well engaged and funding mechanisms are in place for years of work to come. Due to inputs from the international community, there is a good understanding of the situation and hazards at the PChP site, there have been extensive assessments of risks to public and the environment, and plans are in place moving forward toward decommissioning of the site.

4 SAFETY CULTURE

The United States Nuclear Regulatory Commission defines nuclear safety culture as “the core values and behaviors resulting from a collective commitment by leaders and individuals to emphasize safety over competing goals to ensure protection of people and the environment” [7]. The international standards of safety define a set of recommendations for building of national safety culture [1]. In both of the case studies presented herein, a severe lack of safety culture was observed. The following are topics, which have been implemented in the presented case studies that contributed greatly to the building of national safety culture in Iraq and Ukraine. These topics are essential for ensuring that people and the environment are protected in unique and challenging situations that may pose such a threat [1-3].

- Top-level buy-in: Engaging and gaining support from especially high-level decision-makers and funding agencies for mitigation of the hazards present in a given situation.
- Definitive delineation of players and their responsibilities: Ensuring that the required organizations and institutions are given the legal mandate to act in such challenging situations. Ensuring that the responsibilities of each player are very clearly defined in the national legal and regulatory framework.
- Clearly defined dose limits and constraints: Ensuring that the national legal and regulatory framework defines clearly the dose limits and constraints, which will dictate the work to be done (if not in place, then reference should be made to application of international standards of safety).
- Complete understanding of the hazards and appropriate classification of areas: A sufficient level of characterization of the hazards should be implemented in order to gain an understanding of the requirements for decommissioning and remediation. Working areas should be classified appropriately so as to protect potentially exposed individuals.
- Project planning: Detailed planning of proposed actions based upon quantitative safety justifications is essential. Project planning should include the steps to be taken from inception of the decommissioning project through to its completion and license termination.
- Capacity building: The human resources required and their capacities for conducting the proposed decommissioning work needs to be clearly defined in the project planning documents. Personnel conducting work should be properly certified and their capacities verified throughout the project.
- Appropriate barriers in place to prevent and mitigate the consequences of the hazards: According to the project plan and safety justifications, barriers need to be put in place to prevent undue exposures to people and the environment as low as reasonably achievable. The consequences of breakdown of any of these barriers should be foreseen and the potential impacts calculated.
- Safety justifications for facilities/activities: According to the IAEA safety standards, quantitative safety justifications in the form of safety cases and safety assessments should be developed and should dictate the implemented plans. The IAEA has a suite of methodologies and software tools to assist countries in development of such safety justifications.
- Optimization of protection and safety: The system should be optimized and continuously improved to limit the risk of exposure to humans and the environment.
- Radiation monitoring programme including individual dosimetry, workplace monitoring, and environmental monitoring: A monitoring programme should be in place as part of the project planning, which defines the requirements and procedures to be taken to ensure that actual exposures do not exceed those expected in the developed safety justifications.
- Review and audit: All steps in the planning and implementation phases should be independently reviewed and audited to ensure that they are in line with the proposed actions, the national legal and regulatory framework, and international standards of safety.
- Emergency preparedness: As part of the planning process, consideration should be given to preparation for and response to emergency or accident situations. Measures should be taken to minimize the possibility of these events taking place and to mitigate the impacts of the events if they do take place.
- Waste management considerations: Prior to implementation of any decommissioning plan, the management and final endpoint for the resulting waste should be considered.
- Well-established funding mechanisms: Funding mechanisms should be defined and secured to ensure completion of the work from commencement to final release of the site/facility.

5 CONCLUSION

The international community has a series of safety standards, which define the requirements and provide guidance as to the methods of planning and implementing sound safety regimes in the area of decommissioning. These documents, in conjunction with their application through a number of international projects and studies, provide insight into how these standards can be applied to the world's challenging situations. Experiences should be shared between countries through the appropriate mechanisms of international organizations such as the IAEA in order to assist countries in dealing with such challenging situations. Of integral importance is the role of safety culture in planning and implementing decommissioning work. The components of a sound safety culture, as discussed herein, may have a significant impact on the success of a decommissioning project. These components should be considered and adopted into the planning process and implementation of any such project.

6 ACKNOWLEDGEMENTS

The work presented herein is a result of cooperation between over 40 international governments and institutions as well as over 100 international experts. Special thanks must be given to the primary funding agencies: United States Department of State, United States Nuclear Regulatory Commission, European Commission, International Atomic Energy Agency, and the Governments of the United Kingdom and Spain.

7 REFERENCES

[1] IAEA, FAO, ILO, et al. 2014. Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards. IAEA Safety Standards Series No. GSR Part 3. International Atomic Energy Agency, Vienna.

[2] European Commission Coordination Network on Decommissioning, 2008. Institutional, Legal and Regulatory Aspects; Licensing and Decommissioning Plan; Radiological Protection and Industrial Safety: Final Report (September 2008).

[3] Western European Nuclear Regulator's Association (WENRA), 2011. WENRA Working Group on Decommissioning: Decommissioning Safety Reference Levels Report, Version 2.0, November 2011.

[4] IAEA. The Iraq Decommissioning Project: Eight Years of Accomplishments. Proceedings of the 2013 IAEA General Conference. International Atomic Energy Agency, Vienna.

[5] Consortium Facilia AB, Wisutec GmbH, "Development of the method for the remediation activities at the former uranium facility "Pridneprovsky Chemical Plant", Task 1 report, Inception Report", 2015.

[6] Consortium Facilia AB, Wisutec GmbH, "Development of the method for the remediation activities at the former uranium facility "Pridneprovsky Chemical Plant", Task 2 report, Current Situation", 2015.

[7] United States Nuclear Regulatory Commission (USNRC), 2016. Safety Culture. http://www.nrc.gov/about-nrc/safety-culture.html.

Calculation of Dose Rates at the Surface of Storage Containers for High-Level Radioactive Waste

Erik Poenitz[a*], Clemens Walther[a], Thomas Hassel[b]

[a]Institut für Radioökologie und Strahlenschutz, Leibniz Universität Hannover, 30419 Hannover, Germany.
[b]Institut für Werkstoffkunde, Leibniz Universität Hannover, 30823 Garbsen, Germany.

Abstract. In several countries, the high-level radioactive waste that will be disposed in deep geological formations has to be retrievable for a certain time. Exercising the option of retrieval requires an anew handling of the waste containers. Depending on the effort and the feasibility of remote handling, a certain exposure of the involved employees to ionizing radiation is caused. The estimation of the exposure requires the knowledge of the inventory of radionuclides and the radiation field around the storage containers. In this paper, concepts of storage containers for different host rocks (rock salt, argillaceous rock, hard rock) are described with an emphasis on radiation shielding. The calculation of radiation fields and dose rates at the surface of the containers are performed by particle transport simulations using the Monte-Carlo code MCNP.
The calculations show that the gamma radiation of fission and activation products can be efficiently shielded by materials like cast iron and low-alloy steel. Ductile cast iron, however, has a positive influence on the neutron moderation because of the high carbon content. The shielding of neutron radiation strongly depends on both the quantity and position of the polyethylen (PE) rods which are used as neutron moderator. PE has unfavourable features at elevated temperatures. The simulations show that the use of alternative materials with a large hydrogen content like $Al(OH)_3$ and TiH_2 can lead to significantly lower neutron dose rates at the surface of the storage container.

KEYWORDS: radiation transport calculation; final storage container; high-level radioactive waste.

1 INTRODUCTION

1.1 Research project ENTRIA

The research project ENTRIA (German acronym for: Disposal options for radioactive residues: Interdisciplinary analyses and development of evaluation principles) is engaged in finding evaluation criteria for the disposal of high-level (heat-generating) radioactive waste. The research is focused on the radioactive waste management options

- final disposal in deep geological formations without any arrangements for retrieval
- disposal in deep geological formations with arrangements for monitoring and retrieval
- long-term surface storage.

The joint research project consists of twelve institutes at German universities and research facilities as well as one partner in Switzerland. ENTRIA is analysing radioactive waste management subjects from the viewpoint of all involved academic disciplines such as natural sciences, engineering, law and social sciences.

1.2 Motivation

Exercising the option of retrieval requires an anew handling of the waste containers. Depending on the effort and the feasibility of remote handling, a certain exposure of the involved employees to ionizing radiation is caused. The estimation of the exposure requires the knowledge of the inventory of radionuclides and the radiation field around the storage containers.

[*] Presenting author, e-mail: poenitz@irs.uni-hannover.de

2 RADIATION TRANSPORT CALCULATIONS

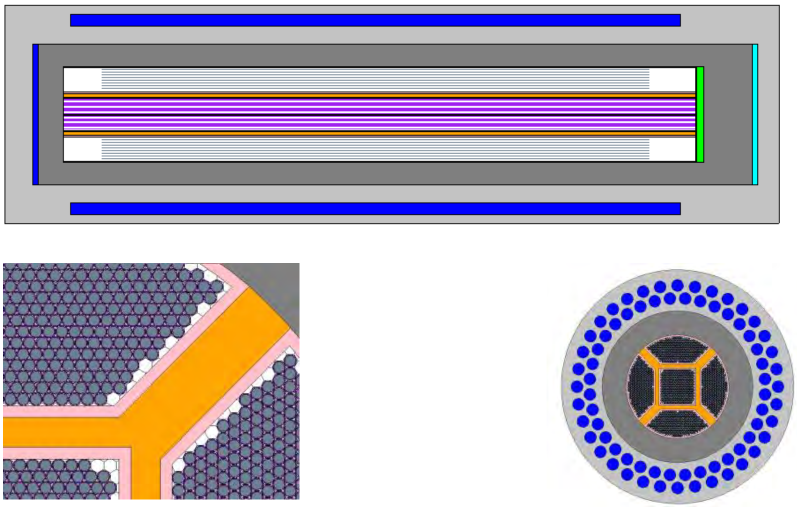
2.1 MCNP

For the calculation of radiation fields around the final storage containers, the Monte-Carlo code MCNP [1] was used. MCNP is a versatile, multi-purpose code for the coupled transport of neutrons, photons and electrons which is widely used for radiation shielding calculations. It allows the modelling of arbitrary, detailed, 3-dimensional objects.

2.2 Modelling of final storage containers for different host rocks

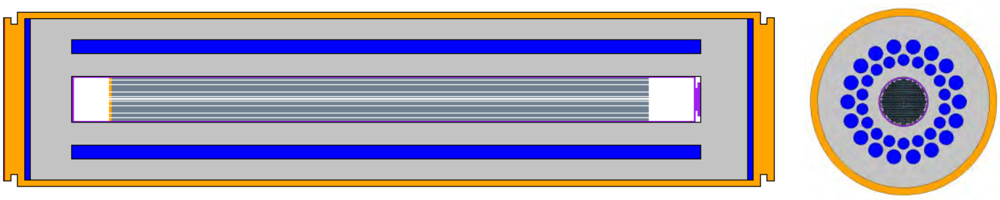
The German concept of a final storage container for rock salt (POLLUX®-10) was modelled in detail according to data taken from [2] and [3]. This model also served as an initial point for different simulations, i.e. variation of the load (spent UO_2 fuel or MOX fuel) and variation of materials of the container. In Fig. 1, the MCNP model of the POLLUX®-10 is shown.

Figure 1: MCNP model for the final storage container POLLUX®-10 for the fuel rods of 10 PWR fuel assemblies. Different colours correspond to different materials, i.e. light grey: cast iron, dark grey: steel 15MnNi6.3, blue: polyethylene, orange: copper.



With the inception of the StandAG (German acronym for: Law for the selection of a site for a repository) in 2014, clay rock/argillaceous rock and hard rock/crystalline rock are also possible host rocks for a repository site. These host rocks have lower temperature limits than rock salt. Final storage containers with a lowered heat production, i.e. smaller radionuclide inventory, are required. Currently, a load with the fuel rods of 3 PWR fuel assemblies is assumed for the final storage containers for both argillaceous rock and hard rock. Additionally, a final storage container for hard rock has to be corrosion-resistant as ingress of water cannot be avoided. Generic concepts for final storage containers for these host rocks are currently under development within the research project ENTRIA. In Fig. 2, a preliminary model of a final storage container for hard rock is depicted.

Figure 2: MCNP model for the final storage container ENCON-K for the fuel rods of 3 PWR fuel assemblies (preliminary)



3 RESULTS

3.1 Radionuclide inventory

The dose rate around a final storage container strongly depends on the radionuclide inventory. In the simulations of this work, data from [3] are taken. A load of the final storage container with spent PWR fuel rods with UO_2 as fuel and a burn-up of 55 GWd per metric ton of initial heavy metal (MTIHM) is assumed. This fits very well to a dwell time of the fuel assemblies of approximately 5 years. A decay time of 50 years is assumed. With these assumptions, the activity of the fission products ^{137}Cs and ^{90}Sr/^{90}Y per final storage container loaded with the fuel rods of 10 PWR fuel assemblies are $1.1 \cdot 10^{16}$ Bq and $6.9 \cdot 10^{15}$ Bq, respectively. The most important source of neutrons is ^{244}Cm with a half-life time of 18 years. This alpha-emitting radionuclide decays via spontaneous fission with a small probability. The overall neutron source strength per POLLUX®-10 is $1.1 \cdot 10^9$ s^{-1}.

Due to the smaller number of fuel rods for the final storage containers for argillaceous rock (ENCON-T) and hard rock (ENCON-K), the activities and neutron source strength are 30 % of the values of the POLLUX®-10.

3.2 Dose rate at the side wall of a POLLUX®-10 final storage container

The dose rate at the side wall of a final storage container for rock salt is depicted in Fig. 3. Additionally to the overall dose rate (black curve), the contributions of neutrons (red), neutron-induced gamma radiation (green) and gamma radiation by fission products and activation products (blue) are shown. The side walls with a thickness of approximately 40 cm provide an efficient shielding against the gamma radiation of fission and activation products. The dose rate is dominated by neutrons which are much more difficult to shield.

In an additional simulation, the neutron moderator (pure) polyethylene was replaced by polyethylene with a boron content of 3 %. This leads to a significantly lowered contribution of the neutron-induced gamma radiation.

Figure 3: Dose rate at the side wall of a final storage container for rock salt (POLLUX®-10)

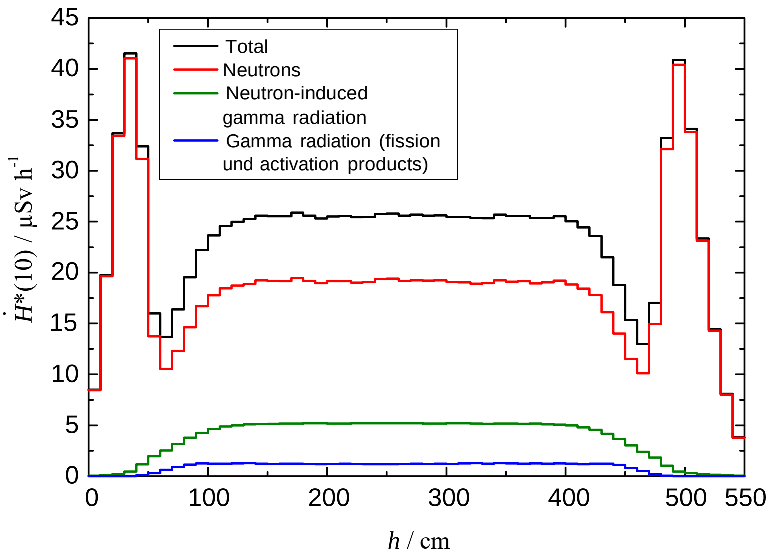
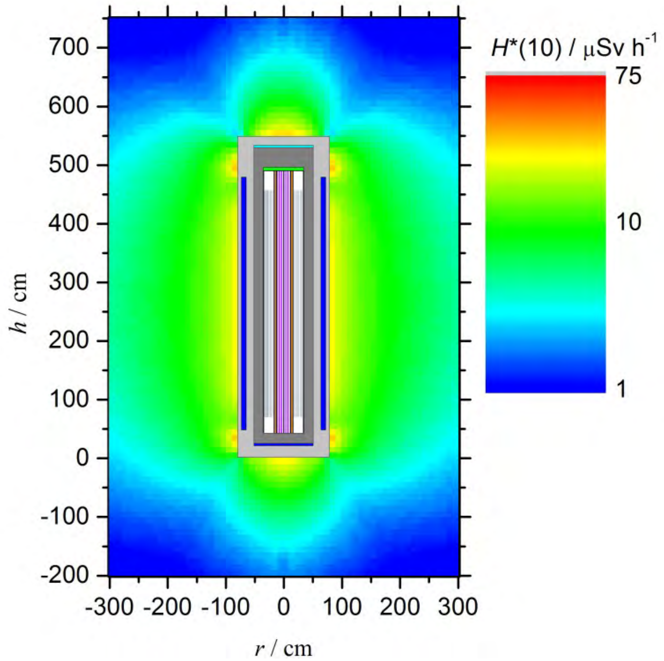


Fig. 3 also reveals that there are two small regions with an increased dose rate at the side wall. This is caused by the limited length of the neutron moderator rods. Fig. 4 shows, however, that the regions with increased dose rate are limited to the direct vicinity of the container's surface.

Figure 4: Dose rate field around the POLLUX®-10 surrounded by dry air



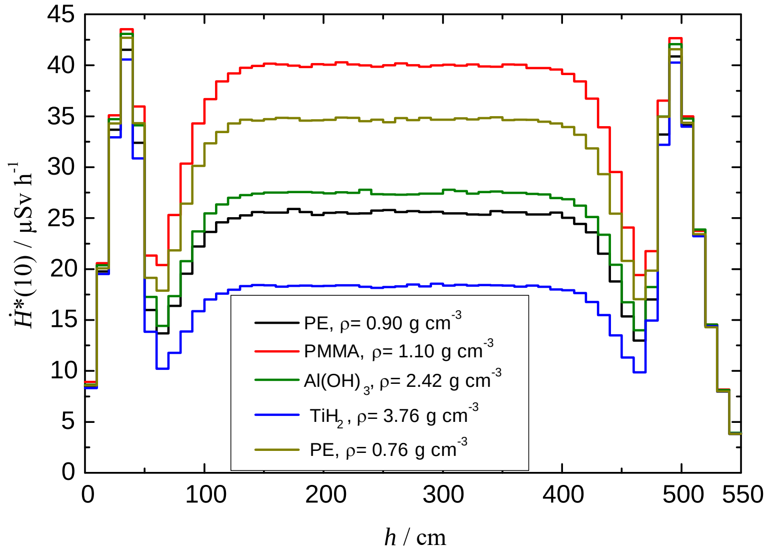
3.3 Variation of the neutron moderator

High-density polyethylen (HD-PE) is a material commonly used as neutron absorber. Important advantages are the large hydrogen content and the low costs. In the scope of final disposal of heat-producing radioactive waste in rock salt with a temperature limit of 200°C, there are also disadvantages. A strong volume increase of 20 % occurs at approx. 130°C – 146°C. The melting temperature is approx. 160°C [4].

To account for the volume increase, simulation with a significantly lowered mass density of HD-PE was carried out. Additionally, simulations for other materials which can provide an alternative to HD-PE were carried out. A constant volume of the neutron moderator was assumed. The results are depicted in Fig. 5.

The mass densities of HD-PE of 0.90 g cm^{-3} and 0.76 g cm^{-3} chosen in the calculations correspond to temperatures of approximately 100°C and 200°C, respectively. The significantly increased dose rate (yellow curve compared with black curve) shows a large influence of the mass density - and therefore the maximum temperature - on the dose rate at the side wall of the storage container.
There are polymers with higher melting temperatures than HD-PE, i.e. PMMA and PA6. These polymers, however, have lower hydrogen contents. A use of these polymers instead of HD-PE leads to larger dose rates. This is shown in Fig. 5 for PMMA (red curve).
The use of TiH_2 as neutron moderator leads to an especially small dose rate. TiH_2 is a material with a large hydrogen content and a large mass density. This material serves as both a neutron moderator and a shielding against gamma radiation. However, one has to keep in mind that the larger mass density also would lead to an increase of the mass of the POLLUX®-10 from 65 to 70 metric tons.

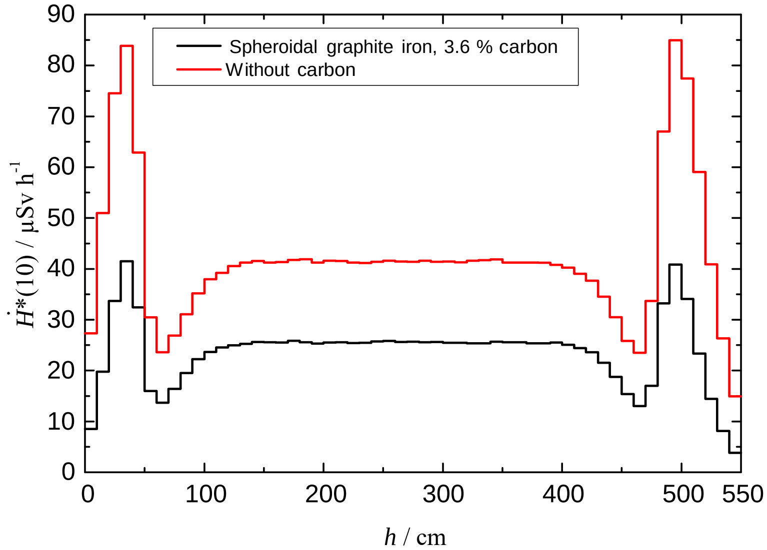
Figure 5: Dose rate at the side wall of a POLLUX®-10 for different neutron moderators



3.4 Carbon content in spheroidal cast iron

Cast iron is a material with large (2.9 % - 3.7 mass-%) graphite content. In the simulations shown in Fig. 2 to Fig. 5, spheroidal graphite iron with 3.6 % graphite content is assumed. Graphite is widely used as a neutron moderator. For the final storage container in other countries, i.e. the Swiss BE-ELB or the Swedish KBS-3, other shielding materials with only a small or no graphite content are foreseen. The influence on neutron moderation in a final storage container was investigated by a separate Monte-Carlo simulation. For this purpose, the graphite content in the cast iron was reduced to 0 % in this simulation. The dose rate at the side wall of the final storage container is depicted in Fig 6.

Figure 6: Influence of the graphite in cast iron on the neutron moderation



The assumption of a shielding container without graphite leads to a significantly increased dose rate at the side wall of the final storage container. In the small regions with increased dose rate at h = 40 cm and h = 500 cm, the dose rate is larger by a factor of 2. This result leads to the choice of a combination of shielding materials for the generic concept of a final storage container for hard rock which is shown in Fig. 2:

- an outer layer of 50 mm copper as protection against (anaerobic) corrosion;
- a thick layer of cast iron for shielding against both gamma and neutron radiation.

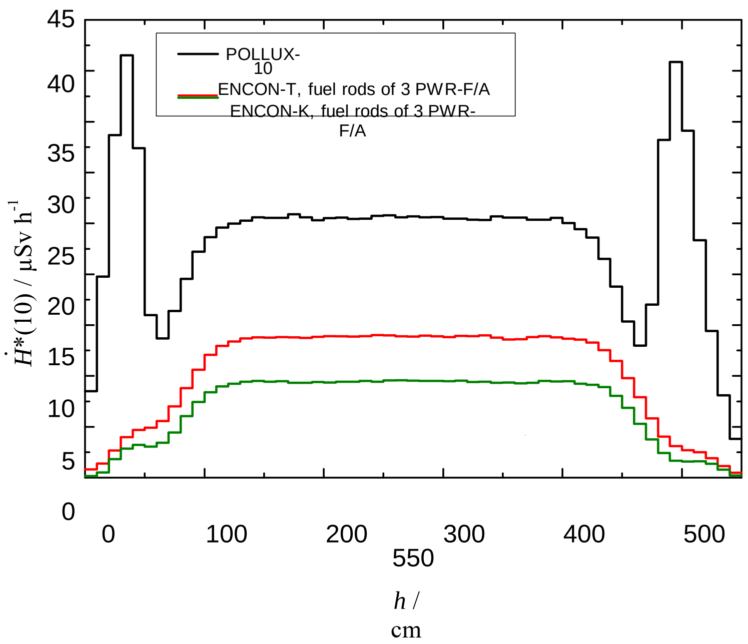
3.5 Generic concepts for final storage containers for argillaceous rock and hard rock

For the disposal of high-level radioactive waste in argillaceous rock or hard rock, final storage containers with a smaller heat production are required. Reasons are the lower temperature limit of approximately 100 °C and the lower thermal conductivity of the host rock and the backfill material Bentonit [5]. Generic concepts for final storage containers for argillaceous rock (ENCON-T) and hard rock (ENCON-K) are currently under development in the research platform ENTRIA. Features with an influence on radiation shielding are:

- Smaller load: fuel rods of 3 PWR fuel assemblies are currently assumed. Compared with the POLLUX-10 for rock salt, more final storage containers are needed;
- Neutron moderator rods are slightly longer. With the moderator plates at bottom and top, this provides a better coverage with neutron moderator;
- The outer ring of moderator rods has a larger diameter. This leads to improved neutron moderation;
- Currently, the wall thickness at the side wall equals to the one of the POLLUX®-10.
- Flanges for handling tools at bottom and top lead to increased wall thickness and lower dose rates at these surfaces;

In Fig. 7, a preliminary result of a simulation of the dose rate at the side wall is shown.

Figure 7: Dose rate at the side wall of final storage containers for rock salt (POLLUX®-10), argillaceous rock (ENCON-T) and hard rock (ENCON-K)



The dose rate at the side walls of the final storage containers for argillaceous rock and hard rock are smaller than of the one for rock salt. This is primarily caused by the smaller radionuclide inventory. One can also see that the dose rate for the container for hard rock is especially small. There are two reasons. One reason is the assumption of a thicker layer of cast iron (see discussion in section 3.4). The second reason is that copper provides a slightly better neutron shielding than iron.

4 CONCLUSION

Dose rates at the surface of final storage containers for different host rocks were calculated. For this purpose, the containers were modelled using the Monto-Carlo code MCNP. The dose rate is dominated by the neutrons which are difficult to shield. The gamma radiation of fission and activation products is efficiently shielded by the approximately 40 cm thick walls of the final storage containers. The actual values of the dose rate at the surface strongly depend on the assumptions of the type of spent fuel, burn-up and decay time. In the examples given in this paper, spent UO_2 fuel with a burn-up of 55 GWd/MTIHM and a decay time of 50 years are assumed. Spent MOX fuel is a larger source of neutrons and therefore leads to larger does rates. The nuclide ^{244}Cm is the primary source of neutrons. It has a half-life of 18 years. An increase of the decay time, i.e. an increase of the interim storage time, will lead to a lower dose rate.
The MCNP model of the final storage container for rock salt was also used for a variation of the materials, i.e. the neutron moderator. Typically, calculation times of less than one week per simulation were required to achieve statistical uncertainties of 1 % on a medium-class multi-core CPU. Radiation shielding codes and modern computers allow the optimization of the radiation shielding at an early stage of the development of a storage container.

5 ACKNOWLEDGEMENTS

The work presented in this paper is financed by the German Federal Ministry for Education and Research (BMBF), support code 02S9082A-E.

6 REFERENCES

[1] Sweezy, J.E. 2003, MCNP – A General Monte Carlo N-Particle Transport Code, Version 5, report LA-UR-03-1987.
[2] Spilker, H., Hüggenberg, R., 1991, Description of the German Concept of Final Storage Cask, RAMTRANS Vol. 2 No. 1/3, pp. 67 – 75.
[3] Peiffer, F., McStocker, B., Gründler, D. et al., 2011, Abfallspezifikation und Mengengerüst, report GRS-278.
[4] Klein, R., 2011, Laser Welding of Plastics, First Edition, Wiley-VCH, pp. 22 – 24.
[5] Stahlmann, J., Leon-Vargas, R., Mintzlaff, V., 2016, Generische Tiefenlagermodelle mit Option zur Rückholung der radioaktiven Reststoffe: Geologische und Geotechnische Aspekte für die Auslegung", report ENTRIA-Arbeitsbericht-03.

Radiological Assessment Approach for Decisions on Mining-Related Remediation Projects

Gert de Beer

P O Box 186, Hartbeespoort, 0216, South Africa

Abstract. Involvement in a variety of mining-related remediation projects will be discussed. These vary from remediation of sites contaminated by and containing contaminated scrap, portions of previous or existing mining sites earmarked for residential, industrial or other use, remediation of sites following the demolition of redundant uranium, heavy mineral, sulphuric acid, gold, pyrite or other mineral extraction plants as well as mines planning for future closure and rehabilitation. The focus will be on assessment approaches for the projects above. These vary from pre-remediation site surveys, sampling and sample-analysis exercises, radiological and non-radiological modelling exercises, as well as final surveys and assessments mainly to assist with decisions on restricted or unrestricted release of the site. The required information detail for the various assessment steps will be indicated.

1 INTRODUCTION

The paper presents an overview of a variety of mining-related remediation projects for which radiological safety assessments have been performed over several years. The projects varied from remediation of sites contaminated by and containing contaminated scrap, portions of previous or existing mining sites earmarked for residential, industrial or other use, remediation of sites following the demolition of redundant uranium, heavy mineral, sulphuric acid, gold, pyrite or other mineral extraction plants as well as mines planning for future closure and rehabilitation.

The focus will be on assessment approaches for the projects above. These vary from pre-remediation site surveys, sampling and sample-analysis exercises, radiological and non-radiological modelling exercises, as well as final surveys and assessments mainly to assist with decisions on restricted or unrestricted release of the site. The required information detail for the various assessment steps will be indicated.

2 ASSESSMENT STRUCTURE

The structure proposed by an IAEA working group on the improvement of safety assessment methodologies (ISAM) for near surface disposal facilities [1] and in the NNR regulatory guide for NORM activities [2] was used as basis for assessments (see Figure 1). This structure was, however, adapted to accommodate assessments for remediation projects including (see Figure 2):

i. The existing site before remediation;
ii. The remediated site after various mitigation options;
iii. Failures in the institutional control applicable during and after remediation.

3 BROAD CONSIDERATIONS

Broader considerations affecting the assessment approach include the following:

i. The assessment should provide input to risk-based decision making and should hence be flexible and address various options (e.g. release options, re-use options, mitigation options, institutional control options, scenario and modelling options, deferring options);
ii. The assessment extent (limited or comprehensive) based on the envisaged decisions.
iii. The applicable radiological conditions based on site history, available information for post-closure and post institutional control conditions, surrounding reference conditions, the need for additional information through surveys, sampling and analysis;

[1] Presenting author, e-mail: gpmdebeer@xanaduecopark.co.za

iv. Whether the site represents an existing (legacy) or planned situation;
v. The assessed radiological risks should allow some comparison with radiological criteria but broadly also with other risks (e.g. chemical, conventional).

Figure 1: ISAM Structure for Near Surface Waste Repositories

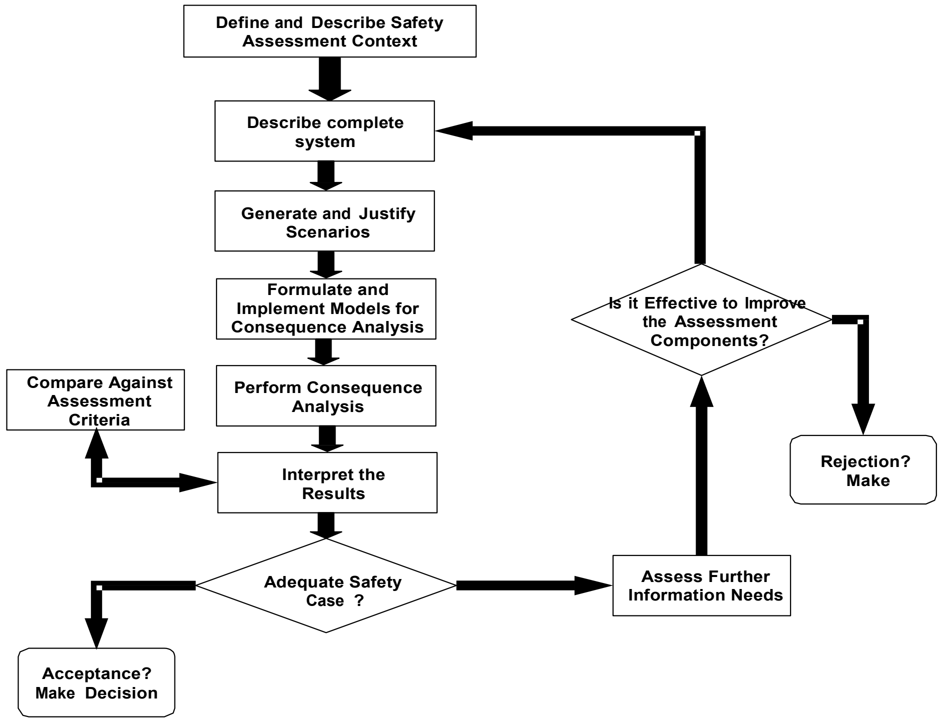
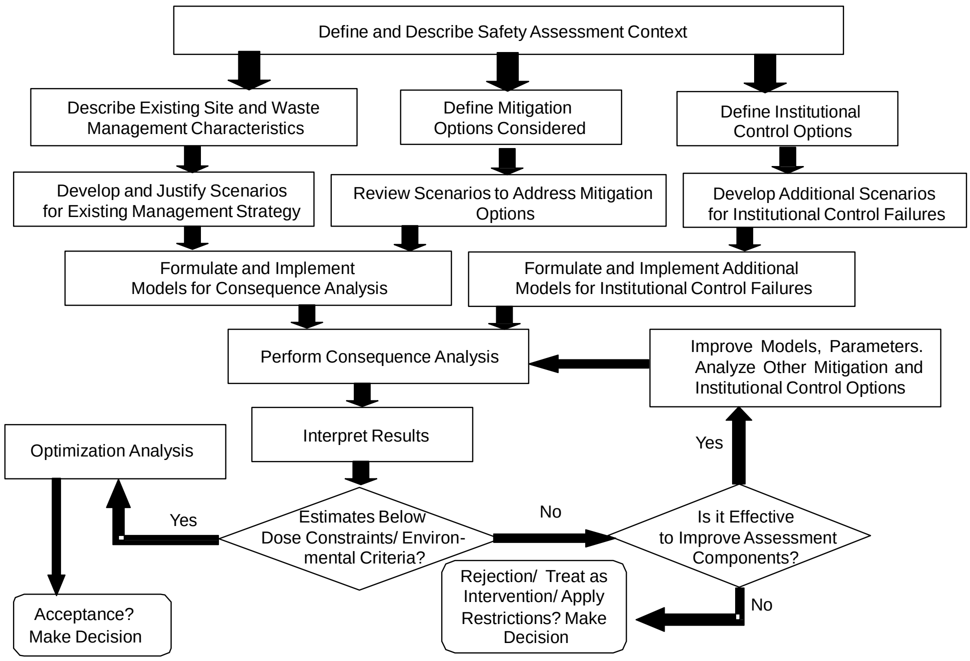


Figure 2: Basic Remediation Safety Assessment Structure



3.1 Source-Pathway-Receptor Analysis

This may assist the decision on the extent of the assessment and involves the following steps:

i. List the sources, pathways and receptors which may contribute to doses:
   - For existing and remediated conditions (mainly public);
   - During remediation (mainly workers);
   - From institutional control failures.

ii. Determine possible interactions between sources and receptors via the identified pathways (Interaction Matrix);

iii. Screen out unimportant combinations.

Sources to consider include the following:

i. External gamma sources (e.g. tailings dams, waste-rock dumps, ore stockpiles, structures, scrap, plumes and contaminated areas around plants but also background or reference areas);
ii. Radon sources (e.g. shafts, open pits, tailings dams, waste rock dumps, ore stockpiles, plumes and contaminated areas around plants and also background or reference areas);
iii. Dust sources (e.g. tailings dams, waste-rock piles, plants, salvage and scrap areas, plumes and salt deposits, including seepage precipitates);
iv. Aquatic sources (e.g. tailings dam run-off and seepage, surface and groundwater releases).

Three pathways are distinguished:

i. External exposure pathway at limited distances from gamma sources;
ii. Atmospheric pathway e.g. vented and exhaled radon and fugitive dust, followed by inhalation, including detectable impacts from deposited dust plumes;
iii. Aquatic pathway e.g. surface water (run-off surface water and seepage to groundwater), groundwater (seepage, decanting and mixing into boreholes and surface streams (both underground flows and transport), followed by direct and secondary ingestion, and external exposure to and ingestion of irrigated soil.

Receptors include:

i. Members of the public (e.g. residents of nearby minor and major towns and informal settlements, users of water from nearby rivers or boreholes, nearby farming communities, consumers of agricultural products from nearby farms and gardens;
ii. Workers (e.g. workers performing remediation, workers when re-using the material or facilities on the mine site, workers in nearby factories).

3.2 Determination of Assessment Extent

The extent of the assessment could by this time be verified and may consist of a:

i. Limited Assessment covering only specific sources and pathways (e.g. surface soil or solid waste) for which a survey could rather be performed on both the site and a reference site;
ii. Comprehensive Assessment, which involves a wider assessment normally requiring computer modelling and predictive exercises (e.g. radon exhalation, fugitive dust emission, atmospheric dispersion, surface-water and groundwater modelling). Modelling exercises are normally for on-site sources and do not require reference site comparison. Site-specific parameters and uncertainties are normally the major obstacles for modelling.

3.3 Survey Planning

The survey should be sufficient to allow a statistical evaluation. A survey should also cover a reference site (e.g. natural background or surrounding mining area). The following should be considered:

i. Sources and pathways to survey;
ii. Survey strategy (systematic random sampling or representative locations);

iii. Random survey grid pattern and grid size (e.g. to identify hotspots), surface or depth profile measurements;
iv. Sampling strategy (e.g. compounding samples for soil, water, sediments, dust);
v. Instrument and measurement/analysis procedures e.g. for:
   - In-situ gamma spectrometry measurements (soil);
   - Radon exhalation or radon concentration measurements;
   - Analysis methods (gross activity or nuclide analysis).

vi. Time-span for survey and sampling:
   - Only one survey for release assessment or two to cover pre- and post-remediation conditions;
   - Extended surveys to cover diurnal and seasonal variations.

4 ASSESSMENT APPROACHES

The assessment relates to existing or prospective exposure conditions for real and hypothetical scenarios developed from the source-pathway-receptor analysis (see Section 3.1) as per the approaches below:

i. Deterministic approach (using fixed parameters with modelling and parameter uncertainties)
ii. Statistical approach (using parameter distributions);
iii. Cover normal evolution scenarios;
iv. Cover disruptive scenarios (e.g. institutional control failures) if sufficient information is available;
v. Calculations could be screening calculations (spreadsheet) or through modelling software;
vi. Obtain calculation (model) and parameter uncertainties (e.g. on source terms, atmospheric dispersion and deposition factors, geo-hydrological transport and transfer factors, demographics, habits, radiological transfer factors and dose coefficients);
vii. A balance between modelling and parameter uncertainties should exist.

Examples of various assessments and scenario options are presented in Sections 4.1 to 4.4 below.

4.1 Examples of Remediation-Related Assessments

i. Restricted or unrestricted land release:
   - Residential, industrial, recreational, agricultural use;
   - Reduction to surrounding mining area levels;
   - Reduction to surrounding natural background;
   - Reduction to lower level of detection (LLD) e.g. for water;
   - Extent of intervention criteria (existing situations).

ii. Re-use of waste materials:
   - Recycling of contaminated metal scrap;
   - Waste rock for road and building construction;
   - Tailings use for brick or paint manufacturing or staining;
   - Tailings treatment (gold, pyrite and uranium extraction and tailings material relocation).

4.2 Example of Representative Persons for Waste-Rock Re-use Scenarios

i. Front-end-loader (FEL) driver 1 (screening operation);
ii. Screen operators;
iii. Transporter (to road-construction site);
iv. FEL driver 2 (road construction site);
v. Tarmac applier driver;
vi. Applier attendants;
vii. Roller driver;
viii. Road painters;
ix. Road users (Taxi driver).

4.3 Examples of Mitigation Options for Soil Contamination to be Considered

i. Option 1:Do nothing or ALARA (below dose constraint);
ii. Option 2: Vegetate tailings dam;
iii. Option 3: Construct wind breaks on tailings dam top surface;
iv. Option 4: Re-slope tailings dam to reduce turbulent flow;
v. Option 5: Cover contaminated areas (tailings dam slopes and/or top surface, salt deposits, seepage precipitates, dust plumes);
vi. Option 6: Total cleanup of dust plumes and total removal of tailings dam.

4.4 Examples of Institutional Control Options

i. Option 1: Re-circulate surface seepage until natural evaporation is sufficient without any groundwater control operations;
ii. Option 2: Re-circulate surface seepage until natural evaporation is sufficient with groundwater control operations active until mixed water quality satisfy criteria;
iii. Option 3: As for Option 2 but with groundwater control operations active until mixed water quality is at background level.

5 MATHEMATICAL MODEL DEVELOPMENT AND CALCULATIONS

5.1 Deterministic Assessment

Mathematical model developments and calculations are largely performed by air-quality and hydro- geological experts and include:
i. Radon exhalation or emission measurements or calculations;
ii. Dust emission data or fugitive dust models with particle size analysis;
iii. Atmospheric dispersion (radon and dust) computer models.
iv. Aquatic mixing or detailed flow and transport computer models.
Parameters from the calculations above or measured parameters are than inserted together with radiological parameters in worker or public dose assessment spreadsheet models containing derived formulas with fixed parameters (deterministic assessment) or parameter distributions (statistical assessment).

The more important parameters for the deterministic assessment are presented below:
i. External dose coefficients [3] or calculated for specific geometries using software [4];
ii. Inhalation and ingestion dose coefficients [5];
iii. Breathing rates for workers and the public (1.2 and 0.93 m^3/h);
iv. Exposure periods for workers and indoors/outdoors for the public (2000, 7000/1760 h/a);
v. Indoor shielding factor (calculated external, 1.0 inhalation);
vi. Radionuclide concentrations in water (radio-analyses);
vii. Water-to-soil, soil-to-plant concentration factors and transfer factors to animal products [6];
viii. Consumption figures (mainly from [2] and [7] but sometimes adapted to local conditions).

5.2 Statistical Assessment

For the statistical assessment, distributions are fitted to available measured or published parameter information using for instance the capabilities of an add-on program to Excel (see [8]). Where insufficient data for fits are available (e.g. only expected values and ranges available) specific distributions (e.g. log-normal) can be fitted. Oher possible solutions are described in the manual of [8]. The parameter distributions are next incorporated into a Monte Carlo simulation program (see [8]) to assess additional-dose distributions for the mining site (calculated from random selections from parameter distributions). The simulation can be repeated for the reference situation (background) as well as for the mine-plus-background and LLD doses. Overlay charts can be generated to compare the background and total (mine-plus-background) dose distributions. Statistical significance tests can then be performed on the mine-plus-background and background-only distributions.

5.3 Statistical Assessment Results

Examples of statistically assessed additional dose results are presented in Figure 3 and Figure 4 and provide information on the mean and spread of the results (percentile values). Examples of overlay charts are presented in Figure 5 and Figure 6 and provide information on the overlap of the two distributions.

Figure 3: Statistical Approach Additional-Dose Result (Scenario 1)

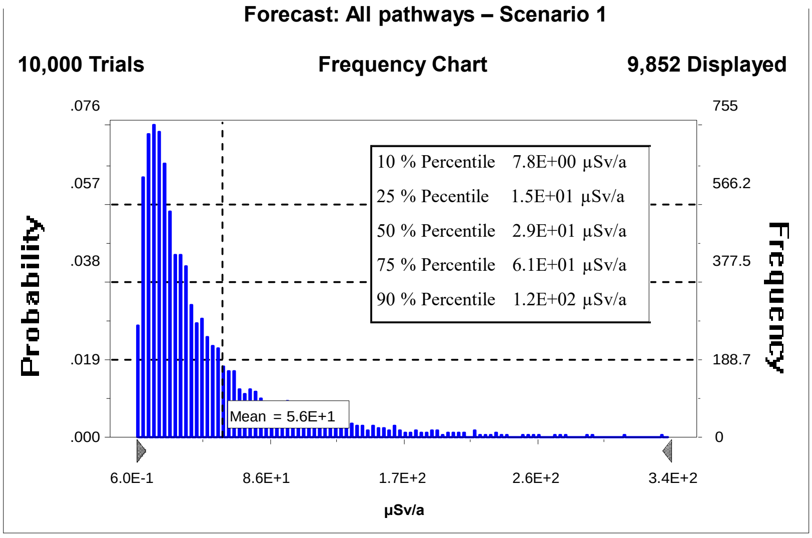


Figure 4: Statistical Approach Additional-Dose Result (Scenario 2)

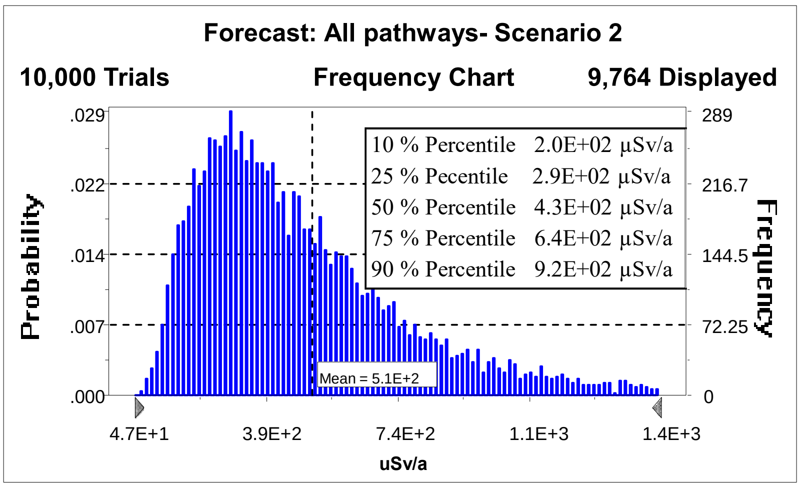


Figure 5: Overlay Chart (Scenario 1)

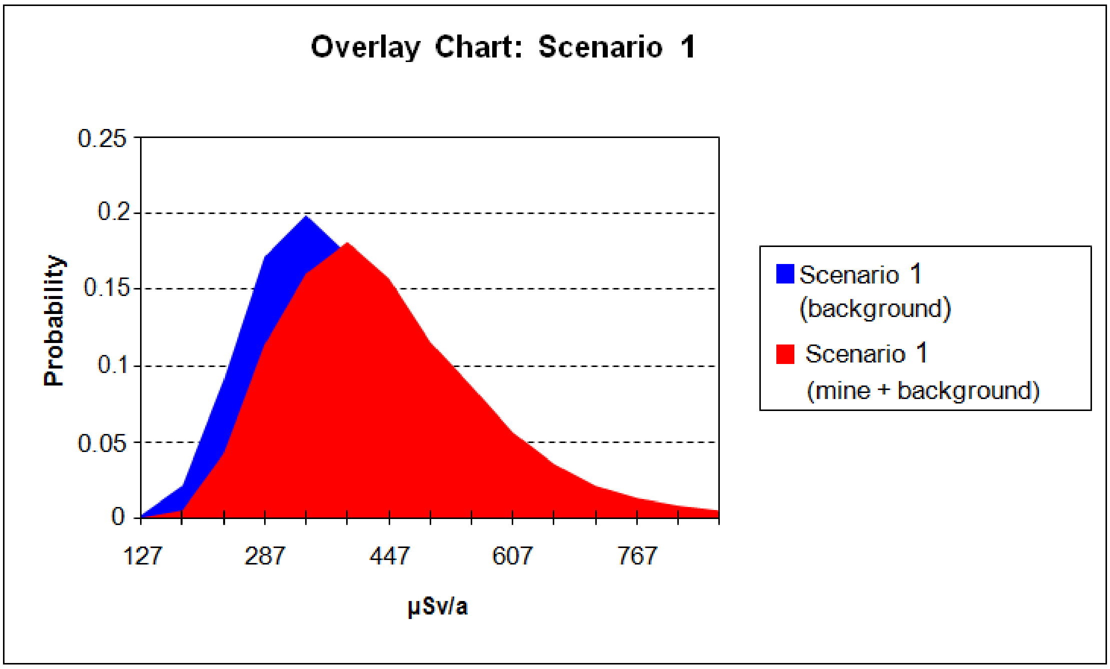


Figure 6: Overlay Chart (Scenario 2)

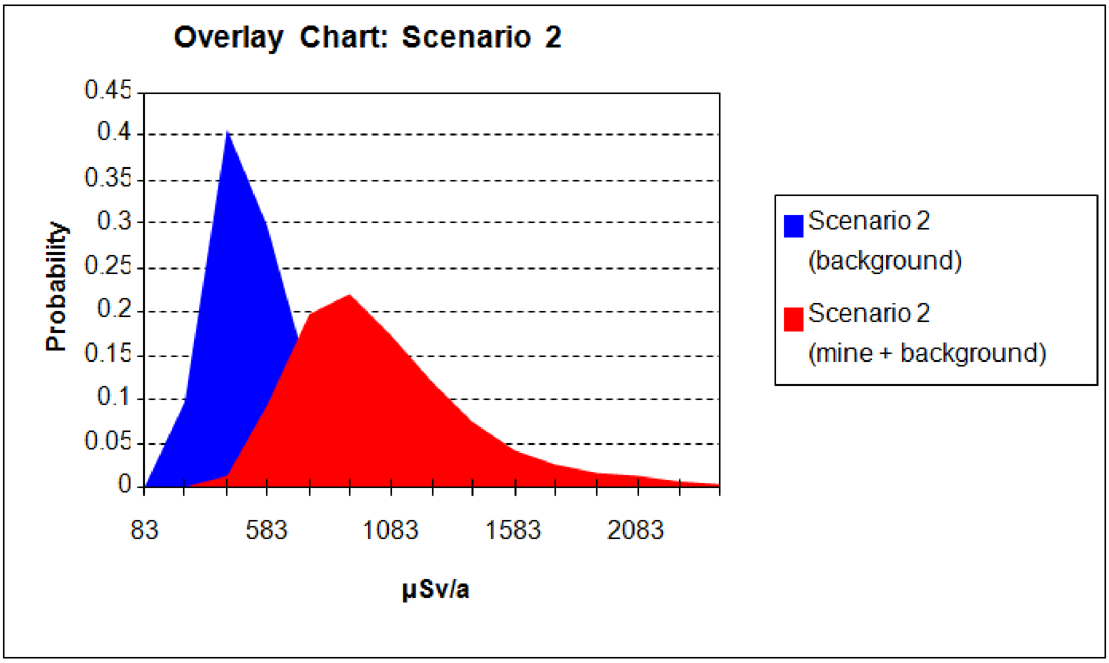
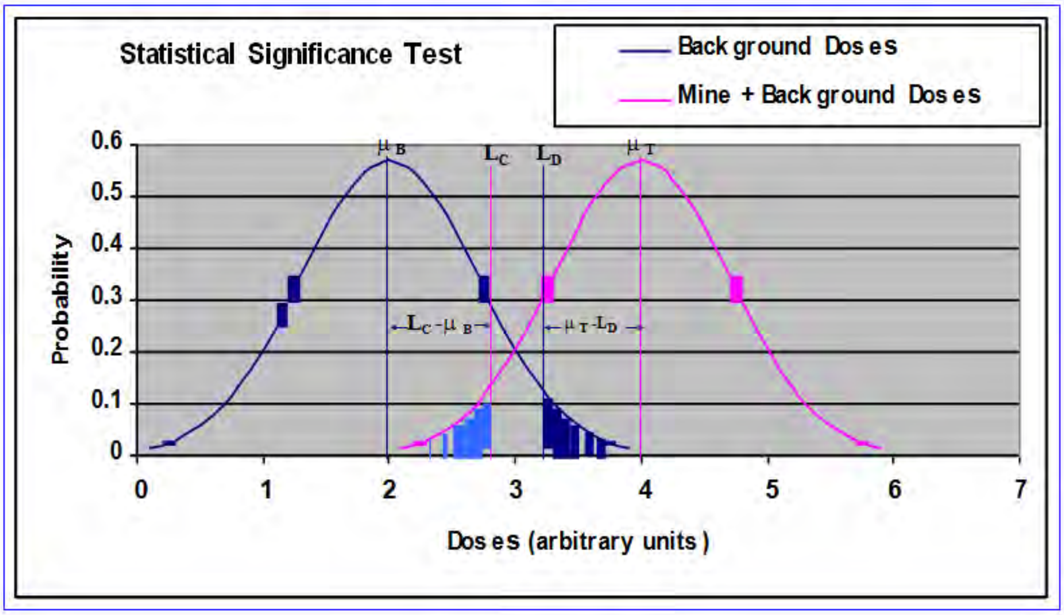


Figure 7: Graphical Representation of Critical and Decision Levels



6 CRITICAL AND DECISION LEVELS

The overlay charts may provide useful tools for decision making by the concept below (see Figure 7).

- Critical level L_C = Level below which one will falsely conclude > 100*α % of the time that a high background dose is not from background but a mine contribution (type 1 error).
- Decision level L_D = Level (above L_C) below which one will falsely conclude >100*(1 - α) % of the time that a real dose from the mine is actually background, indicated as a (type 2 error).
- α is the confidence level approximated by the percentile value of the distribution.

For the data presented in the overlay charts in Figure 5 and Figure 6 the following levels are calculated with μ_B the background mean and μ_T the mine-plus-background mean of the distributions.

Scenario 1
μ_B = 3.9E+02 L_C (25 %) = 3.4E+02 (low confidence); L_C (10%) = 2.9E+02 (high confidence) Hence $\mu_B > L_C$ (25 %) and > L_C (10 %); Hence [L_C (25 %) - μ_B] < 0 and [L_C (10 %) - μ_B] < 0
Distributions are not significantly different at low and high confidence levels for type 1 errors.
μ_T = 4.5E+02, L_D (75 %) = 4.6E+02, L_D (90%) = 5.5E+02,
Hence $\mu_T < L_D$ (75 %) and < L_D (90 %); Hence [μ_T - L_D (75 %)] < 0 and [μ_T - L_D (90 %)] < 0
Distributions are not significantly different at low and high confidence levels for of type 2 errors.

Scenario 2
μ_B = 4.5E+02 L_C (25 %) = 7.1E+02 (low confidence) L_C (10%) = 5.9E+02 (high confidence) Hence $\mu_B < L_C$ (25 %) and < L_C (10 %); Hence [L_C (25 %) - μ_B] > 0 and [L_C (10 %) - μ_B] > 0
Distributions are significantly different at low and high confidence levels in terms of type 1 errors.
μ_T = 9.6E+02 L_D (75 %) = 5.2E+02 (low confidence) L_D (90%) = 6.3E+02 (high confidence)
Hence $\mu_T > L_D$ (75 %) and > L_D (90 %); Hence [μ_T - L_D (75 %)] > 0 and [μ_T - L_D (90 %)] > 0
Distributions are significantly different at low and high confidence levels in terms of type 2 errors.

Based on results like those above, remediation tasks may be prioritized.

7 RADIOLOGICAL CRITERIA

Recommendations for unconditional release of sites hinge around a public dose constraint of 250 - 300 $\mu Sv.a^{-1}$, below which only ALARA considerations are recommended. For conditional release, recommendations depend on the conditions (e.g. 1 mSv/a for industrial use). For existing sites, intervention criteria may be recommended. Radon exposures are normally evaluated separately against a reference level. For workers, occupational-exposure criteria for area classification and individual exposure control are recommended.

8 ACKNOWLEDGMENTS

The work above formed part of the tasks completed over many years within the PelRad group of Necsa and represents a team effort with Gert Liebenberg as team leader for most of the time. Surveys were mainly performed by Neels Boshoff and Rean Swart. The ISAM assessment structure was established by Japie van Blerk. The statistical assessment approach was introduced by Alastair Ramlakan. Activity and nuclide analyses were performed by the Radioanalysis group at Necsa.

9 REFERENCES

[1] IAEA (2004). Safety Assessment Methodologies for Near Surface Disposal Facilities. Results of a co-ordinated research project. Volume 1. Review and enhancement of safety assessment approaches and tools, International Atomic Energy Agency, Vienna.

[2] NNR (2013). Safety Assessment of Radiation Hazards to Members of the Public from NORM Activities. NNR Regulatory Guide RD002, Rev 0.

[3] Eckerman, K.F., Sjoreen, A.L. (2013). Radiological Toolbox User's Guide. NUREG/CR-7166/ ORNL/TM-2013/16.

[4] Grove Software, Inc. (2007). MicroShield Software V7.02.

[5] IAEA (1996), International Basic Safety Standards for Protection against Ionizing Radiation and the Safety of Radiation Sources, International Atomic Energy Agency Safety Series, No. 115, International Atomic Energy Agency, Vienna. (In 2014 revised as IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards).

[6] IAEA (2010), Handbook of Parameter Values for the Prediction of Radionuclide Transfer in Terrestrial and Freshwater Environments, IAEA Technical Reports Series No. 472, International Atomic Energy Agency, Vienna.

[7] Nel, J.H., Steyn, N.P. (2002), Report on South African Food Consumption Studies Undertaken Amongst Different Population Groups (1983–2000): Average Intakes of Foods Most Commonly Used. Department of Health, Directorate: Food Control, Private Bag X828, PRETORIA, 0001, South Africa, e-mail: caseya@health.gov.za.

[8] Decisioneering (2000), Crystal Ball 2000 User Manual, Decisioneering, Inc., 1515 Arapahoe St., Suite 1311, Denver, Colorado, USA.

Application of an Artificial Neural Network for Evaluation of Activity Concentration Exemption Limits in NORM Industry by Gamma-ray Spectrometry

Hannah Wiedner[a*], Virginia Peyrés[b], Teresa Crespo[b], Marcos Mejuto[b], Eduardo García-Toraño[b], Franz Josef Maringer[a]

[a]BEV – Federal Office of Metrology and Surveying, Arltgasse 35, 1160 Vienna, Austria

[b]CIEMAT – Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas, Avda. Complutense 40, 28040 Madrid, Spain

Abstract. Naturally occurring radionuclides like ^{40}K and the decay products of the primordial radionuclides ^{232}Th, ^{235}U and ^{238}U are present in many natural resources. Naturally occurring radioactive materials (NORM) containing these radionuclides are exploited industrially and often exceed the exemption limits of the activity concentration. These industrial activities generate a significant portion of waste, possibly enhancing the potential of exposure of workers and the public and the management and deposition of material above the exemption limit is very costly. The European Metrology Research Project MetroNORM focuses on creating traceable, accurate, and standardized measurement methods, reference materials and systems for application in the concerned industries. The main problem with NORM lies in the variety of densities and compositions of the materials. NORM emits many (interfering) gamma-rays of different energies that have to be analyzed by an expert. An alternative way to approach this problem is the use of artificial neural networks (ANNs). ANNs are mathematical software tools that emulate the way the human brain works. They are trained, tested and validated using sample datasets and can generalize the “knowledge” gained from the content of the training set, applying it to new problems. This can be viewed as a new calibration tool where no expert knowledge of gamma-ray spectrometry is needed by the end-user. In this work an ANN was created in the frame of MetroNORM that is able to decide from the input data of a raw gamma-ray spectrum if the activity concentrations in a sample are above or below the exemption limits. Six NORM reference materials have been analyzed. To widen the applicability of the algorithm, a set of artificial gamma-ray spectra with varying densities and activity concentrations and material compositions have been created by Monte Carlo simulation and used in the training, testing and validation of the ANN.

KEYWORDS: artificial neural network; exemption limit; NORM; Monte Carlo; gamma-spectrometry.

1 INTRODUCTION

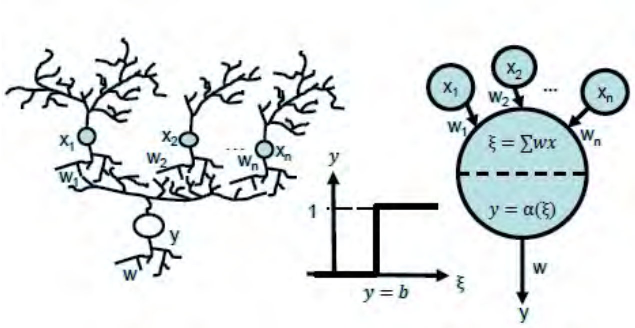
One of the main applications and challenges of gamma-ray spectrometry and especially when dealing with natural radioactivity, is to determine if a certain material is above or below the exemption limit stated in the national legislation. With the latest Basic Safety Standards Directive entered into force on 6 February 2014 these limits are 1 Bq/g for all natural radionuclides with the exception of ^{40}K (10 Bq/g) (Council of the European Union, 2013). Naturally occurring radioactive materials (NORM) containing these radionuclides are exploited by industrial endeavours and often exceed the exemption limits of the activity concentration for radionuclides of the U and Th series, depending on the mineral composition and geological origin. Industrial activities are generating a significant portion of waste and can enhance the potential of exposure of workers and the public.

The aim of this work was to create an artificial neural network (ANN) that is able to decide from the data of a raw gamma spectrum if a NORM material is above or below the exemption limit, avoiding the rather complex analysis associated to spectral deconvolution. The main problem with analyzing NORM lies in the variety of densities and compositions of the materials. NORM emits many (interfering) gamma-rays of different energies that have to be analyzed by an expert. The evaluation of activity content in a sample also requires the detector to be properly calibrated in terms of energy and efficiency response. The detection efficiency highly depends on source-to-detector and sample geometry, amount and composition of sample material and energy of the gamma-rays to be measured. Self-attenuation of the gamma-rays due to the sample material has to be considered. The detection efficiency for each sample has to be determined. This can be done with a standard source – a source with the same geometry and material as the sample that is to be measured, but with a traceably known activity concentration. This is a time-consuming process that in most cases requires a large number of standard sources or is only feasible for a small number of sample situations. Therefore, in many cases, mathematical models are used to calculate the efficiency of the detector. With mathematical models a large variety of sample situations can be used. The use of Artificial Neural Networks can be seen as an alternative way of calibrating the detector.

2 ARTIFICIAL NEURAL NETWORKS (ANNS)

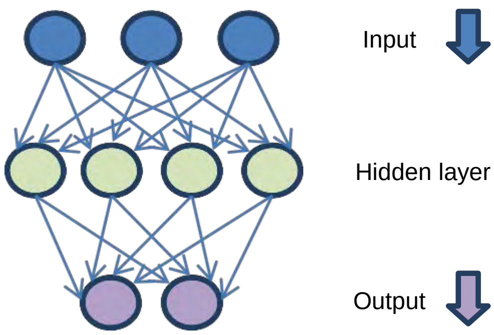
ANNs try to imitate the way the human brain works. In the human brain a number of biological neurons each generate a signal of an intensity x and a synaptic feeding strength *w*. These signals feed into a neuron with a threshold b using dendrites and axons. If the product of *x* and *w* is below the threshold, the neuron does not recognize the input. If the product is above the threshold, the neuron computes the inputs and generates a signal *y* that can again be an input to another neuron. Fig. 1 illustrates the process [1].

Figure 1: Basic processes in human brain.



An ANN consists of a number of nodes that represent the computing units (or neurons), connections (axons and dendrites) between those nodes, connection weights (synapses) and thresholds (activity in the soma). A non-linear transfer function, often a sigmoidal function, is applied to the weighted sums of each neuron. Generally speaking, an ANN consists of an input and an output layer and can also contain one or more hidden layers. Networks with no hidden layer are only able to perform linear tasks [2] while a second hidden layer is only necessary for discontinuous problems [3]. Depending on the problem, an even greater number of hidden layers may be necessary. Fig. 2 shows a schematic of an ANN.

Figure 2: ANN schematic.



The number of input neurons is defined by the number of input parameters and the number of output neurons reflects the number of output variables. The optimal number of neurons in the hidden layer has to be obtained. This is usually done by employing a trial-and-error method. Each connection comes with a connection weight that signifies the importance of the input. The network is trained by providing it with a number of inputs and the corresponding outputs. Unless reliable a-priori information is available, the network algorithm starts out with random connection weights that are changed after each training cycle to reflect the wanted output. Literature research shows that the most commonly used algorithm to adapt the weights is the backpropagation algorithm. This algorithm uses the difference between the ANN solution and the actual solution of the training example that is provided for the training process to change the connection weights. ANNs trained in that style are capable of learning and can apply their "knowledge" to unknown situations. The main advantage of the network is its ability to generalize and handle imprecise and noisy information. This process can be considered as a robust alternative to a classical calibration method. ANNs come in different shapes and many different kinds of ANNs exist. They are classified by their purpose (e.g. solving classification problems, forecasting, etc.), learning algorithm and other characteristics of their use but for working with all kinds of ANNs there are three major steps to follow:

- *Training:* The training phase consists of designing and building the neural network as well as providing it the relevant training data. A set of training examples consists of the input data as well as the respective output or target data. With the use of the chosen training algorithm the connection weights and biases are adjusted until the desired output is reached;
- *Validation:* This phase is used to evaluate if the network is correctly trained and working properly. Additionally, it is used to minimize overfitting. An over fitted network loses the ability to generalize the data. If the accuracy over the training data set increases, but the accuracy over the validation data set stays the same or decreases, the network is overfitting and training should stop;
- *Testing:* This phase is to check if the output is correct and to evaluate predictive power of the ANN.

3 MATERIALS & METHODS

3.1 ANN specifications

After examining the alternatives found in literature, it was decided to create an ANN able to predict if a material is below or above the exemption limit for a predefined list of radionuclides and materials relevant to NORM. This restriction is necessary as the number of materials available is limited and the aim of this work is to study the feasibility of using ANNs in this context. A number of seven materials available at CIEMAT where used to train the ANN. Five of those materials come from the JRP MetroNORM and two are reference materials available at CIEMAT (Table 1). The spectra are analyzed for six radionuclides typically found in NORM materials and representing the three naturally occurring decay chains. Although the nuclides considered for analysis emit several gamma lines, it was decided to analyze a subset of those shown in Table 2 as they are free from interferences with other lines.

Table 1: List of NORM reference materials used to creating the sample materials.

CIEMAT	MetroNORM
Phosphogypsum Huelva	Phosphogypsum (D1.2.2)
Ilmenite Huelva	Tuff 1 (D1.2.2)
	Tuff 2 (D1.3.2)
	Sand (D1.3.2)
	TiO_2 (D1.3.2)

Table 2: List of radionuclides and gamma lines analyzed by the ANN.

Radionuclide	Gamma-ray energies (keV)		
^{210}Pb	46,65		
^{234}Th	63,31	92,59 *	
^{235}U	143,77	163,36	
^{212}Pb	238,63	300,09	
^{214}Pb	242,00	295,22	351,93
^{228}Ac	911,20	968,96	

*combination of 2 lines that are not well separated: 92.38 keV and 92.80 keV

All materials have been analyzed at CIEMAT's chemical laboratories. The information provided by the chemical analysis was used to produce artificial spectra of the available materials with the Monte Carlo code PENELOPE v. 2014 and CIEMAT's add-on PENNUC. The artificial spectra differ from the original gamma-ray spectra in density of the sample, radionuclide composition and activity concentration. These spectra are used in order to cover a wider range of experimental conditions and to provide the network with more training examples. In order to create an ANN that correctly predicts the output, representative samples have to be used as training input. This requires intensive study of equilibrium and disequilibrium situations and careful sample preparation.

3.2 Measurement setup

The measurement system used for this project is based on a Canberra Industries GX4020 extended-range coaxial detector with 45.4 % relative efficiency and a carbon-epoxy window. This kind of detectors can extend the usual energy range of Ge detectors down to a few keV. The electronic setup includes a high-voltage power supply from BERTAN, preamplifier, spectroscopy amplifier and pulse generator from CANBERRA and a successive-approximation analog-to-digital converter from SILENA driven by a PC. A prismatic shielding structure of about 80 x 80 x 80 cm surrounds the detector. It is composed by layers of Pb (5 cm), Cd (3 mm) and Cu (1.5 mm) (V. Peyres, E. García- Toraño, 2007). Fig. 3 shows the detector system. The software used for spectrum acquisition is SILENA International SpA EMCA2000 MCA Emulation Software (2000). The software used to calculate the peak area is a non-linear code (GRILS) included in the GANAAS [4] package, which is freely distributed by the International Atomic Energy Agency.

Figure 3: Detector and cooling system at CIEMAT gamma laboratory.

Once the gamma spectrum has been acquired, the activity of a sample can be calculated using the following formula:

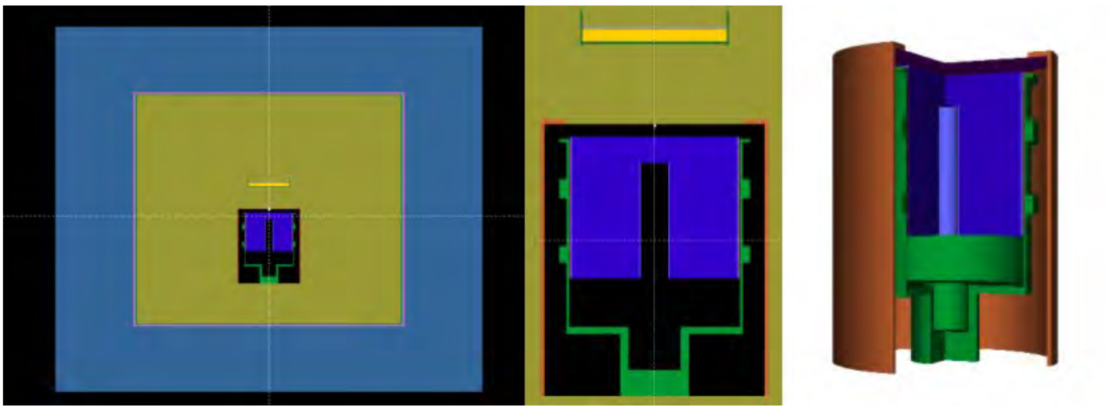
$$A = \frac{N}{t\, \varepsilon(E)\, p_\gamma} f_c\, f_a \tag{1}$$

A	sample activity	$\varepsilon(E)$	efficiency at energy E
N	counts in the peak area	p_γ	photon emission probability
t	measurement time	f_a	pile up correction factor
f_c	coincidence summing correction factor		

3.3 Monte Carlo calculations

In order to cover a wider range of real conditions, make up for the limited number of real sample material and make the ANN applicable to a greater number of situations, a large number of artificial spectra have been created using the Monte Carlo code PENELOPE v.2014 [5] with CIEMAT's add-on PENNUC [6]. In accordance with literature values, these spectra vary significantly in density and activity concentration of the analysed radionuclides and are used as training material for the ANN. With PENNUC the simulation process involves all particles in a cascade, whereas PENELOPE on its own simulates each particle separately. Therefore, coincidence summing is not separately calculated, but is an integral part of the whole detection efficiency calculation. Fig. 4 shows the detector model.

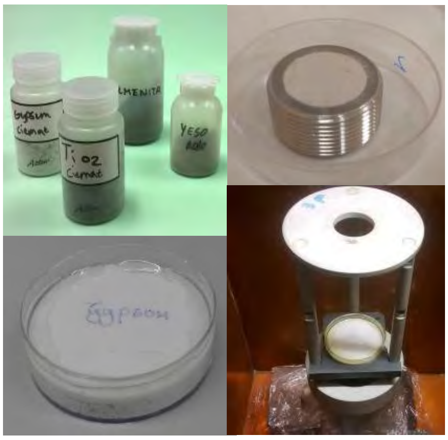
Figure 4: 2D and 3D virtual detector images of CIEMAT detector (PENELOPE).



3.4 Samples

The samples used as input data for the ANN are samples from MetroNORM Work Package 1 as well as NORM samples available at CIEMAT. The samples were carefully weighted and prepared using polypropylene containers as can be seen below. After equilibrium had been reached (21 days) the samples were measured on CIEMAT's GX4020 detector. Fig. 5 shows the samples at different stages in the preparation and measurement cycle.

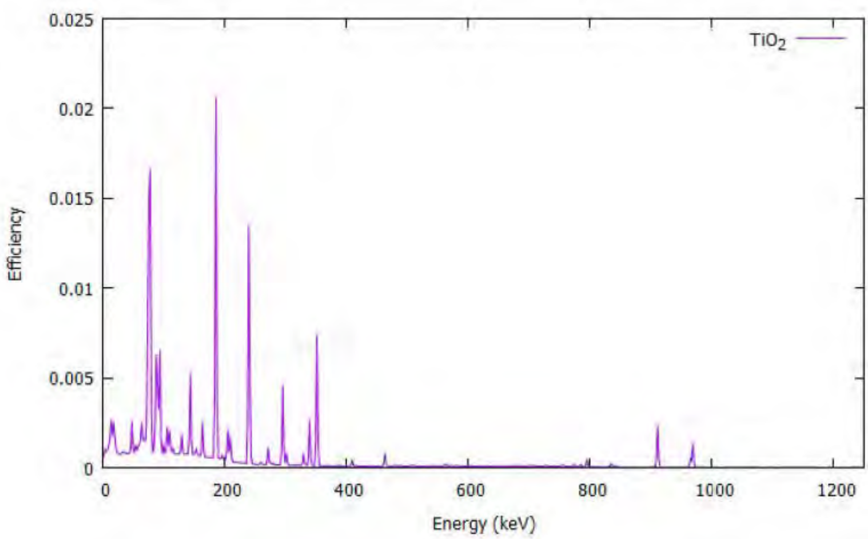
Figure 5: Samples at various stages.



The samples were analyzed for major elements and trace elements by Wavelength dispersive X-ray fluorescence spectrometry (XRF) using a PANAlytical AXIOS automated XRF spectrometer with Rh radiation. The x-ray diffraction (XRD) data were collected using a PANalyticalX'Pert PRO diffractometer operating in θ-θ configuration, with Cu Kα radiation. The data was collected from 20-120º 2θ (θ…Bragg angle). Elements were determined by simultaneous Inductively Coupled Plasma Optical Emission Spectrometer (ICP-OES), model Varian (now Agilent) 735-ES in a radially viewed configuration and a VistaChip image-mapped Charged Coupled Device (CCD) detector. The determination of concentration was performed using an Inductively Coupled Plasma Mass Spectrometer (ICP-MS) Thermo i CAP Q (Thermo Scientific, Bremen, Germany) equipped with a high-performance quadrupole analyser, He KED collision cell, and electron multiplier detector. A LECO CS-244 elemental analyser was used for determining carbon content. The amount of carbon dioxide was measured by an infrared detection method (Method ASTM 415.1, EPA-600/4-79-020). Total uranium analysis was performed by laser kinetic phosphorimetry. The kinetic phosphorimetry measurements were carried out using the kinetic phosphorescence analyser KPA-11 (Chemchek).

To gain significant results, the real materials have been measured for 200000 s on the detector while a number of 5.00E+08 showers have been simulated for each radionuclide and density, amounting to a total number of 126 simulations. After separately calculating the efficiency for each radionuclide, material and density using PENELOPE and PENNUC, the spectra are convoluted. This means, that the calculated data is convoluted with a Gaussian curve whose width is a function of energy in order to reproduce the system's response and include the electronic noise. Figure 6 shows the simulated spectrum after convolution.

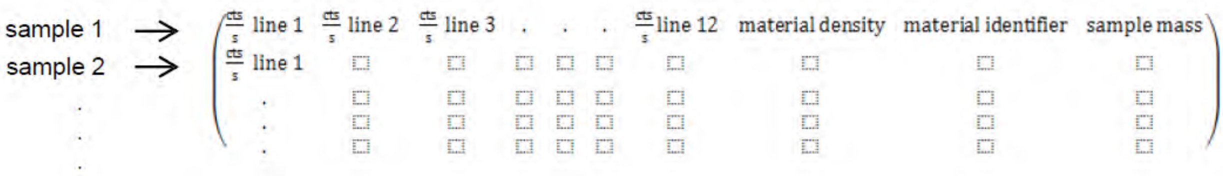
Figure 6: Exemplary PENELOPE/PENNUC simulation data for TiO2.



Afterwards, the calculated data of each material and radionuclide is multiplied by the target activity and combined in order to gain one synthetic spectrum per material, density and activity concentration that includes all radionuclides as in the real spectrum. This gives a spectrum containing the counts per second for each energy bin according to equation (1). Correction factors are included in the efficiency calculated by PENELOPE and PENNUC. In this way a total number of 635 artificial spectra with varying distributions of activity concentration have been generated as training input to the ANN. To generate the artificial spectra and train the ANN, the software package Matlab has been used.

The structure of the input is a matrix (635 x 15) where the rows represent the sample and the columns represent the individual lines to be analyzed, followed by information on density, material and sample mass. Figure 7 shows a schematic of the input matrix.

Figure 7: Schematic of input matrix.



The ANN's output layer consists of 12 nodes, each one corresponding to one analyzed line. After studying possible scenarios of disequilibrium, the artificial samples for both equilibrium and disequilibrium state were classified using five categories of activity content. This means, that samples were produced to correspond to the activities shown in Table 3, therefore minimizing the total number of samples and relying on the power of the ANN to intrapolate to the missing data. These categories are also used to classify the output of the ANN. Category 1 corresponds to an activity well below the exemption limit, while categories 2 and 3 correspond to an activity a little below or of 1 Bq/g (the exemption limit), respectively. Categories 4 and 5 correspond to activities a little above and well above the exemption limit.

Table 3: Activity categories assigned in the sample preparation stage.

Activity	Category
0,1 Bq/g	1
0,7 Bq/g	2
1 Bq/g	3
1,2 Bq/g	4
20 Bq/g	5

4 RESULTS

The best results were obtained using an ANN with 15 input nodes, one hidden layer, 31 hidden neurons and 12 output nodes using a backpropagation algorithm and a sigmoidal transfer function based on a logarithm. Introduction of the sample weight as an input parameter caused regression to get significantly better. The output of the ANN agrees with the manually assigned activity categories. Training the ANN for 91 epochs and then retraining with the same training data resulted in an ANN with an overall regression factor of 0.9975. Good convergence of the output can be observed. Fig. 8 shows the training plots.

Figure 8: Performance, regression and training state plots after retraining.

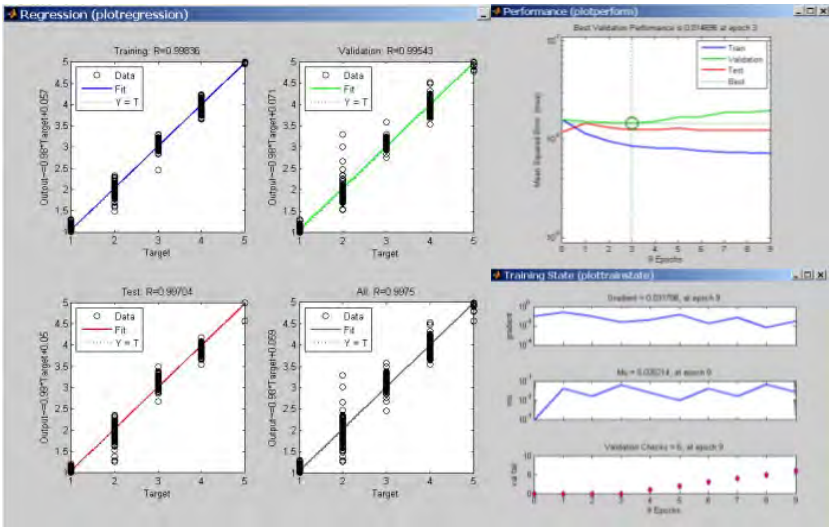


Table 4 shows the results of the final ANN when presented with the unfamiliar testing samples.

Table 4: ANN output for each of the six testing materials (exact ANN output).

Radio-nuclide	γ-ray energy (keV)	TiO_2	Sand	Phosphog. Huelva	Phosphog. MetroNORM	Ilmenite	Tuff
^{210}Pb	46.65	4.129	1.1106	2.3224	1.1028	1.0873	1.4027
^{234}Th	63.31	1.0005	1.1162	1.2007	1.0976	1.0721	1.576
	92.59*	1.0006	1.1165	1.1992	1.0967	1.072	1.5747
^{235}U	143.77	1	1.0843	1.2183	1.0696	1.0565	1.1353
	163.36	1	1.0841	1.2213	1.0701	1.0562	1.1353
^{212}Pb	238.63	5	1.0487	1.0569	1.0966	1.2138	1.2004
	300.09	5	1.1069	4.1687	1.1735	1.0686	1.2905
^{214}Pb	242.00	5	1.1106	4.1404	1.2009	1.0744	1.3819
	295.22	5	1.0478	1.0586	1.1147	1.2061	1.2637
	351.93	5	1.1073	4.1697	1.1724	1.0694	1.2908
^{228}Ac	911.20	5	1.0458	1.0774	1.1386	1.2256	1.2389
	968.96	5	1.0459	1.0774	1.1386	1.2243	1.2399

*combination of 2 lines that are not well separated: 92.38 keV and 92.80 keV

5 CONCLUSION

This study was undertaken to find out if ANNs can be applied to the problem of determining if the activity concentration in a sample is above or below the exemption limit of 1 Bq/g. In the course of this work it was shown, that the ANN was able to correctly classify all of the testing materials and ANNs are well-suited to carry out this task. For specialized industries where only one material has to be analysed and a large number of sample spectra are available, the authors propose the use of an ANN specifically trained to that purpose without the use of other materials. The biggest limiting factor for the use of ANNs is the availability of real sample material. In the course of this work this problem has been sidestepped by calculating artificial spectra from Monte Carlo simulations but this, in turn, necessitates the complicated and time-consuming study of disequilibrium situations. It is necessary to note that this constraint only applies to the creation of training material and of the ANN itself, not the usability of the ANN. For the end-user only a gamma spectrum in ASCII format and no specialized knowledge whatsoever in the field of gamma-ray spectrometry is required.

6 ACKNOWLEDGEMENTS

This work has been supported by the European Metrology Research Programme (EMRP), JRP-Contract IND57 MetroNORM (www.emrponline.eu). The EMRP is jointly funded by the EMRP participating countries within EURAMET and the European Union. Hannah Wiedner received an Early Stage Researcher Mobility Grant associated to MetroNORM to complete this work. Thanks are given to Alberto Quejido, Belén Gómez, Miguel Sanchez, Isabel Rucandio, Marta Fernández and Pilar Galán of the analysis laboratories of CIEMAT that cooperated in the characterization of the material used in this study.

7 REFERENCES

[1] Basheer, I. A., Hajmeer, M., 2000, Artificial neural networks: fundamentals, computing, design, and application. Journal of Microbiological Methods 43 (2000) 3-31.

[2] A. Negarestani et al., 2002, Layered neural networks based analysis of radon concentration and environmental parameters in earthquake prediction. Journal of Environmental Radioactivity 62 (2002) 225-233.

[3] Rafiq, M.Y., et al., 2001, Neural network design for engineering applications. Computers and Structures 79 (2001) 1541-1552.

[4] International Atomic Energy Agency, Nuclear analysis software, Part 2. Gamma spectrum analysis, activity calculations and neutron activation analysis (GANAAS), 1991, Vienna.

[5] Salvat, F., et al., 2001, PENELOPE, A Code System for Monte Carlo Simulation of Electron and Photon Transport. Proceedings of a Workshop/Training Course, OECD/NEA 5-7 November 2001 NEA/NSC/DOC(2001)19.

[6] García-Toraño, E., Peyres, V., Be, M.-M., Dulieu, C., Lepy, M.-C., Salvat, F., Simulation of decay processes and radiation transport times in radioactivity measurements, to be submitted to Nucl Instr and Meth B.

No Nuclear Power - No Disposal Facility?

J. Feinhals

DMT GmbH & Co. KG, Gr. Bahnstr. 31, 22525 Hamburg, Germany, Secretary of the Working Group “Disposal and RadWaste Management” of the Swiss German Fachverband, für Strahlenschutz.

Abstract. Countries with a nuclear power programme are making strong efforts to guarantee the safe disposal of radioactive waste. The solutions in those countries are large disposal facilities near surface or in deep geological layers depending on the activity and half-life of the nuclides in the waste. But what will happen with the radioactive waste in countries that do not have nuclear power plants but have only low amounts of radioactive waste from medical, industrial and research facilities as well as from research reactors? Countries producing only low amounts of radioactive waste need convincing solutions for the safe and affordable disposal of their radioactive waste. As they do not have a fund by an operator of nuclear power plants, those countries need an appropriate and commensurate solution for the disposal of their waste. In a first overview five solutions seem to be appropriate: (i) the development of multinational disposal facilities by using the existing international knowhow; (ii) common disposal with hazardous waste; (iii) permanent storage; (iv) use of an existing mine or tunnel; (v) extension of the borehole disposal concept for all the categories of radioactive wastes.

1 THE CHALLENGE

Nuclear power plants are operated worldwide in 30 countries, while 71 countries are operating research reactors [1]. Even if the spent fuel is returned to the manufacturer and the production rate of radioactive waste is much lower than at a nuclear power plant (NPP), the radioactive waste from operation and future decommissioning cannot be neglected. As there is not commercial power generation, there is also not the levy on power consumption that goes to a waste fund. There is not the money set aside for the disposal of radioactive wastes generated in these countries. Therefore these countries look for more cost effective disposal routes for the wastes that they produce. The European Commission stipulates each state needs to develop a national programme for the safe disposal of radioactive waste (Council Directive 2011/70/ EURATOM) [2]. Similar requirements do exist outside the European Union, with every nation responsible for the safe management of radioactive waste, including the need to have a disposal plan. The challenge for the future is: which alternatives for the safe disposal of radioactive waste are possible for countries generating small amounts of radioactive waste?

2 RADIOACTIVE WASTE IN COUNTRIES WITHOUT NUCLEAR POWER PROGRAMME

Countries without nuclear power plants or any nuclear fuel cycle facilities do not have high level waste (HLW), particularly when the fuel from research reactors is returned to the country of origin. In these countries, only small amounts of radioactive waste are produced. The main sources of this waste are the use of radioactive material in medicine, industry and research as well as the operation and decommissioning of nuclear research facilities like research reactors. Usually a large part of the waste can be cleared as non-radioactive waste after storage or decontamination. The amount of remaining radioactive waste that is suitable for near-surface disposal (LLW) is less than ten thousand tons while the amount that is not suitable for near-surface disposal (ILW) is less than a few hundred tons. Furthermore, in all countries there are disused sealed radioactive sources, including long-lived sources such as lightning conductors containing mainly ^{241}Am (432.2 a) or ^{226}Ra (1.6×10^3 a), and ionization chamber smoke detector (ICSD) containing mainly ^{241}Am, ^{226}Ra and sometimes ^{239}Pu (2.41×10^4 a) which are not suitable for near surface disposal in large quantities due to their long half-life.

In countries without a nuclear programme, significant amounts of radioactive waste arise from the operation and decommissioning of research reactors [3]. The radioactive waste streams depend on the reactor type, the implemented applications and the schedule of operation. They can be activated and include contaminated materials. The most activated part of the reactor structure is the core, while the biological shield, usually made of concrete and steel reinforcements, is exposed to relatively low neutron fluxes. Contamination arises from the activation of the corrosion/erosion products as well as from the dispersion of the irradiated fuel and fission products through cladding breaches and conveyed by the coolant. Fission products in contaminated materials generally become significant in the case of failure of fuel elements. A large variety of radionuclides can be produced by neutron activation at nuclear reactors. The radionuclides which are important from the viewpoint of disposal are the long-lived radionuclides (half-lives higher than 30 a). The major long-lived nuclides are: ^{14}C (5730 a) which is significant in concretes and graphite; ^{36}Cl (3.01×10^{5} a) is present in some stainless steels and aluminum reactors components; ^{41}Ca (1.03×10^{5} a) is one of the main constituents of bioshield concrete; ^{59}Ni and ^{63}Ni (7.6×10^{4} a and 100.1 a respectively) is found in nickel alloys and stainless steel; ^{93}Mo (3500 a) is present in some stainless steels; ^{93}Zr (1.5×10^{6} a) is important in irradiated cladding and in moderator tubes; ^{108m}Ag (130 a) is significant in control rods with large amounts of silver.

Common examples of solid very low level waste (VLLW) and low level waste (LLW) are items contaminated during handling of radioactive materials such as personnel protection items, cleaning materials and tools as well as components exposed to neutron beams such as containers for production of radioisotopes or for irradiation of samples. Low and intermediate level waste (LLW and ILW) can be materials used for cleaning of water, such as ion exchange resin or materials in the ventilation systems as well as irradiated components of the reactor such as the materials at the reactor core, monitoring equipment (ionization and fission chambers, thermocouples etc.), control rods and startup neutron sources.

Liquid radioactive wastes during operation are usually coolant from the reactor pool or vessel, liquids used for decontamination and liquids produced from hot chemistry laboratories. In case the aqueous wastes cannot be discharged, they are concentrated to minimize the volume and the residues usually solidified in cement. Other liquid wastes like organic solvents are solidified in cement directly or incinerated together with other radioactive waste.

Tritium in liquid wastes is of higher importance in reactors cooled and/ or moderated with heavy water. In gaseous radioactive wastes, the main radionuclides are ^{41}Ar and ^{14}C which are produced by activation of the air present in the reactor coolant/moderator and irradiation facilities.

A significant application in research reactors is the production of radioisotopes for medicine, agriculture, industry and research. Radioisotopes are produced at research reactors by neutron capture in targets or by nuclear fission of ^{235}U [4]. In the case of radioisotope production by neutron capture, target encapsulation is an important stream of solid radioactive waste. The use of zircaloy for encapsulation yields waste with ^{93}Zr while the use of stainless steel results mainly in waste with ^{55}Fe, ^{63}Ni, ^{60}Co. The nuclear fission of ^{235}U produces the full set of fission products and some actinides.

The main decommissioning wastes are activated and contaminated metals (e.g., stainless steel, carbon steel, lead, aluminum) and concrete from the biological shield. More than 50% of materials from dismantling of research reactors are exempt waste and a small amount, less than 10%, are ILW. In research reactors some specific materials like graphite or beryllium are also used. Graphite is used as a moderator and reflector. Some research reactors have a stacking of graphite in one of their irradiation facilities, the thermal column. The long lived ^{14}C isotope can be produced by neutron activation in the graphite. The activity of this isotope determines the management/ disposal options of graphite. Beryllium is used in research reactors as a source of neutrons, moderator and reflector. The material itself is extremely toxic. The main radionuclides in beryllium are ^{3}H and the long lived ^{10}Be (1.6×10^{6}a).

3 EXISTING CONCEPTS FOR DISPOSAL OF RADIOACTIVE WASTE

Europe is running a very intensive research in the area of disposal facilities [5]. For coordination of all the research projects for an effective exchange of information technical platforms were established [6]. The IGD-TP (Implementing Geological Disposal – Technical Platform) was launched for the research for deep geological disposal facilities, where the concept of near surface disposal facilities is described as sufficient for low level and intermediate level waste with short half-life. Geological disposal is recommended for intermediate level and high level waste especially containing isotopes with long half-lives. All these projects for disposal facilities have one thing in common: They are very money and time consuming, because they are designed for large amounts of radioactive waste. Such solutions seem to be not adequate for the disposal of some thousand drums with radioactive waste. Nevertheless, countries with a high progress in such disposal projects shall take over a lighthouse function for those countries, which have just started planning for a disposal facility.

Existing concepts are:

- Near Surface burial – low level waste is buried within 10 m of the surface in a conventional style landfill.
- Shallow burial – low level waste is packaged and buried within 100 m of the surface.
- Engineered structures and concrete vaults – typically for 100,000 m^3 of waste or more.
- Engineered boreholes for disused sealed radioactive sources.
- Geological caverns for the disposal of intermediate level waste or high level waste.

4 ALTERNATIVES

Fully aware of this challenge the following alternative solutions are also discussed.

4.1 Multinational disposal facility

A multinational disposal facility is a disposal facility, which is used by several countries (sometimes also called “regional disposal facility”). This approach investigated by the World Nuclear Association (WNA) [7] and the International Atomic Energy Agency (IAEA) [8] makes sense from the technical as well as from the economical view. In EU, the European Repository Development Organization (ERDO) works for the implementation of one or more shared regional repositories for radioactive waste. The idea is compelling, but the political challenges are very difficult. The definition of the area of competence for a supervising and licensing authority might be easy, although it has to be active beyond state borders to control waste packages in other countries and to decide whether waste packages are acceptable or not. There are many challenges:

- How will the costs be shared for the participating countries if the project has a significant delay (which is very normal in those projects) or has to be abandoned?
- What will happen if acceptance of foreign waste is suddenly unenforceable due to a lack of public acceptance?
- How stable will the country, its government and borders be for the life of the control period?
- Would the site become a security risk for all the countries around it?

These challenges are only some of the reasons why politicians see only little chances for a multinational disposal facility.

Groups have nominated Australia as the site of a multi-national repository, and discussed the concept of uranium leasing; the country which mines the uranium has to take the uranium back at the end of its useful life, along with whatever other wastes were produced with it. There is no political will or public support for either idea within Australia, and there is a minority viewpoint that if radiopharmaceuticals are exported then Australia should be exporting that portion of the radioactive waste to the country using the radiopharmaceuticals. There is an idea for a south-east Asian repository, however the issues are still over who will have control, where it will be situated and how this will impact on the regional tensions between countries. This would be a long term goal (100 years) for the region.

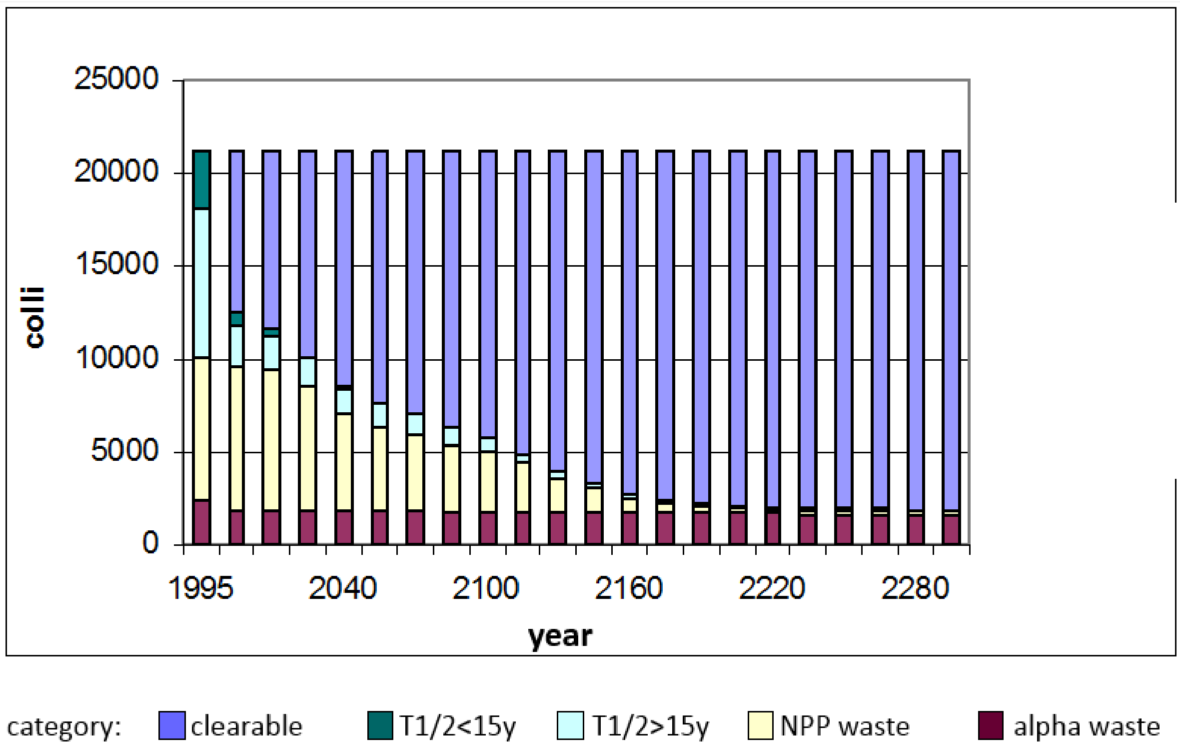
4.2 Common disposal with hazardous waste

This alternative idea seems to be smart, as requirements for technical barriers at landfills for hazardous (toxic, harmful, dangerous goods) waste are comparable to near surface disposal facilities for very low level waste. Already existing capacities at landfills for toxic waste might be usable for low level waste [9]. But it has to be considered, that in case of a failure of the technical barriers of the landfill the impact on the environment will significantly increase. The health effects by incorporation of radioactive substances might be of minor importance compared to the toxic substances, but the effort for remediation will be much higher. In any case an additional safety assessment is required. This would not be suitable for intermediate level wastes. One advantage is that radiation will eventually disappear, unlike the other hazardous and toxic wastes. For some governments, the very low level radioactive waste can be stored in hazardous waste facilities.

4.3 Permanent Storage

A different solution can be found in the Netherlands. Radioactive waste has to be stored in a central interim storage (COVRA), designed for an operation of 100 years. Actually, already conditioned waste (supercompacted and cemented) will be checked for their specific activity. Those drums below the Dutch clearance values, are opened, sorted and cleared as conventional waste. By this way COVRA could increase their capacity significantly (s. fig. 1) [10]. For some countries generating only small amounts of waste this strategy of permanent storage with subsequent clearance might be fully sufficient, especially, if the half-life of the nuclide is short. Additional individual considerations for a specific clearance might be helpful, if an enhancement of clearance values is radiological acceptable on the basis of the de minimis concept. Independent of the consideration about the amount of radioactive waste such a solution makes sense even for countries with high amount of radioactive waste, because permanent storage enables the use of the option clearance and goes easy on the resource capacity of a disposal facility. Such a strategy is under discussion for example in Switzerland.

Figure 1: Number of clearable drums in Dutch permanent storage COVRA [10]



4.4 Small scale disposal facility

If the above mentioned alternatives are not applicable or do not fulfill the radiological requirements, the following alternatives should also be investigated:

- New construction of a small scale near surface disposal facility for radioactive waste
- Use of an already existing mine or tunnel
- Development of a borehole disposal concept appropriate for more categories of wastes besides the sealed sources The already existing concept published by IAEA [11] is related only to the disposal of sealed sources

A commensurate solution is possible on the basis of a for each country individual consideration of the following parameters:

- The waste properties; like specific activity, half-life, amount, chemical waste form etc.
- The technical conditions; like an appropriate disused mine or permanent storage already existing
- The geological conditions; like site selection for a new near surface or geological disposal facility
- The legal conditions; like use of specific clearance values.

4.4.1 Use of an already existing mine

For discussion of the use of an already existing mine as a disposal facility the following aspects have to be considered:

- Geological situation, the system of natural barriers: Is a proof for long term safety possible and for which time duration is it necessary? Which additional measures are necessary to keep the safety requirements? For this case additional concrete structures for sealing the drums with radioactive waste from the host rock can be helpful. The safety parameters have to be calculated on the basis of the radioactive inventory, which might be brought into the disposal facility in future. The aim is to prevent radioactive material from coming into contact with groundwater in which it could dissolve, as this is the principal route by which radionuclides could be transported from a disposal facility through the host rock to the near surface, where it can affect humans.
- History of the mine, rock stability: stability of caverns and pillars especially in old mines has to be checked under consideration of the planned operational life time. Additionally, two shafts, a good ventilation and ways for rescue and emergency are state of the art requirements. In general, the use of the mine for the time of operation and closure as disposal facility has to be added into the safety assessment. This can cause a significant effort for repair, reconstruction, and maintenance. Additionally, measures for backfilling of empty caverns have to be taken into account.
- Robustness against incidents and events: In old mines shafts are often not appropriate for transport of radioactive waste. A design including the drop of waste packages into the shaft as well as earthquakes is necessary. The results of calculations of the potential dose in case of such incidents and events must demonstrate that the legal requirements are not exceeded.
- Site selection: In case of selection between different sites logistical aspects for the transport to the site and the infrastructure at the site have to be considered as well as the public acceptance in the surrounding communities.

In consideration of all these points it becomes clear that a disused mine could meet all the safety requirements with a small need for reconstruction would be a very good choice.

4.4.2 Extension of the borehole disposal concept

Borehole disposal is to dispose of items in a vertical cylindrical hole underground. There are two types – shallow and geological boreholes. If the waste is below a depth of 150 m it is considered as geological disposal. The current use of a borehole is designed for the disposal of disused sealed radioactive sources generally. The extension of this concept is to make the whole diameter slightly bigger and have waste canisters placed into the hole. The borehole could be up to 5 km deep and it would be lined to prevent water from filling the borehole. The waste would be placed in the whole, fill placed around the waste, a spacer to the next waste canister and it would be filled up to an appropriate level depending on groundwater levels. The advantage of this method is that it does not rely on creating tunnels, inspection systems or ventilation systems. With the mining knowledge and capabilities a borehole down 1 or 2 km is possible now which could be used. This is a much cheaper form of geological disposal. The packages will have to be stronger as there will be tons of force on each package.

4.5 Other disposal concepts

Other disposal concepts are also discussed:

- Subsea burial: boreholes under the ocean as another level of protection. The boreholes could be shallower and the capping will increase over time through sedimentation. This method has very little public support, and is more complicated and costly than land based borehole disposal. This method is banned by international treaties
- Subduction zone burial: emplacement of waste in land, which is slowly moving under another tectonic plate. The idea is that eventually the waste will be in magma and dissolved in the fluid rock. This method has never been implemented as the uncertainties around earthquakes and eruptions are too high.
- Use of already contaminated areas (nuclear weapons testing sites): Use of an area within a nuclear test site (above or below ground) for disposal of radioactive wastes, or use of contaminated tunnels for waste placed by robots. This could only be used by a small number of countries, and would have to demonstrate adequate radiation protection to all workers to be enacted. This option is used by Kazakhstan.

5 CONCLUSION

Countries without a nuclear power programme may produce radioactive waste, and have to responsibly deal with that waste. As there is not commercial power generation, there is also not the levy on power consumption that goes to a waste fund. There is not the money set aside for the disposal of radioactive wastes generated in these countries. Therefore these countries look for more cost effective disposal routes for the wastes that they produce.

With the wide variety of radioactive wastes which are produced, the simpler forms of conditioning and disposal are more suitable for countries with a small radioactive waste inventory. The first step is to understand the waste that the country has, and will generate. This should all be reported in the national reports to the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management as coordinated by the IAEA. The next step is to understand the options that are available to the country.

The most practical solution is a multi-national repository, where the countries pool resources to build a larger facility than any one country could do on their own. However there are many political problems which are not yet solved. The states may look to co-disposal with other hazardous wastes, or for permanent storage until a disposal option becomes viable, whether that be exemption or a radioactive waste disposal site.

If these possibilities are not feasible, the next option is to create a small scale disposal facility based on existing technologies. This could be a smaller engineered concrete vault structure, the use of an existing disused mine for geological disposal or extending the borehole concept to take in other wastes. These smaller scale structures will still cost money, but not as much as for waste facilities for nuclear power plants.

There is a chance to combine existing possibilities and to fit them individually for each country. But it has to be considered, that:

- Governments in some of these countries have not realized the necessity of a final solution for the radioactive waste.
- Some countries might have proceeded in treatment of waste without knowing the final disposal solution; the problems may increase as the waste may need to be re-conditioned.
- At the moment there is no way for funding of a disposal facility.

Especially, the last item hampers small scale and affordable solutions. A support by the European Community in this direction can be useful for many countries.

6 REFERENCES

[1] IAEA http://nucleus.iaea.org/RRDB/RR/ReactorSearch.aspx?rf=1

[2] Council Directive 2011/70/Euratom of 19 July 2011 establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste.

[3] International Atomic Energy Agency (1998). Radiological Characterization of Shut Down Nuclear Reactors for Decommissioning Purposes. Technical Report Series No. 389, IAEA, Vienna.

[4] International Atomic Energy Agency (2003). Manual for Reactor Produced Isotopes. TECDOC-1340, IAEA, Vienna.

[5] D. Bosbach: Research for the safe management of radioactive waste in the Helmholtz Association FS-Symposium Zwischenlager-Dauerlager-Endlager, 22.-24.09.2014, Mainz.

[6] www.igdtp.eu

[7] http://www.world-nuclear.org/info/Nuclear-Fuel-Cycle/Nuclear-Wastes/International-Nuclear-Waste-Disposal-Concepts/

[8] Developing multinational radioactive waste repositories: Infrastructural framework and scenarios of cooperation, IAEA TECDOC 1413, Vienna 2004.

[9] M. Dützer: Very low level waste disposal in France: a key tool for the management for decommissioning wastes in France, StrahlenschutzPraxis 2/2015

[10] J. Wellbergen, E. Verhoef, R. Wiegers: Radioactive waste: from cradle to cradle TÜV NORD Symposium Provisions for Exemption and Clearance 21-23 Sept. 2009, Wiesbaden, Germany.

[11] Borehole Disposal Facilities for Radioactive Waste, IAEA SSG-1, Vienna 2009.

Radiation Protection In The Dismantling of Nuclear Power Plants

Oscar González[a], José Campos[a], Teresa Ortiz[b*]

[a]Radiation Protection Service in the Dismantling of José Cabrera Nuclear Power Plant, Emilio Vargas, 7 28043 Madrid, Spain

[b]Radiation Protection Technical Unit, Emilio Vargas, 7 28043 Madrid, Spain

Abstract. This paper discusses the problematics of protection from radiation when dismantling a nuclear power plant with a water-pressure reactor of 150MW, designed by Westinghouse, which went into operation in 1968. A French-designed, natural uranium and graphite gas reactor has already been dismantled at level 2, as have an experimental pool-type reactor and various research facilities related with irradiated fuel. This paper summarizes the experience acquired by facing the challenges of applying radiation protection when dismantling of nuclear facilities. It also identifies a few solutions and lessons learned.

KEY WORDS: radiation protection; ALARA; dismantling; hot particles.

1 APPLICATION OF RADIATION PROTECTION IN DISMANTLING OPERATIONS

1.1 Introduction

ENRESA, the Spanish Radioactive Waste Management Agency (Empresa Nacional de Residuos Radiactivos) has dismantled the Vandellós 1 Nuclear Power Plant's experimental reactor and several research facilities at the Energy, Environment and Technology Research Center (Centro de Investigaciones Energéticas Medioambientales y Tecnológicas, CIEMAT). Currently, the José Cabrera Nuclear Power Plant is being dismantled. The plant houses a water pressure reactor. From the point of view of radiation protection, dismantlings have a series of peculiarities that span from continual change in the sort and level of the radiation and conventional risks, up through actions on equipment and systems that have never before been interfered with. To this should be added the gradual decommissioning of protection systems that were available throughout the operations, which must be replaced by portable devices.

Many of the workers who participate in these projects are not used to working in dismantling activities; in fact, many of them are not even used to working with ionizing radiations. Finally, the volume of residual materials to be managed, both radioactive and conventional waste, is very high, since good segregation and characterization of the same is necessary. This article describes the application of Radiation Protection (RP) in dismantlings, and goes into detail on the most relevant aspects of the dismantling currently underway at the José Cabrera Nuclear Power Plant. References [1] to [9] include more information on specific aspects of RP and the lessons learned in the dismantling processes already carried out by Enresa.

1.2 Radiation Protection Organization

During the execution of a dismantling process it is necessary to strengthen the organization of RP that has been implemented during the operational stage, since the RP actions are heightened, on both the planning and execution levels. The organization of RP can be structured in the following manner:

- RP Leadership: should possess at least two diplomas as RP Leaders issued by the Nuclear Safety Council (CSN) specifically for dismantling.

* Presenting author, e-mail: torr@enresa.es

- Technical office: monitors the requirements of workers exposed, training and issuance of reports. It may also be involved in the evaluation of measures taken in the declassification of materials, parameters and floors.
- Operational RP: in charge of radiological monitoring and control of work, of the issuance and control of Radiation Work Permissions (RWP), of operational doses and of all aspects related with the application of ALARA criteria.
- Laboratory and Radiological Measures: includes measurement in the laboratory of operative RP samples and of effluent, of control and verification of the laboratory equipment and of the portable equipment, and declassification measures.
- Environmental area: carries out activities associated with the Environmental Radiological Monitoring Program and with the Environmental Monitoring Program, and with the calculation of the exterior dose.

1.3 ALARA Program

A very important aspect of a dismantling process is the application of ALARA criteria, in which all of the staff at the facility need to be involved. They must be aware of the radiation risk of the measures that are required so that it is kept to a minimum. Furthermore, contracting companies must take on the commitment and promote ALARA culture, of proposing and implementing ALARA techniques and methods and of ensuring proper training and education for their staff.

The organizational structure, on which the ALARA program is based, should be multidisciplinary, and include participation of execution, maintenance, material management, contractor and worker services. The structure is made up of an ALARA Committee and of ALARA groups. The ALARA Committee is made up of the site management and of the different organization managers. Their chief tasks are establishing and tracking the ALARA goals, along with revising and analyzing the ALARA work and techniques that may be implemented. ALARA groups made up of RP personnel and of the execution managers for all projects are created for specific projects with higher radiation risk, which require ALARA study.

ALARA study is another very useful tool for minimizing the dose involved in projects. It is required based on the criteria established beforehand and which are related to the estimated collective dose, the current level of superficial or environmental contamination, the dose rate effectiveness factor in the area, the length of the project, etc.

1.4 Staff radiation monitoring

Unlike normal dismantling operations, there is a stipulated need for monitoring of bioanalysis of excreta, in the case of presence of pure alpha or beta emitters. The scope of these bioanalysis programs is defined by the number of isotopes present. In the case of alpha emitters, the detection of trace-level activity, with the associated uncertainty, may spiral into a heightened dose assignment, when, as is common, the moment of incorporation or physical-chemical characteristics of the contaminant are unknown, which may imply the use of conservative factors that imply higher doses than those that the worker has really received.

1.5 Project radiation monitoring

Radiation monitoring in dismantling projects is not fundamentally different than the monitoring conducted in normal operations or in refueling, although it does have a few peculiarities that should be borne in mind, as is indicated below.

- Simultaneous, continuous, evolving dismantling projects in parts of the facility's radiation areas, leading to a radiological, zonal and access reconfiguration that is nearly continuous, as well as a reconfiguration of the associated radiation monitoring systems.

- The reduction, and even cancellation, of the fixed radiation monitoring systems insofar as dismantling is taking place, along with the need to monitor the risks with greater importance in the dismantling processes with respect to normal operations, such as environmental contamination risk, which is noted to be rising both by the cutting operations and by the disappearance of fixed ventilation systems in the plant.
- The changing and hardly predictable evolution of the irradiation levels and contamination that must be continually monitored, especially the superficial and environmental alpha levels of contamination.
- Increase in the production of both primary waste materials from the dismantling and technological waste of a secondary nature. This situation is best dealt with by increasing the measuring capacity, in both the number and type of equipment, and in the number of staff members in charge of radiation monitoring.

Furthermore, strictly monitoring the conditions imposed in the work orders and in the Radiation Work Permissions (PTR) helps ensure that the radiation status of the plant does not change without knowing what direct monitoring of projects requires. The use of techniques and special monitoring equipment, such as portable flares to measure in real time the environmental contamination, with the ability to detect alpha-beta-gamma and control and remote visualization in all of the areas of maintenance liable to present a risk of environmental contamination are very helpful when carrying out radiological control of work. Also, the use of television cameras enables monitoring of projects from the control position of RP, reducing the need for continual presence in the cuts.

1.6 Material radiation monitoring

The management of materials that are produced in dismantling processes is another key aspect of RP. All of these materials can be initially divided into three large groups: conventional materials from non-radiological areas, unclassifiable materials from radiological areas (conventional or with radioactive content) and radioactive waste. Materials from the unclassifiable group must be subjected to the process of declassification that is associated with a series of controls from the moment in which they are produced.

The functions and responsibilities of the RP Service in relation to the management of materials are as follows:

- Making radiological determinations in the process of material control, as well as the accompanying determinations.
- Carrying out periodical checks of the equipment and measurement systems for which they are responsible.
- Updating and correcting files of the material management units and their delivery to the Classification and Control area.
- Preparation of documents for which they are responsible as outlined in the Material Management Plan and in its procedures.
- Issuing final certifications of the content or the absence of radioactive material. To that end, use of portable tools typical to RP (radiometers, contamination gauges with gas or flicker probes, or portable spectrometers with INa detectors, etc., along with spectrometry equipment with portable germanium detectors (ISOC) and with fixed (Box-Counter style) spectrometric equipment for such activity. Detailed information on this entire process can be found in reference [3].

2 DISMANTLING OF THE JOSÉ CABRERA NUCLEAR POWER PLANT

The dismantling and decommissioning project for the Nuclear Power Plant known as José Cabrera is the first that corresponds to a power plant that uses light water in Spain. It is also the first whose objective is to reach Level 3 of dismantling (complete restoration of the site of the facility).

2.1 Antecedents and particularities

The planning and design of the projects currently underway, which has so far reached 70% completion, include attention to certain particular aspects that are derived both from special characteristics of the setting and from the existence of the specific regulations being newly applied.

The José Cabrera Nuclear Power Plant is a first generation plant, designed and built in accordance with standards in force at the time, when priority was given to nuclear security considerations, rather than specific aspects of radiological protection. At that time, the ALARA philosophy was yet to be developed. Despite the significant efforts made throughout its operational history, the aforementioned circumstance has imposed certain limitations on the possibility of modifying the original design and modes of operation, whose objective was radiological optimization.

The added existence of a Temporary Individualized Warehouse for the storage of spent fuel and special waste implies something new that was not present in earlier projects. This Temporary Individualized Warehouse, built on the site and first occupied with the transfer of the facility's title, will remain operational for most of the temporary project execution period.

One of the pre-dismantling activities that was carried out was a chemical decontamination process for the inside of the primary refrigeration circuit and annex systems. This decontamination's objective was to extract from the systems the greatest possible amount from the activity of liquid circuits that presented high dose levels. The disposal of this "source term" achieved a major reduction in the levels of radiation in the environment of the systems affected and has meant substantial savings in the collective and individual dose for the workers involved in dismantling tasks and management of material (waste) from the disassembly process.

Another aspect of particular importance has been the construction, equipment and startup of the Auxiliary Dismantling Building, a facility designated for the conditioning of those wastes that come from the dismantling of part of the larger components of the primary refrigeration circuit and of those parts inside of the reactor that may be accepted for disposal as intermediate- and low-activity waste (LILW).

For radiation protection, of special concern are the design and operation of the controlled ventilation, the monitoring limits on the level of environmental activity in the work areas, the methodology employed in the categorization and monitoring of waste materials, the declassification process and the implementation of the Culture of Safety within the project. The additions to these regulations have led to a significant effort in the planning, design and documentary development that make the project compatible with the circumstances imposed by the dismantling scenario.

2.2 Challenges and applied solutions

The focus of Radiation Protection and of the ALARA policy during dismantling, on both the planning and the design level, must be to remember all aspects such as those mentioned above.
The experience acquired in earlier dismantling processes offers an important advantage in terms of basic knowledge of the organizational models of management, coordination and instrumentation.
Nevertheless, certain aspects such as the physical configuration of the facilities and components, the radiological characteristics in terms of distribution, levels and particular types of risks differ substantially and condition many operational aspects of the implementation of Radiation Protection.

The characteristics of dismantling work, methodologies and techniques are determined by:

- The project's needs, such as the waste conditioning criteria, segmentation and size requirements, physical and radiological uniformity of the material to be conditioned, etc.
- The nature of the equipment and structures, their dimensions and geometry, their composite material, their location and accessibility, etc.
- The RP, Quality and Prevention requirements that regularly affect any work.

This conditional framework of obligatory compliance must be compatible with the efficiency and the safety in which development of a project, its program, its budget and, as far as is possible, within the global estimation of collective dose of the project, which, in this case, reaches 6.4 p.Sv.

The majority of the components, equipment and systems to be dismantled in the José Cabrera Nuclear Power Plant are metal in nature, of large dimensions and, to a greater or lesser degree, contaminated, and present significant dose rate effectiveness. Beyond the possibility of manual mechanical disassemblies, or "cold" cuts through the use of diamond wire saws, there are no "delicate" segmentation/material cutting techniques and we must turn to abrasive or thermal methods such as radial, oxy-fuel, plasma, etc.; all of these are liable to generating significant levels of environmental and superficial dispersion of radioactive contamination. The monitoring requirements for the ventilation and of the extremely low levels of activity required in the work area mean the obligatory design of solutions that are compatible with both needs and further guarantee an optimal individual protection of the workers involved.

Based on the options available and the characteristics of the job, some of the independent or simultaneous alternatives applied have passed for:

- Manual or remote decontamination prior to execution of the cut.
- Fixation of the contamination through application of paint or adhesive coverings.
- Installation of HEPS aspiration and filtering systems on-site at the intervention points.
- Construction of monitored watertight confinement vessels equipped with recirculated interior filtration and filtered aspiration driven to plant ventilation.

In many cases, these systems enable one to carry out heavy, aggressive, high-yield cuts in safe and controlled conditions. On the other hand, their use may require direct and complicated interventions by staff at hot radiation points and require intermediate movements and maneuvers with large pieces (Fig.1 and Fig. 2) from their original location to the cutting confined enclosures (SAS), leading to radiological and physical risks, which must be considered ahead of time.

Figure 1: Cut and transfer of contaminated walls

Figure 2: Transfer of the reactor vessel

One of the most representative activities of this dismantling, new in our country and in its level of technicality required, was the segmentation and conditioning of the internals of the reactor. The extremely high levels of radiation produced by these metallic elements activated (up to 1200Sv/h with majority presence of Co 60), demand that the segmentation areas, manipulation and per-a conditioning in baskets be carried out entirely in a remote manner at the bottom of the fuel pool and cavity of the reactor and with a shield between 7 and 9 m of water (Fig. 3 and Fig. 4). The techniques selected for segmentation have been those cut by cable saw, cutting discs and drills operated from a control bridge on the surface.

Figure 3: Underwater cut of the upper internal

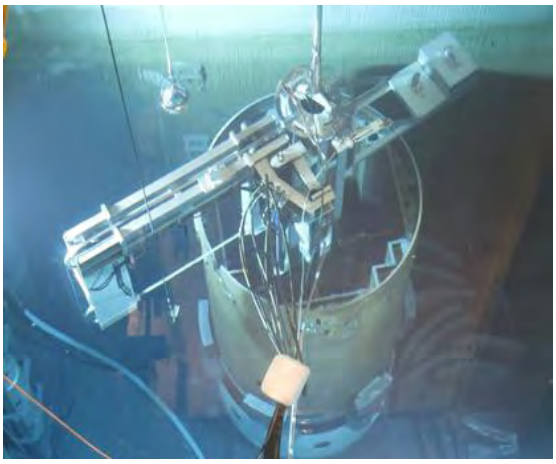
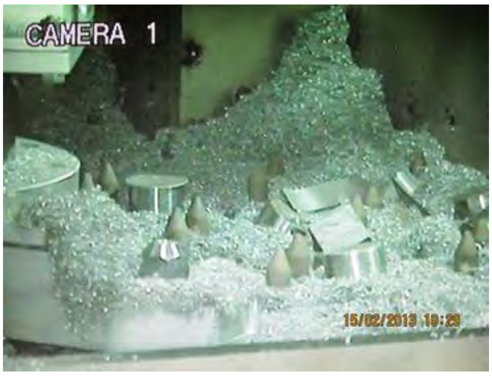


Figure 4: Cherenkov Effect

The main radiation risk associated with this activity is derived from the significant amount of metallic materials that are fragmented into very fine pieces produced by cutting tools. These hot shavings, fragments and particles, which are variable in size (they may be as small as one one-hundredth of a millimeter) can represent local dose rates between hundreds of mSv/h to several Sv/h (Fig. 5), liable to produce deep exposures and/or superficial exposures that are important to dispose of, in the second case, the threshold of deterministic effects, this circumstance requires a stepping-up of monitoring and surveillance of situations that may give rise to such exposures, such as:

Figure 5: Shavings of internal cuts (hot particles)



- Unnoticed or uncontrolled extraction of tools submerged with particles or fragments applied with doses in the area from dozens to hundreds of mSv/h. • Direct manipulation of material in the repairing of equipment, substitution of cutting elements, changes of device position, etc., extracted from the water that have been improperly decontaminated or checked radiologically and present hidden hot particles.
- Uncontrolled dispersion of secondary cleaning waste and decontamination with particles or fragments of activation.
- Direct contamination for prolonged periods of time (minutes to hours) on the skin or clothing of the workers involved in the work.

The evidence of these radiation risks and their prevention requires establishment of rigorous planning procedures, control and monitoring of people and materials, protocols on the use of personal and collective protections and the specific instrumental means of detecting, measuring and conducting radiological characterization, and methodologies for evaluating the personal dosage by contamination by "hot particles."

3 CONCLUSIONS AND LESSONS LEARNED

ENRESA, as the party legally responsible for the execution of dismantling and nuclear projects, has acquired substantial technical and managerial experience. This experience is naturally to be shared with the organizations and businesses that collaborate with the Regulatory Body, enabling the development of stronger improvements; it advances and gives content to the application of Safety Culture and materializes references for the optimization of projects that may be usable on a global level. The application of these lessons learned in the sphere of Radiation Protection covers numerous aspects, including:

- Anticipation of the identification of the particular risks derived from the various methodologies of cutting and quartering complicated materials and equipment from the dimensional and radiological point of view, including the development of programs and procedures focused on the prevention and control of non-habitual risks deriving from these activities: high levels of environmental contamination, the presence of hot particles, the presence of artificial alpha contaminants, etc.
- Harmonization of prevention and protection methods when safety and hygiene are added to the radiological risks.
- Optimization of the processes of design, construction and of controlled confinement spaces for quartering with more thermal or aggressive techniques (Fig. 6).

Figure 6: Thermal cut of the Vapor Generator's casing

- Improvement of the chemical and/or mechanical decontamination criteria to reduce the source term.
- Establishment of criteria for the selection, acquisition and use of measuring instruments, radio protection and laboratory (robustness, detection capacity, indication in real time, wireless control, transportability, autonomy in repairs and maintenance, etc.).

- Systematic implementation of dosimetric techniques and programs staff that consider the risks of internal exposure by transuranics and external localized exposure from hot particles.
- Adaptation of content and methods of training and entertainment for workers exposed to the particularities and specificity of the jobs to be done.
- Adaptation of the criteria for the application of the ALARA policy, converting it into a tool for the efficiency of projects, simplifying the analytical methods and prioritizing optimization and safety above compliance with the dose estimations.

4 REFERENCES

[1] T. Ortiz, M. Ondaro, I. Irún. "ALARA durante las operaciones de desmontaje de equipos y descontaminación de las piscinas de combustible gastado en la central nuclear Vandellós I". Radioprotección 24: 20-26; Vol VIII (2000).
[2] T. Ortiz, M. Ondaro, I. Irún, J. Just. "Evolución del estado radiológico de una central nuclear en fase de desmantelamiento y clausura". Radioprotección 25: 21-29; Vol VIII (2000).
[3] T. Ortiz, M. Ondaro, I. Irún, J. Just. "Gestión radiológica de materiales de desmantelamiento". Radioprotección 28: 28-39; Vol IX (2001).
[4] T. Ortiz, M. Ondaro. "Aspectos de Protección Radiológica del desmantelamiento de C.N. Vandellós I". Radioprotección 37: 22-30; Vol X (2003).
[5] T. Ortiz, O. González. "Radiological protection during the dismantling of Vandellós I nuclear power plant". 12th Congress of the International Radiation Protection Association (IRPA-12).2008. 19- 24 October. Buenos Aires. Argentina.
[6] T. Ortiz, O. González. "Protección radiológica durante el desmantelamiento de C. N. Vandellós 1". III Congreso SEFM-SEPR. Alicante 2-5 de junio de 2009.
[7] T. Ortiz, O. González. "Lessons learned in Radiological Protection during the dismantling of Nuclear Facilities". Proceedings of Third European IRPA Congress. 2010. June 14-16. Helsinki. Finland.
[8] T. Ortiz, O. González. "Radiological Protection during the dismantling of nuclear facilities". 13th Congress of the International Radiation Protection Association (IRPA-13). 2012. 13-18 May. Glasgow. Scotland.

German Guidelines put into Practise: Inhalation or Ingestion?
A study of a specific case of incorporation in an incident at a facility for dismantling nuclear installations

Peter Hill*, **Martina Froning, Burkhard Heuel-Fabianek**

Forschungszentrum Jülich GmbH, S, D-52425 Jülich, Germany

Abstract. The German legal frame work for radiation protection includes a guideline on incorporation monitoring. It foresees standard procedures for the assessment of dose. For occupational radiation workers the standard path of incorporation is assumed to be inhalation. At some working places ingestion might occur though and for certain isotopes the difference of dose for a specific intake even might differ by three orders of magnitude depending on the path of intake. For a specific example at a dismantling facility involving the nuclides ^{90}Sr,^{137}Cs and ^{241}Am it is shown how the German Guideline on incorporation monitoring can be put into practice in complicated cases. Using both information from urinary and fecal excretion as well as follow-up measurements the question 'inhalation or ingestion?' can be solved and the path of ingestion verified. In the specific case the effective dose obtained for incorporation has been about 1400 times lower than the dose obtained using the guidelines standard evaluation procedure based on inhalation.

KEYWORDS: incorporation monitoring; ingestion; special monitoring; dismantling facility.

1 INTRODUCTION

The german legal frame for incorporation monitoring at nuclear installations consists essentially of the nuclear law [1], the radiation protection ordinance [2] and a guideline on incorporation monitoring [3]. The guideline foresees standard procedures for the assessment of dose. For occupational radiation workers the standard path of incorporation is assumed to be inhalation. Only in the case of H-3 the incorporation path through the skin is considered additionally.

It is known, that in some cases incorporations through ingestion at occupational workplaces might occur. For many isotopes a wrongly assigned incorporation path will lead to a faulty dose which is not too different from the true dose. However for isotopes with a very low absorption rate in the intestines the difference in dose between the inhalation path and the ingestion path for a specific intake might reach 3 orders of magnitude. In such cases it is essential to distinguish the incorporation paths with respect to deciding whether dose limits have been kept or whether additional measures such as decorporation or medical treatment might have to be taken.

Using the example of a specific incorporation case at a dismantling facility it is shown how the German guideline on incorporation monitoring can be put into practice even in complicated cases involving the verification of the incorporation path (ingestion or inhalation?).

2 THE LEGAL FRAMEWORK

In the legal framework for occupational dose assessment and incorporation monitoring all relevant regulations are subordinated to the Nuclear Law (AtG) [1]. The most relevant document is the Radiation Protection Ordinance (StSchV) [2]. In agreement with European Basic Safety Standards [4] it regulates e.g. dose limits and the need for monitoring dose by competent monitoring bodies.

* Presenting author, e-mail: p.hill@fz-juelich.de

Subordinated to the Radiation Protection Ordinance are guidelines. They describe the practical implementation of general rules laid down in the Radiation Protection Ordinance. The guideline on incorporation monitoring implements §§40, 41 and 42 StSchV. It defines the general concepts of internal dose monitoring, the realization of monitoring, requirements on competent incorporation monitoring bodies and requirements on analytical and measuring procedures. Of special relevance to this study is the described procedure to evaluate body dose from measurement results.

New versions of international basic safety standards have been released recently [5,6]. For compliance with the EU directive the German regulations have to be adapted till 6 February 2018. Since radiation protection is becoming more important outside nuclear facilities[1] it will in the future no longer be subordinated to the Nuclear Law but be subject of a Radiation Protection Law. This work is still in progress. It can be expected that the procedures laid down in the guideline on incorporation monitoring [3] for occupational radiation workers will prevail.

3 ASSESMENTS ACCORDING TO THE GUIDELINE: A SPECIFIC CASE

3.1 The case

At the facility involved a decommissioned nuclear installation is being dismantled. Occupational radiation workers take part in a routine incorporation monitoring scheme which has been set up based on a nuclide vector determined beforehand. Reference nuclides include Sr-90, Cs-137 and Am-241. Feces and urine samples are taken and analyzed at a monitoring interval of 180 days. Body counter measurements are performed as well with the same time interval.

A temporary worker took part in some routine operations at a controlled area in the facility. Beside other means safety measures involved wearing of protective cloth and working under respiratory protection. The work was closely surveilled by radiation protection staff. The assignment of the worker was only for a short period of time and hence became subject to a special monitoring scheme. The same sampling and measuring scheme did apply as for the permanent staff. The assessments were however carried out at the beginning and at the end of the working period of 11 days. Due to sampling the monitoring period was effectively 10 days for excretion analysis. All measurements have been performed by the radioanalytical laboratory and the whole-body counter facility of the Competent Incorporation Monitoring Body Jülich (CIMB-J).

At the beginning of the monitoring period all results were below the detection limit with the one exception of Sr-90 in feces. However, the value was within the range of excretion of Sr-90 taken in from natural sources.

The samples taken at the end of the monitoring period showed quite different results. The fecal sample contained 140 Bq/sample Sr-90 and 8.1 Bq/sample Am-241. The urine sample contained 1.8 Bq/Sample Sr-90 and no detectable traces of Am-241. The body burden of Cs-137 was 120 Bq.

The evaluation of body dose from the measuring values followed strictly the procedures laid down in the German guideline on incorporation monitoring [3].

[1] e.g. in medicine, in air transportation or working with natural radiation sources

3.2 A first assessment of body dose

The German guideline for incorporation monitoring foresees a reference procedure in evaluating doses from the results. Standard parameters and assumptions are to be used. Those include:

- The incorporation path being inhalation
- calculating dose with an aerosolize of 5µm(AMAD)
- to use a specific solution class (as given in table 1)
- standard biokinetic behaviour of the nuclides in the body
- incorporation occurring in the middle of the monitoring interval.

The reference procedure intakes and effective doses have been used to calculate three nuclides (Sr-90, Cs-137 and Am-241). For Cs the calculation has been based on the results of in-vivo monitoring at a whole-body counter laboratory (body content). Otherwise the calculations have been based on the measurement results of the feces samples. The doses obtained are summarized in table 1.

Table 1: Result of dose assessment using the standard procedure (Inhalation)

Nuclide	Solubility class	Intake	Eff. Dose	Organ dose	Critical organ
Sr-90	S	10000 Bq	0.77 mSv	6.3 mSv	lung
Cs-137	F	120 Bq	1 µSv	1 µSv	(Uterus)*
Am-241	M	620 Bq	16.8 mSv	685 mSv	Bone surface

* Please note: in Germany dose assessment is unisex

The total effective dose of 17.6 mSv clearly exceeds the investigation threshold (6 mSv/a). The organ dose for the bone surface of 0.69 Sv in the case of Am-241 clearly exceeds the dose limit (0.3 Sv/a).

If investigation thresholds or dose limits are exceeded further measures have to be taken and the validity of the standard parameters for the specific case has to be tested. The dose evaluation has to be adopted to all known influencing factors.

A first check investigates the compatibility of urinary and fecal excretions. Intakes derived from the measuring results of the feces samples have been used to calculate expected urinary excretion values (table 2).

Table 2: Comparison of expected and measured urinary excretion (Inhalation)

Nuclide	Solubility class	Intake (from feces)	Urinary excretion (expected)	Urinary excretion (measured)
Sr-90	S	10000 Bq	1.3 Bq/d	1.8 Bq/d
Am-241	M	620 Bq	45 mBq/d	<0.25 mBq/d

In the case of Sr-90 there is at least some agreement between expected urinary excretion values and the measurement results. Solubility class F is not to be considered since there would be no relevant fecal excretion.

However, in the case of Am-241 there is a great discrepancy between the expected urinary excretion values and the measurement results. The discrepancy can neither be explained by changing the inhaled particle size nor by varying the date of intake. Could the incorporation possibly have been on the ingestion instead of the inhalation path? Hence the calculation has been repeated for the ingestion pathway (table 3).

Table 3: Comparison of expected and measured urinary excretion (Ingestion)

Nuclide	Solutibility class	Intake (from feces)	Urinary excretion (expected)	Urinary excretion (measured)
Sr-90	f_1=0.01	4500 Bq	1.3 Bq/d	1.8 Bq/d
Am-241	-	260 Bq	0.25 mBq/d	<0.25 mBq/d

For Sr-90 there is again a certain agreement between expected urinary excretion values and the measurement results. Unfortunately the urinary excretion rates of Sr-90 seem not to be sensitive to the ingestion path.

In the case of Am-241 there is now a much better agreement between the expected urinary excretion values and the measurement results. Please note that these calculations still assume an incorporation at the middle of the monitoring interval.
To evaluate the body dose more information has been required. To follow up an additional urine sample and two feces samples were taken a fortnight after the first sampling. The goal has been to finally establish ingestion as incorporation path and possibly get a better fix on the time of incorporation.

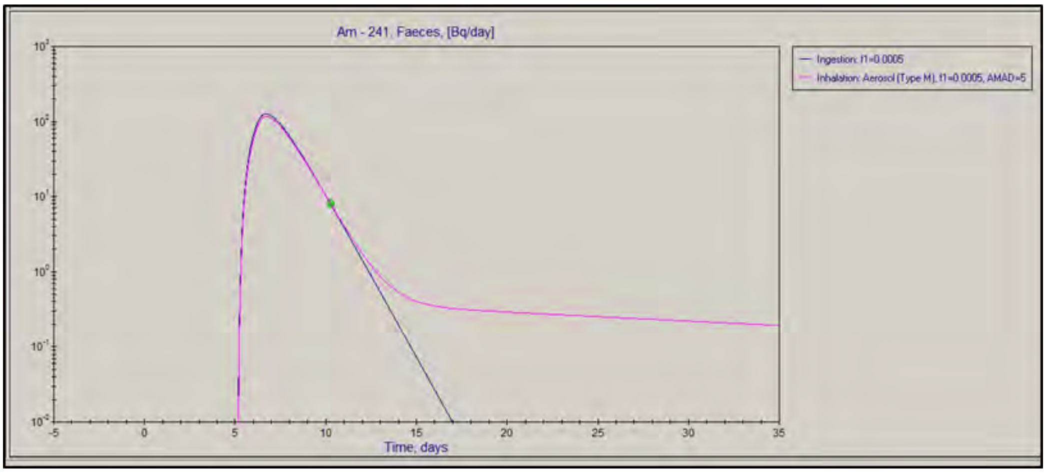
3.3 Assessment after follow-up monitoring

The follow-up samples have again been analyzed in the radioanalytical laboratory of CIMB-J. Fecal samples now contained 0.077 Bq/d and 0.073 Bq/d Sr-90 and no detectable traces of Am-241. The urine sample contained 0.036 Bq/d Sr-90 and also no detectable traces of Am-241.

A combined assessment of the first and the follow-up samples has been performed using the incorporation monitoring and dose assessment software IMIE 9.4 [7], which is based on ICRP78 [8]. The start of the monitoring period has been defined as day 0 of the time scale. Day 5 corresponds to the middle of the monitoring interval.

Figure 1 shows the time evaluation of fecal excretion rates for Am-241. Curves have been adjusted to the first measuring point. There have been no detectable traces of Am-241 in the follow-up fecal samples. The limit of detection for the sample taken on day 24 has been 0.5 mBq. This is about three orders of magnitude lower than the measurement value to be expected in the case of inhalation. This clearly proves the *path of incorporation* to be *ingestion*.

Figure 1: Fecal excretion curves for Am-241 (IMIE 9.4)

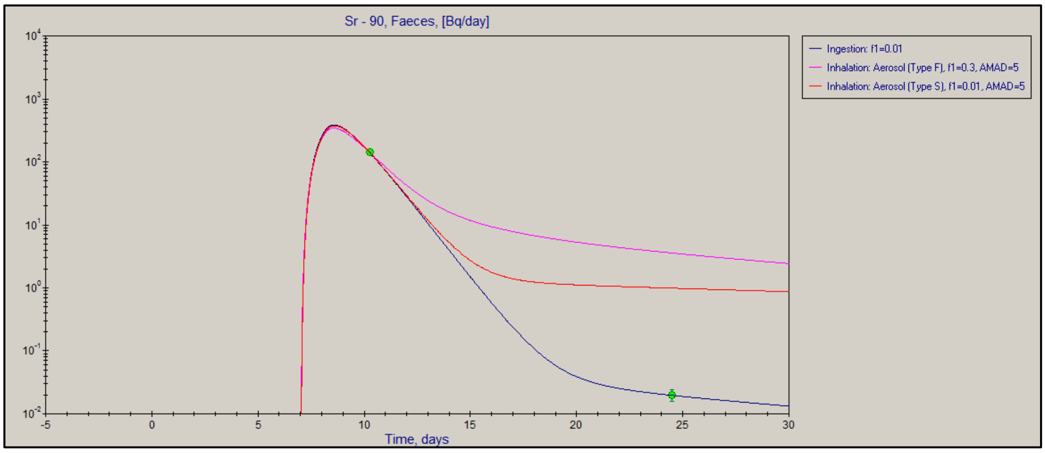


Obviously Am-241 results cannot be used to learn more about the time of incorporation since the second data point is below the detection threshold. Could eventually the data for Sr-90 yield more information?

For Sr-90 the time evaluation of fecal excretion rates is shown in figure 2. Again curves have been adjusted to the first measuring point. In the case of inhalation the curves do not agree with the additional data point at day 24 even if the time of intake is varied. The measuring value is about two orders of magnitude lower than it should be in the case of inhalation. Ingestion as the path of intake is confirmed.

Since the Sr-90 measurement results for the follow-up fecal samples are well above the limit of recognition the data points could be used to get information on the time of intake. The best agreement between the excretion curve and measured data is obtained, if *incorporation occurred on day 7* after start of the monitoring period. This is compatible with available information on the working schedules and the daily documentation of external dose.

Figure 2: Fecal Excretion curves for Sr-90 (IMIE 9.4)



To summarize: the data prove ingestion as the path of intake and suggest day 7 as time of intake. No further adjustments to the standard parameters are required.

Table 3: Result of an individual dose assessment (Ingestion)

Nuclide	Solubility class	Intake	Eff. Dose	Organ dose	Critical organ
Sr-90	f_1=0.01	900 Bq	2 µSv	5 µSv	Red bone marrow
Cs-137	-	120 Bq	2 µSv	2 µSv	(ovaries)*
Am-241	-	46 Bq	9 µSv	410 µSv	Bone surface

* Please note: dose assessment is unisex

Table 3 shows the result of the final dose calculation. The total effective dose is 13 µSv. This is 1400 times lower than inhalation would yield[2].

[2] For Am-241 the effective dose is reduced even by a factor of 1900 compared to inhalation.

4 FINAL REMARKS

It could be shown how the German guideline on incorporation monitoring is used in the assessment of a complex incorporation case.

Using both information from urinary and faecal excretion as well as information from follow-up samples the incorporation path could be proven to be ingestion. The time of intake could be verified.

The final assessment showed ingestion of Sr-90, Cs-137 and Am-241 and yielded a total effective dose of just 13 µSv. This is well below any dose limits or the investigative threshold. In the specific case the dose obtained for incorporation by the ingestion path has been about 1400 times lower than the dose obtained preliminary using the standard assumption of incorporation by inhalation.

Other cases of ingestion occurring during dismantling work are known and most likely connected with the process of changing protective garment.

The example underlines once more the importance of individual assessments in specific cases, which upcoming international guidelines also will foresee.

5 ACKNOWLEDGEMENTS

The authors acknowledge the contribution of the staff of the Competent Incorporation Monitoring Body JÜLICH (CIMB-J) who performed all sample preparations and measurements with great expertise and dedication. They thank Andreas Holz for assistance in the preparation of the poster.

6 REFERENCES

[1] FRG, 1959. Nuclear law. Gesetz über die friedliche Verwendung der Kernenergie und den Schutz gegen ihre Gefahren (Atomgesetz - AtG) vom 23. Dezember 1959, Neufassung vom 15. Juli 1985 (BGBl.I 1985, Nr. 41, S. 1565, zuletzt geändert durch Artikel 1 des Gesetzes1 vom 20. November 2015 (BGBl.I 2015, Nr. 46, S. 2053). (in German).

[2] BMUB, 2001. Radiation Protection Ordinance. Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung - StrlSchV) vom 20. Juli 2001 (BGBl. I 2001, Nr. 38, S. 1714, BGBl. I 2002, Nr. 27, S. 1459), zuletzt geändert durch Artikel 5 der Verordnung vom 11. Dezember 2014 (BGBl.I 2014, Nr. 58, S. 2010). (in German).

[3] BMUB, 2007. Guideline on physical radiation protection supervision in evaluating body dose; Part 2: Evaluation of body dose by internal radiation exposure; Incorporation monitoring (§§40, 41 and 42 Radiation Protection Ordinance) GMBl 2007 No 31/32 pp. 624ff (in German).

[4] Council of the European Union, 1996. COUNCIL DIRECTIVE 96/29/EURATOM of 13 May 1996 laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionising radiation; Official Journal of the European Union L 159, 29.6.1996, pp. 1ff.

[5] IAEA, FAO, ILO, et al., 2014. Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards. IAEA Safety Standards Series No. GSR Part 3. International Atomic Energy Agency, Vienna.

[6] Council of the European Union, 2013. COUNCIL DIRECTIVE 2013/59/EURATOM of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, and repealing Directives 89/618/Euratom, 90/641/Euratom, 96/29/Euratom, 97/43/Euratom and 2003/122/Euratom; Official Journal of the European Union L 13, 17.1.2014 pp. 1-73.

[7] IMIE 2007 (Version 9.4 GE) © H.Ratia,Y.Bonchuk, V.Berkovski.

[8] ICRP, 1997. Individual Monitoring for Internal Exposure of Workers (preface and glossary missing). ICRP Publication 78. Ann. ICRP 27 (3-4).

Investigation into the Pu uptake pathway in Corn (*Zea mays*)

Stephanie Hoelbling[a*], Fred Molz[a], Brian Powell[a], Nishanth Tharayil[b], Nicole Martinez[a]

[a]Department of Environmental Engineering and Earth Sciences, Clemson University, Clemson, South Carolina.

[b]Department of Agricultural and Environmental Science, Clemson University, Clemson, South Carolina.

Abstract. This paper investigates the potentially competitive uptake and translocation of plutonium (Pu) and iron (Fe) in corn (*Zea Mays*) to gain insight into the Pu uptake pathway. Plutonium has no known biological function in plants yet it has many chemical properties similar to iron (an essential nutrient) implying that Pu might be taking up into a plant through a Fe pathway. Consequently, a series of experiments was conducted in which two hydroponically grown corn species (one healthy and one deficient in the transporter protein for Fe) were exposed to complexed Pu and Fe. New phytoremediation strategies are useful in long-term environmental stewardship and remediation, and can be determined through understanding the factors contributing to the mobility of Pu. Preliminary results suggest plutonium may inhibit corn growth and uptake and translocation of plutonium in the absence of iron can occur.

KEYWORDS: plutonium; corn; uptake; translocation; phytoremediation.

1 INTRODUCTION

Iron (Fe) is an essential nutrient for plants, and it is responsible for redox reactions that drive plant respiration and photosynthesis [1]. Plutonium (Pu) upward migration in soil has been observed [2], a phenomenon that may be attributed to plant uptake of plutonium due to the similarities between Pu and Fe, such as the charge to ionic radius ratios for Pu(IV) and Fe(III), hydrolysis constants, and complexation constants with chelating agent DFOB [3]. These similarities may result in plutonium uptake if the plant requires iron or is iron deficient [3].

Understanding the mechanisms of plutonium and iron uptake in plants allows for the use of phytoremediation as a means to remove contaminants that may be found in the environment. Phytoremediation is an environmentally friendly and cost efficient solution, which allows for the removal of contaminants from the soil through degradation, stabilization, or translocation of contaminants from the soil into the plant [4]. Investigation of plant uptake mechanisms allows for more efficient phytoremediation techniques, and the targeting of specific contaminants, such as plutonium.

Phytoremediation success is primarily attributed terrestrial plants and their able to grow long, fibrous root systems [5]. The use of a chelating agent, such as citrate, nitrate or DTPA, has been shown to increase plant uptake of heavy metals, and even demonstrated increased uranium uptake potential [6]. The absence of chelating agents has shown to prevent translocation of plutonium into the plant all together, causing plutonium to remain in the root xylem [7]. Lee et al (2002) investigated the influence nitrate, citrate, and DTPA had on uptake success uptake of plutonium in sunflower and Indian mustard grown in soil and a hydroponic solution. It was concluded that the higher the plutonium concentrations in solution, the higher the plutonium uptake in both the shoots and the roots [5,8], and there was little difference between nitrate and citrate uptake levels, demonstrating either nitrate or citrate would be successful chelating agents for plutonium uptake.

Corn (*Zea mays*) has also demonstrated phytoremediation potential through phytoextraction, in which contaminants are removed from the soil by plant uptake and translocation of contaminants from the roots into the shoots. Corn is also heavy metal tolerant and can accumulate heavy metals like cadmium and lead [9]. Thompson et al (2010) conducted an investigation into the competition between Pu and Fe uptake in

* Presenting author, e-mail: shoelbl@g.clemson.edu

corn, concluding more plutonium was found in the roots rather than the shoots and iron deficiency results in increased plutonium uptake.

In this study, two different strains of corn were used to investigate the uptake potential of ^{242}Pu with the aid of two different chelating agents, DFOB and citrate. The similarities between the Pu and Fe complexation constants for DFOB [3] and the presence of Fe-citrate complexes in plants [1] allows for the comparison of the Pu uptake influence of two widely used chelating agents. The first corn strain is typical yellow corn, referred to as Trucker's Favorite, a normal, healthy strain. The second strain is referred to as Yellow Stripe 1 (YS1), which displays iron deficient characteristics, specifically yellowing on the leaves of the corn. YS1's iron deficiency is due to a lack of the YS1 gene, which is responsible for uptake of Fe into the roots from the rhizosphere [10], which is often referred to as the plant-root interface [11].

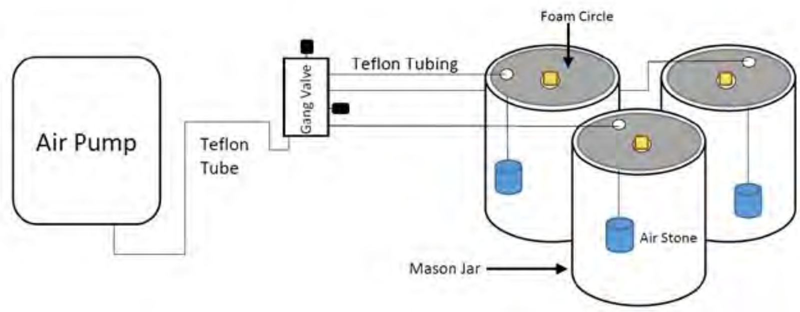
If plutonium is present in the shoots of YS1, then it can be concluded that the pathway between iron and plutonium in corn is different. However, if YS1 does not take up plutonium, then the pathways are the same. The specific objectives of this study include:

1. Quantify the uptake of plutonium (Pu) in hydroponically-grown corn, considering both healthy and iron-pathway-deficient strains.
2. Compare the uptake of Pu to that of iron (Fe) and determine if coincident exposure influences this uptake.
3. Compare the ability of the complexing ligands citrate and DFOB to facilitate plutonium uptake

2 METHODS

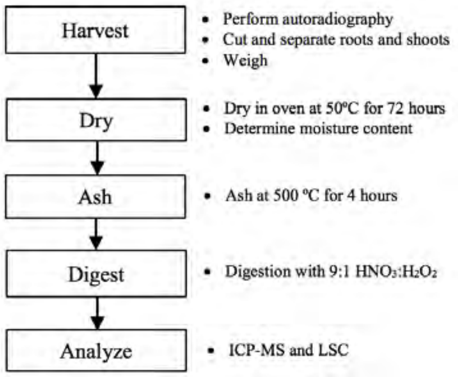
Corn kernels were allowed to germinate for approximately five days before being placed in the hydroponic solution to grow for approximately seven days. The base hydroponic solution consisted of the following compounds in milligrams/liter (mg/L): 946 $Ca(NO_3)_2{\cdot}4H_2O$, 150 KCl, 120 $MgSO_4$, 68 KH_2PO_4, 0.69 H_3BO_3, 0.06 $ZnSO_4{\cdot}7H_2O$, 0.024 $Na_2MoO_4{\cdot}2H_2O$, 0.022 $MnCl_2{\cdot}4H_2O$, 0.017 $CuCl_2{\cdot}2H_2O$, 0.60 $FeCl_3$ [12]. Note that the final compound was purposefully omitted in some experiments as described below. The pH of the hydroponic solution was be maintained around 6.0 [3]. Due to evaporation and plant uptake of the hydroponic solution, corn was set in foam sheets that floated on top of the hydroponic solution. A physical representation of this is shown in Figure 1.

Figure 1: Representation of the experimental set up for one group of mason jars, and each experiment had either three of four groups of mason jars. The air pump and air stones are used to provide oxygen for the roots of the plant in the hydroponic solution.



Harvesting of corn will occur approximately 7 days after being placed in the hydroponic solution. When harvesting, the roots will be rinsed with DDI, cut, and separated from the shoots. Figure 2 represents a harvesting and analysis process flowchart.

Figure 2: Sampling and analysis flow chart.



2.1 Determination of Plutonium and Iron Pathways

Two rounds of experiments were conducted to investigate plutonium and iron pathways. As previously mentioned, corn strains were grown in a hydroponic solution. The first round of experiments investigated iron uptake in four experimental groups of three samples (i.e. mason jars) each. The first group consisted of TF in iron-free HP solution with foliar fertilization, to 'mimic' the YS1 strain. The second group consisted of TF in iron-free HP solution yet did not receive foliar fertilization, so that no iron would be available to the corn. The third group consisted of TF in iron-supplemented HP. The final group consisted of YS1 grown in iron-supplemented HP and also provided foliar fertilization.

The second round of experiments investigated plutonium uptake and consisted of a set up that mirrored the iron experiments. The only difference is that the hydroponic solutions contained ^{242}Pu in place of Fe. Plutonium was complexed with either citrate or DFOB to investigate the impact of the particular complexes on the uptake and translocation of Pu in corn. Due to similarities seen in growth between TF with foliar fertilization and TF without foliar fertilization in the iron competition experiments, the latter group was removed, allowing for a larger sample size in each group.

Three experiments were conducted using this experimental setup. In the first two experiments 1 part per billion (ppb) and 10 ppb plutonium were complexed with citrate, respectively, and spiked in the hydroponic solutions. In the third experiment 1 ppb plutonium was complexed with DFOB and spiked into the hydroponic solutions. A summary of each experiment is represented in Table 1.

Table 1: Summary of the experiments conducted in this study. Please note the "experimental set up" is for all experiments, with TF (1) indicating one Trucker's Favorite group and TF (2) indicating the second Trucker's Favorite group.

Complexation Ligand	Concentration (ppb)	Activity Concentration (Bq mL^{-1})	Experimental Set up (all)
Citrate	1	18	TF: (1) HP (no Pu) with FF
	10	164	(2) Pu HP with FF
DFOB	1	18	YS1: Pu HP and FF

2.2 Preliminary Autoradiography Imaging

Autoradiography was used as a preliminary observation of plutonium uptake through the roots and into the shoots. One TF plant that grew in Pu HP solution and one YS1 plant were saved from the DFOB experiments. The TF plant was placed on an autoradiography plate and exposed for 5 days. After seeing the results, the YS1 plant autoradiography plate was exposed for 6 days. The plates were scanned and analyzed on a Typhoon FLA 7000.

3 RESULTS AND DISCUSSION

3.1 Growth Comparisons

The shoot and root growth comparisons for the iron uptake experiments are represented in Figure 3, and the root and shoot growth for 1 ppb (18 Bq mL^{-1}) ^{242}Pu complexed with citrate is represented in Figure 4.

Figure 3: Median shoot length (left) and median root length (right) for Fe uptake experiment of Trucker's Favorite and YS1 corn.

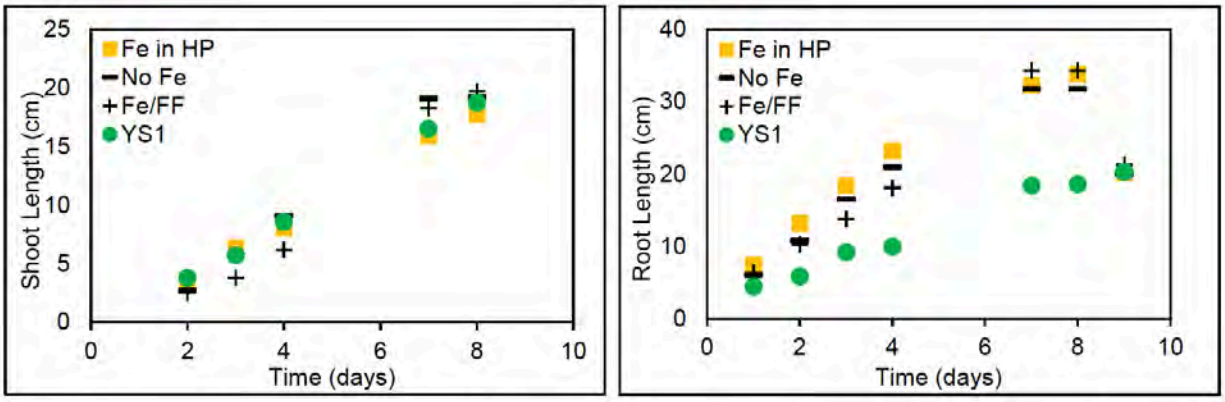
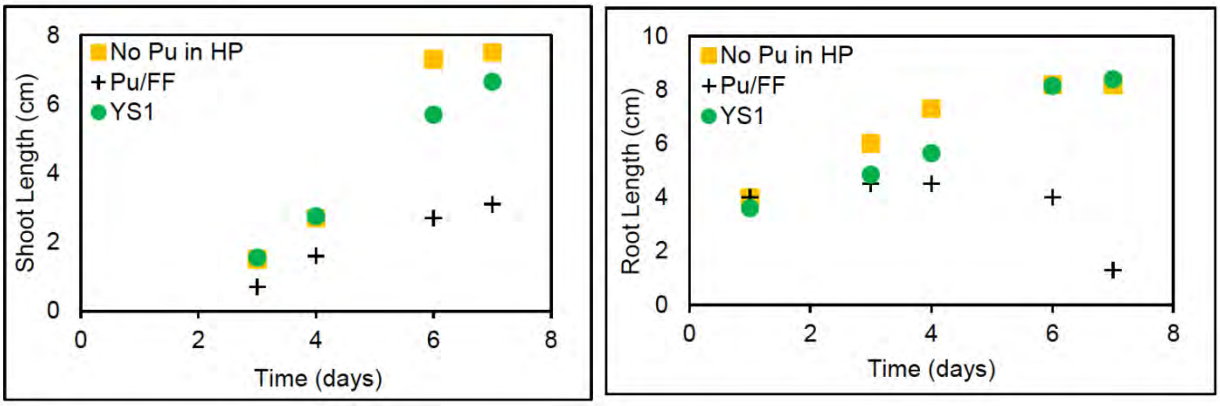
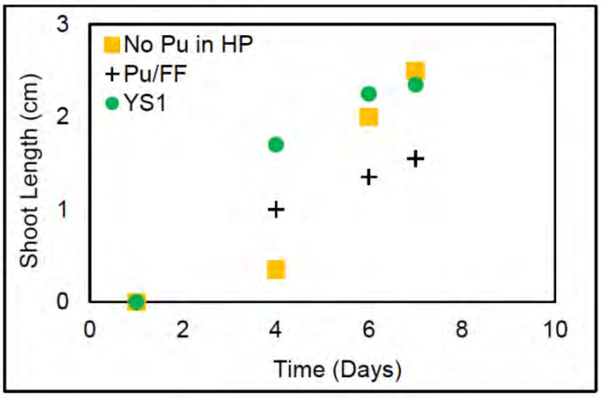


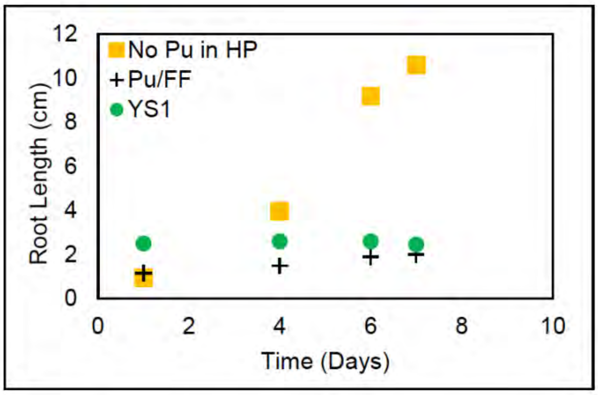
Figure 4: Median shoot length (left) and median root length (right) for Pu uptake experiment of Trucker's Favorite and YS1 corn grown in 18 Bq mL^{-1} Pu-citrate HP solution.



The iron uptake experiments demonstrated the shoots of TF and YS1 grew at a similar rate and length, indicating the foliar fertilization methods were sufficient for YS1 growth. Interestingly, the YS1 roots grew much less than the Trucker's Favorite. YS1 and TF (no Pu) roots and shoots grew similarly throughout the whole experiment; however, the roots and shoots did not grow as much as in the iron uptake experiments, likely due to the absence of iron (an essential nutrient) in the case of TF. The TF grown in Pu solution demonstrated less growth in the shoots compared to YS1 and even decreased in root length, which was due to unhealthy roots breaking off. The 10 ppb (164 Bq mL^{-1}) ^{242}Pu-citrate uptake experiments showed the root length for TF in no Pu HP grew significantly more than the root growth of either corn strained grown in Pu, as shown in Figure 5.

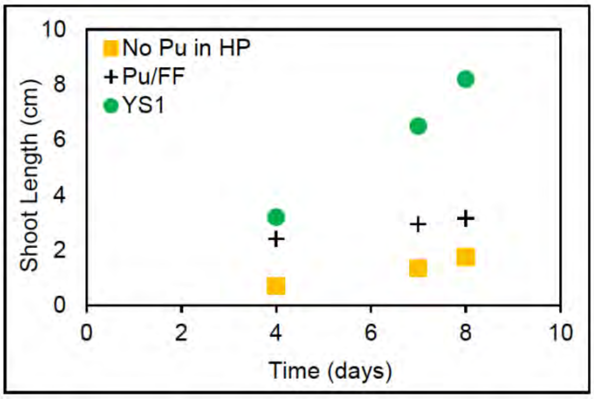
Figure 5: Median shoot length (left) and median root length (right) for Pu uptake experiment of Trucker's Favorite and YS1 corn grown in 164 Bq mL^{-1} Pu-citrate HP solution.

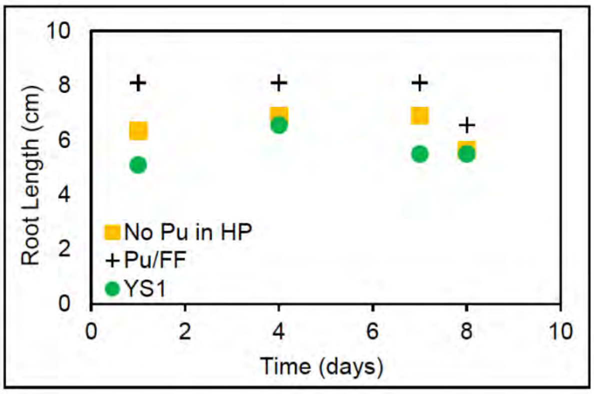




The shoot growth for YS1 and TF (no Pu) grew similarly during the last two days of the experiment, while TF in Pu grew slightly less. Compared to the 1 ppb ^{242}Pu-citrate experiment, the shoot growth of TF (no Pu) and YS1 grew similarly, but the YS1 roots grew more like the TF (Pu) when grown in a higher plutonium concentration. The 1 ppb (18 Bq mL^{-1}) ^{242}Pu-DFOB uptake experiments showed the root length for TF and YS1 remained relatively constant throughout the experiment, but grew significantly less than in the iron uptake experiments, which was similar to the ^{242}Pu-citrate experiments. YS1 shoots grew faster than TF, but growth was also inhibited. These results are represented in Figure 6.

Figure 6: Median shoot length (left) and median root length (right) for Pu uptake experiment of Trucker's Favorite and YS1 grown in 18 Bq mL^{-1} Pu-DFOB HP solution.





Overall, the growth comparisons demonstrated Pu inhibits growth for both strains of corn when comparing Pu versus Fe growth. Corn grown in 1 ppb Pu and 10 ppb Pu demonstrated different growth patterns, indicating higher Pu concentrations may further inhibit YS1 growth. Additionally, there was a large negative influence on root growth from Pu-citrate HP solution for TF(Pu), while there was little influence on root growth from the Pu-DFOB HP solution. These results suggest DFOB and citrate may allow corn to uptake Pu differently.

3.2 Preliminary Autoradiography Images

The preliminary autoradiography images provided insight into the uptake of plutonium into the shoots of corn. Trucker's Favorite grown in 18 Bq mL^{-1} ^{242}Pu-DFOB hydroponic solution compared to YS1 grown in the same solution shows less translocation from the roots to the shoots, as represented in Figures 7 and 8.

Figure 7: Autoradiography images of TF grown in 18 Bq mL^{-1} ^{242}Pu-DFOB hydroponic solution. A photo of the corn on the autoradiography plate (left) and the autoradiography image (left) is shown. The orange circle shows where the plutonium was found in the root of the corn plant.

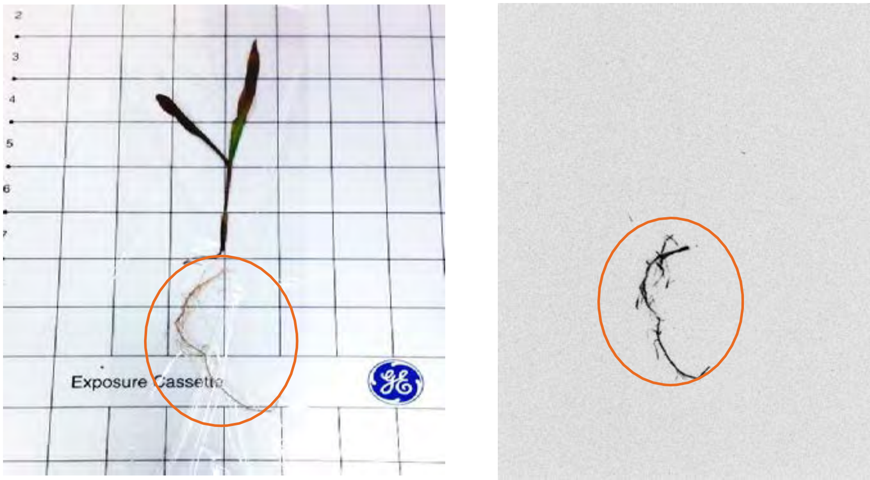
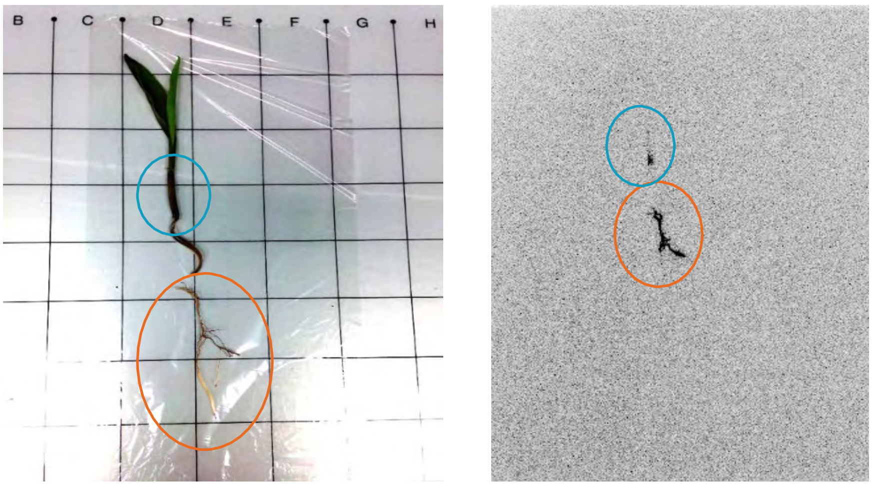


Figure 8: Autoradiography images of YS1 grown in 18 Bq mL^{-1} ^{242}Pu-DFOB hydroponic solution. A photo of the corn on the autoradiography plate (left) and the autoradiography image (left) is shown. The orange circle shows where the plutonium was found in the root of the corn plant. The blue circles show where the plutonium was found in the shoot of the plant.



However, the Trucker's Favorite autoradiography plates were only exposed for five days, and a longer exposure time may have been necessary to see Pu in the shoots.

YS1 grown in 18 Bq mL^{-1} ^{242}Pu-DFOB hydroponic solution demonstrates there is a large amount of plutonium present in the root region, which could be a result of sorption onto the roots. A small amount of plutonium is present in the low part of the shoots, indicating uptake and translocation. The exposure time for the YS1 autoradiography plate was 6 days, allowing plutonium to be visible in the YS1 corn strain.

Translocation was evident in the YS1 autoradiography images which suggest the Pu-iron pathways are not the same, or there may be more than one pathway for Pu. The autoradiography images also allowed for the establishment of suitable exposure times. For example, 5 days was long enough to see Pu in the roots but was too short to see Pu in the shoots and 6 days was when the first trace of Pu in the shoots was visible in YS1. Therefore, the plats may need to be exposed longer than 6 days to see more plutonium that could be present in the shoots.

4 CONCLUSIONS AND FUTURE WORK

The growth comparisons and preliminary autoradiography images provide significant insight into the uptake of Pu in corn. Plutonium inhibited growth for both strains of plant when compared to the growth of plants exposed to iron, and there was a difference in growth between Pu-citrate and Pu-DFOB. The little influence Pu-DFOB compared to the negative influence of Pu-citrate on root growth could be contributed to the similar DFOB complexation constants for Pu and Fe [3], and these differences would be important to consider when determining phytoremediation techniques, including which chelating agent should be used. Translocation was evident in YS1 corn which suggests the Pu-iron pathways are not the same, or the plant may uptake plutonium through multiple pathways. Also, a priori knowledge is essential for appropriate autoradiography image and preliminary autoradiography image work established suitable exposure times for future imaging.

However, future work on elemental and nutrient analysis for the roots and shoots will be conducted to quantitatively determine the concentration of Fe and Pu in the roots and shoots, and will further explain the growth trends and Pu visible in YS1 shoots. Also, competition experiments will be conducted in which Trucker's Favorite will be grown in HP solutions containing various Fe/Pu ratios.

5 ACKNOWLEDGEMENTS

I would like to acknowledge Dawn Montgomery and Dr. E. Miller Wylie for their help on this project. Funding for this project was provided by United States Nuclear Regulatory Commission Nuclear Education Grant #NRC-HQ-G-38-0002.

6 REFERENCES

[1] Kim, S., & Guerinot, M. L. (2007). Mining iron: Iron uptake and transport in plants. *FEBS Letters,* (581), 2273-2280.

[2] Kaplan, D.I., Demirkanli, D.I., Molz, F.J., et al. (2010). Upward movement of plutonium to surface sediments during an 11-year field study. *Journal of Environmental Radioactivity*, 101, 338-344.

[3] Thompson, S. (2010). *Mobility of plutonium in zea mays (corn): Determination of transport velocities, spatial distribution and correlations with iron.*

[4] Ruggerio, C., Deladurantaye, E., Duran, B., et al. (2003). Researching the phytoremediation of plutonium. *Actinide Research Quarterly,* (3rd4th quarter), 29-34.

[5] Lee, J. H., Hossner, L. R., Attrep Jr., M., et al. (2002). Uptake and translocation of plutonium in two plant species using hydroponics. *Environmental Pollution,* (117), 61-68.

[6] Huang, J. W., Blaylock, M. J., Kapulnik, Y., et al. (1998). Phytoremediation of uranium-contaminated soils: Role of organic acids in triggering uranium hyperaccumulation in plants. *Environmental Science and Technology, 32*(13), 2004-2008.

[7] Cataldo, J.R., McFadden, K.M., Garland, T.R., et al. (1988) Organic constituents and complexation of nickel(II), iron(III), cadmium(II), and plutonium(IV) in soybean xylem exudates. *Plant Physiology*, 86, 734-739.

[8] Lee, J. H., Hossner, L. R., Attrep Jr., M., et al. (2002). Comparative uptake of plutonium from soils by *Brassica juncea* and *Helianthus annus*. *Environmental Pollution,* (120), 173-182.

[9] Wuana, R. A., & Okieimen, F. E. (2010). Phytoremediation potential of maize (zea mays L.). A review. *African Journal of General Agriculture, 6*(4), 275-287.

[10] Nozoye, T., Nakanishi, H., & Nishizawa, N. (2013). Characterizing the crucial components of iron homeostasis in the maize Mutans *ys1* and *ys3*. *Plos One, 8*(5), 1-11.
[11] McNear, D. H. (2013). The rhizosphere - roots, soil and everything in between. *Nature Education Knowledge, 4*(3).
[12] Garland, T., Cataldo, D., & Wildung, R. (1981). Absorption, transport, and chemical fate of plutonium in soybean plants. *Journal of Agricultural and Food Chemistry, 29*(5), 915-920. doi:10.1021/jf00107a007.

Application of USNRC Regulatory Guide 4.21 for Decommissioning Feasibility and Life-Cycle Planning
KHNP APR1400 Design Certification Project

Seunggi Lee[a*], Sangho Kang[a], Irving Tsang[b], Sara Amitrani[b]

[a]KEPCO E&C, Republic of Korea.

[b]DERADS, USA.

Abstract. KHNP (Korea Hydro & Nuclear Power) submitted an APR1400 design for United States (US) Design Certification application (APR1400 DC Project) in 2014. KHNP, assisted by KEPCO E&C, developed a Regulatory Guide (RG) 4.21 Program to include design features and commitments to develop operating programs and procedures in accordance with the requirements outlined in 10CFR20.1406 and RG 4.21. This paper provides a summary description of the RG 4.21 program and examples of the design features embedded in the APR1400 DC Project. The intent of the incorporation of these design features, coupled with the development and implementation of operating procedures and programs, is to minimize the spread of contamination within the facility and from the facility to the environment. Through the minimization of contamination, these design features and programs offer another degree of radiation protection to the workers and the public, thereby satisfying regulatory compliance and providing life-cycle planning of the facility.

KEYWORDS: Prevention of Unintended Release; Early Leak Detection; Prompt Assessment and Timely Response.

1 INTRODUCTION

There are three basic principles that form the foundation of RG 4.21 [1]:

- The design shall prevent unintended releases to minimize the spread of contamination;
- If unintended releases cannot be prevented, the design shall provide early leak detection to initiate operator actions; and
- The design of the structures, systems, and components (SSCs) shall facilitate the prompt assessment of leakages to support a timely and appropriate response.

RG 4.21 focuses on the minimization of contamination of the facility, minimization of contamination of the environment, facilitation of decommissioning, and minimization of waste generation. RG 4.21 provides over sixty control methods for the US NRC to examine the adequacy of a nuclear power plant's compliance with the requirements of 10 CFR 20.1406 and RG 4.21. These regulations and guidance are developed based on lessons learned from the issues pertaining to contamination that have arisen at operating plants as well as issues discovered during the decommissioning of old nuclear power plants at the end of their life cycle. Hence, RG 4.21 requires the facility design to provide due emphasis on the prevention/minimization of the spread of contamination from unintended leakage, thus minimizing the uncertainty of contamination levels during operation and decommissioning.

[*] Presenting Author, e-mail: lowo@kepco-enc.com

Unintended leakages may occur from the following areas:
- Embedded piping inside the facility
- Buried yard piping
- Overflows of equipment (e.g. tanks) containing radioactive material
- Spills and residues of cleanup liquids
- Component cross-leakages (e.g., heat exchangers, mechanical seals, etc.)
- Building underground penetrations

2 APR1400 RG 4.21 PROGRAM

The APR1400 RG 4.21 Program consists of four design and two operating objectives and involves sound engineering principles and nuclear industry operating experience for the implementation of these objectives as follows:
- Prevention/Minimization of Contamination
- Early Leak Detection
- Minimization of Cross-Contamination, Decontamination, and Waste Generation
- Decommissioning Planning
- Operating Programs and Procedures, and
- Site Radiological Environmental Monitoring Program

2.1 Objective 1: Prevention/Minimization of Contamination

Objective 1 involves design considerations to prevent and/or minimize radiological contamination at the system level through segregation by system characteristics and contamination levels. The systems with the highest contamination levels, such as the reactor coolant system, are housed inside containment for maximum protection and are designed with special materials considerations and chemistry control for life-cycle planning. Additionally, for systems located within containment, the spread of contamination is further controlled through the provision of a steel liner as a barrier for the positive containment of leaked fluids and the provision of sumps for the active management of leaked fluids inside containment. Other supporting systems, such as the chemical and volume control system, are segregated and housed in cubicles inside the auxiliary building with active liquid level control and leak management features. These systems are isolated from other systems, especially from systems with lower contamination levels, through segregation and the use of containment walls or dikes around contaminated SSCs. Leakage control for these systems is further supported by the provisions of leak detection instrumentation, drain collection, and double-walled piping where necessary.

In order to prevent unintended leakage, piping between buildings is routed in piping tunnels with liquid leak detection capability to indicate and alarm from leaks or infiltration water. Building penetrations are provided with a special seal design, piping sleeves, and drain collection for contaminated piping. As embedded or buried piping is one of the most susceptible sources for unintended and undetected leakage, the design emphasizes on the minimization of buried and embedded piping. As an alternative design, piping carrying highly contaminated materials within the facility is routed inside pipe chases as much as practicable and buried pipes outside the facility are routed in underground concrete tunnels which are provided with leak detection and worker access. The design also utilizes the use of nuclear-industry-proven technologies, reinforced quality control, and strict adherence to the applicable codes and standards.

2.2 Objective 2: Early Leak Detection

Objective 2 delineates the requirements for the incorporation of design features for early leak detection, in addition to the normal tank level controls and overflow protection through cross-tie piping. The APR1400 design incorporates early leak detection features for individual and separate components based on the detection of minimal volumes of leakage. The facility design includes the provisions of smooth surfaces to facilitate drainage, drainage collection points within close proximity to contaminated components, leak detection instruments located as close as practicable to the sources where potential leaks may occur, and adequate access pathways and spaces to allow for the prompt assessment of detected leakage and the execution of appropriate operator responses and mitigation as required.

2.3 Objective 3: Minimization of Cross-Contamination, Decontamination, Waste Generation

Objective 3 targets the design considerations that facilitate the segregation of radiological contamination at the component level. Components containing large quantities of contaminated materials during normal operation, such as tanks, are provided with proper cross-ties to manage overflows, operational interlocks and alarms, and administrative controls to avoid inadvertent bypasses and releases. The minimization of the potential for component failures is facilitated through the selection of proper materials and chemistry controls. Component layout for systems that contain significant radiological contamination, is designed with multiple features to minimize cross- contamination, including the use of sloped and epoxy coated floors and drainage trenches for the collection of drainage.

In order to minimize cross-contamination within the facility, the components are segregated in accordance with the contamination types (e.g., floor drains versus equipment drains) as well as by the levels and characteristics of radiological contamination. In this approach, tanks are grouped together and placed at the basemat level of the facility in individual cubicles. System filters and ion exchangers are separately grouped and placed in suitable areas for contamination control, waste handling, and solid waste transporting for internal storage. The components and their associated piping are provided with flushing and decontamination capabilities, and components are specified to have smooth and cleanable surfaces for easier decontamination. This approach meets the requirements for life-cycle planning for the facility from design, through construction, operation, license termination, and decommissioning.

2.4 Objective 4: Decommissioning Planning

Objective 4 focuses on decommissioning planning that facilitates decontamination and minimizes waste generation through modular construction, the minimization of buried and embedded components and piping, and the use of effective but compact SSCs that minimize the generation of radioactive waste during decommissioning activities. The decommissioning planning includes the provision of removable block walls to facilitate the removal of SSCs and the commitment to provide documentation and information in a centralized area for ready recovery.

2.5 Objective 5: Operating Programs and Procedures

Objective 5 relates to the identification of procedures and programs that support the operation, inspection, and maintenance of the features described in Objectives 1 through 4. This Objective further describes the requirements for the documentation of any incidents that may occur during the operating life of the facility. The documentation of these incidents, including spillages, leakages, overflows, and associated cleanup requirements, is an integral part of decommissioning planning. Since APR1400 is undergoing design certification of the standard design, this objective is met by establishing the commitments through combined operating license information items for the plant/facility utilities to develop effective procedures and programs for the implementation of RG 4.21. These procedures and programs include commitments to provide proper operation, inspection, calibration, and maintenance of the SSCs as well as to delineate operator actions to prevent/minimize the spread of contamination. The commitments also include performing periodic review of operational procedures to provide reasonable assurance that the procedures reflect lessons learned and other relevant updates, including personnel qualifications and training.

Accurate records and documentation of design, construction, modifications, and operational incidences facilitates safer operation as well as more effective decommissioning.

2.6 Objective 6: Site Radiological Environmental Monitoring Program

Lastly, Objective 6 provides the commitment to develop a Site Radiological Environmental Monitoring Program (REMP) that includes the evaluation of site-specific hydrogeology and potential contamination migration and groundwater transport pathways. The REMP provides the capability to assess the impacts of the construction and operation of the nuclear facility based on the hydrogeological characteristics of the site environment in order to establish a site-specific contamination monitoring program.

3 RG 4.21 KEY DESGN FEATURES

From the implementation of these design and operational objectives, the APR1400 design provides control measures to prevent and/or minimize leakages. As leakages and spills do occur occasionally, the design provides specific features to manage the unintended leakages, through the collection of liquid drainage and the provisions of high level signals to alarm for timely notification for operator actions.

The tanks in the APR1400 design are equipped with level instruments to indicate and alarm in the instance of overflow. For the tanks that are paired, the overflow lines are cross-connected for added flexibility to minimize leakage to the facility. Tanks containing and handling radiologically contaminated fluids are provided with individual leak detection features for prompt and early detection. The emphasis of the leak detection features is not only the detection instruments but the integrated approach of the design and SSC layout along with the supplemental operating procedures and programs that facilitate prompt detection of leakage as close as possible to the leakage source to allow timely intervention and to prevent the unintended and wider spread of contamination.

The integrated design approach consists of the following:

- Drain pipes are implemented to collect leakage from individual tanks. The drain pipe is designed with a level switch and a drain valve inside a valve pit. The drain pipe is sloped to facilitate faster collection of liquid to activate the alarm signal for operator action. The pipe is also sized to collect a small amount of liquid (about 0.5 liters) before the alarm is initiated. Since this feature is tank-specific, the design is customized to be able to facilitate prompt and early detection of the leakage from the individual tank. Please refer to Fig.1.
- The drain pipes are embedded within the concrete of the cubicle floor. To minimize the risk of the drain pipes leaking to the basemat, double wall piping is used. The outer piping collects and routes any leakage from the inner piping to the concrete trench.
- The valve pits are designed for easy access for inspection and calibration of the level instruments. The drain piping is connected to a drain header to direct drains to a nearby sump. The drain header pipe is routed inside a concrete trench to facilitate the collection and routing of floor drains to the sump.
- The tank cubicles have sloped and epoxy coated floors to facilitate easier drainage of minor and sustaining leaks. Please refer to Fig.2.
- Because leakage detection is only the first step in minimizing the spread of contamination, the design features are also supplemented with procedures for operator actions to provide a timely assessment and prompt response based on the location and characteristics of the leak.

Figure 1: Leak Detection Schematic.

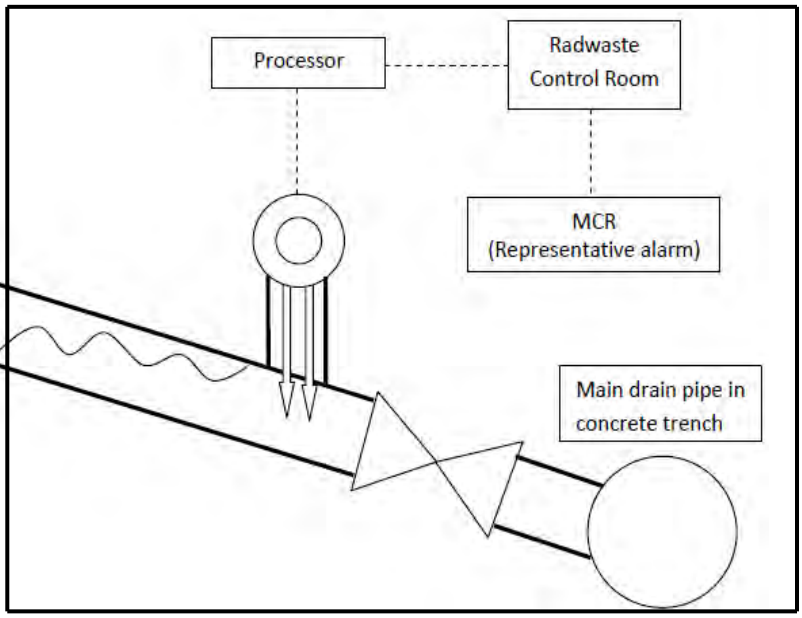
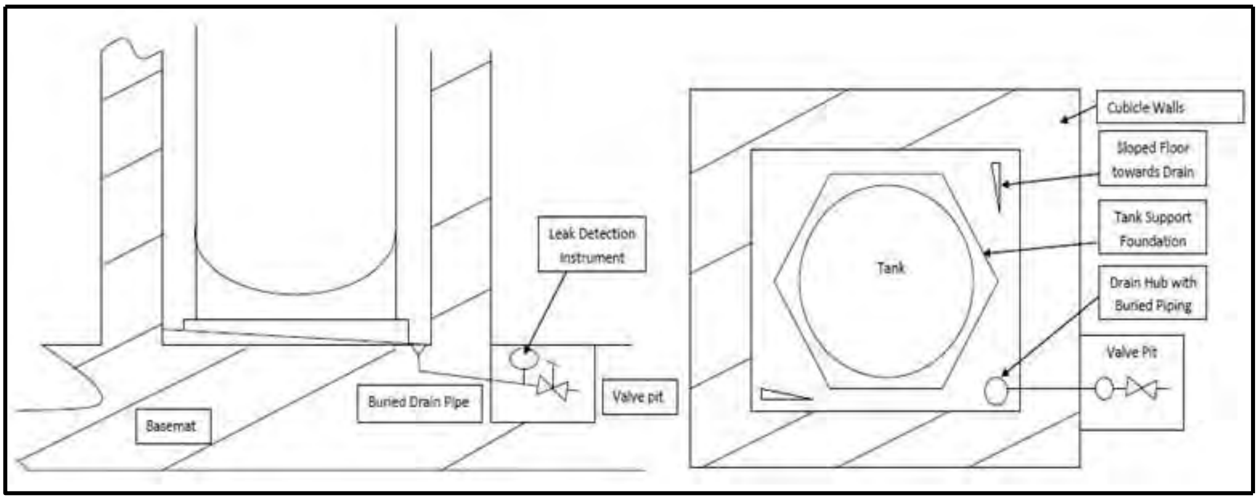


Figure 2: Cubicle Layout Design with Leak Detection.



In summary, the APR1400 RG 4.21 Program contains the following four elements:

- Comprehensive design features to minimize leakage through component design and material specifications;
- Active leak detection and management features to minimize the spread of contamination and waste generation;
- Adequate and accessible spaces for inspection, calibration, and maintenance; and
- Supplemental procedures and programs to provide guidance for fast and appropriate response and/or mitigation.

These four elements represent the primary design features to address the requirements from 10 CFR 20.1406 and RG 4.21 which specify facility design and procedures for operation that shall minimize, to the extent practicable, contamination of the facility and the environment, facilitate eventual decommissioning, and minimize, to the extent practicable, the generation of radioactive waste.

4 RG 4.21 IMPLEMENTATION APPROACH

Since the RG 4.21 Program involves various disciplines, from Project Management to the Engineering and Design team, to develop and implement design details, the team of project personnel worked closely during the early stage of the preparation of the Design Control Document (DCD). During the early stage of the APR1400 RG 4.21 Program, the concepts of primary-to-secondary leakage and the risks associated with the spread of potentially contaminated fluids were discussed in detail to establish the risk-based design requirements and evaluation methodology on the radiological impacts against the system performance and cost effectiveness. High risk items also include defense- in-depth design approaches and supplemented by administrative control through operational procedures and programs. In-depth training was provided to all project personnel to be familiar with the RG 4.21 principles and the design objectives. Those systems that contain radioactive, or potentially radioactive fluids were identified through the assessment via screening criteria. Detailed reviews of the SSCs were conducted to identify the existing design features that exhibit compliance to RG 4.21 and additional features were also evaluated and implemented to provide the early detection requirements. To enhance early detection, the cubicle layout and the design of concrete trenches, pipe routing, and underground concrete tunnels were modified to facilitate the collection of leakages.

The early leak detection feature is used for those components that contain substantial contaminated fluids in terms of volume and radiological activities. The Liquid Waste Management System (LWMS) floor drain tanks are used as the example to illustrate the comprehensive design approach and the early leak detection capability.

The floor drain waste tanks are located in separate cubicles which are provided with sloped floors (Fig.2). The floors and walls are epoxy coated to facilitate the draining of potential leakage and to facilitate decontamination and cleaning. A drainage hub and a drain pipe are provided to collect leakage within each cubicle. The drain pipe is sloped and is provided with a level switch and isolation valve (Fig.1). When fluid is accumulated to a certain level inside the drain pipe, the level switch (or conductivity instrument) is activated to alarm for operator action. Operator actions include the investigation and isolation of the leakage source and the cross transfer of fluid, decontamination, and repair, as required.

When a leakage source is identified and mitigation actions are required, the isolation valve is manually opened to drain the fluid to a drain header which routes the fluid to a collection sump. The leak detection instrument and the valve are located in a valve pit inside the pump room to facilitate access. The drain pipe is embedded into the basemat and is constructed of double-walled pipe to minimize the contamination of the basemat. Leakage from the inside pipe is collected in the outside pipe and is drained to a concrete trench. The concrete trench is designed to house the drain header and to facilitate the collection of floor drainage for transfer to a nearby sump.

In summary, the individual leak detection design facilitates prompt identification of the tank that is the source of the leak, the sloped drain pipe facilitates the collection of a small quantity of leaked fluid for early detection, and the cubicle design facilitates drainage that minimizes the infiltration of leaked fluid into the basemat concrete. The valve pit, in which the level switch and the isolation valve are housed, provides an accessible location for quick operator response. The drain header that connects all leak detection piping facilitates drainage of the tank contents in the event that the tank needs maintenance or repair. To complete the compliance to RG 4.21, a RG 4.21 Program (as outlined in NEI 08-08A [2] or equivalent) is to be established by the Combined Operating License applicant to identify the following:

- Alarm response procedures and actions
- Leak detection instrument calibration frequency and procedures (ODCM, NEI 07-09A [3])
- Epoxy coating inspection, repair, and maintenance programs and procedures (Epoxy Maintenance Program)

There are also other programs that are supplemental to the leak detection capability:

- The Radiological Environmental Monitoring Program is developed to detect the spread of contamination through the monitoring of hydrology data and the collection of soil and groundwater measurements, from the pre-construction stage through the end of the plant's operation.
- The documentation program is established to provide and maintain records on the updated design, design changes, and incident and mitigation details; and
- The process control program is developed to manage liquid and solid wastes from the operation of the nuclear facility.

The APR1400 design also incorporates other design features to prevent and/or minimize the spread of contamination and radioactive waste generation, especially considerations to improve decommissioning feasibility and economics, as follows:

- Segregation of SSCs based on the types, characteristics, and the radiological level of contaminations to simplify decontamination and decommissioning planning;
- Concrete trenches and drain header system to minimize contamination of basemat and concrete floors;
- Minimization of embedded piping, both in numbers and lengths, to minimize unintended leakages to the facility and the environment and therefore minimize decommissioning uncertainties;
- Use of underground concrete tunnels for the routing of contaminated or potentially contaminated piping to minimize the spread of contamination and decommissioning waster generation;
- Provision of a specific design for piping tunnels between building to detect leakage and prevent the infiltration of groundwater to minimize the spread of contamination to the environment;
- Use of double-wall piping where risks for contamination is high;
- Provision of a comprehensive drain piping design and layout;
- Consideration of modularization of construction such as layered concrete pavement and modularized component and building segment design to facilitate easier removal for decommissioning planning; and
- Provision of operating directions through commitments by COL applicants for RG 4.21 programs, including the use of mobile technologies, a site-wide decontamination facility, and a site-wide interim radwaste storage facility.

5 CONCLUSION

In conclusion, the APR1400 design incorporates an integrated and active approach for RG 4.21 compliance through the implementation of specific design features and operating procedures and programs. The primary objective of the APR1400 RG 4.21 Program is to establish life-cycle planning of the facility to prevent and/or minimize the spread of contamination; actively manage leakage potentials to reduce radiological consequences; and to facilitate more efficient decommissioning, thus providing environmentally cleaner and safer plant operations and reducing decommission costs and schedule.

6 REFERENCES

[1] Regulatory Guide 4.21, 2008. Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning. U.S. Nuclear Regulatory Commission, Washington DC.

[2] NEI 08-08A, 2009. Generic FSAR Template Guidance for Life Cycle Minimization of Contamination. Nuclear Energy Institute, Washington DC.

[3] NEI 07-09A, 2009. Generic FSAR Template Guidance for Offsite Dose Calculation Manual (ODCM) Program Description. Nuclear Energy Institute, Washington DC.

The Post Fukushima Events Development and Severe Accidents Overview: Activities to Enhance Safety and Radiation Protection Regarding Academy Perspective

Tadashi Narabayashi*

Tadashi Narabayashi, Hokkaido University, N13, W8, Kita-ku, Sapporo 060-8628, Japan.

Abstract. Many lessons can be learned from the Fukushima Daiichi Nuclear Power Plant (NPP) accident. After the loss of Emergency Core Cooling System (ECCS) and IC core cooling, fuels in the core melted down. Leak of fission product and hydrogen began because of high-temperature damage to the PCV packing. A hydrogen explosion occurred in the upper floor in the reactor building in Units 1, 3 and 4. At Unit 2, reactor core isolation cooling (RCIC) continued to function for about 3 days. Soon after the loss of RCIC water injection, the water level in the RPV declined. Drywell pressure increased from 400 kPa to 750 kPa, and PCV top flange might leak began through silicon rubber O-ring. It was an initiation of severe contamination around the NPS. In the afternoon on March 15, wind blew toward Iidate village. Melted core relocation into lower plenum caused the radiation level increased. The radiation level was measured by containment atmospheric monitoring system (CAMS). Hokkaido University developed filtered containment venting system (FCVS) to remove not only CsI but also methyl iodine (CH_3I). Radioactive iodine is captured on the surface of a molecular sieve or AgX, made from zeolite particles with silver coating. FCVS will be installed in all Japanese NPPs. By using the FCVS technology, we had started a high decontamination air cleaning system to remove multi-nuclides for radiation protection to conduct decommissioning the Fukushima NPP.

***KEYWORDS**: Fukushima Daiichi accidents; filtered containment venting system; decontamination; air cleaning system; multi nuclides; radiation protection; decommissioning; Silber zeolite; AgX.*

1 INTRODUCTION

The author has been involved in investigating the causes of the Fukushima Daiichi accidents and developing countermeasures for the other NPPs in Japan, as a member of advisory meeting member of NISA and NRA (Nuclear Regulatory Authority) for Fukushima Daiichi Accident Investigation Team, regarding academy perspective. NRA enforced the New Regulatory Requirements, based on the concept of "Defence-in-Depth", for Commercial Nuclear Power Plants from July 8, 2013. It is hoped that the lessons learned from this accident will improve the safety of nuclear power plants worldwide. Many lessons can be learned from the Fukushima Daiichi Nuclear Power Plant (NPP) accident. After the loss of Emergency Core Cooling System (ECCS) and IC core cooling, fuels in the core melted down.

2 INVESTIGATION OF FUKUSHIMA DAIICHI ACCIDENTINVESTIGATION OF ACCIDENTS

Although other NPPs such as Fukushima Daini, Onagawa and Tokai Daini were also struck by the tsunami, they all were able to safely terminate operation, until the cool-down condition. The Fukushima Daini NPP succeeded in safe shutdown, even though Unit 1 was affected by water flooding through hatches and an emergency diesel generator (EDG) air intake. AC power was restored by changing the power cable and the seawater pump motors were replaced by bringing in new motors from the Toshiba Mie Works and Kashiwazaki-Kariwa NPP by helicopter. At the Fukushima Daiichi NPP, Unit 5 was brought under control by using EDG power from Unit 6 [1,2,3].

Figure 1 shows a comparison of the flood damage to EDGs. At Units 1 through 4, there was a complete loss of both AC power from the EDGs and DC power, and this was the main cause of the ensuring severe accidents [3]. At Unit 2, reactor core isolation cooling (RCIC) continued to function for about 3 days.

* Presenting author, e-mail: tnaraba@eng.hokudai.ac.jp

Figure 1: Comparison of flood damage to emergency diesel generators for Fukushima Daiichi and Daini NPPs [3].

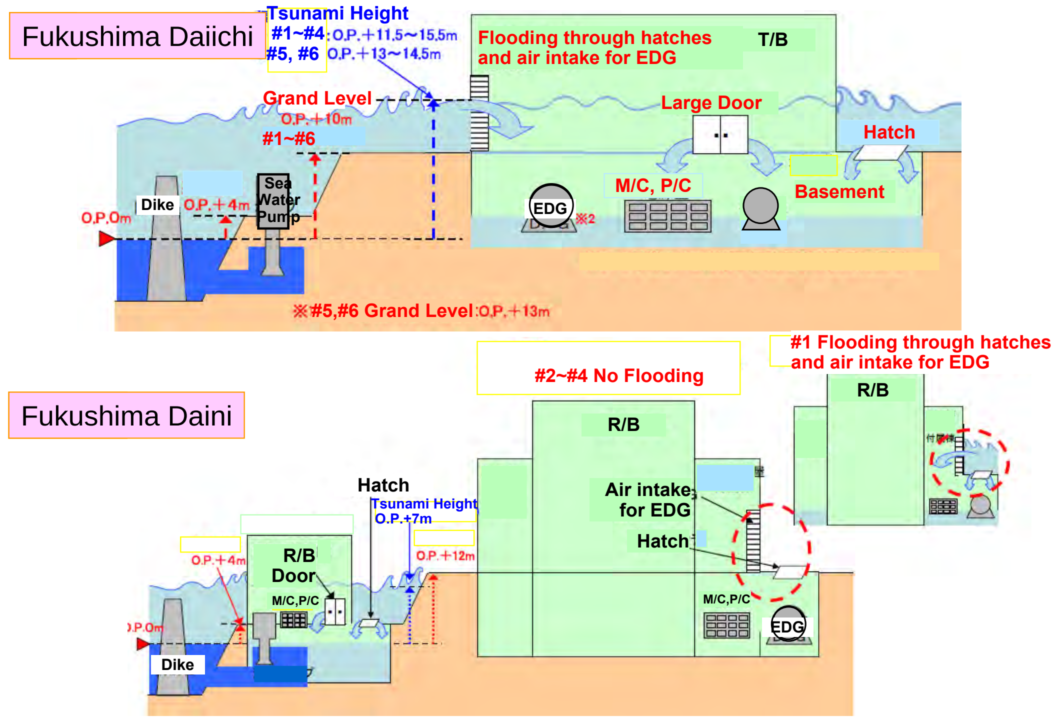


Figure 2: Measured pressure and water level in RPV and PCV of Unit 2 [4].

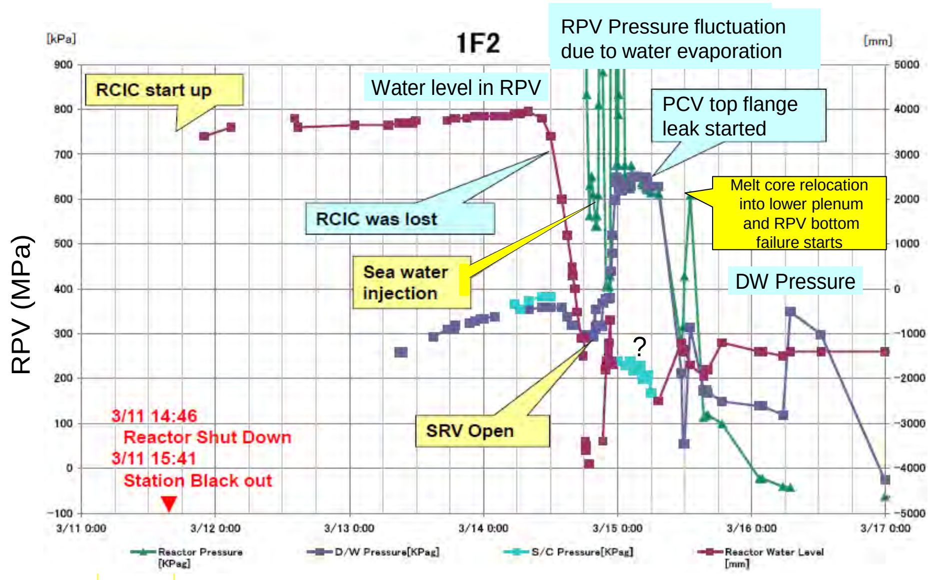


Figure 2 shows that soon after the loss of RCIC water injection, the water level in the RPV declined. The safety relief valve (SRV) was opened and sea water injection started. But, RPV pressure shows fluctuation due to water evaporation and metal-water reaction in core. Drywell (DW) pressure increased from 400 kPa to 750 kPa (abs.), and PCV top flange leak began through silicon rubber O-ring. It was an initiation of severe contamination around the NPS. In the afternoon on March 15, wind blew toward Iidate village. Melted core relocation into lower plenum caused the RPV bottom CRD pipe failure and PCV pressure and radiation level increased (Fig. 3). The radiation level was measured by containment atmospheric monitoring system (CAMS) [10].

Figure 4 shows H_2 explosion in Unit Nos. 1, 3 and 4. Upon the Unit 3's detonation occurred the blowout panel of Unit 2 was opened. Hydrogen in Unit 2 was released through the opened blowout panel and there was no explosion/detonation in that unit. The sound of an explosion was reported near the S/C of Unit 2. However, examination of the data showed that this was due to a hydrogen detonation in the reactor building (R/B) of Unit 4. Soon after this detonation, DW pressure in the Unit 2 decreased (Fig. 3).

Figure 3: MAAP analysis results compared with actual plant data for Unit 2[10].

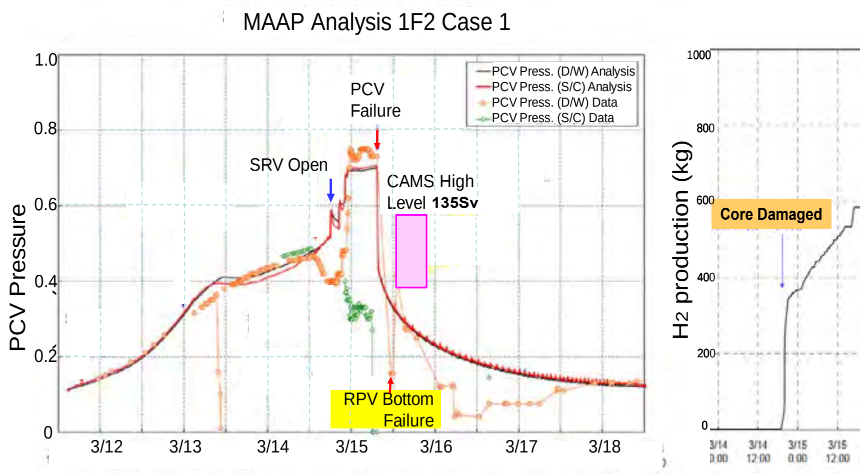


Figure 5 shows trends in monitored radiation dose levels for Units 1, 2, 3 and 4, which can be compared with events illustrated in Fig. 6. It appears that the explosion occurred after venting operations. The radiation level increased soon after the Unit 2 PCV rupture on March 15. A loss of core cooling occurred because of the IC trip in Unit 1, and the RCIC steam turbine also tripped owing to loss of battery power in Units 2 and 3. Suppression pool (S/P) temperature and pressure became so high that water injection actions for accident management took a long time. This was the reason for the chain of severe accidents in the four units of Fukushima Daiichi NPS, as shown in Fig. 6 [11].

Figure 4: H_2 detonations occurred after vent operations (Units 1, 3 and 4) [11].

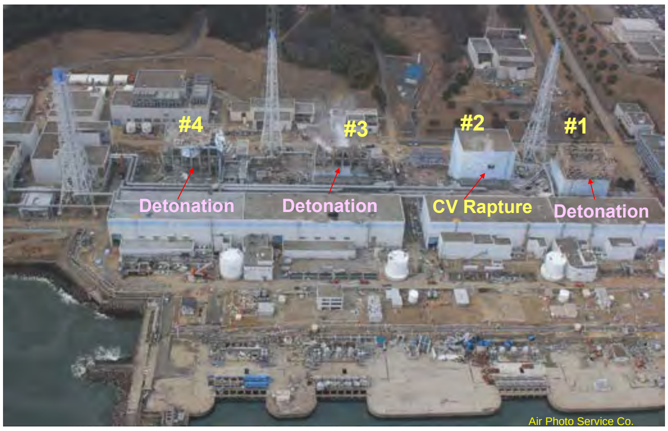


Figure 5: Monitored radiation levels for Fukushima Daiichi Units 1, 2, 3 and 4 [2]

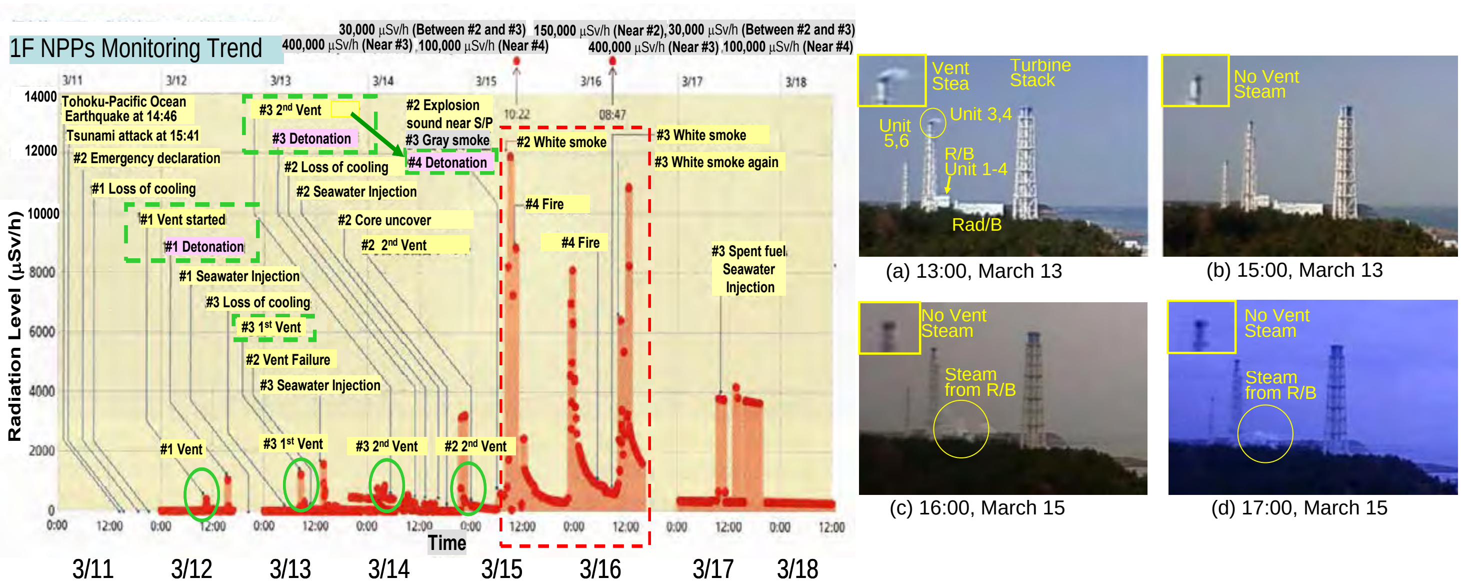
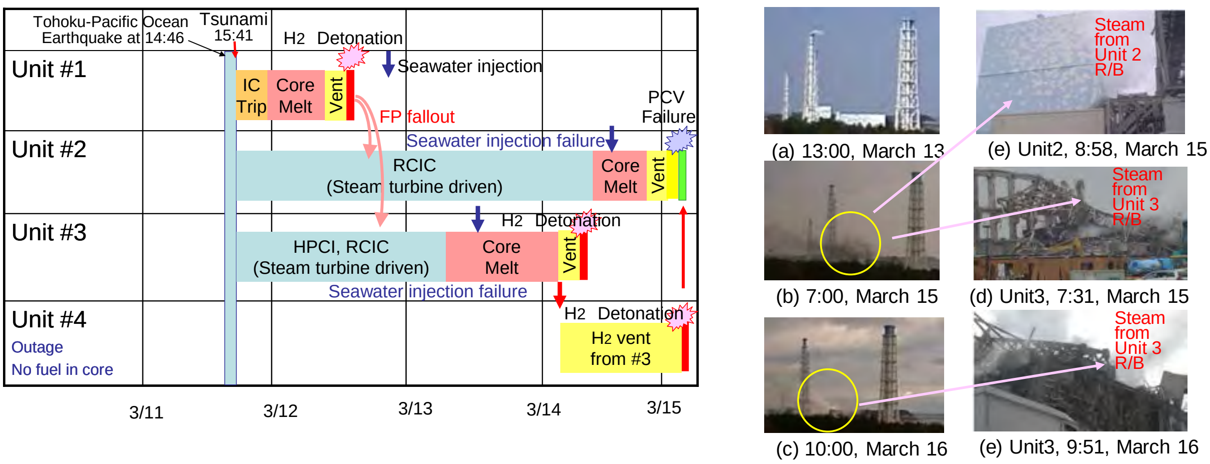


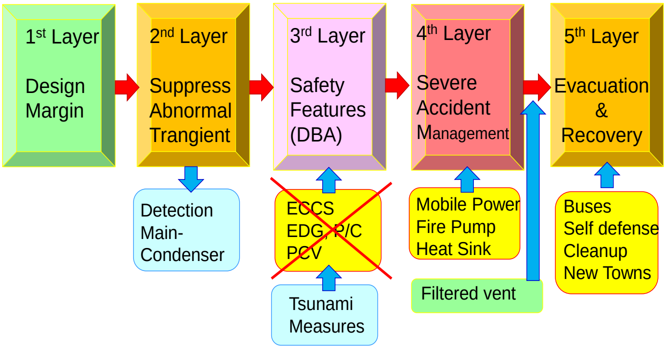
Figure 6: Chain of major events at Units 1 through 4 causing severe accidents at Fukushima Daiichi NPS [8].



3 MEASURES FOR SEVERE ACCIDENTS INSTALLED IN WESTERN NPPS

There are many good practices of countermeasures to prevent FP release in the world. Based on the "Defense-in-Depth" (DiD) concept (Fig. 7), essential safety features were incorporated in the third layer for design bases accident (DBA) and prevention of simultaneous loss of all safety functions owing to common causes, such as tsunamis. Mobile safety features for the fourth layer such as mobile fire pumps should be deployed for core and containment cooling or corium cooling [6].

Figure 7: Concept of "Defense-in-Depth"



As shown in Fig. 8, after the Three Mile Island (TMI) Unit 2 and Chernobyl Unit 4 NPP accidents, countries such as France, Germany, Switzerland, Finland and Sweden decided to install filtered containment venting systems (FCVS) to protect against radioactive material exhaust (Fig. 9) [5].

Figure 8: Filtered containment venting system [8].

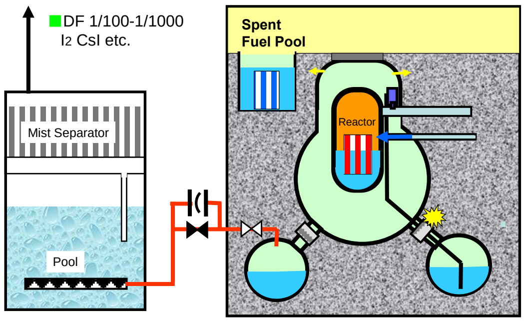
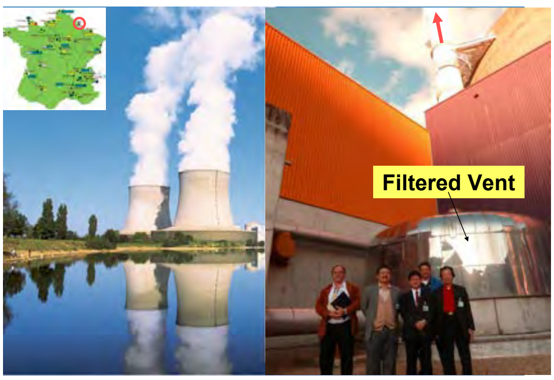


Fig. 9 Filtered containment venting systems in Chooz NPP (PWR), France and in Leibstadt NPP (BWR), Switzerland [8].



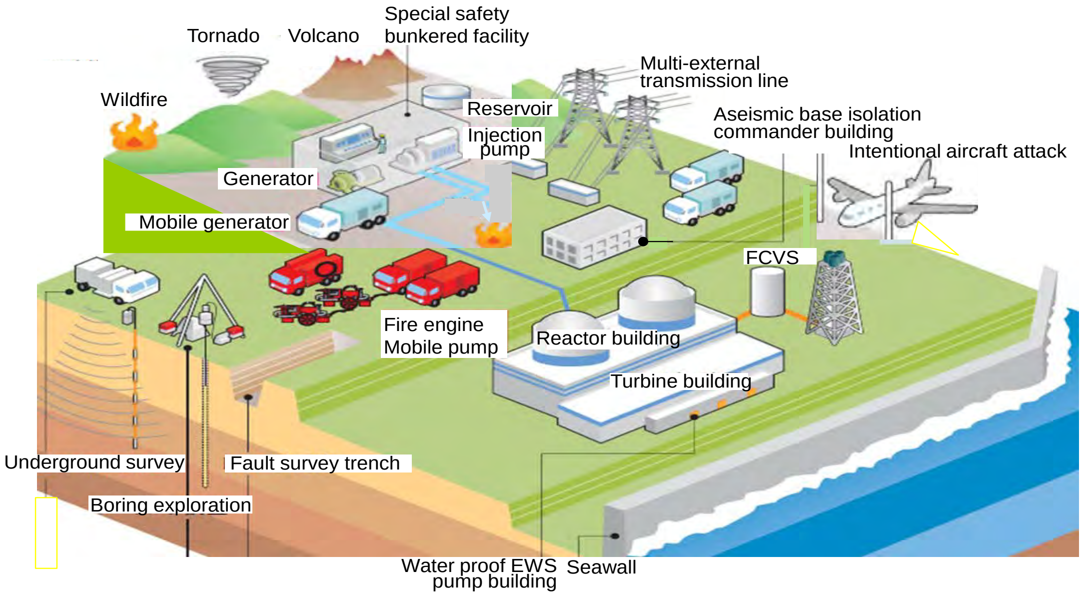
4 COUNTERMEASURES BASED ON THE NEW REGULATORY ENFORCEMENT

4.1 New Nuclear Regulatory Requirements in Japan

A new nuclear regulatory body, the Nuclear Regulation Authority (NRA) was established on September 19, 2012. The NRA performed a complete review of safety guidelines and regulatory requirements [7].

On July 8, 2013, new regulatory requirements for commercial power reactors came into force. These requirements stipulate that all Japanese utilities conform to the regulatory requirement before restarting NPP. Design requirements must treat natural phenomena, such as volcanoes, tornados and forest wildfires (Fig. 10). The new regulatory guidelines require direct deployment of mobile power, mobile pumps, fire engine, and installation of tsunami protection. Design requirements should be prepared to protect against cable fire between the reactor building and main control room, and against internal inundation by water proof areas of important safety components and systems.

Figure 10: Enforcement of new regulatory requirements, from July 2013 [14].



4.2 BWR NPS to be Reviewed for the New Requirements or Restarting

Following the design reviews of 12 PWRs, nine BWRs with enhanced safety measures were in ongoing design review for restarting.

Upon occurrence of a SA, vent gas with radioactive fission products is blown out to a scrubbing pool through numerous venturi nozzles (Fig.11). Mist in steam moves upward to a metal fiber filter through a multi-hole baffle plate. After the mist is removed by that filter, radioactive methyl iodine (CH_3I) is captured on the surface of a molecular sieve or AgX, made from zeolite particles with silver coating [11,12].

Figure 11: Filtered containment venting system (FCVS) with silver zeolite [11].

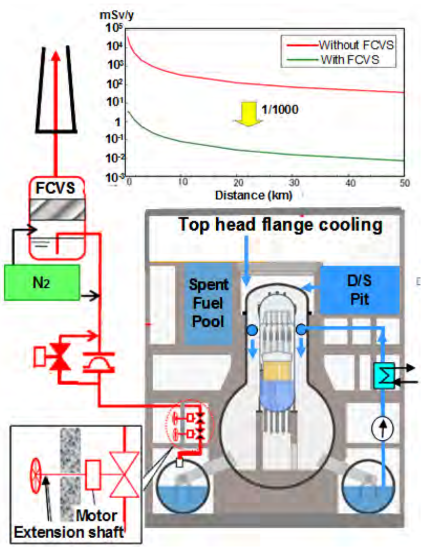
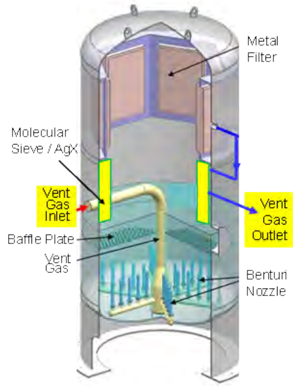


Figure 12: Installtion of Filtered containment venting systems (FCVS) [11].

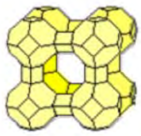


(a) FCVS pit at Hamaoka NPS (b) Installation of FCVS at Kashiwazaki-Kariwa NPS

Figure 12 shows the FCVS pit at Hamaoka NPS of Chubu electric and installing FCVS at Kashiwazaki-Kariwa NPS of TEPCO, respectively.

Figure 13 shows the FCVS visualization test facility at Hokkaido University. An AgX filter is used down-stream of the scrubbing pool and metal fiver filter. This study was conducted by Kakenhi (B) funded No. 2436038802. Thickness of AgX filter is very important parameter to obtain enough decontamination factor (DF). As shown in Table 4 of TUV test result in Germany, the DF for the radioactive iodine exceeds 10,000 at bed depth (AgX filter thickness) greater than 75mm [11].

Figure 13: Development of high DF FCVS using silver zeolite at Hokkaido Univ [11].



(a) AgX filter test section, (b) Zeolite model

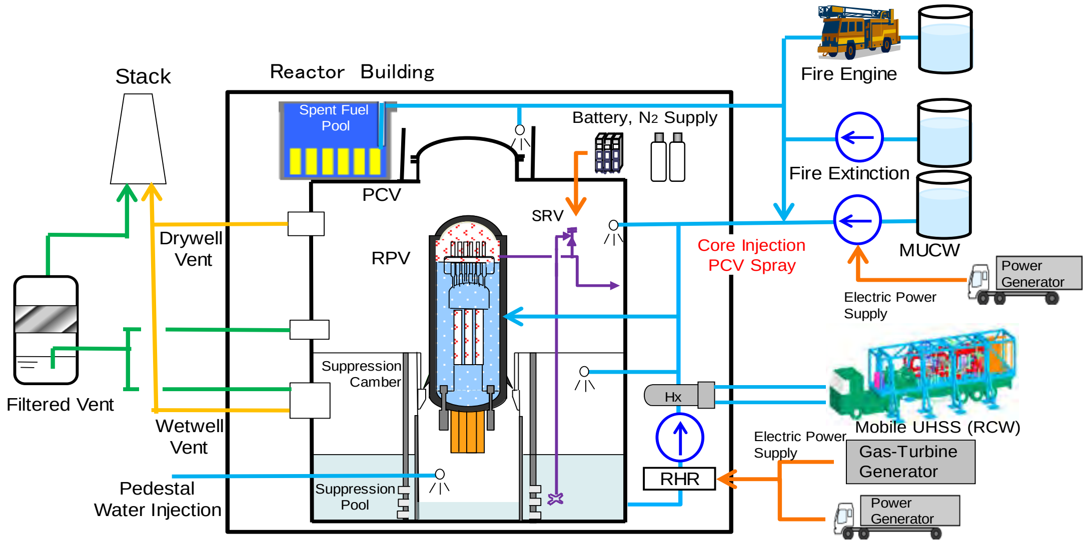
(c)FCVS test facility at Hokkaido Univ. (d) Review meeting of FCVS-WG/JSME

Table 1: Test result of adsorption efficiency of CH_3I under various bed depth [12].

Bed depth (mm)	Residence time (sec.)	Adsorption efficiency of CH_3I (%)
50	0.246	99.967
75	0.369	> 99.999
100	0.492	> 99.999

Testing conditions
Temperature: 130 °C, Pressure: 399 kPa, LV: 20 cm/sec.;
Relative humidity (RH): 95 %, CH_3I concentration: 1.75 mg/m³ (I-131).

Figure 14: Full cooling strategy for SBO to protect against SA for BWR [9].



5 ACTIVITIES TOWARDS DECOMIMISIPMING FUKUSHIMA DAIICHI

Figure 15 shows the current status of the reactors in unit 1 through 4. It is assumed that the reactor cores of Units 1 through 3 are melted and that some portion dropped onto the pedestal floor of the CV. It is very important to survey the melted core debris distribution from the core to pedestal floor. Some of debris may be attached on surface of the control rod drive outer casing.. About 4 ton/h of water are injected into the cores of units 1 through 3, and temperature in the pressure vessel is maintained around 30 to 40 °C. Temperature in the spent fuel pool (SFP) is maintained around 30 °C [13,14].

Figure: 15 Current status of reactors in Unit 1 through 4 [13,14].

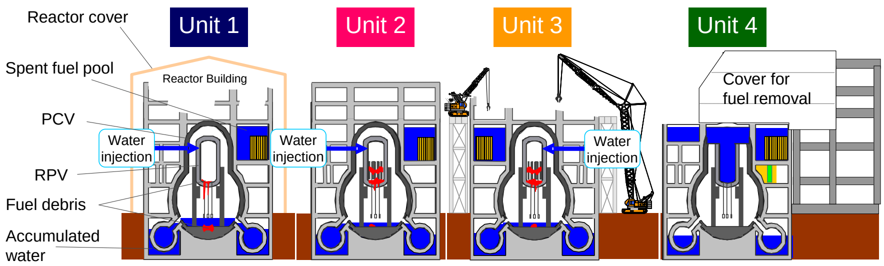
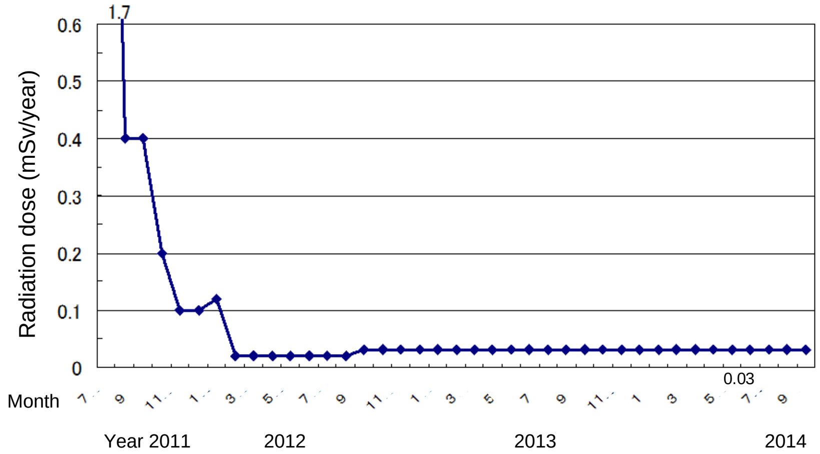


Figure 16 shows the annual radiation dose from radioactive materials at the site boundary of Fukushima Daiichi NPS, released from Units 1 through 4. At the time of the accident, vast amount of these materials were released, but the maximum annual dose is now 0.03 mSv/y equivalent to 1/70th that from background radiation.

Figure 16: Annual radiation dose at site boundary of Fukushima Daiichi NPS [13,14].



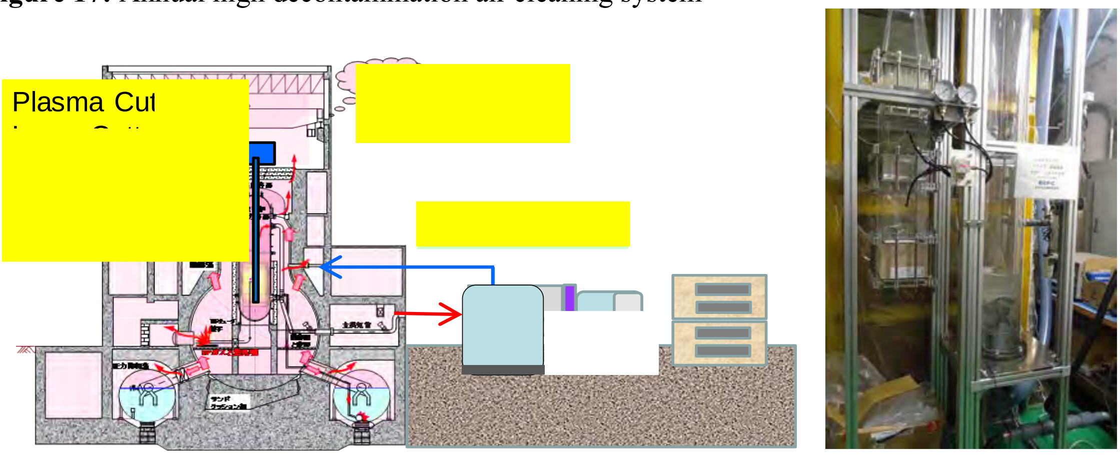
（**Reference**）

* Concentration limit in the air of environment surveillance area :
[Cs-134] : 2×10^{-5}Bq/cm^3
[Cs-137] : 3×10^{-5}Bq/cm^3

* Dust concentration in the area surrounding 1F site boundary :
[Cs-134] : ND（Detection limit: approx. 1×10^{-7}Bq/cm^3）,
[Cs-137] : ND（ Detection limit: approx. 2×10^{-7}Bq/cm^3）

By using the FCVS technology, we had started to devwlop a high decontamination air cleaning system to remove multi-nuclides for radiation protection to conduct decommissioning the Fukushima NPP (Fig. 17). Development of high efficiency multi-nuclide aerosol filters for radiation protection duringa process of cutingcore debris at Hokkaido University. A plasma cutter, laser cutter, wire cutter, drilling machine, etc, will be used and will gennerate aerosols. Therefore, the air cleaning system should be needed for removing core debris.

Figure 17: Annual high decontamination air cleaning system



6 REFERENCES

[1] Narabayashi, T., and Sugiyama, K.: "Fukushima Daiichi NPPs accidents caused by the Tohoku- Pacific Ocean Earthquake and Tsunami", AESJ Atoms, vol.53, No.6, P.387-400 (2011).

[2] Nakajima, H., Narabayashi, T., “Start from Level 7”, Nikkei Science, July, (2011) , (in Japanese), (http://www.nikkei-science.net/modules/journal/index.php? vol=201107).

[3] NISA, Technical knowledge of Fukushima-Daiichi NPP's Accidents and countermeasure (Interim report), (2012), (in Japanese), (http://www.meti.go.jp/press/2011/03/20120328009/20120328009.html).

[4] NISA, JNES: The 2011 Pacific coast of Tohoku Pacific Earthquake and the seismic damage to the NPPs, (2011).

[5] Narabayashi, T. Lessons of Fukushima-Daiichi NPP's Accidents for Achievement of the 1st Class Safety in the World, Fukushima Severe Accident Dose Management & Global Lessons Learned in Occupational Dose Reduction, 2012 International ISOE ALARA Symposium, Fort Lauderdale, Florida, (Jan, 2012).

[6] Gavrilas, M., et. al., Safety Features of Operating Light Water Reactors of Western Design, CNES (2000).

[7] Nuclear Regulation Authority, Nuclear Regulation Authority Enforcement of the New Regulatory Requirements for Commercial Nuclear Power Reactors, (July 8, 2013), (http://www.nsr.go.jp/data/000067212.pdf).

[8] Narabayashi,T., Lessons learned from the Fukushima Daiichi Nuclear Power Plant Accident, Turbulence, Heat and Mass Transfer 7, Begell House, Inc.(2012). (www.gepr.org/en/contents/20121231-01/KeynoteDrNarabayashi-THMT-12r2.pdf).

[9] FVCS working group report, Power and Energy System Division, JSME, (Dec., 2013), (http://www.jsme.or.jp/pes/Research/A-TS08-08/index03.html).

[10] TEPCO, Second Progress Report for Fukushima Daiichi, (Aug. 8, 2014), (http://www.tepco.co.jp/cc/press/2014/1240099_5851.html).

[11] Narabayashi, T., et al, Development of High Efficiency Filtered Containment Venting System by using AgX, Short paper of ICMST-Kobe,(2014).

[12] Wang, J., Kobayashi, T. and Endo, K., Properties and Applications of Silver Zeolite (AgX), Short paper of ICMST-Kobe,(2014).

[13] Yamashita, N., Activities towards the Decommissioning of Fukushima-Daiichi (TEPCO), ICMST Kobe (2014).

[14] Narabayashi, T., Lessons Learned from the Fukushima Daiichi Accident to Establish Resilience Technology for Nuclear Plants Based on the Defense in Depth Philosophy (Invited Plenary), North American ISOE ALARA Symposium, Ft. Lauderdale, Florida USA (Jan. 2015).

Radioactive waste management in case of incidental melting of a Co-60 source

Teresa Ortiz Ramis[a*], Elena Alcaide Trenas[a], Pablo Belinchon[b]

[a]Radiological Protection Technical Unit, National Radioactive Waste Company (ENRESA), C/Emilio Vargas,7, 28043 Madrid.

[b]Radiological Protection Technical Unit, Adqualitas (AdQ).

Abstract. To date there has been no incident in the Spanish steel industry involving the melting of a Co-60 source with a transfer factor to the melt of 93.5%. As a preventive measure, recommendations were drawn up for actions to follow in the event of the melting of a source. On detection of radioactive material contaminated with Co-60, the recommendation was to melt until a semi-product was obtained. Subsequently, after several meetings with Spanish steel industry representatives, it was considered that continuing to melt until a semi-product was obtained was an unviable option, and dumping the melt into the steelworks' spill pits was considered more realistic. This study has been developed in this context and using the data provided with regard to the weight of the melt and the options and dimensions of the spill pits/slag pots.

KEYWORDS: accidental melting of sources; radioactive waste; radiological protection.

1 INTRODUCTION

The presence of radioactive material in scrap metal had been considered a potential risk in Spain until 30 May 1998, when a Caesium-137 source was accidentally melted in one of the Acerinox factory furnaces in Los Barrios (Cadiz). The incident showed that this risk was real and that it could impact on people and the environment and, above all, have significant economic consequences.

The Acerinox incident directly motivated the creation of the 'Protocol for Radiological Monitoring of Metallic Materials'. This is a voluntary agreement signed on 2 November 1999 between the Ministry of Industry and Energy, the Ministry of Public Works, the Nuclear Safety Council (CSN), the National Radioactive Waste Company (ENRESA), the Union of Steel Companies (UNESID) and the Spanish Recycling Federation (FER). These entities were subsequently joined by the Mining and Metallurgy Federation of Workers' Commissions, the State Metal Federation, Construction and Allied Workers of the UGT, the Spanish Association of Aluminium Refiners (ASERAL), the National Union of Copper Industries (UNICOBRE), the Union of Lead Industries (UNIPLOM) and the Spanish Federation of Smelters' Associations (FEAF).

Some incidents have occurred in which a radioactive source has entered the furnace hidden among scrap metal and has been melted, causing the dispersion of the radioactive material. The majority of these sources have been Cs-137, which partitions 100% to the off-gas dust, according to studies carried out by the University of the Basque Country (UPV) in partnership with ENRESA and the CSN.

To date there has been no incident in the Spanish steel industry involving the melting of a Co-60 source with a transfer factor to the melt of 93.5%. As a preventive measure, UNESID drew up recommendations for actions to be followed in the event of choosing to cast a semi-product on detection of radioactive material contaminated with Co-60 in the process. This document contains an Excel sheet which can be used to calculate, for each facility and workplace, the value for the specific activity of Co-60 which would have to be registered in the melt test so that the effective dose received by each of the workers in different jobs, taking into account exposure time, was 1 mSv (limit set by the Regulation on the Protection of Health against Ionising Radiation for members of the public). This specific activity is considered as an activity limit for a worker to remain at their post throughout the entire melt process.

[*] Presenting author, e-mail: torr@enresa.es

Subsequently, after several meetings with steel industry representatives, it was considered that continuing to melt until a semi-product was obtained was an unviable option, and dumping the melt into the steelworks' spill pits was considered more realistic. This study has been developed in this context and using the data provided with regard to the weight of the melt and the options and dimensions of the spill pits/slag pots.

2 PROBLEMS WITH MELTING OF Co-60 SOURCES

To date, 21 Co-60 sources have been detected at the entrances of facilities and have been removed by ENRESA, following the steps laid down in the Protocol. Table 1 details these sources by facility, date and activity.

Table 1: Co-60 sources removed by ENRESA

DATE	YEAR	No. SOURCES	ACTIVITY (MBq)	IAEA Category[(a)]	TOTAL ACTIVITY (MBq)
27/06/2007	2007	1	456	4	456
03/02/2005	2005	1	11.4	4	11.4
03/02/2005	2005	1	370	4	370
14/11/2006	2006	2	1,850	4	3,700
22/12/2005	2005	1	3.7	4	3.7
16/12/2004	2004	1	370	4	370
12/11/2003	2003	1	4.32	4	4.32
05/05/2004	2004	1	1.52	5	1.52
03/10/2007	2007	1	66.8	4	66.8
03/06/2004	2004	1	10,800	4	10,800
12/06/2007	2007	1	2.18	5	2.18
14/03/2002	2002	1	37	4	37
04/08/2004	2004	1	403	4	403
17/01/2007	2007	1	1,110	4	1,110
07/02/2002	2002	2	509	4	1,018
29/07/2004	2004	1	1.94	5	1.94
08/08/2002	2002	1	0.339	5	0.339
26/03/2004	2004	1	1.87	5	1.87
30/06/2000	2000	1	14.22	5	14.22

[(a)]Category 5: Most unlikely to be dangerous, Category 4: Unlikely to be dangerous, Category 3: Dangerous, Category 2: Very dangerous, Category 1: Extremely dangerous

The highest activity source was detected in 2004, registering 10.8 GBq. According to IAEA categorisation, this was a Category 4 source, although close to Category 3 (from 30 GBq). Of the total number of detected sources, 15 were Category 4 and 6 were Category 5. Overall, the percentage of detected Co-60 sources is low, in the region of 6%. Most of the detected sources are Ra-226 (62 %), of very low activity and these are followed by Cs-137 (16 %).

As mentioned in the introduction, there has been no accidental melting of a Co-60 source in a steelworks furnace in Spain, but there have been cases in other countries. The most significant incident reported happened in Ciudad de Juárez (Mexico) in 1983, when a disused cobalt therapy unit was stolen. It was disassembled in a recycling plant and most of the source was melted at another facility. The source had an activity of 37 TBq (37E6 MBq). The material produced in this facility was used to make girders, tables and chairs. The incident was not discovered until a lorry carrying a batch of these products triggered an alarm in the radiation monitoring system at the Los Alamos facility in the USA. In total, 841 houses built with contaminated girders had to be demolished and 2,500 sets of tables and chairs had to be recovered.

According to the reference material consulted, the majority of sources melted in the USA (1983-1998) have been Cs-137 (50%) and 10% were Co-60. Other incidents around the world show similar percentages.

At the present time, incidents of melting of Co-60 sources are reported indirectly, on detection of products made with contaminated steel at radiation portal monitors installed at borders and ports. In 2008, for example, the French regulatory authority issued a statement in which it reported that a French company that manufactured and distributed lift buttons had detected buttons contaminated with Co-60, some of which could have come from a Spanish manufacturer.

In Spanish ports, where radiation detectors are installed to monitor inbound and outbound containers, there have been various detections of objects made with steel contaminated with Co-60, including springs in plastic buttons and decorative trays, as well as raw materials. Internationally, there are references to stools, cutlery, cooking pans, paper towel dispensers, and shoe and bag buckles which have been contaminated with Co-60. While all these cases involved very low levels, this is still an indicator that some cases of melting of Co-60 sources have occurred undetected.

3 PROCEDURE TO FOLLOW IN THE EVENT OF DETECTION OF RADIOACTIVE MATERIAL CONTAMINATED WITH CO-60 IN THE MELT

The procedure is as follows:

- **A sample of steel from each melt must be taken and analysed in the spectrometer. This laboratory equipment must be operated by qualified personnel who are trained in its operation and it must be calibrated.**

- **If the alarm is confirmed, the steel sample is considered contaminated and the facility, in accordance with its procedures, will determine the specific activity of the sample and identify the isotope or isotopes present. Potential situations are as follows:**

  a) If the activity of the sample triggering the alarm is below 1 Bq/g for Co-60, either the melting process will be continued until a final product or semi-product is obtained, which is properly labelled by melt and separated for subsequent control, or the melt will be poured into the spill pits. This will be at the discretion of each facility, since it will not be considered radioactive waste.

  b) If the activity of the sample triggering the alarm is between 1 Bq/g and 10 Bq/g for Co-60, the melt will be poured into the spill pits and it will be considered very low level radioactive waste.

  c) If the activity of the sample triggering the alarm is greater than 10 Bq/g for Co-60, the melt will be poured into the spill pits for subsequent conditioning and it will be considered low and intermediate level radioactive waste.

In the last two cases, and in accordance with Section 3.6 of the Annex to the Protocol (cited below in Figure 1), the following actions are carried out.

Figure 1: Section 3.6 of the Annex to the Protocol

6. Actions in the event of detection of radioactive material

1. The signatory company will carry out the following actions:

   b)In the event of radioactive material being detected during the process or in the resulting products, the signatory company will carry out the following actions, with immediate support from the CSN:

   1) Notify the CSN of the detection immediately, using the fastest possible means and relaying all available information.
   2) With advice from the CSN, try to confirm if the detection is real. This involves the signatory company carrying out the following actions, with their own personnel or with the support of a Radiological Protection Technical Unit contracted for the purpose, applying various operational measures:
   3) In the event that the detection is real and with the advice of the CSN, the signatory company will carry out the following actions:
      - Stop all stages of the process that are understood to be affected, except those which help to ameliorate the consequences, as well as cleaning and decontamination tasks;
      - Immediately suspend the release of products from the facility which have been in contact with the affected stages of the process;
      - Report the situation to any recipient of products suspected of being affected by the incident;
      - Ensure that a Radiological Protection Technical Unit determines the extent of the contamination in the processing line and immediate surroundings.

4 CONDITIONING OF WASTE GENERATED IN THE MELTING OF CO-60 SOURCES

a) Contaminated material with activity between 1 Bq/g and 10 Bq/g

The melt will be poured into the spill pits and will be considered very low level radioactive waste.

The conditioning of this block should be undertaken in accordance with the acceptance criteria of the disposal facility, according to the different types of waste. This is decided on a case-by-case basis and must fulfil the conditions for transport, in accordance with current regulations (ADR). These conditions are as follows:

- Specific activity: <10 Bq/g
- Contact dose rate: below 2 mSv/h
- Dimensions of the blocks:
  a) For transport in an ISO container:
     - Length: 11 m
     - Height: 1.22 m
     - Width: 1.994 m
     - Maximum weight: 15 tonnes
  b) For transport in a specific container (C2A):
     - Length: 2 m
     - Height: 0.6 m

- Width: 2 m
- Maximum weight: 20 tonnes

b) Contaminated material with activity over 10 Bq/g

The melt will be poured into the spill pits for subsequent conditioning and will be considered low and intermediate level radioactive waste.

The conditioning of this block must be undertaken in accordance with the acceptance criteria of the El Cabril disposal facility for low and intermediate level waste, according to the different types of waste. This is decided on a case-by-case basis and must fulfil the conditions for transport, in accordance with current regulations (ADR). These criteria and conditions are more stringent than those for very low level waste. In the event of an accidental melting, this will be analysed on a case-by-case basis to determine the most feasible option, in accordance with the requirements of the regulatory body.

Table 2 shows what the activity of the melted source would have to be in each facility, according to the weight of the melt, to generate the different types of waste.

Table 2: Generation of different types of waste based on the weight of the melt in each facility and the activity of the source

Facility	**Melt weight (t)**	**Melted source activity**		
		Conventional waste (< 1 Bq/g)	**Very low level waste (1- 10 Bq/g)**	**Low and intermediate level waste (> 10 Bq/g)**
Plant 1	100	100 MBq	100-1,000 MBq	> 1,000 MBq
Plant 2	130	130 MBq	130-1,300 MBq	> 1,300 MBq
Plant 3	110	110 MBq	110-1,100 MBq	> 1,100 MBq
Plant 4	100	100 MBq	100-1,000 MBq	> 1,000 MBq
Plant 5	150	150 MBq	150-1,500 MBq	> 1,500 MBq
Plant 6	140	140 MBq	140-1,400 MBq	> 1,400 MBq
Plant 7	63	63 MBq	63-630 MBq	> 630 MBq
Plant 8	90	90 MBq	90-900 MBq	> 900 MBq
Plant 9	150	150 MBq	150-1,500 MBq	> 1,500 MBq
Plant 10	115	115 MBq	115-1150 MBq	> 1,150 MBq
Plant 11	50	50 MBq	50-500 MBq	> 500 MBq
Plant 12	150	150 MBq	150-1,500 MBq	> 1,500 MBq
Plant 13	135	135 MBq	135-1,350 MBq	> 1,350 MBq
Plant 14	40	40 MBq	40-400 MBq	> 400 MBq
Plant 15	200	200 MBq	200-2,000 MBq	> 2,000 MBq
Plant 16	100	100 MBq	100-1,000 MBq	> 1,000 MBq

Facility	Melt weight (t)	Melted source activity		
		Conventional waste (< 1 Bq/g)	**Very low level waste (1- 10 Bq/g)**	**Low and intermediate level waste (> 10 Bq/g)**
Plant 17	75	75 MBq	75-750 MBq	> 750 MBq
Plant 18	140	140 MBq	140-1,400 MBq	> 1,400 MBq
Plant 19	120	120 MBq	120-1,200 MBq	> 1,200 MBq
Plant 22	285	285 MBq	285-2,850 MBq	> 2,850 MBq

In the data recorded on Co-60 sources which have been detected and recovered, two of them would generate in one case very low level waste and in two cases low and intermediate level waste, depending on the weight of the melt at the plant.

5 DOSE CALCULATION AT DISTANCES FROM THE SPILL PITS

This analysis was performed with data provided by UNESID, collated from various facilities.

The assumptions used for the calculation of dose rates are as follows:

- The activity of the radiation source has been calculated according to the furnace capacity for a specific activity of 10 Bq/g and 1 Bq/g.
- The volume of the source has been calculated according to the size of the spill pits reported by the facilities.

The dose rates were calculated using the program Microshield 8.01, the results of which are presented in Table 3.

Table 3: Dose rate at distances from the spill pits based on melt level

Facility	Weight of steel (kg)	Source volume (m^3)	Dose rate for 10 Bq/g (uSv/h)				Dose rate for 1 Bq/g (uSv/h)			
			1 m	**2 m**	**3 m**	**Distance limit for 0.5 uSv/h (m)**	**1 m**	**2 m**	**3 m**	**Distance limit for 0.5 uSv/h (m)**
Plant 1	10,000	1.27	2.23	1.01	0.55	3.25	0.22	0.1	0.055	0.25
	45,000	5.73	1.61	0.83	0.49	3	0.16	0.083	0.049	contact
	45,000	5.73	1.61	0.83	0.49	3	0.16	0.083	0.049	contact
Plant 2	130,000	16.54	5.88	4.43	3.36	12.5	0.51	0.38	0.29	contact
Plant 3	110,000	13.99	3.25	1.69	1.01	5	0.32	0.19	0.1	0.5
Plant 4	100,000	12.72	3.78	1.94	1.14	6	0.37	0.19	0.11	0.75
Plant 5	45,000	5.73	2.5	1.28	0.75	4	0.25	0.12	0.075	0.25
	45,000	5.73	2.5	1.28	0.75	4	0.25	0.12	0.075	0.25
	60,000	7.63	1.98	1.17	0.7	4	0.19	0.11	0.07	contact
Plant 6	140,000	17.81	3.27	2.26	1.6	7.5	0.32	0.22	0.16	contact
Plant 7	63,000	8.02	1.68	0.82	0.5	3	0.16	0.082	0.05	contact
Plant 8	90,000	11.45	2.43	1.29	0.81	4	0.24	0.12	0.08	0.25
Plant 9	150,000	19.08	3.6	2.08	1.3	6.25	0.36	0.2	0.13	0.5
Plant 10	120,000	15.27	1.89	1.04	0.63	3.5	0.189	0.102	0.06	contact
Plant 11 Ladle 1	50,000	6.36	0.75	0.41	0.25	2	0.075	0.041	0,025	contact
Plant 11 Ladle 2	50,000	6.36	0.95	0.49	0.29	2	0.095	0.049	0.029	contact
Plant 13	110,000	13.99	3.23	1.56	0.87	4.5	0.32	0.15	0.087	0.5
	25,000	3.18	1.77	0.7	0.36	2.5	0.17	0.07	0.036	0.25
Plant 14	100,000	12.72	3.25	1.65	0.98	4.5	0.32	0.16	0.098	0.5

Facility	Weight of steel (kg)	Source volume (m^3)	Dose rate for 10 Bq/g (uSv/h) 1 m	2 m	3 m	Distance limit for 0.5 uSv/h (m)	Dose rate for 1 Bq/g (uSv/h) 1 m	2 m	3 m	Distance limit for 0.5 uSv/h (m)
Plant 15	200,000	25.45	2.27	1.28	0.8	4.25	0.22	0.12	0.08	contact
Plant 17	75,000	9.54	1.1	0.5	0.3	2	0.11	0.05	0.03	contact
Plant 18	140,000	17.81	3.9	1.2	0.67	2.5	0.39	0.12	0.06	contact
Plant 19	120,000	15.27	3.9	2.4	1.6	7	0.4	0.24	0.16	contact
Plant 22	285,000	36.26	2.59	1.48	1.01	6	0.25	0.14	0.1	0.25

6. REFERENCES

[1] Protocolo de colaboración sobre la Vigilancia Radiológica Materiales Metálicos (Protocol for Radiological Monitoring of Metallic Materials).

[2] RP 89. Recommended radiological protection criteria for the recycling of metals from the dismantling of nuclear installations.

[3] Council Directive 2013/59/Euratom of 5 December 2013, laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation.

[4] Royal Decree 783/2001 of 6 July, approving the Regulation on the Protection of Health against Ionising Radiation.

[5] Royal Decree 551/2006 of 5 May, regulating the transport of dangerous goods by road within Spanish territory, amended to comply with the ADR (Official State Gazette A-2015-4093).

[6] Categorization of radioactive sources (IAEA safety standards series no. RS-G-1.9).

Comparative Study of Radioactive Waste Management Standards in Brazil

Ana Cristina Lourenço[a], Wagner de Souza Pereira[a,b]

[a]Industrias Nucleareas do Brasil, Resende, Rio de Janeiro, Brazil

[b]Universidade Veiga de Almeida, Tijuca, Rio de Janeiro, RJ, Brazil

Abstract. Radioactive Waste Management (RWM) is an activity that includes the generation, separation, transport and deposition of radioactive waste. The first Brazilian standard of RWM was published in 1985 as an experimental standard and was reviewed in 2014, as nuclear standard. This study aims to compare the changes of the standards listed before from a technical and operational point of view. Some of these changes are described as following. About the goals, the experimental standard (ES) aimed to establish general criteria and basic requirements of RWM, while the nuclear standard (NS) ensure the safety and radiological protection in RWM. The classification of radioactive waste (RW) evolve from type of emission and physical state into a classification based on radiological characteristics, as activity and half life, providing a more objective classification from the point of view of radiation protection. About the containers for segregation, collection and transport of the RW, the main change was the replacement of the term “container” for "packaging” adapting this standard for the standard of transport of radioactive material (CNEN-NE 5.01). Concerning the treatment of RW, the approval of the National Nuclear Authority (CNEN) is still required. The transfer of RW between facilities must still be authorized by CNEN. While the import is not allowed, the export of RW can be authorized by CNEN. The records and inventories have to be kept as before. As a conclusion we can notice an evolution in RWM in Brazil, sometimes adapting to the practice, others to new standards and international guides. A very well established RWM and an improved characterization of the RW were important to ensure the safety and security of individuals occupationally exposed and the members of the public, making the RWM safer in Brazil.

Radioecological Monitoring at the Areas of the Former Military Technical Bases at the Russian Far East

Akhromeev Sergey, Kiselev Sergey, Titov Alexey, Isaev Dmitry, Seregin Vladimir, Gimadova Tamara

SRC – Burnasyan Federal Medical Biophysical Center, Moscow, Russia

Abstract. In the 1960s, the coastal technical base of the Pacific Fleet was created in the Far East region of the USSR in Sysoeva Bay at Primorskiy territory 40 km away from the Vladivostok port. This technical base was used to service nuclear submarines performing reception and storage of spent nuclear fuel (SNF) and radioactive waste (RW).After termination of operation at the serviced facilities of the nuclear fleet of the former Soviet Union, the Military Technical Base in Sysoeva Bay has been reorganized to the site for SNF and RW temporary storage (STS). According to the Russian Federation Government Directive, in 2000, this base was transferred under authority of Russian Ministry of Atomic Energy (the Minatom of Russia, since 2008 - the State Atomic Energy Corporation "Rosatom") for the purpose of its remediation. The main activities of STS are receipt, storage and transmission to radioactive waste reprocessing. The nature and peculiarity of the STS area radioactive contamination are the following: 1) high levels of radioactive contamination on the industrial site; 2) non- uniformity of the contamination distribution; 3) spread of contamination in the area of health protection zone. To get unbiased comprehensive information on the current radiation conditions at the STSs and provide the effective response to changing radiation situation, the complex environmental monitoring (radiological, chemical and biological) of the DalRAO facilities has been carried out during 2009-2014. The following environmental components are contaminated: soil, vegetation, bottom sediments and seaweeds at the offshore seawaters. The dominant radionuclides are cesium-137 and strontium-90. The monitoring findings served as a basis for development of regulatory documents aimed at assurance of public and environment radiation protection and safety during RW management.

The Radiological Risk Assessment for Workers Involved in Liquid Waste Transfer Operations

Ana Stochioiu, Carmen Tuca, Mitica Dragusin, Daniela Gurau, Felicia Mihai

"Horia Hulubei" National Institute for Physics and Nuclear Engineering, IFIN-HH, Magurele, Ilfov, Romania

Abstract. The paper aims to assess the radiological risk of workers which are involved in liquid waste transfer operations to the treatment station. These effluents arise from decommissioning activities of a nuclear research reactor (NR) – source 1- and from the decommissioning of the Spent Nuclear Fuel Storage (SNFS) pools – source 2. The radiological inventory consists to the following radionuclides: i) source 1: ^{60}Co, ^{134}Cs, ^{137}Cs, ^{108m}Ag, ^{54}Mn, ^{235}U, ^{238}U and ii) source 2: ^{60}Co, ^{134}Cs, ^{137}Cs, ^{108m}Ag, ^{54}Mn, ^{152}Eu, ^{154}Eu, ^{241}Am. The risk assessment consists of effective dose calculation received by workers, based on the activities concentration measured for one cubic meter of liquid effluents, for each source and specific dose coefficients for ingestion. The coefficients were taken from IAEA Safety Standards Series No. GSR PART 3, IAEA 2014 and from Romanian Fundamental Norms for Radiological Safety (NSR-01), CNCAN 2000. The dose could be received by workers for normal and abnormal situations during the transfer process of radioactive liquid effluents. For the source 1 the dose value is 8.82E+00 mSv and for source 2 the values are: 3.92E-01mSv pool 1; 1.44E+00 mSv pool 2; 5.71E+02 mSv pool 3 and 1.67E+01mSv for pool 4. Thus, the higher risk for workers involved in transfer process is generated by radioactive liquid effluents arising from pool 3.It was estimated the radiological risk of workers due to the external exposure during the decommissioning activities of SNFS. For this purpose it was determined the dose ambient equivalent - H*(10) – using the environment thermoluminescent system (TLD). The maximum value (4387±259) nSv/h was obtained for the third pool, due to the radioactive leakage from a defect fuel assembly, during the operation period. For the other three pools the values are ranging between (92÷ 800) nSv/h.

KEYWORDS: radiological risk; worker; activity concentration; dose; decommissioning.

Slaying the Dragon – The story of one FPSO, twenty odd Vietnamese and 3 concrete mixers.
Decontamination and disposal of NORM

Annelize van Rooyen, Anthony O'Brien

Radiation Professionals, PO Box 3231, East Perth, WA 6004, Australia

Abstract. End of life of an FPSO requires a lot of planning and management. One of the major challenges is the issue of decontamination and waste management. Waste disposal is a very touchy subject and with agreements like the London Protocol and differences in legislation between countries, it can become a major stumbling block. Radiation safety in the oil and gas arena is something not often on the mind of an FPSO operator or even the management. The only radiation hazard that they are aware of is the presence of a source in the density gauges/profilers in the separator vessels. The planning and layout of such a vessel and its processing plant have usually not gone through any ALARA process during its design. The fact that the workers can be operating a plant that has a myriad of waste traps, elbows, flow restrictors and filters that can become significant radiation sources due to scale and schmoo (a gelatinous organic/inorganic mixture of oil, water and fine silt, often caused by the addition of treating chemicals to production waters) build up is not something they think about. Planning the decontamination of such a vessel should start long before the actual decommissioning date appears on the calendar. Doing regular vessel clean outs and radiological profiling of the entire plant can be beneficial in reducing the stress of clean up in the end. Finding a workable solution in getting NORM contaminated waste out of the vessels and tanks and effectively reducing the waste volumes for end of life clean-up is very important. This is the story of one such a vessel and its end of life

KEYWORDS: FPSO; decontamination; oil and gas; waste disposal.

Radiation Protection Dosimetry (2017), Vol. 173, No. 1-3, pp. 268–273

doi:10.1093/rpd/ncw331

Issues Related to Regulation, Control, and Waste Management of Natural Radioactive Scales with Low Specific Activity in Oil Producing Establishments in Libya

Bulgasem El-Fawaris

Tajura Nuclear Research Center, Tripoli-Libya, Libya

Abstract. During the last four decades of the 20th century, literature reviews were full with regulations and control documents related to exposure, safe handling, and waste management of man- made radioactive sources. In contrast there is still a lack of consistent regulatory control of natural radioactive scales with low specific activity in mining and oil producing equipment, leading to the existence of Technologically Enhanced Naturally Occurring Radioactive Materials (TE-NORM). So, different countries apply different version of TE-NORM regulatory control, depending on type and industry causing such phenomena. This paper illustrates the different regulatory effort promulgated by Libyan authorities for controlling utilization of ionizing radiation sources and prevention of risk related to such utilization, including environmental protection. The paper addresses the regulatory control of TE-NORM, source characterization and generation, radiation protection concerns, solid waste management and its safe disposal based on some observations that were obtained through many field trips and investigating scenarios that were conducted and carried out within many operating oil fields in Libya for more than eight years starting in 1998.

KEYWORDS: TE-NORM, promulgated Libyan regulations.

A comparison of remediation after the Chernobyl and Fukushima Daiichi accidents

Brenda J Howard[a], Sergey Fesenko[b], Mikhail Balonov[c], Gerhard Pröhl[b], Shinichi Nakayama[d]

[a]Centre for Ecology and Hydrology, Lancaster, UK

[b]International Atomic Energy Agency, Vienna

[c]Institute of Radiation Hygiene, St Petersburg, Russian Federation

[d]Japan Atomic Energy Agency, Japan

Abstract. Extensive remediation was conducted on contaminated landscapes after the Chernobyl accident in 1986 and the Fukushima Daiichi accident in 2011. A comparison is made of a range of different features relevant to each accident including the characteristics of the contamination and the landscapes affected, the radiological criteria, the designation of areas to be remediated and the remediation measures adopted.

KEYWORDS: remediation; Chernobyl; Fukushima Daiichi.

Radiation Protection Dosimetry (2017), Vol. 173, No. 1-3, pp. 170–176
doi:10.1093/rpd/ncw312

Radioactive Waste Management without Adherence to Standards at the Laguna Verde Nuclear Power Plant, Mexico

Bernardo Salas

National Autonomous University of Mexico, Mexico City, Mexico

Abstract. Waste Management at the Laguna Verde Nuclear Power Plant (LVNPP) has been questioned by the National Commission of Nuclear Safety and Safeguards (NCNSS)- the regulatory body in nuclear matters in Mexico, after making the "Inspection of Activities Related to the Waste Management from the Laguna Verde Nuclear Power Plant 1 and 2" and the activity entitled: “Verification the isotopic composition of the Lots prior to initiating the process of solidification”, it was observed anomalies in the implementation of the working procedures that establish the ways how they must perform these activities, such as "Verifying the operation of the main monitoring systems stores”, “Technical Instructions for Process Control Program” and in particular the proceeding concerning radioactive waste compaction. This is based on the letter No. AOO.130 / 017/2015 dated 14 April 2015 issued by the NCNSS with the subject: “Administrative Punishment Reprimand with the Warning". This paper outlines the recommendations issued by NCNSS to overcome these irregularities and are discussed other incidents about Waste Management, which aims to ensure the safety of personnel, the public and the environment, as it concluded by the letter of the NCNSS: “Emphasis is placed on that possess an Operating License from a Nuclear Power Plant, granted by the Ministry of Energy of the Mexican government, it gives the LVNPP responsibility to perform to the highest safety standards, fulfilling all and each of the regulatory requirements and license conditions which apply, always staying vigilant to ensure the safety of personnel, the public and the environment.” This paper was developed in the Department of Physics at the Faculty of Sciences of the National Autonomous University of Mexico.

Statistical Learning Approaches Applied to the Calculation of Scaling Factors for Radioactive Waste Characterization

Biagio Zaffora[a,b], Matteo Magistris[a], Francesco Paolo La Torre[a], Gilbert Saporta[b], Catherine Luccioni[a], Jean-Pierre Chevalier[b]

[a]European Organization for Nuclear Research (CERN), Geneva, Switzerland

[b]Conservatoire national des arts et métiers, Paris, France

Abstract. In the frame of radiological characterization of radioactive waste, scaling factors are typically used to quantify the activity of difficult-to-measure radionuclides from the direct measurement of dominant gamma emitters. The application of this characterization method relies on statistical considerations, such as the probability distribution of the scaling factors themselves, along with a deep understanding of the activation mechanisms in place. The present study assesses the use of statistical learning techniques for the prediction of scaling factors, for the characterization of very-low-level radioactive waste produced at CERN (European Organization for Nuclear Research, Geneva, CH). In particular, decision trees are used to identify sorting criteria of future radioactive waste based on the minimization of the Residual Sum of Squared errors coupled with binary splitting. Other methods (e.g. pruning, bagging, random forest, boosting and linear models) are also used for the optimization of the calculation of the scaling factors together with the prediction intervals. The estimation of scaling factors for the present study is based on over 1 million analytical calculations. These calculations cover the most typical activation scenarios and chemical composition of metals used at CERN. The calculations were performed with the code ActiWiz, which was developed at CERN based on FLUKA Monte Carlo simulations of reference particle spectra. The analytical predictions were benchmarked with extensive radiochemical and gamma- spectrometry measurements of samples taken from activated metals. The results demonstrate the applicability of statistical learning techniques on waste characterization, especially for the prediction of the activity of difficult-to-measure radionuclides. Average scaling factors and prediction intervals are systematically calculated and can be easily applied on routine characterization processes at CERN. A first analysis of the errors introduced by the use of scaling factors is also presented.

KEYWORDS: scaling factor; radioactive waste; decision tree; random forest; bagging; boosting; linear models.

The Swiss approach to deal with radium legacies from the watch industry *The Radium action plan (2016-2019)*

Christophe Murith, Sybille Estier, Martha Palacios, Sebastien Baechler

Radiation Protection Division, Federal Office of Public Health (FOPH), CH-3003 Bern, Switzerland.

Abstract. Following the discovery of radioactive radium contaminations in a former landfill site it was decided to search for and remediate those sites that had possibly been contaminated with radium as a result of its use in the watchmaking industry between 1920 and 1960. This approach is based on the concept of existing exposure situations as developed by the International Commission on Radiological Protection, and is supported by an action plan for radium 2016 - 2019 approved by the Federal Council in 2015. This plan comprises 4 steps: the search for potentially contaminated sites, the measurement and assessment of each site, the remediation of those sites where the public would be exposed to an annual dose greater than 1 mSv, and actions to secure the landfill sites. The arrangements for each step are described in the present article. The measurement and remediation procedures imply intrusions into the privacy of the inhabitants. This requires the public authorities to actively inform the population and to develop an effective and transparent means of communication. The actions developed for this are also described.

KEYWORDS: Radium; contamination; remediation; existing exposure situation.

Radiation Protection Dosimetry (2017), Vol. 173, No. 1-3, pp. 245–251
doi:10.1093/rpd/ncw335

Post closure safety of the SFR repository for short-lived low- and intermediate level waste

Eva Andersson, Fredrik Vahlund, Klas Källström, Ulrik Kautsky

Swedish Nuclear Fuel and Waste Management Co. (SKB), Stockholm, Sweden

Abstract. In December 2014 Swedish Nuclear fuel and waste management company (SKB) submitted an application to extend the existing repository for low- and intermediate level waste (SFR) to the authorities. SFR, the existing part and planned extension, is placed at 60-120 meter depth in archean rock. The nuclear power plants in Sweden will be decommissioned and dismantled which require additional repository capacity. Additional disposal capacity is also needed for operational waste from nuclear power units in operation since their operating life- times have been extended compared with what was originally planned. For the application an evaluation of long-term safety is required. This is done by conducting a detailed safety analysis and evaluating the compliance with the Swedish Radiation Safety Authority's regulations concerning safety and protection of human health and the environment in the long-term perspective. The annual risk criterion for humans is 10^{-6}, which corresponds to 1 % of the background radiation at the site. The time frame of the safety assessment is 100,000 years under which there is an development of both the repository system and the external conditions (climate and ecosystem). The extended SFR repository is shown and the methodology for the safety assessment is described. Some examples of major results for the dominating radionuclides (C-14, Mo-93, Ni-59) are presented. The central conclusion of the safety assessment SR-PSU is that the extended SFR repository meets regulatory criteria with respect to long-term safety.

Development of a Standardised Screening Procedure for the Evaluation of Sites Potentially Contaminated with NORM in Austria

Eva Maria Lindner, Claudia Landstetter, Michael Tatzber, Fabian Rechberger, Michael Dauke, Christian Katzlberger

AGES (Austrian Agency for Health and Food Safety), Vienna, Austria

Abstract. Due to Austria's access to the Joachimsthal uranium mines several factories and institutes in Austria started to work with materials containing heightened levels of radioactivity at the end of the 19th century. In a study carried out by the Austrian Agency for Health and Food Safety the premises of a former chemical factory were investigated. According to historical research the company, which was founded around 1890, used to process pitch blende residues (uranium ore residue) for the production of radium as well as monazite sands for the production of thorium and subsequent manufacturing of incandescent gas mantles. The site of the factory was situated next to a river and nowadays serves as a recreational area. The results of the radiological survey showed elevated levels of radioactivity and upon detailed examination of the soil samples, elevated activity concentrations of thorium, uranium and lead. After the decontamination of the site further research showed that several other locations in Austria may have also been contaminated due to historical activities with natural radionuclides. In order to be prepared for the evaluation of these sites the results and experience gained from the above mentioned investigation were reviewed for the purpose of developing a standardised screening procedure for potentially contaminated sites. After taking into consideration the available radiochemical procedures and measurement methods a detailed 'plan of action' was defined. As a first step on-site measurements are carried out and on the basis of the results a sampling strategy is developed. There are various sample matrices that may be encountered at relevant locations, such as mud, soil, sediments, building materials and water. All subsequently used measurement methods must be applicable to these types of sample matrices or adapted accordingly. Samples are first assessed by gamma spectroscopy. Further sample treatment/separation through physical and radiochemical procedures followed by low-level liquid scintillation counting, alpha spectrometry and ICP-MS (Inductively Coupled Plasma Mass Spectrometry) may be necessary to investigate non gamma-emitting radionuclides and accurately predict an exposure scenario. For the implementation of remediation measures detailed results are needed to classify the encountered materials and correctly dispose of them (land-fill, radioactive waste, etc).

Critical Factors for Radiological Closure Criteria for Uranium Mine Remediation

Frank Harris

Rio Tinto, Brisbane, Queensland, Australia

Abstract. Historically some of the most significant radiological issues associated with uranium mining have resulted from poor remediation practices. Modern uranium mines generally consider rehabilitation through all operational stages and hence have far less potential for environmental and public impact. The control of all potential exposure pathways and long term containment periods are required. The importance of considering and subtracting the pre-mining natural background is critical to determining the closure criteria for the site. An overly conservative approach to the consideration of closure is not appropriate as it may lead to poor utilisation of natural resources and unnecessary restrictions on future land use. Optimisation of the closure criteria is the key with strong consideration of local site specific factors and the future land use being critical. Appropriate closure criteria are essential to both minimise the future impact and allow for successful handover of a mine-site post closure.

The International Standards for the Safety of Radioactive Waste Management

Gerard Bruno

International Atomic Energy Agency, Vienna, Austria

Abstract. The purpose of the presentation is to give an overview of the international standards for the safe management of radioactive waste. The International standards cover both the predisposal management and the disposal of radioactive waste. Safety requirements on the management of radioactive waste provide the requirements, using "shall" statements that must be met to ensure the protection of people and the environment, both now and in the future. The safety requirements are supported by safety guides providing, using "should" statements, the recommendations and guidance on how to meet the safety requirements. While the safety requirements on predisposal management of radioactive waste is a general publication covering all facilities and activities, the safety requirements on the disposal of radioactive waste is a facility specific publication providing 26 requirements for all types of disposal facilities for all types of radioactive waste. The presentation will address the safety requirements in the field of radioactive waste in particular highlighting the elements related to the demonstration of safety, but will also make a specific focus on the international classification for radioactive waste which has been established in the view of long term safety and identifies the suitable disposal facilities for each of the class of radioactive waste.

Talking to the Future: Sustainable Solutions for Radioactive Waste

Gordana Lastovicka-Medin

University of Montenegro, Podgorica, Montenegro

Abstract. This paper presents research on currently used and developed solutions for radioactive waste and gives a proposal for future sustainable solutions. Paper presents research on nuclear waste disposal, the potential of reprocessing and recycling. Additionally paper provides an overview of the relationship between reprocessing used nuclear fuel and recycling it, as well as the impact that reprocessing would have on radioactive waste disposal. Arguments for and against sustainable solutions for radioactive waste are carefully chosen and presented.

The Dose Calculation on Graphite Waste Samples of the Decommissioned KRR-2

Heereyoung Kim, Deokjung Lee

Ulsan National Institute of Science and Technology, Ulsan, Republic of Korea

Abstract. The radioactivity of radiocarbon and tritium of graphite samples from the decommissioned Korea Research Reactor-2 (the KRR-2) was analyzed by using high-temperature oxidation and liquid scintillation counting method. The graphite waste was estimated to be disposed as a low-level radioactive waste rather than reuse or recycle. The KRR-2 has been operated for 23 years since 1972 and shutdown in 1995. The KRR-2 facilities with a thermal output of 1 MWth, were completely dismantled in 2009 where the graphite waste with the weight of about seven tons was generated during dismantling. Graphite was used as a moderator which consisted thermal column in the KRR-2 and irradiated by thermal neutron. graphite is one of the sources producing radioactivity in addition to the nuclear reactor core. The graphite with the large volume from the KRR-2 was expected to include considerable amounts of low and intermediate radioactive wastes. The radioactivity of the graphite of thermal column is originated from various nuclides including beta nuclides of ^{3}H, ^{14}C, ^{63}Ni, ^{56}Fe, ^{36}Cl, ^{90}Sr; gamma nuclides of ^{152}Eu, ^{137}Cs, ^{60}Co; and others. Most of the radioactivity is known to be due to ^{14}C and ^{3}H, which are released as a gas state from the graphite waste. The graphite samples were oxidized at a high temperature of 800 degrees centigrade, and their counting rates were measured by using a liquid scintillation counter (LSC). About 99 % of the graphite was oxidized on the sample with a maximum weight of 1 g. The high-temperature furnace showed the recoveries of around 100 % and 90 % in ^{14}C and ^{3}H, respectively. The minimum detectable activity was 0.04-0.05 Bq/g for the ^{14}C and 0.13-0.15 Bq/g for the ^{3}H at the same background counting time. Its annual dose was analyzed in terms of the internal exposures including inhalation and ingestion where the pure beta emitting nuclides of ^{14}C and ^{3}H, have dominant effects. The dose estimation due to the ^{14}C and ^{3}H radionuclides showed that the graphite waste from the sampled region of the thermal column of KRR-2 would have difficulty in taking clearance. Hence, it was thought that the graphite waste from the dismantled research reactor could not be reused or recycled and should be disposed as radioactive waste in spite of no dose estimation by nuclides including gamma emitting ones.

Characterization of Concrete Structures to Determine the Strategy for the Decommissioning of the 250 MeV CGR Cyclotron of the Ghent University in Belgium

Isabelle Meirlaen[a], Myriam Monsieurs[a], Hubert Thierens[a], Luc Ooms[b], Sven Boden[b], Luc Noynaert[b]

[a]University of Ghent, Ghent, Belgium

[b]Belgian Nuclear Research Center SCK•CEN, Mol, Belgium

Abstract. In the framework of the decommissioning of the 250 MeV CGR cyclotron of the Ghent University in Belgium, that will be carried out in the coming years, the first step is preparing a final decommissioning plan. This task was assigned to the Belgian Nuclear Research Center (SCK•CEN). The plan needs to define the decommissioning strategy and determine the end stage of the decommissioning. The major strategic decision is the destination of the activated concrete bunkers. Considering the activated concrete as radioactive waste would result in a considerable additional cost, therefore decay storage of the bunkers down to the clearance level might be the preferred option. The feasibility of this option depends on the duration to reach the clearance levels, and so the half-life of the radionuclides present and their activity concentration. The activation products of the concrete structures are mainly ^{152}Eu and in smaller amounts ^{60}Co. A total of 30 sampling points were selected based on a dose rate mapping and a judgemental approach. At each location a sample was taken using wet core drilling. Each core of about 35 cm length and 55 cm diameter was cut into slices of about 1,5 cm thickness. These slices are radiologically characterized. The measurements are performed at Ghent University using two gamma spectrometry devices: the Low Level Radioactive Waste System, which is a fixed laboratory setup, and an In Situ Object Counting System (ISOCS), a mobile measuring system. Both systems are based on a High-Purity Ge-detector with a relative efficiency of 25%. The measuring methods were developed to allow for the minimal detectable activity for ^{152}Eu and ^{60}Co to be significantly below the legal clearance level of 0.1 kBq/kg.

Waste Management at the Decommissioning of the Ghent University Research Reactor Thetis in Belgium

Isabelle Meirlaen[a], Myriam Monsieurs[a], Hubert Thierens[a], Patrick Maris[b], Luc Noynaert[b], Luc Ooms[b]

[a]University of Ghent, Ghent, Belgium

[b]Belgian Nuclear Research Center SCK•CEN, Mol, Belgium

Abstract. In the framework of the decommissioning of the research reactor Thetis of the Ghent University in Belgium, carried out in 2012-2014, a waste management program has been set up. Based on the selective collection and radiological characterization of all material within the reactor building, waste categories were determined with different destinations depending on increasing levels of radioactive contamination or activation: free release, conditional release following melting or radioactive waste storage. Starting from the physical and radiological inventory from the installation, the waste categories were defined in which all material that must be removed, was classified. Each type of material was, as much as possible, separately collected in a waste category. The waste packages used for collecting the waste were mainly steel drums of 200 liter for solid waste and jerrycans for liquid waste. The first step in the radiological characterization was neutron activation calculations performed by the Belgian Nuclear Research Center (SCK•CEN), who managed the decommissioning project. By these calculations the different activation products produced in different reactor components, as well as their activity, were determined. For activity measurements, ^{60}Co was used as key nuclide and activities of other radionuclides were determined relative to ^{60}Co using the specific radiological vector obtained from the activation calculations. The ^{60}Co activity was measured by gamma spectrometry using a High-Purity Ge-detector with a relative efficiency of 25%. The measuring procedures allowed for a minimal detectable activity for ^{60}Co to be significantly below the legal clearance level of 0.1 kBq/kg. All information about the waste from the decommissioning was stored in a material management system developed by SCK•CEN. That way, traceability was guaranteed throughout the process and a practical overview of all waste was always available. The waste management program allowed removal of all radioactive material from the reactor, while also reducing the amount of radioactive waste, using contemporary measurement techniques and methodologies.

Decontamination and Recovery of a Nuclear Facility to allow Continued Operation

Josh Cavaghan

Sellafield Ltd, Seascale, Cumbria, CA20 1PG, UK.

Abstract. A power supply failure caused a loss of power to key ventilation systems in an operating nuclear facility. During this time, the in-cell depression was lost which led to an egress of activity through prepared areas and into the normal operating areas. After an initial program of radiological monitoring to quantify and categorise the activity in the operating areas, a team was established consisting of subject matter experts and plant managers to establish a plan for the decontamination and remediation of the plant. The scope of the recovery plan was substantial, with many stages to ensure decontamination was undertaken in a methodical and logical manner. Examples of some of these stages are: return to service of the unfiltered ventilation system in on-plant once areas and access corridors, decontamination of normal operating areas with loose contamination levels up to 1000 Bq/cm^2, entry into cells with unknown contamination levels, and subsequent decontamination of these areas with loose contamination found up to 9000 Bq/cm^2, over 1000s of square metres throughout the facility. The activity was found to be almost entirely ^{137}Cs, which reflects the α:β/γ ratio indicated by the facility isotopic fingerprint. In addition to the physical remediation work, several administrative controls were introduced to further improve the safety of the decontamination teams, such as new Local Rules, safety signage to indicate abnormal radiological conditions in certain areas, and training of the decontamination teams on the radiological hazards and associated controls. Upon completion of the recovery operation, all areas of plant which were contaminated were returned to normal access arrangements and the plant was successfully returned to full operational capability, less than 12 months from the date of the event.

KEYWORDS: decontamination; nuclear; ALARP.

Radiation Protection Dosimetry (2017), Vol. 173, No. 1-3, pp. 118–123
doi:10.1093/rpd/ncw296

General Principles for the Short and Long Term Management of Former Uranium Mining Sites in France

Jérôme Guillevic, Marie Odile Gallerand, Gwenaelle Loriot, Christophe Serres

Institut de Radioprotection et de Sureté Nucéalire, Fontenay aux Roses, France

Abstract. As part of managing the remediation of repository for uranium mill tailings in France, a working group led by the French Ministry of Environment has published a guide in May 1999. This document was intended to introduce the options currently available for the remediation of a repository of uranium mill tailings and the principles and methods used to assess the short and long term impact on both man and environment. As part of its mission to support public authorities, the "Institut de Radioprotection et de Sûreté Nucléaire" (IRSN) was asked to contribute to the updating of this guide by extending its scope to all mining sites. The work was: to update the objectives and general principles of protection of man and the environment in accordance with regulations; to update the methodology for evaluating the radiological and chemical impact by extending it to all the mining sites, and clarifying the sites remediation principles; to complete the initial document with the essential elements of a site monitoring and surveillance plan and its environment. This update is in the perspective of administrative closure process required by the French mining code that will lead to the output of the French mines police regime of a number of sites and the transfer of their responsibility to the state. The extension of the scope of this guide requires taking into account other sources of pollutant than tailings themselves and, in particular, waste rock dumps, water treatment plants but also radiametric abnormalities associated with mining materials (waste rocks, low grade ore or tailings or uranium concentrate) left on the grip of the sites. The update of this document is also an opportunity to integrate the principles of the reference texts published in recent years in the field of Radiation Protection (ICRP Publication 103, 2013/59/Euratom Directive) and integrate the feedback acquired through the actions undertaken by the government on the management of former uranium mining sites.

Safety Standards for the Management and Disposal of Radioactive Waste from Uranium Mining in France

Jérôme Guillevic, Marie-Odile Gallerand, Gwenaelle Loriot, Christophe Serres

Institut de Radioprotectin et de Sûreté Nucélaire, Fontenay aux Roses, France

Abstract. Uranium was mined in France between 1948 and 2001. During this period some 250 mine sites were exploited and 76 000 tons of ore extracted. The waste generated from these activities includes 50 000 000 tons of tailings and 200 000 000 tons of waste rock. The tailings are disposed in 17 disposal facilities. A doctrine was issued in 1999 addressing the safety of these residues from the uranium mining and processing activities. In view of the developments in international radiation safety recommendations and standards and the recently updated European Basic Safety Standards, the need has arisen to update this doctrine. The update will cover the objectives and general principles of protection of humans and the environment, the methodology for evaluating the radiological and chemical impact and site monitoring plans. The impact assessment methodology will be extended to all the mining sites; it will cover remediation activities and will cover the long term impact of mine tailings facilities. Because of the related nature of the issues, the updated doctrine will also cover the issue of radon in buildings. The work on updating the doctrine is in progress and the paper will present the proposals under consideration and the issues that have arisen.

Enhancing Regulatory Oversight of Radioactive Sources through International Cooperation

Jack Ramsey

U.S. Nuclear Regulatory Commission, Washington, DC, USA

Abstract. The Code of Conduct on the Safety and Security of Radioactive Sources (the Code) was approved by the Board of Governors of the International Atomic Energy Agency (the IAEA) in September 2003. Since then, over 120 countries have written to the Director General of the IAEA expressing their desire to both adopt and implement the provisions of Code. This is a remarkable achievement. Nonetheless, there are still a number of remaining IAEA Member States and non-Member States that have not yet expressed support for the Code. To support both further adoption and implementation of the Code, the Nuclear Regulatory Commission (NRC), the IAEA and others are working to strengthen cooperation between international regulatory counterparts for safety and security oversight of radioactive sources. Such cooperation supports efficient and effective global adoption and implementation of the Code. This presentation describes the bilateral cooperative programs established by the NRC with international regulatory counterparts, programs that enhance safety and security regulatory oversight radioactive sources consistent with the Code. This presentation also highlights the synergy and coordination between the respective IAEA and NRC radioactive source-related outreach programs.

New Measurement Systems for Clearance of Radioactive Materials from Nuclear Facilities Decommissioning

Jiri Suran[a], Jana Smoldasova[a], Petr Kovar[a], Lukas Skala[b], Bent Pedersen[c], Dirk Arnold[d], Daniel Zapata[d], Pierino de Felice[e], Doru Stanga[f], Simon Jerome[g]

[a]Czech Metrology Institute, Prague, Czech Republic

[b]Envinet Nuvia Group, Trebic, Czech Republic

[c]Joint Research Centre-European Commission, Ispra, Italy

[d]Physikalisch-Technische Bundesanstalt, Braunschweig, Germany

[e]Agenzia Nazionale per le nuove tecnologie, l'energia e lo sviluppo economico sostenibile, Roma, Italy

[f]Institutul National de Cercetare Dezvoltare pentru Fizica si Inginerie Nucleara, Horia Hulubei, Magurele, Romania

[g]National Physical Laboratory, Teddington, UK

Abstract. The world faces a major challenge of great urgency: the enormous costs of decommissioning many outdated nuclear facilities. Nuclear decommissioning covers all activities from shutdown to the environmental restoration of the site. More than 200 power reactors are presently being decommissioned or will be in some phase of the process by the year 2025. Therefore, it is essential to achieve a significant reduction in the enormous decommissioning costs based on the development and implementation of new measurement techniques. The European Metrology Research Program (EMRP) is a coordinated research-and- development program that pursues the closer integration of national research programs. The EMRP Metrology for Decommissioning Nuclear Facilities (MetroDecom) Joint Research Project (JRP ENV54), a 36-month effort that became active in September 2014, delivers research addressing all aspects of the decommissioning process, including the characterization of waste materials present on the decommissioning site, pre-selection prior to free release or repository acceptance measurement, free-release measurement, the measurement of waste packages thermal power prior to the repository storage, and the monitoring of stored wastes and repositories in the very long term. This paper describes the track record of two JRP ENV54 working packages (WP2, WP3), which address the measurement facilities for: pre-selection of waste materials prior to measurement for repository acceptance or possible free release (segregation facility); and free release (free release measurement facility), based on a single standardized concept characterized by unique, patented lead-free shielding, modularity, mobility, high accuracy and high throughput. The key objective is to quickly improve the throughput, accuracy, reliability, modularity and mobility of segregation and free-release measurement, which will result in more reliable decision-making with regard to the safe release and disposal of radioactive wastes into the environment and, in the process, ensure positive economic outcomes.

Development of Postulated Initiating Event's Scenarios for the Decommissioning of Korean 1400 MWe PWR

Kyomin Lee[a], Sangho Kang[a], Keunsung Lee[b]

[a]KEPCO Engineering & Construction Companny Inc., 269 Hyeoksin-ro, Gimcheon-si, Gyeongsangbuk-do, Republic of Korea

[b]Korea Hydro & Nuclear Power Central Research Institute Co., 1312-70 Yuseong-dae-ro, Yuseong-gu, Daejeon, Republic of Korea

Abstract. It is widely accepted that radiological hazards associated with a nuclear facility under decommissioning are less than when in normal operation, since the reactor has been defueled following the permanent shutdown. However, the significant release of airborne radioactivity during decommissioning could be caused by the events associated with accidental cutting operations, dropping of radiological contaminated component or fire and/or explosion. Considering the planned decommissioning activities of the Korean 1400 MWe PWR, therefore, the postulated initiating event's (PIEs) scenarios that could result in releases of radioactive material as airborne particles small enough to be respirable were developed in this study. First of all, a list of the anticipated PIEs is established using the process of the hazard identification specified in the IAEA, WS-G-5.2, "Safety Assessment for Decommissioning of Facilities Using Radioactive Material". The initiating events are sorted into three categories: internal events; external events including events initiated by natural phenomena and human-made event; on-site (or off-site) transportation events. The initiating event scenarios within each of these categories are further subdivided into ten (10) groups in accordance with the extent of similarity of the cause/evolution/effect of the events, where each group can be represented by a single event scenario, of which the consequences will represent (or exceed) those of other scenarios in the group. The event scenarios include the followings: accident type, accident duration and termination events, cause and activities, definition of conservative source terms, and release rate. The qualitative analysis for each event scenario has been undertaken based on an unmitigated consequence, where "unmitigated" means that radionuclide concentrations and total inventory, form, location, release fraction and dispersibility are considered without any safety measure including air filtration system, fire suppression, and minimization of radioactive dust. As a result of the qualitative analysis, the event scenarios associated with the fire and/or explosion, and the failure of the control envelope were determined as representative ones that could have an effect on significant radiological consequences for members of the public and workers.

Impact Assessment on Rn-222 in a Radioactive Waste Disposal Facility

Luis Fuentes[a], José-Luis Pinilla[a], Teresa Ortiz[b]

[a]ENRESA-CABRIL, Córdoba, Spain

[b]ENRESA, Madrid, Spain

Abstract. Radioactive sources may be temporarily stored in disposal facilities for low and intermediate level radioactive waste. These sources include disused Ra-226 sources, which do not meet the criteria for disposal. The presence of these sources inside concrete buildings, to ensure the shielding of the plant's other activities, can cause elevated levels of radon in the facility. This paper presents the planning of measures taken to ensure that there is no significant radiological impact on the workers in the facility.

The Process of Industrialization of the Management of Radioactive Waste: The Example of High Energy Accelerators Waste at CERN

Luisa Ulrici, Yvon Algoet, Luca Bruno, Doris Forkel-Wirth, Francesco Paolo La Torre, Matteo Magistris, Christian Theis, Ralf Trant, Helmut Vincke, Nick Walter

European Organization for Nuclear Research (CERN), Geneva, Switzerland

Abstract. The European Organization for Nuclear Research (CERN) has operated high-energy accelerators for more than 60 years for research in fundamental physics. This activity generates radioactive waste as a result of maintenance, repair, upgrading and dismantling of experiments or accelerators. The radioactive waste produced at CERN mainly consists of metallic low-level-radioactivity components, like machine components, and parts of the neighboring infrastructures such as cables, metallic supports and iron or concrete from the shielding elements. The waste is stored at CERN in a disused accelerator tunnel that has been converted in a facility for temporary storage of radioactive waste. Substantial quantities of radioactive waste are produced during the long shutdowns periods, which take place every three years in order to perform preventive maintenance and to upgrade the accelerators. In view of the required availability of temporary storage during these periods, the limited available space and the legal obligation to deliver radioactive waste to the final repositories in the two CERN Host- States (France and Switzerland), CERN decided to launch a project to ramp up the radioactive waste treatment and disposal. The industrialization of the management of radioactive waste (RW) at CERN is a process that relies on the following main actions: 1) considering the aspects related to radioactive waste disposal already during the design phase of a facility or experiment, e.g. by the development of computing tools allowing the appropriate choice of materials 2) fostering the awareness of the waste producers on the constraints for the temporary storage and disposal of waste, e.g. with the introduction of internal acceptance criteria, 3) developing efficient and reliable techniques for the characterization of radioactive accelerator waste, which may present non- homogeneous distribution of induced radioactivity, 4) constructing a new state-of-the-art radioactive waste treatment center, which will considerably increase the throughput of radioactive waste for disposal while optimizing the disposal costs through separation and volume reduction. The contribution will describe the entire life cycle of radioactive waste at CERN, from the design phase until its disposal, emphasize the new, overall approach and provide details on the new facilities.

NORM Waste in Oil & Gas Industry (Challenges & Solutions)

Mohammad Aref

Weatherford, Abu Dhabi, United Arab Emirates

Abstract. Objectives & Process: Due to growing of worldwide Oil & Gas Production activities, NORM waste volume is growing and became a real challenges to eth industry. The proper handling, transport & Disposal of NORM is a challenges in most part of the world. Oil & Gas field development and increasing of production have increased the NORM waste. The society & Industry facing a major challenges in NORM Waste Management. NORM Waste Management is one of the major environmental challenges being faced equally by all Oil & Gas Production countries. Sustainable NORM Waste Management is a process focusing on reducing & Managing NORM waste from the sources; and it's a way of thinking that profoundly changes our approaches to resources and production. NORM Waste management is something that should be a joint effort between government and industries. Our planet suffered tremendous damage after the industrial revolution when people mass-produced and then mass dumped products. Environmental contamination liabilities pushing for more effectively manage waste to minimize the impact on the environment. Methods: In this presentation I'll try to explain the Effective NORM Waste Management, what's the Challenges, Methods of Disposal & Technology as part of solutions. What's' are the responsibilities of regulatory authority & Industry, how Effective NORM Waste Management can support the legislations compliance and Environment, and explain the Government & Industry Partnership relationship on NORM Waste Management. Conclusion: Highlights of Discussion: NORM Waste Management & Key Factors. What are the Challenges?

Disposal of Norm Waste from Oil and Gas Industry by Underground Injection (Case Study)

Michael Cowie[a], Roman Bilak[b], Steve Wasson[c], Kelly Hatch[c]

[a]Saudi Aramco, Dhahran, Saudi Arabia

[b]Terralog Technologies, Calgary, Canada

[c]NORM Survey of Canada, Alberta, Canada

Abstract. Naturally Occurring Radioactive Material (NORM) is an unwanted by-product of oil & gas production. Saudi Aramco has established a strategy for the management of NORM associated with its operations, an integral part of which is the identification and disposal of NORM waste. This paper details the operational and radiological controls associated with the handling, storage, and disposal by a process of underground injection in a dedicated NORM disposal well.

Waste Management Protocols for Iridium-192 Sources Production Laboratory Used in Cancer Treatment

Maria Elisa C. M. Rostelato[a], Carla Daruich de Souza[a], Daiane C. Barbosa de Souza[a], Carlos A. Zeituni[a], Rodrigo Tiezzi[a], Osvaldo L. da Costa[a], Bruna Teiga Rodrigues[a], João A. Moura[a], Anselmo Feher[a], Anderson Sorgatti[a], Eduardo Santana de Moura[a], José Ronaldo de Oliveira Marques[a], Rafael Melo dos Santos[a], Dib Karam Jr.[b]

[a]Instituto de Pesquisas Energéticas e Nucleares (IPEN/CNEN-SP), São Paulo, SP, Brazil

[b]Universidade de São Paulo, USP Leste., São Paulo, SP, Brazil

Abstract. Introduction: Brachytherapy is a form of treatment that uses radioactive seeds placed in contact or inside the region to be treated, maximizing the radiation dose inside the targeted areas. Iridium-192 is being used in brachytherapy since 1955. It presents emission energy in the "therapy region" (370keV) and is easily produced in a nuclear reactor ($^{191}Ir\ (n, \gamma) \rightarrow {}^{192}Ir$). Wires are an iridium-platinum alloy with 0.36 mm diameter and they can be cut in any needed length. They can be used in several types of cancer. The linear activity is between 1 mCi/cm (37 MBq/cm) and 4 mCi/cm (148 MBq/cm) with variations of 10% in 50 cm maximum. This activity values classified the treatment and low dose rate (0,4 à 2 Gy/h). The propose of this work is to present a waste management system in a cancer treatment radioactive sources production laboratory. Methodology and Results: The solid waste is previously characterized in the analysis phase. The contaminants are already known and they are insignificant due to their fast half- life. The iridium-192 half-life is 74.2 days, classified as very short half-life waste. The waste activity is adds to 8mCi (2.96×10^{8} Bq) per wire. According to the CNEN-NN 6.08 standard, that presents the discharge levels, the limit is 1 $kBq.kg^{-1}$ (2.7×10^{-5} $mCi.kg^{-1}$). The radioactive waste generated during the I^{192} wires production has a weakly activity of 9.7 $GBq.g^{-1}$. According to the standards, this activity is too high to be discarded into the environment. The waste must be managed following the ALARA principal using the R&R (retain e retard) system, that means, temporary storage and posterior discharge. Since every 4 months, maintenance is performed inside the hot cell used for production, the waste must be removed. Using the equation: $A = \frac{L}{\lambda}(1 - e^{-\lambda t})$, the total calculated activity is 1.68 x 10^{16} Bq and 4.8 g mass at the end of each 4 months period. This amount is stored inside a shielding device that has 212.37 cm^3 volume. The waste will take 9.8 years (calculated by $A = A_0(e^{-\lambda t})$) to decay to the discharge levels. To store 30 devices during 10 years, a space with 6,370 cm^3 is necessary. The laboratory has enough space for this storage. Thus, the radioactive waste management can be performed through the R&R (retain and retard) system safely.

Clearance and Declassification of Research Reactor Thetis at Ghent University, a Prime for Belgium

Myriam Monsieurs[a], Isabelle Meirlaen[a], Hubert Thierens[a], Karel Strijckmans[a], Geert Cortenbosch[c], Luc Noynaert[b], Luc Ooms[b], Sven Boden[b]

[a]Ghent University, Ghent, Belgium,

[b]The Belgian Nuclear Research Centre (SCK-CEN), Mol, Belgium

[c]Bel-V, Brussels, Belgium

Abstract. The Thetis research reactor of the Ghent University has been operational from 1967 until 2003. Maximal thermal power capacity was 250 kWth. In 2010 the spent fuel was unloaded. In 2012-2015 the decommissioning of the reactor took place. The final phase of the dismantling of the Thetis facility consisted of the clearance of the reactor pool. In a first step, a complete mapping of the stainless steel liner of the reactor-pool was performed using an In-Situ Counting System (ISOCS) based on Ge gamma-spectrometry (Canberra). The vertical wall of the cylindrical liner was found to be activated above the clearance limit of 0.1 Bq/g Co-60 from the bottom up to 0.5 m height with max value 0.85 Bq/g. The liner of the reactor floor was completely activated with max activity level of 90 Bq/g. Based on these characterisation measurements the liner was removed from the reactor pool up to 2.5 m above floor level and packed for conditional release (melting). Secondly, for characterisation of the concrete with the liner removed again a mapping using the ISOCS system was performed. To obtain information on the activity in depth drilled samples cut in 1.5 cm slices were measured with Ge spectrometry. Only the bottom concrete plate was found to be activated above clearance levels, mainly due to Eu-152 (2.46 MBq total, 0.54 Bq/g). Removal of the activated concrete of the reactor pool structure was not indicated in view of the possible structural instability of the reactor building and contamination risk. Therefore, in-situ physical decay was chosen as the best strategy. This decision lead to a change of the envisioned end-status of the decommissioning: immediate unconditional release could no longer be obtained. Finally, all other rooms within the reactor building could be cleared using a combination of surface contamination mapping with hand-held monitors and performance of wipe tests, measured by Ge gamma-spectrometry and liquid scintillation counting. Except for the concrete reactor pool floor, the reactor building could be unconditionally released. On September 8th 2015 the Belgian Federal Agency for Nuclear Control agreed with the the declassification of the reactor Thetis, a prime for Belgium. The concrete reactor-pool floor will be included into the existing class II license for the site, up to its unconditional release expected in 2047.

Methods used for Clearance of the Rooms of Former Research Reactor Thetis at Ghent University, Belgium

Myriam Monsieurs[a], Isabelle Meirlalen[a], Hubert Thierens[a], Geert Cortenbosch[c], Luc Ooms[b], Sven Boden[b]

[a]Ghent University, Gent, Belgium

[b]The Belgian Nuclear Research Centre SCK-CEN, Mol, Belgium

[c]Bel-V, Brussels, Belgium

Abstract. The Thetis research reactor of the Ghent University has been operational from 1967 until the end of 2003. In 2010 the spent fuel was unloaded. In 2012-2014 the dismantling of the reactor took place. The final phase of the dismantling consisted of the clearance of the rooms inside the reactor building. First, the rooms were divided into 4 categories determining their likelihood to be contaminated based on a combination of historical data and contamination measurements performed before the dismantling of the reactor. Depending on the category, the number of samples (150 cm^2 wipe tests) to be performed per square metre was determined. However, during the dismantling only very few and very small contaminations were found, even in the highest category rooms (cat 1). Therefore, the authorities agreed to adjust the categories of some of the rooms in order to reduce the number of wipe tests. Each wipe test was analysed by gamma spectrometry using a high purity Germanium detector in low background circumstances and also by liquid scintillation counting in case of possible contamination with pure beta-emitters (^3H, ^{14}C). A room was considered to be unconditionally cleared when no sample showed an activity above the clearance level of 0.4 Bq/cm^2 on both techniques. Using this method, no contaminations were found. However, during a check-up measurement by the authorized inspection organization Bel-V, one spot contamination could still be determined. Both the radioisotopes present in the spot, as well as its location, determined this had to be a historical contamination in a laboratory that was originally considered to be cat1. It was determined by all parties that the adjustment of cat1 rooms had been too rash. Therefore, all original cat 1 rooms were completely re-measured. This time 100 % direct measurement of the surfaces (to rule out missing spot contaminations) was performed using 30 second integrate measurements by Berthold contamination detectors on the flat surfaces, and gamma spectrometry using the NaI probe Inspector 1000 for the interior surface of the cylindrical pits used to store radioactive waste. Since no additional contaminations were found, as a Belgian precedent, at September 8 2015, the Belgian authorities consented to a declassification of the dismantled reactor.

Removal of Radiocesium Aqueous Solution using Activated Carbon Modified with Anionic Surfactant

Michael Olatunji, Mayeen Khandaker, Yusoff Amin, Ekramul Mahmud

University of Malaya, Kuala Lumpur, Malaysia

Abstract. The increasing discharge of heavy metals and radioactive material waste solutions into the natural environment is a matter of major concern to the environmental scientists across the globe. This is due to the bioaccumulation of these contaminants in the body tissues when ingested, leading to various complicated health effects even at trace amount. Cesium radioisotopes are major products of nuclear fission and form common constituents of liquid wastes from medicine, mining and milling industries and it is very toxic, non-biodegradable and easily assimilated in body tissue. Hence, it is important to reduce the volume of waste solution from different industrial processes before disposal. However, the high cost of materials involved in synthesizing adsorptive materials of high sorption capacity has impeded the use of many sorbents reported in the literature particularly in the developing countries. Among the materials of low cost which find applications in wastewater treatments is activated carbon. Activated carbons are produced from biomass wastes (wood, coconut shell, crab shell, etc) which are abundantly free and safe in the environment. In this study, we investigated the effect of activated carbon modification with anionic surfactant on its sorption capacity for ^{137}Cs. The effects of temperature, contact time, sorbent loading, solution pH, competing ions and concentration of the contaminants were studied to understand the mechanism of the adsorption process. The as-prepared modified activated carbon was characterized by FESEM-EDAX, FTIR, XRD, XRP and TGA-DSC for chemical structure, composition, morphology and thermal stability. As an adsorbent, activated carbon proved to yield enhanced sorption capacity when modified with anionic surfactant.

Norwegian-Russian Cooperation in Nuclear Legacy Regulation: Continuing Experience and Lessons

Malgorzata K. Sneve[a], Natalya Shandala [b], Alexey Titov [b], Vladimir Seregin [b], Sergey Kiselev [b]

[a]Norwegian Radiation Protection Authority, N-1332 Østerås, Norway

[b]Federal Medical Biophysical Center of FMBA, Russian Federation

Abstract. Safe management of nuclear legacies arising from past activities is a critical issue in maintaining confidence in the continuing and future use of radioactive materials. Effective and efficient regulatory supervision of nuclear legacy management is a critical part of that process. The Norwegian Radiation Protection Authority plays an active role in bi-lateral regulatory cooperation projects with parallel authorities in the Russian Federation, as part of Norway's plan of Action to improve radiation and nuclear safety. Based on this experience and by reference to specific legacy sites and facilities, this paper provides an overview of the substantial progress made in remediation at the Site of Temporary for spent fuel and radioactive waste at Andreeva Bay and presents radiation protection the issues arising from the experience and lessons learned in this work. It is suggested that this experience can be useful in the further development of international recommendations, standards and guidance in relation to remediation of nuclear legacies.

KEYWORDS: legacy supervision; regulation; optimization; communication; international cooperation.

Radiation Protection Dosimetry (2017), Vol. 173, No. 1-3, pp. 73–79

doi:10.1093/rpd/ncw340

Rationale for a Comprehensive Assessment of Radio-Ecological Safety of Near Surface Radioactive Waste Storage Facilities

Alexander Samoylov[a], Nataliya Shandala[a], Igor Korenkov[a], Tatyana Lashchenova[a], Vladimir Romanov[b]

[a]State Research Center – Burnasyan Federal Medical Biophysical Center of Federal Medical Biological Agency, Moscow, Russia

[b]Federal Medical Biological Agency, Moscow, Russia

Abstract. According to information provided by the State control and account of radioactive material, today Russia has more than 500 million m3 of liquid waste and more than 180 million tones of solid ones. At that, 99% are contributed by Low Level Waste (LLW) and Intermediate Level Waste (ILW), which are located at the premises of about 1400 near surface storage facilities. These storage facilities were built 30-50 years ago and over this time, the protective engineering barriers were subjected to degradation due to temperature fluctuations of atmospheric precipitation, resulting in increased filtration properties of radioactive waste (RW), leaching of radionuclides from the RW matrix, chemical corrosion of the cement matrix, leaching and migration of radionuclides into the environment. The goal of this paper is to study changing physical and mechanical, hydro-geological, geo- chemical and radiation parameters of shielding barriers of the storage facilities, on the basis of the interaction assessment of "the storage facility – environment" radio-ecological system. According to many-year studies, geo-chemical anomalies in the nearest area of the storage facilities and the pH value serve as indicators of shielding barriers and areas of radionuclide migration into the environment. The paper demonstrates the changing filtrate composition of the nearest area of the storage facilities depending on the duration of Na+, K+, Ca+2, NO3-, SO4-2 ion operation. We used the pH values are an indicator of the degradation process of the concrete constructions of the storage facilities.

Assessing the Environmental Impact of Man-made Radioactive Contamination at the Andreeva Bay Site for Temporary Storage of Spent Nuclear Fuel and Radioactive Waste

Natalia Shandala[a], Vladimir Seregin[a], Anna Filonova[a], Aleksey Titov[a], Vladimir Shlygin[a], Malgozhata Sneve[b], Graham Smyth[b]

[a]SRC – Burnasyan Federal Medical Biophysical Center, Moscow, Russia

[b]Norwegian Radiation Protection Authority (NRPA), Oslo, Norway

Abstract. Within international cooperation between FMBA and NRPA, the impact of man-made radioactive contamination of the site for temporary storage (STS) of spent nuclear fuel and radioactive waste at Andreeva Bay on the biota was assessed. The study covered three areas of the site, specified by different pathways of exposure to biota: external gamma radiation from buildings and constructions, exposure from radionuclides in soil. Radiation exposure was predicted for representatives of flora and fauna under compliance with the criteria for the STS area remediation. The sufficiency of the previously developed remediation criteria has been assessed in terms of the environmental protection. The performed assessment demonstrates that for any selected option of future use of the STS area, in compliance with the remediation criteria, negative impact of radiation factor on the representative organisms within the supervision area is not expected. Nevertheless, negative effects on some populations of representative organisms (lemmings, birch, motley grass) can arise.

Radiation Situation around the Shipyards Involved in Decommissioning/Dismantlement of Nuclear Submarines

Nataliya Shandala, Alexey Titov, Maria Semenova, Vladimir Seregin, Anna Filonova

SRC - Burnasyan Federal Medical Biophysical Center, Moscow, Russia

Abstract. After the end of "Cold War", about 200 nuclear submarines were withdrawn from the Navy of the Russian Federation. The operation life of the majority of submarines has been terminated by that time. There was the spent nuclear fuel on-board; therefore, these submarines posed a serious threat to the environment during the storage afloat. In Russia, some shipyards, such as the “Nerpa” shipyard in the Northwest Russia and the “Zvezda” plant in the Far Eastern region, were involved in the nuclear submarine decommissioning and dismantlement (D&D) operations. Up to now, the D&D programme of the decommissioned nuclear submarines is almost completed. This paper describes the results of a study of the radiation situation prevailing around these shipyards, after the completion of the decommissioning operations. The study covered the site and marine environment in the vicinity of the “Zvezda” and “Nerpa” shipyards. The subjects of the study included: soil, plants, sea water, seaweeds, bottom sediments, local foodstuffs (milk, potato), wild berries and mushrooms. It was shown that the impact of D&D operations on the population and environment was negligible

The study of the Ground Water Contamination. The Study of the Environmental Conditions of the Region during Remediation of the Andreeva Bay STS

Natalya Shandala[a], Vladimir Seregin[a], Sergey Kiselev[a], Stanislav Geras`kin[b], Malgorzhata Sneve[c], Graham Smith[c]

[a]SRC - Burnasyan Federal Medical Biophysical Center, Moscow, Russia

[b]Russian Institute of Agricultural Radiology and Agroecology RAAS, Obninsk, Russia

[c]Norwegian Radiation Protection Authority, Oslo, Norway

Abstract. Within the FMBA-NRPA international cooperation, the comprehensive assessment of ecosystem conditions is being carried out at the Andreeva Bay site for temporary storage of spent nuclear fuel and radioactive waste. This paper includes the results of the study of ground water contamination with radionuclides and chemicals. In some cases, concentrations of heavy metals in water of the observation wells exceed the established norms. The highest specific activities of 137Cs and 90Sr are 154 Bq/L and 890 Bq/L, respectively. Bio-testing of ground water being collected from the inspected area under the monitored conditions has been carried out using the allium-test. Cyto- and genotoxic effects of the analyzed samples on cells of the onion root meristem have been found. This helped to identify some sampling points, water from which has increased mutagenic activity. The dependence of the observed biological effects on the level of ground water contamination with radionuclides and heavy metals has been analyzed. The obtained information will be used for the purpose of monitoring optimization and decision making on organization of remedial activities to reduce negative environmental impact of the site.

The Status of Occupational Exposure Source Term Measurement with In-situ Gamma-ray Spectrometry for NPPs in China

Qinjian Cao[a,b], Xueqi Chang[b], Liye Liu[b], Wanchun Xiong[b], Yunshi Xiao[b], Sanqiang Xia[b]

[a]China Institute of Atomic Energy, Beijing, China

[b]China Institute for Radiation Protection, Taiyuan, China

Abstract. The reduction of collective dose is a constant challenge for all nuclear power plant operators. More than eighty percent of the occupational collective doses are received during the outage for PWRs due to activated corrosion products deposited on the out- of-core surface. An occupational exposure source term characterization program was performed by CIRP and most of NPPs in China. It is very important to determine radioisotope concentrations in contaminated pipes for the reduction of collective dose. The radiation environment can be analyzed in a number of ways, among which the non-destructive measuring method would be a good choice for the operating NPPs. A radiological characterization method based on in-situ gamma spectrometry had been developed. The in-situ gamma spectrometry measuring system and the technique of virtual efficiency Monte Carlo calibration were constructed. The measuring system consists of HPGe detector, collimator, MCA and vehicle. The in situ gamma spectrometry was collected by the measuring system. The surface activity of source term and dose rate can be calculated with the Monte Carlo calibration method. As an addition to the gamma-spectrometry, contacted gamma-ray dose rates were also measured. The comparisons between calculation and measuring dose rates showed that the relative deviations were mostly within 40%. From 2005, about 18 measurement campaigns has been performed in China. The detected nuclides include Cr-51, Mn-54, Fe-59, Co-58, Co-60, Zr-95, Nb-95, Ag-110m, Sb-124.The primary contributors to out-of-core radiation fields in PWRs have been identified as Co-58 and Co-60.

Study on spraying water soluble resin for Fukushima Dai-ichi nuclear power plant accident

Qiong Zhang, Ruiping Guo, Bo Wang, Liang Wang, Fudong Liu, Xiaoqiu Chen

Nuclear and Radiation Safety Center, Beijing, China

Abstract. The damage caused by the earthquake, tsunami and the release of radioactive material was summarized after the Fukushima nuclear accident in Japan. Meanwhile, the policy, regulations, programs and progresses of the environmental remediation on off-field of nuclear accident in Fukushima were introduced to propose the problems about environmental remediation on off-side of radioactive pollutants after nuclear accident worth to discuss and pay attention to in the future of China.

Achievements by the NORM and Legacy Sites Working Group of the IAEA MODARIA Project

Rodolfo Avila

Facilia, Stockholm, Sweden

Abstract. The general aim of the IAEA MODARIA Programme was to improve capabilities in the field of environmental radiation dose assessment by means of acquisition of improved data for models, model testing and comparison, reaching consensus on modelling philosophies, approaches and parameter values, development of improved methods and exchange of information. The WG 3 of the project, Norm and Legacy Sites, was dedicated to improve, further develop and testing models and approaches for Safety Assessment in support to remediation of legacy sites contaminated with NORM and artificial radionuclides. The work of the group consisted of three main tasks. In Task 1 a Framework and a Methodology for Safety Assessment in support to remediation of legacy sites contaminated with NORM and artificial radionuclides was developed. The framework and methodology are based on results from a similar work previously done within the IAEA EMRAS II Programme and the framework for Safety Assessment of Near Surface Disposal Facilities developed within the IAEA ISAM Programme. The proposed Framework and Methodology provide guidance on how to perform safety assessment in an integrated systematic way; in order to support decision making for remediation of contaminated legacy sites. In Task 2 the NORMALYSA software package was developed and tested. NORMALYSA is a tool that can be used for performing screening dose assessments considering all relevant pathways from Contamination Sources to Receptors: atmospheric dispersion, groundwater transport and surface runoff. The NORMALYSA software supports a broad range of sources (uranium tailings, contaminated lands, waste rocks, etc.) and potentially impacted receptor environments, such as crop lands, pasture lands, forests, garden plots, fruit lands, lakes, rivers, wells, sea coastal basins and buildings. In Task 3 there Cases Studies were developed based on existing NORM legacy sites: a site for processing of phosphate ore for the production of dicalciumphosphate (contamination with Radium) in Tessenderlo, Belgien, a tailings repository in Bellezane, France and a legacy site with tailings and contaminated buildings in Dnirpodzerchynsk, Ukraine. These three case studies were used for testing the methodology and the models.

NORMALYSA – A Tool for Risk Assessment to Support Remediation of Legacy Sites Contaminated with NORM and Artificial Radionuclides

Rodolfo Avila

Facilia, Stockholm, Sweden

Abstract. The general aim of the IAEA MODARIA Programme was to improve capabilities in the field of environmental radiation dose assessment by means of acquisition of improved data for models, model testing and comparison, reaching consensus on modelling philosophies, approaches and parameter values, development of improved methods and exchange of information. The WG 3 of the project, Norm and Legacy Sites, was dedicated to improve, further develop and testing models and approaches for Safety Assessment in support to remediation of legacy sites contaminated with NORM and artificial radionuclides. One of the tasks of this group was to develop a software package (NORMALYSA) to assess doses to the public from environments contaminated with radionuclides as result of past activities, such as uranium mining and milling. NORMALYSA consists of a model simulator and a library of models for making predictions of the transport of radionuclides from Sources to Environmental Receptors and performing Dose Calculations. The sources of contamination could be uranium tailings, covered and uncovered, waste rock piles and contaminated lands. Different pathways of radionuclide transport are supported: atmospheric dispersion, groundwater transport and surface runoff. The model library includes models for a broad range of receptor environments, such as: crop lands, pasture lands, forests, garden plots, fruit lands, lakes, rivers, wells, sea coastal basins and buildings. The software package supports calculation of doses by all important exposure pathways: external irradiation from the cloud and the ground, inhalation of dust and gases (Radon) indoors and outdoors, ingestion of water and food. The model simulator supports both deterministic and probabilistic simulations, as well as sensitivity analyses. NORMALYSA can used for integrated screening assessments covering all potential exposure pathways and exposure conditions with a minimum of data requirements. The results from these assessments can be used as a basis for making decisions about the need for remediation of the sites and for identifying areas were more detailed modelling and further site investigations are required.

Information Management System Supporting a Multiple Property Survey Program with Legacy Radioactive Contamination

Ron Stager[a], Douglas Chambers[a], Gerd Wiatzka[a], Monica Dupre[b], Micah Callough[b], John Benson[c], Erwin Santiago[c], Walter van Veen[c]

[a]Ron Stager, Arcadis Canada Inc., 121 Granton Drive, Suite 12, Richmond Hill, ON L4B 3N4, Canada

[b]ARCADIS-US, Highlands Ranch

[c]Port Hope Area Initiative Management Office Canadian Nuclear Laboratories

Abstract. The Port Hope Area Initiative (PHAI) is a project mandated and funded by the Government of Canada to remediate properties with legacy low level radioactive waste (LLRW) contamination in the Town of Port Hope, Ontario. Radiological characterization of exterior gamma radiation, interior gamma radiation, contaminated surfaces and soil investigation on some 4,800 properties in Port Hope are required. The management and use of large amounts of data from the surveys is a significant task and is critical to the success of the project. A large amount of information is generated through the surveys, including: scheduling individual field visits to the properties, capture of field data, photos, comments, laboratory sample tracking, QA/QC, report generation and internal and client project reporting at regular schedules. Field data collection of exterior gamma radiation surveys included direct and interactive loading of data into the Information Management (IM) system. Screening soil measurements including X-ray fluorescence (XRF) for uranium and arsenic soil concentrations from all 20 cm increments from each borehole in addition to down- hole gamma radiation measurements which were used in documentation and interpretation of the property condition. Electronic chain of custody (eCOC) forms were used to facilitate tracking, coding consistency and completeness of the soil and swipe samples from the field office to the external laboratory to the IM database. Automated completion of property report sections allowed the sections to be carefully reviewed and supported the development of the individual property reports. Web-mapping tools were used to track and display temporal progress of various tasks ranging from property scheduling to property status and allowed consideration of spatial associations of contamination levels to the client and the project team. The IM system facilitated the management and integrity of the large amounts of information collected and provided a consistent application and traceable execution for this project.

***KEYWORDS:** low level radioactive waste; radiological survey; information management.*

Radiation Protection Dosimetry (2017), Vol. 173, No. 1-3, pp. 138–144

doi:10.1093/rpd/ncw353

Uranyl Ions Adsorption to Na-GMZ and Interactions with FA Adsorption: experiments and modelling

Yuanlv Ye, Liang Wang, Fudong Liu, Yahua Qiao

Nuclear and Radiation Safety Center, Beijing, China

Abstract. In the environment, an important factor of radionuclide mobility is their interaction with mineral–water interfaces. To predict radionuclide mobility, it is necessary to understand fundamental processes such as surface precipitation and surface complexation. Studies of uranium sorption onto mineral surfaces have great practical importance for risk assessment. In addition to the effects for sorption, NOM may influence U(VI) distribution via some other mechanisms. In this work, data for the interactions between fulvic acid (FA) with uranyl ions at the surface of Na-GMZ are presented. U adsorption to Na-GMZ in the presence of FA can be well predicted with the SCD model (surface and complex distribution). According to the model calculations, the nature of the interactions between FA and U at Na-GMZ surface is mainly surface complex.

Occupational Dose Evaluation during Decommissioning of a Radiological Facility

Zamazizi Dlamini, Frik Beeslaar

South African Nuclear Energy Corporation, Pretoria, South Africa

Abstract. Decommissioning is a regulatory requirement during the planning of a radiological facility. Decommissioning of a radiological facility involves complete dismantling of technologies and demolishing of the building to a point where radiation protection measures are no longer a requirement. A prospective hazard assessment was done for a facility which was originally used for chemical cleaning of large process components and later as a decontamination facility; where a projected occupational dose was calculated. Internal exposure was monitored using in-vitro method of urine sample and calculated using results from smear sampling. External exposure was monitored by area dose rate surveillance in conjunction with TLDs. The internal occupational dose for the highest exposed worker was found to be within the dietary intake levels. External exposures as monitored with TLD was below detection levels. The external and internal dose was compared to the projected doses. It was found that external (as calculated with area monitoring) agreed with the projected doses. However the internal exposures was higher than the projected doses.

The Proceedings of the 14th International Congress of the International Radiation Protection Association
Volume 5 of 5

Area 12: Societies

Argentine Radiation Protection Society (SAR): 50 years Straightening Radiation Protection

Ana María Bomben

Argentine Radiation Protection Society, Buenos Aires, Argentina

Abstract. In 2016, the Argentine Radiation Protection Society (SAR) celebrates its 50th anniversary. Five decades ago a group of pioneers of radiation protection decided, with great vision and tremendous enthusiasm, lay the foundation for the SAR. In this initial group they were adding radiation protection colleagues and related professions imbued with the purposes of society. The objectives of SAR are: a) to promote the execution of works and exchange of expertise in radiation protection and related issues, b) to promote awareness of radiation protection principles in regard to the existence and use of radioactive material and sources of radiation and c) to promote radiation protection as a professional expertise and to contribute to its progress. To fulfil these foundational purposes SAR carried out and continues to do every year many activities: organizing and promoting conferences, symposia, congresses, courses and other events on radiation protection and supporting the active participation of Argentine professionals and technicians in these events as well as the international exchange of knowledge on the matter. One of the most important milestones in relation to the exchange of expertise in radiation protection was the International Congress IRPA 12, organized by SAR, in October 2008, wherein 1,500 professionals from 89 countries shared their knowledge and information within the framework of the international congress. More recently, in April 2015, SAR organized the X Latinamerican IRPA Regional Congress where colleagues from the region and other countries presented the state of the art in radiation protection. In this paper SAR will present the activities to promote radiation protection: its role in education and training by increasing the number of courses covering a wider spectrum of radiation protection topics; the promotion of radiation protection in Latin America, reaching professionals in their own language, by translating into Spanish relevant radiation protection documents; and the results of the increasing relations with scientific associations in Argentina and in the region as a founder member of the Latin American and Caribbean Federation of Radiation Protection Societies (FRALC). With all these activities and the plan for the next decade, SAR continues sowing in a fertile soil to harvest a significant improvement in all radiation protection fields.

The History of the South African Radiation Protection Society

Christoph Trauernicht

University of Cape Town, Cape Town, South Africa

Abstract. The South African Association of Medical Physicists was established in Cape Town on 2 February 1960, making it one of the older medical physics associations in the world. In 1968 the association was expanded the include health physicists and the South African Association of Physicists in Medicine and Biology was born. The SAAPMB contains three societies, namely the South African Medical Physics Society (SAMPS), the Radiobiological Society (SARS), as well as the South African Radiation Protection Society (SARPS). SARPS was established in 1970 under the umbrella of the SAAPMB. SARPS's emphasis is to promote Radiation Protection in Health Care. The Society seeks to reduce the radiation dose received by the public, workers and patients by promoting awareness of the radiation protection procedures and techniques. This is achieved through congresses, lectures and ongoing training among its members. SARPS has an elected council of four members and currently has ~ 75 members.

Belgian Society for Radiation Protection

Tom Clarijs, Patrick Smeesters, Claire Stievenart, Andrzej Polak, Pascal Froment

Belgian Society for Radiation Protection, Brussels, Belgium

Abstract. The Belgian Society for Radiation Protection was founded in 1963 to study and provide independent advices on scientific issues regarding the individual and collective protection against ionising radiation. It aims at promoting collaboration between the different scientific disciplines, and to facilitate the participation of Belgium to international organisations and events related to ionising radiation. An important focus of the Society is communication about radiation protection, where the main activities involve scientific lectures, debates, education and training. The periodic publications such as newsletters and the Annals are addressing the latest issues in radiation protection. Future challenges include the creation of a new website for improving the member-interaction and outreach. The Society aims at increasing and follow-up of young future members and gender equilibrium and facilitating their involvement in scientific meetings, as well as the involvement of people outside the radiation protection field.