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The analysis of management of radioactive waste arisen from steam generator's dismantling from radiological and economical point of view

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A B S T R A C T

As a result of the application of decontamination and dismantling techniques the radioactive waste of various form and activity content is generated. These hazardous materials are mostly in solid form (primary waste) but also in liquid form (secondary waste) which has to be fixed in a solid matrix. From this reason the complex analysis of the dismantling process has to take into account also this phenomenon. The paper analyses the expected radioactive waste arisen from the dismantling of steam generator used in Slovak nuclear power plant V1 in Jaslovské Bohunice from the perspective of waste streams and evaluates the expected exposure of the workers. Moreover, the basic cost analysis is presented in order to analyse the waste management scenarios also from economical point of view. The results show that the maximal collective effective dose is at the order of ones man-mSv which is below the respective exposure limit. In the case of the relative costs of waste management scenarios, the influence of the application of decontamination techniques is significant. The calculation methodology applied in the case of assessment of external exposure reflects the expected changes of the input data, for instance the amount, activity and nuclide composition of secondary waste.

1. Introduction

The dismantling of steam generator (SG) as a large-scale primary circuit component used in nuclear power plants (NPP) is one of the most important tasks of the decommissioning process (together with the dismantling of activated components, e.g. reactor pressure vessel and reactor internals). The complex analysis of this process involves among the optimisation of radiation protection during dismantling also the analysis of the material management scenarios as well as the analysis from economical point of view.

Currently in Slovak Republic the V1 NPP in Jaslovské Bohunice (Russian type of pressurised water reactor – PWR) is in the second and final decommissioning stage with planned duration from 2015 to 2025 (National Nuclear Fund, 2015). Within this stage, the large components will be cut in-situ and the fragmented parts will be stored and/or conditioned and disposed in the repository (Nuclear and Decommissioning Company). Since the detailed analysis of the SG's dismantling from the perspective of exposure and radiation impact on the public and environment was published in (Horňáček and Nečas, 2016), this paper deals with the next steps, i.e. the management of resulting radioactive waste (treatment, conditioning, transport and final disposal) as well as cost analysis.

This set of actions is strongly influenced by the application of decontamination techniques which can decrease the class of radioactive waste and lower the exposure when primary waste is being treated and conditioned. On the other hand, the secondary waste is also a subject of radioactive waste management. The effectiveness of the decontamination can be quantitatively characterised by decontamination factor (DF) which can be defined as the ratio of the measurements' results (of activity or dose rate) before and after decontamination (Bregani, 1995a), (Taboas et al., 2004).

Currently there is practical experience of the **pre-dismantling decontamination** of the primary circuit components in NPPs with PWR. Some of the examples are listed below:

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Abbreviations

SG	steam generator
NPP	nuclear power plant
PWR	pressurised water reactor
DF	decontamination factor
ENV	environment
VLLW	very low level waste
LLW	low level waste
IRAS	International Radiation Accident Scale
ALARA	As Low As Reasonably Achievable
FCC	fibre-concrete container
NRWR	National Radioactive Waste Repository in Mochovce
ISDC	International Structure for Decommissioning Costing
IAEA	International Atomic Energy Agency
HET	heat exchange tubes

- German NPP Unterweser (1410 MW) – the total DF of 158 of the primary circuit, **DF of 147** of SG's heat exchange tubes (HET) were achieved in 2012 (Topf et al., 2013).
- German NPP Stade (672 MW) – total DF of more than 58, **DF of more than 160** of SG's heat exchange tubes were achieved (Stiepani, 2011).
- German NPP Obrigheim (357 MW) – total DF of 625, **DF of 1409** of SG's heat exchange tubes (Topf, 2007).

In the case of **post-dismantling decontamination** of SGs, the following techniques can be applied (Bregani, 1995b):

- Electrochemical decontamination
    - Application "in-tank" – most common, the item is immersed in a tank filled with a suitable electrolyte.
    - Application "in-situ" – the use of mobile devices.
- Jetting decontamination techniques based on the impingement of either liquid or solid media (or a liquid-solid mixture). These techniques can be realised by:
    - High pressure water – the **DFs are about 50** or more.
    - Abrasives – the abrasive medium is propelled by jet of air (dry blasting) or water (wet blasting). The **DFs of 13 to 125** can be achieved in the case of abrasives in suspension.
- Ultrasonic techniques – are commonly based on the immersion of the component in a tank containing liquid (water or chemical agents) with application of ultrasonic waves to the walls of the tank. The **DFs** are strongly dependent on the physical and chemical properties of the liquid; usually they are **about 100.**

Note: In the case of metallic radioactive waste also the **melting** can be considered as decontamination technique (given the fact that volatile isotopes like ^{137}Cs are distributed to slag or aerosols since the isotopes like ^{60}Co will remain in ingot). Since in this studied case the most dominant nuclides are chemically similar to iron (^{60}Co, ^{63}Ni and also ^{55}Fe – Table 1), the melting as decontamination technique is not suitable. For the illustration, with the input parameters described in the next chapter and the distribution coefficients stated in (Anigstein et al., 2003) it can be seen that the DF of about 1.05 could be achieved. On the other hand, melting can be also considered as conditioning technique since it homogenises the volume. The detailed analysis regarding the melting of metallic waste in Slovakia can be found in (Slimák and Nečas, 2016).

Table 1
Nuclide vector considered in the analyses.

Nuclide	T1/2 [years]	Relative share		
		Year		
		2015	2020	2025
^{14}C	5.73E+03	5.19E-05	8.62E-05	1.14E-04
^{41}Ca	1.03E+05	1.99E-03	3.30E-03	4.37E-03
^{54}Mn	8.55E-01	1.26E-03	3.62E-05	8.31E-07
^{55}Fe	2.73E+00	3.76E-01	1.75E-01	6.53E-02
^{57}Co	7.45E-01	9.21E-06	1.46E-07	1.84E-09
^{59}Ni	7.60E+04	4.46E-03	7.42E-03	9.82E-03
^{60}Co	5.27E+00	2.33E-01	2.00E-01	1.37E-01
^{63}Ni	1.00E+02	3.44E-01	5.52E-01	7.06E-01
^{65}Zn	6.68E-01	2.19E-05	2.04E-07	1.51E-09
^{79}Se	6.50E+04	8.52E-04	1.42E-03	1.87E-03
^{90}Sr	2.87E+01	1.69E-03	2.50E-03	2.93E-03
^{93}Mo	3.50E+03	1.33E-02	2.21E-02	2.93E-02
^{93}Zr	1.50E+06	1.28E-03	2.12E-03	2.81E-03
^{94}Nb	2.03E+04	1.61E-03	2.68E-03	3.54E-03
^{107}Pd	6.50E+06	1.42E-03	2.36E-03	3.13E-03
^{110m}Ag	6.82E-01	1.93E-04	2.00E-06	1.65E-08
^{125}Sb	2.76E+00	1.15E-03	5.44E-04	2.05E-04
^{126}Sn	1.00E+05	1.56E-03	2.60E-03	3.44E-03
^{129}I	1.60E+07	4.27E-05	7.10E-05	9.39E-05
^{135}Cs	2.30E+06	3.55E-03	5.90E-03	7.81E-03
^{137}Cs	3.02E+01	1.51E-03	2.24E-03	2.65E-03
^{144}Ce	7.81E-01	7.54E-05	1.48E-06	2.32E-08
^{151}Sm	9.00E+01	5.29E-03	8.47E-03	1.08E-02
^{238}Pu	8.77E+01	3.44E-05	5.49E-05	6.99E-05
^{239}Pu	2.41E+04	2.65E-05	4.41E-05	5.84E-05
^{240}Pu	6.56E+03	2.65E-05	4.41E-05	5.83E-05
^{241}Pu	1.44E+01	6.25E-03	8.17E-03	8.50E-03
^{241}Am	4.32E+02	1.23E-04	2.02E-04	2.65E-04
^{244}Cm	1.81E+01	3.24E-05	4.45E-05	4.86E-05

2. Methods

2.1. Input data and general preconditions

The steam generator was used in each of the 6 loops of the primary circuit within one unit and consists of the following main parts:

- SG casing – total mass 113.4 tonnes, part of the secondary circuit.
- Heat exchange tubes – 5536 U-tubes, total mass 34.7 tonnes, contaminated part of the primary circuit.
- 2 collectors – total mass of both collectors 25.4 tonnes, contaminated part of the primary circuit.

The total length of SG is 11.8 m, the outer diameter is approx. 3.4 m.

From the construction point of view the vessel is made of carbon steel 22 K; the collector material as well as the heat exchanging tube material is titanium stabilized austenitic steel with 0.08% of carbon, 18% of chromium, 10% of nickel and less than 1% of titanium (International Atomic Energy Agency, 1997).

The considered **nuclide vector** was derived from the radiological characterisation, consisting of 29 radionuclides and is depicted in Table 1. The nuclide vector is based on the radiochemical analysis of samples from the whole primary circuit of both units of NPP V1. The values represent relative share of the activity of each nuclide on the total activity.

The total activity of the SG components including alpha and beta emitters without application of pre-dismantling decontamination is shown in Table 2: