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Practical Approach for Assessment of End-State Radiological Criteria for Remediation of Radioactively Contaminated Sites

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An approach is described for assessment of the end state radiological criteria for remediation of radioactively contaminated sites. The target criteria are set in a form of prospective effective doses for members of the population who are subject to the higher exposures (representative persons). Brief review of international best practice in setting risk based remedial criteria is presented. The site-specific release criteria for activity concentrations in released material (e.g., Bq/g of soil) are derived using tabulated values of radionuclide activity from IAEA Safety Guide RS-G-1.7 (corresponding to the effective dose of 10 μSv/a). These tabulated values are scaled with the relevant target dose criteria for remediation of the specific site. Applicability and limitations (e.g., with regard to volume of released material) of proposed approach are discussed. The procedure for incorporating complimentary site-specific scenarios is described. The article further illustrates the approach by application of the methodology to the specific radioactively contaminated site (i.e., radioactive waste storage site with clean-up wastes of Chernobyl origin situated in Kiev Region). The proposed approach is generally applicable to a wide range of similar problems.

Keywords: safety assessment, remediation, end-state criteria, Chernobyl accident

Early radioactive waste management practices (that have not complied to modern safety standards) and nuclear accidents have created worldwide numerous radioactively contaminated legacies ranging in scale from individual facilities and/or sites to large contaminated areas (e.g., areas contaminated by Chernobyl and Fukushima accidents), and remedial efforts are undertaken currently in many countries in order to bring these sites to condition that is safe for humans and the environment [1—5].

Selection of remedial option and developing a remedial design for radioactively contaminated site is a complex process that usually weights safety, technical, economic, and social factors [5—7]. One of key elements of remedial design are end-state radiological criteria for the remediated site. Such radiological criteria provide safety goals that need to be achieved as the result of remedial works, and may eventually determine the technological requirements, extent of remedial works and the amount of the retrieved waste material [3, 5].

In this article we describe practical approach for assessment of the end state radiological criteria for activity concentrations in the material which remains on the remediation site (e.g., soil) to be achieved upon completion of remedial works. The presented approach was developed in the project "Remediation of Radioactive Waste Storage Sites Resulting from the Chernobyl Nuclear Power Plant Accident and Situated Outside the Exclusion Zone" (Project U4.01/12D), which was implemented through the Instrument for Nuclear Safety Cooperation (INSC) Programme by the European Commission, DG DEVCO [8, 9].

No specific guidance on the procedure for setting end-state remedial criteria are available currently in the Ukraine. Therefore, the proposed methodology is based on relevant International Atomic Energy Agency (IAEA) safety standards and guidance documents, and relies upon review of international best practices. The article further illustrates the proposed approach by application to the specific radioactively contaminated site that was selected as a "Pilot Facility" in the Project U4.01/12D for developing the "standard" remedial design for the radioactive waste storage sites considered in this project.

Risk-based approach for developing the end state criteria for radioactively contaminated sites

General framework. The general approach for developing the end-state criteria for radioactively contaminated sites followed in this study is described in the IAEA Safety Guide No. WS-G-5.1 [10]. This safety standard recommends that target criteria are set in a form of prospective effective doses for members of the population who are subject to the higher exposures (representative persons). The site-specific release criteria for activity concentrations (e.g., Bq/g of soil) can be then back-calculated from doses through evaluation of potential radiological consequences through all relevant exposure pathways. The safety standard further recommends a dose constraint for the released site of less than 300 μSv per year. A limit below which further dose reduction measures are unlikely to be warranted is 10 μSv per year. The zone between 10 and 300 μSv per year is considered to be a "zone of optimization" (see [10, Fig. 1]).

It should be noted that the IAEA Safety Guide No. WS-G-5.1 considers release of sites in the context of "planned exposure" situations. In case of "existing exposure" situations, remediation process relies on optimization principles in a generally similar way. However, as recommended by IAEA GRS Part 3 [6] radiation protection and safety of population is ensured in this last case by establishing more flexible "reference levels" (instead of "dose constraints") that are essentially dependent on the feasibility, costs, and other relevant aspects of controlling the "existing exposure" situation. In case of Project U4.01/12D, the reference level of 300 μSv/a was coordinated by contractor with the Ukrainian regulatory authority as the relevant dose criteria.

Eventually, the respective dose end-state criteria shall be considered in comparison to background contamination level (e.g., as doses from contaminated site *exceeding* the background doses to representative persons).

Review of international experiences in setting risk-based remedial goals. In this paragraph we present a brief review of international practices in setting the clean-up criteria including numerical values of relevant criteria. The review presented below relies on the recently published compilations of European, Asian and U.S. American remediation experiences [1—5]

European and Asian experiences in setting the end-state remedial criteria are summarized in Table 1.

Table 1. Summary of experiences in Europe and Asia (S.Korea) in setting the end-state remedial criteria for radioactively contaminated sites [1, 2, 5]

Facility	Remedial criteria
CEA's Grenoble STED Facility (France)	Residual impact below 0.1 mSv/a for industrial reuse, without technical restriction; if reasonably achievable: residual radioactivity below 0.4 Bq/g (or Bq/cm^2) for β/γ- emitters and below 0.04 Bq/g (or Bq/cm^2) for α-emitters
Uranium conversion facility, Daejeon (Republic of Korea)	Dose based release criteria of 0.1 mSv/a by considering the future unrestricted use of the site and the urbanization of the surrounding area
PIMIC "Lenteja" at CIEMAT (Centre for Energy-Related, Environmental and Technological Research) (Madrid, Spain)	As the site has a restoration plan approved, the general criteria for the release of land and spaces is 0.1 mSv/a; Values greater than 0.1 mSv/a must be justified by an optimization study
Riverbanks contaminated with the waste water (^{137}Cs) from Bohunice NPP (Slovak Republic)	For the dose criterion of 1 mSv/a the max. accept. level of ^{137}Cs estimated at 3.0 or 4.4 Bq/g (for large volume of soil; assuming residential scenario)
Waste disposal site "I'Orme des Merisiers" at St Aubin (Esonne, France)	The criterion for rehabilitation of this site (contam. ^{137}Cs, ^{90}Sr, $^{239/240}$Pu, ^{241}Am) was chosen to be equal to ten times the surrounding background (due to gamma radiation)

The U.S. experiences (see Table 2 below) are reviewed based on information given in [3]. This last report summarizes the various regulatory standards and requirements that dictate the clean-up at radioactively contaminated sites, and presents case studies from 12 selected sites in the U.S. Remediation end state criteria (remedial goals) are usually established by assessing radiological health effects using a risk-based approach for CERCLA (Comprehensive Environmental Response, Compensation and Liability Act) sites or a dose-based approach for NRC (Nuclear Regulatory Commission) sites. Both approaches require selecting appropriate scenarios, models (equations), and site-specific input parameters. It should be noted that dose criteria of 0.15 mSv/a listed in Table 2 compares to the lifetime risk criteria of 10^{-4}. In particular, the US EPA guidance documents have stated that a 0.15 mSv annual dose corresponds to the 3×10^{-4} risk [3].

Table 2. Summary of US experiences in setting end-state remedial criteria for radioactively contaminated sites [3].

Facility	Remedial end-state criteria	
	Dose, mSv/a	Risk
Hanford Site	0.15	
Johnston Atoll		10^{-4} — 10^{-6}
Clean Slate Sites, Nevada	1	
Rocky Flats Cleanup Agreement	0.15	
Rocky Flats (Oversight Panel)	0.15	
Rocky Flats (Revised Soil Action Levels)	0.25	10^{-4} — 10^{-6}
Brookhaven National Laboratory, New York	0.15	
Fort Dix, New Jersey	0.15	
Oak Ridge Reservation—Melton Valley Watershed, Tennessee	0.25	10^{-4}

The presented brief review shows that the most common international practice in European countries, the US and worldwide is to set the end-state remedial criteria for radioactively contaminated sites in the range of doses to relevant critical group of ~0.1—0.15 mSv/a above background contamination level (if reasonably achievable). Lower end-state dose criteria are usually not feasible due to technological, economic or background contamination issues. In a number of reviewed cases higher end state dose criteria of 0.25—1 mSv/a were used.

Method for calculating site-specific release criteria for radionuclide activity concentrations in the material (soil) of the site

The conceptual basis. The proposed method for derivation of specific remedial criteria is based on the IAEA Safety Guide RS-G-1.7 on the application of the concept of exclusion, exemption and clearance [11] which contains tabulated radionuclide specific activity values in released material corresponding to the effective dose of 10 μSv/a. In order to develop site-specific release criteria corresponding to particular dose limit, the tabulated values of radionuclide activity in released materials from RS-G-1.7 (corresponding to the effective dose of 10 μSv/a) can be scaled with the relevant target dose criteria for release of the specific site. The calculation procedures are detailed below.

The bases for radionuclide specific activity values tabulated in Safety Guide RS-G-1.7 [11] are described in the IAEA Safety

Series Rep. no.44 [12]. The activity concentration values in [12] are determined such that individual effective doses to a critical group (i.e. the public and workers) would be of the order of 10 μSv/a (using realistic parameter values). The procedure is based on evaluation of a selected set of typical exposure scenarios for all material, encompassing external irradiation, dust inhalation and ingestion (direct and indirect). List of scenarios used in the IAEA SRS no.44 to develop clearance levels is quite comprehensive including (see [12, Table 2]): workers involved with various operations with the contaminated material, residence and farming near (or immediately within) the area containing contaminated material, using contaminated groundwater, surface water etc. All relevant pathways are implemented for a large list of exposure situations. It is stated that the derived values are sufficient to ensure an adequate protection in both occupational and public exposure situations. The large list of scenarios provides some level of "conservatism" and "safety margin" in application of "scaling" procedures using clearance levels to calculate end-state remedial criteria for remediated sites.

The same radionuclide specific activity values as in the IAEA RS-G-1.7 are included to Ukrainian regulatory document on clearance levels [13]. The last document states that listed clearance levels among other applications can be used during:

- Decommissioning of facilities related to radioactive waste management, and
- In situation of intervention related to remediation of territories contaminated due to nuclear accidents.

Applicability and limitations. The important issue when analysing applicability of release activity criteria listed in the IAEA RS-G-1.7 [11] for setting the end state remedial criteria, is volume (or mass) of contaminated material assumed in underlying risk assessment calculations. The activity concentrations listed in the IAEA RS-G-1.7 for radionuclides of artificial origin apply to "bulk quantities" of radioactive materials [11]. The "Bulk quantity" is defined as "any amount of material that is greater than a moderate quantity", where "moderate quantities" are defined as those "of the order of a tonne". The amount of material involved in calculation of release criteria can be assumed as high as 25 000 m^3 (but typically less than 100 000 m^3) [12, p.42].

Assumptions about mass (volume) of the released material are incorporated to the calculation procedures for release criteria for relevant exposure scenarios described in [12] by means of assumed "dilution factors - D_f" (where D_f represents ratio of released contaminated material to surrounding non-contaminated material). Dilution factors are typically less than 1 (e.g., $D_f = 0.1$).

It is important to note that the values of activity concentration provided in IAEA RS-G-1.7 are not intended to be applied to "radioactive residues in the environment" (e.g., in case of "contaminated land" — i.e. throughout contamination of the environmental media) [11, p.4]. This implies that relevant release criteria can be applied to a large enough mass (volume) of released material, but this amount cannot be "unlimited" (e.g., whole "contaminated land").

Example application of methodology

Description of the Pilot Facility. In this section, the outlined approach is applied to derive the end-state criteria for remediation of the Decontamination Waste Storage facility (DWSF) "Pisky-1". This is a trench-type disposal facility containing radioactive materials from post — Chernobyl accident clean-up operations carried out in 1986-89 in the small village Pisky situated in the Ivankiv District of Kiev Region in the close vicinity of the Chernobyl Exclusion zone. The DWSF "Pisky-1" is situated within the "Zone of Guaranteed Voluntary Resettlement" (defined by the "Law of Ukraine on the Legal Status of the Territory Exposed to the Radioactive Contamination Resulting from the ChNPP Accident" [14]). Population is allowed to reside in this area, however the law imposes requirements with regard to the enhanced monitoring program and restrictions with regard to industrial activities that can lead to the increased exposure of population.

The radioactive material storage conditions in DWSF "Pisky-1" do not comply with applicable regulations and safety requirements and pose potential unacceptable risks to the public [8, 9]. Therefore, this facility was selected in the Project U4.01/12D as "Pilot facility" for developing the remedial design. This project task included among other issues development of the end-state criteria.

The main radioactive contaminant of concern in waste material stored within the DWSF "Pisky-1" is ^{137}Cs (maximum activity 53 kBq/kg, mean activity 3 kBq/kg as in 2015, based on data of State Enterprise "KORO", Zhovty Vody). The waste contains also ^{90}Sr in activity comparable to activity of ^{137}Cs (^{90}Sr to ^{137}Cs activity ratio varies for different samples from 0.7 to 2), as well as significantly smaller specific activity concentrations of ^{241}Am and Pu isotopes (see Table 3 for more detail). Radionuclide ratios in waste are within the range typical for fallout particles originating from the dispersed nuclear fuel of Chernobyl nuclear power plant Unit 4 at the time of the accident. The total volume of stored waste (known to be mainly contaminated soil and construction debris) is about 190 m^3. The background surface contamination of topsoil by Chernobyl fallout in the vicinity of DWSF "Pisky-1" constitutes ~0.4 Bq/kg for ^{137}Cs and ~0.2 Bq/kg for ^{90}Sr [8, 9].

The following objectives were pursued when developing the end state remedial criteria for Pilot Facility: (1) they should provide relevant level of radiation safety to population and environment, and (2) they should be balanced with background contamination levels of the environment by Chernobyl fallout.

Table 3. Radionuclide scaling factors with respect to ^{137}Cs, clearance levels and radionuclide dose conversion coefficient values (derived using eq.(1)) used in calculation of end-state criteria for Pilot Facility (DWSF "Pisky-1")

Radio-nuclide	Radionuclide scaling factor with respect to ^{137}Cs (K_i)	Clearance level, Bq/g	DCC_i (Sv a^{-1})/(Bq kg^{-1})	K_i x DCC_i (Sv a^{-1}) /(Bq kg^{-1})
^{137}Cs	1	0.1	1.00E-07	1.00E-07
^{90}Sr	2	1	1.00E-08	2.00E-08
^{241}Am	0.018	0.1	1.00E-07	1.80E-09
^{238}Pu	0.004	0.1	1.00E-07	4.00E-10
^{239}Pu	0.004	0.1	1.00E-07	4.00E-10
^{240}Pu	0.006	0.1	1.00E-07	6.00E-10
^{241}Pu	0.18	10	1.00E-09	1.90E-10
Sum $\{DCC_i$ x $K_i\}$				1.23E-07

It appears reasonable to assume that the potential post-remedial radiological exposure scenarios for Pilot Facility (for example excavation of the remediated site for house construction etc.) would usually involve some mixing of residual materials with surrounding non-contaminated environmental materials. The volume of residual contaminated material to remain at the Pilot Facility can be estimated not to exceed ~100 m^3. This relatively small value complies with the relevant assumptions on volume of contaminated material used in derivation of activity criteria listed in the IAEA SRS no.44 [12].

Calculation procedures for derivation of end-state criteria. The procedure for calculating the end-state criteria for the Pilot Facility uses the Dose Conversion Coefficients (DCC-s), which are based on the tabulated radionuclide specific activity values from the IAEA RS-G-1.7 report [11] corresponding to the dose constraint of 10 μSv y^{-1} (i.e., clearance levels). Formula to calculate the Dose Conversion Coefficients for radionuclide "*i*" (DCC_i, (Sv a^{-1})/(Bq kg^{-1})) is as follows:

$$DCC_i = \frac{Dose_{Constraint}}{CL_i}, \qquad (1)$$

where $Dose_{Constraint}$ is relevant dose constraint value (i.e., 10 μSv/a), and CL_i is the clearance level for radionuclide "*i*" from the IAEA RS-G-1.7 (Bq/kg). Thus DCC_i represents a yearly dose received by an reference individual per unit activity concentration of radionuclide "*i*" in the source material.

The formula utilizing the defined above DCC_i values to calculate the exposure dose from facility ($Dose_{Facility}$, Sv/a) is as follows:

$$Dose_{Facility} = \sum_i DCCi \times Ci; \qquad (2)$$

Where C_i (Bq/kg) is activity of radionuclide "*i*" in contaminated materials related to facility.

The equation for the target end-state remedial dose criteria for facility ($Dose_{Criteria}$, Sv/a) can be written as follows:

$$Dose_{Facility} \le K_{sf}\, Dose_{Criteria}; \qquad (3)$$

Where K_{sf} is a "safety factor" (K_{sf} <1) accounting for measuring uncertainties in contaminant concentration values in released materials related to facility (e.g., analytical uncertainties, statistical variability of contamination, etc.).

Substituting (2) to equation (3) yields the constraint for contaminant concentration values in released materials (C_i) which guarantees that relevant dose criteria is satisfied:

$$\left\{\sum_i DCCi \times Ci\right\} \le K_{sf} \times Dose_{Criteria}; \qquad (4)$$

It can be further assumed that radionuclide activities in material related to facility (C_i) can be scaled with ^{137}Cs activity in the same material:

$$Ci = Ki \times Ccs; \qquad (5)$$

Where C_{cs} is activity of ^{137}Cs in released materials (Bq/kg), and K_i is scaling coefficient of activity of radionuclide "*i*" to activity of ^{137}Cs (unitless) (see Table 3). The resulting formula establishing a constraint on the concentration of ^{137}Cs in material remaining on the site is as follows:

$$Ccs \le \frac{Ksf \quad Dose_{Criteria}}{\sum DCCi\, Ki}; \qquad (6)$$

In case background contamination levels are needed to be taken into account, the following expression for "dose criteria" should be substituted in equations

Should be substituted in equations (3) or (6)

$$Dose_{Criteria} = Dose_{Inc} + Dose_{Bg}, \qquad (7)$$

$$Dose_{Bg} = \sum_i DCCi \times Ci,bg; \qquad (8)$$

Where $Dose_{Inc}$ is the incremental dose criteria above background levels (e.g., 0.1 mSv/a), $Dose_{Bg}$ is dose associated with the background contamination, and $C_{i,bg}$ (Bq/kg) is background concentration of radionuclide "*i*" in soil.

Results and discussion. Calculations of the radionuclide DCC_i values (based on IAEA RS-G-1.7) and the sum of DCC_i values scaled with ^{137}Cs ratios in waste material of Pilot Facility are summarized in Table 3. The higher end value of ^{90}Sr to ^{137}Cs activity ratio (i.e., 2) is chosen to provide a conservative dose assessment. Analysis of data of Table 3 suggests (considering listed DCC_i and K_i numerical values) that the main radionuclide determining radiological hazard from waste material is ^{137}Cs, while ^{90}Sr activity will be a second parameter by importance. Taking into account comparatively low specific activity of ^{241}Am and Pu isotopes in waste material, these radionuclides have relatively low impact on overall radiological hazard from waste material.

Release criteria for ^{137}Cs in waste material of Pilot Facility for different target dose criteria calculated using formula (6) are summarized in Table 4. Calculation assumes background concentrations in soil of 0.4 Bq/kg for ^{137}Cs and ~0.2 Bq/kg for ^{90}Sr. Calculation employs safety factor value K_{sf}= 0.8. This value is based on data of publication [15] regarding accuracy of field measurements of soil radioactivity assuming that ^{137}Cs activity in waste material (C_{cs}) is averaged on 5 samples, and analytical measuring error of ^{137}Cs is 10—20 % (which is in agreement with the procedure of analytical measurements of waste material, that is foreseen by the remedial project design).

Table 4. Activity criteria for ^{137}Cs in residual waste material of Pilot Facility for different target dose criteria.

Target dose criteria (dose above background), mSv/a	Concentration of ^{137}Cs in waste material*, Bq/g
0.1	0.92
0.2	1.6
0.3	2.2
0.4	2.9
0.5	3.5

Note: * — other radionuclides are included implicitly assuming respective K_i ratios listed in Table 3

Estimated amount of waste material in DWSF "Pisky-1" corresponding to various threshold ^{137}Cs activity values is listed in Table 5. This table is based on statistical parameters of the data set of ^{137}Cs activity measurements in DWSF "Pisky-1" inferred from gamma-logging characterization works [8].

Table 5. Estimated amount of waste material to be retrieved from DWSF "Pisky-1" corresponding to various threshold ^{137}Cs activity criteria.

^{137}Cs activity in waste, Bq/g	Target dose criterion, mSv/a	% of gamma-logging measurements	Estimated waste volume to be retrieved, m^3
> 1	~0.1	59	110
> 2	~0.3	38	71
> 3	~0.5	27	51
Total waste volume		100	187

Based on analyses of information contained in Table 4 and Table 5 it appears that a target dose criteria of 0.1 mSv/a (above background contamination) is a justified end-sate criterion for DWSF "Pisky-1". This dose criterion corresponds to a target activity concentration of ~1 Bq/g of ^{137}Cs (it is implicitly assumed also that other radionuclides are included in waste material with respective K_i ratios to ^{137}Cs listed in Table 3). Material with contamination above the target activity concentration must be removed from the remediation site as waste material and disposed off elsewhere.

Data of Table 4 show that decreasing the target dose criterion from 0.3 mSv/a to 0.1 mSv/a results in an increase of the estimated amount of waste material to be retrieved from Pilot Facility by ~40 m^3 and the total amount of waste material to be retrieved is estimated at ~110 m^3. This is a feasible amount of waste to be managed. The end state criteria of ^{137}Cs activity in soil of 1 Bq/g is ~2.5 times above average background contamination of the DWSF "Pisky-1" location area by Chernobyl fallout. Some adjacent areas (e.g., Karpilovka Village) have fallout ^{137}Cs hot-spots event with approximately twice higher specific activity of topsoil (e.g., ~0.8 Bq/g). Therefore, a lower value of the ^{137}Cs target activity (dose) criteria for DWSF "Pisky-1" is not justified because of relatively high background radioactive contamination levels of the environment. The target criterion is feasible from the point of view of on-site in-situ gamma spectrometry measurements of bulk material for waste sorting in the course of the waste retrieval process [8]. Lastly, the proposed end state criterion for the Pilot Facility conforms to the best international practice in remediation of radioactively contaminated "legacy" sites.

It is assumed that upon completion of remedial works the long-term administrative regime of site will fully conform to requirements of the territory of the "Zone of Guaranteed Voluntary Resettlement", where the DWSF "Pisky-1" is situated. In particular, no construction works (or other similar disturbances) will be carried out without appropriate justification, and the site will be covered by a comprehensive radiation monitoring program.

Remark on incorporation of site-specific scenarios. The assessment procedure for remediated site may potentially require consideration of site-specific scenario(s) in addition to those that have served the basis for derivation of clearance level listed in the IAEA Safety Series Rep. no.44 [12]. Let's assume, that such complimentary scenario results in dose conversion coefficient for radionuclide "i" $DCC_{i,com}$ (Sv/a). In this case dose conversion coefficient to be used in formula (6) to calculate the dose from facility shall be replace by the following one:

$$DCC_i = \max\{DCC_{i,CL}, DCC_{i,com}\},$$

where $DCC_{i,CL}$ is dose conversion coefficient value calculated based on clearance level using formula (1).

Conclusions

The method for assessment of the end-state criteria for remediation of radioactively contaminated sites described in this publication has the following advantages:

- It uses simple and transparent calculation procedures;
- It is based on the reputable international references, well documented assessment procedures and dose model parameters (i.e., IAEA SRS no.44 [12]);
- It is based on balanced approach to dose calculations using, at one hand, a large list of exposure scenarios combined, at the other hand, with the realistic (rather than conservative) values of dose model parameters;
- Additional site-specific scenarios can be potentially easily integrated to the assessment procedure.

The presented approach can be easily transferred to other radioactively contaminated sites (e.g., similar to 'Pilot Facility' described in this article), keeping in mind limitations regarding the size of the site and volume of the residual radioactively contaminated material.

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Практичний підхід для оцінки радіологічних критеріїв кінцевого стану при реабілітації радіоактивно-забруднених об'єктів

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Описано підхід для визначення радіологічних критеріїв кінцевого стану при реабілітації радіоактивно-забруднених ділянок. Цільові критерії встановлюються у формі прогнозних ефективних доз для груп населення, що зазнають підвищеного опромінення (представницьких осіб). Представлено короткий огляд найкращого світового досвіду у визначенні критеріїв реабілітації на основі оцінки ризиків. Специфічні для об'єкту критерії звільнення у формі концентрації активності у матеріалі, що звільняється від контролю (наприклад, в Бк/г для ґрунту) визначаються за допомогою табличних значень питомої активності радіонуклідів згідно керівництва з безпеки МАГАТЕ RS-G-1.7 (що відповідають ефективній дозі 10 мкЗв/рік). Ці табличні значення перераховуються з відповідним коефіцієнтом пропорційно до обраного цільового дозового критерію для реабілітації конкретного об'єкту. Обговорюються межі придатності та обмеження запропонованого підходу (наприклад, щодо обсягу звільненого матеріалу). Описано процедуру врахування додаткових сценаріїв опромінення, що є специфічними для конкретного об'єкту. Наприкінці, описаний у статті підхід з метою ілюстрації застосовано до конкретного радіоактивно забрудненого майданчика (тобто, до пункту зберігання радіоактивних відходів дезактивації чорнобильського походження, розташованого у Київській області). Запропонований підхід може бути застосований до широкого кола аналогічних проблем.

Ключові слова: оцінка безпеки, реабілітація, критерії кінцевого стану, аварія на ЧАЕС

Практический подход для оценки радиологических критериев конечного состояния при реабилитации радиоактивно загрязненных объектов

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Описан подход для определения радиологических критериев конечного состояния при реабилитации радиоактивно загрязненных участков. Целевые критерии устанавливаются в форме прогнозных эффективных доз для групп населения, подверженных повышенному облучению (представительных лиц). Представлен краткий обзор лучшего мирового опыта в определении критериев реабилитации на основе оценки рисков. Специфические для объекта критерии освобождения в форме концентрации активности в материале, освобождаемом от контроля (например, в Бк/г для грунта) определяются с помощью табличных значений удельной активности радионуклидов согласно руководства по безопасности МАГАТЭ RS-G-1.7 (соответствующих эффективной дозе 10 мкЗв/год). Эти табличные значения пересчитываются с соответствующим коэффициентом пропорционально избранному целевому дозовому критерию для реабилитации конкретного объекта. Обсуждаются пределы пригодности и ограничения предложенного подхода (например, относительно объема освобождаемого материала). Описана процедура для учета дополнительных сценариев облучения, которые являются специфическими для конкретного объекта. В конце, описанный в статье подход с целью иллюстрации применен к конкретному радиоактивно загрязненному объекту (а именно, к пункту хранения радиоактивных отходов дезактивации чернобыльского происхождения, расположенному в Киевской области). Предложенный подход может быть применен к широкому кругу аналогичных проблем.

Ключевые слова: оценка безопасности, реабилитация, критерии конечного стостояния, авария на ЧАЭС

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