21 June 2022

IAEA SAFETY STANDARDS

for protecting people and the environment

Status: Step 8a

Submission for 120-day Member States review

Radiation Protection Aspects of Design for Nuclear Power Plants

DS524

DRAFT Revised SAFETY GUIDE

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide on radiation protection aspects of design for nuclear power plants provides recommendations on how to meet the requirements of IAEA Safety Standard Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [1], in particular Requirements 5, 12, 19, 81 and 82 and supporting requirements provided in paras 2.6 - 2.10, 5.4 and 5.31A. It addresses the provisions that should be made in the design of nuclear power plants to protect site personnel, the public and the environment against radiological hazards for operational states, accident conditions and decommissioning.

1.2. The recommendations on radiation protection provided in this Safety Guide are consistent with the IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [2].

1.3. The recommendations provided in this Safety Guide are intended to meet other relevant requirements of IAEA Safety Standards Series regarding radiation protection aspects of design for nuclear power plants, including the following:

- SSR-2/2 (Rev. 1) [3], Safety of Nuclear Power Plants: Commissioning and Operation;
- SSR-1 [4], Site Evaluation for Nuclear Installations;
- GSR Part 2 [5], Leadership and Management for Safety;
- GSR Part 4 (Rev. 1) [6], Safety Assessment for Facilities and Activities;
- GSR Part 5 [7], Predisposal Management of Radioactive Waste;
- GSR Part 6 [8], Decommissioning of Facilities;
- GSR Part 7 [9], Preparedness and Response for a Nuclear or Radiological Emergency.

1.4. This Safety Guide is a revision of the IAEA Safety Standard Series No. NS-G-1.13, Radiation Protection Aspects of Design for Nuclear Power Plants, which it supersedes.

1.5. Effective radiation protection is a combination of good design, high quality construction and proper operation. Procedures that address the radiation protection aspects of operation are covered in the safety guides on Occupational Radiation Protection (GSG-7) [10] and Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors (SSG-40) [11].

OBJECTIVE

1.6. The purpose of this Safety Guide is to provide recommendations for ensuring radiation protection in (1) the design of new nuclear power plants, (2) design modifications to operating plants, and (3) safety reviews and assessments of operating plants (for example as part of the comprehensive evaluation of safety or the periodic safety review of the plant). The recommendations are provided to assist in meeting the applicable safety requirements, in particular those established in SSR-2/1 (Rev. 1) [1], under the fundamental safety objective, that is “to protect the people and the environment from harmful effects of ionizing radiation", by establishing and maintaining in nuclear installations effective defences against radiological hazards.

1.7. This Safety Guide is for use by organizations responsible of designing, manufacturing and constructing nuclear power plants, by operating organizations and contractors, including plant operators who are involved in planning, managing and carrying out the design and design modification of nuclear power plants, and by regulatory bodies and technical support organizations.

1.8. This Safety Guide includes recommendations on radiation protection of design for nuclear power plants that have interface with other IAEA safety guides; in those cases, references to those safety guides are indicated, to help facilitate consistent implementation of the recommendations provided in different safety guides about radiation protection of workers, the public and the environment.

SCOPE

1.9. This Safety Guide:

(1) Describes the applicable safety requirements of the system of dose limitation and optimization as a basis for the radiation protection measures that should be implemented in the design of nuclear power plants.

(2) Describes the measures to be taken in the design for the radiation protection of site personnel and the public and the environment.

(3) Outlines the methods that are used to calculate on-site and off-site radiation levels and to verify that the design provides an adequate level of radiation protection.

(4) Describes in one of the annexes the important sources of radiation and contamination against which protection for site personnel, the public and the environment has to be provided in the design and provides examples of zoning for radiation protection.

1.10. In addition to the measures that are required to protect site personnel and members of the public when the plant is in operational states and during decommissioning, this Safety Guide also deals with accident conditions, including severe accidents.[1]

1.11. This Safety Guide is intended for use primarily for land based, stationary nuclear power plants with water cooled reactors designed for electricity generation or for other heat generating applications

[1] This Safety Guide does not address the design measures that are necessary to reduce the probability of the occurrence and to prevent the development of accidents. These aspects are considered in the Safety Requirements for Design [1] and in other safety guides.

(e. g. district heating, desalination). It is recognized that for other reactor types, including future plant systems having limited operating experience, some recommendations may need interpretive judgement applied or may not be appropriate.

1.12. This Safety Guide addresses radiation protection aspects of the handling, treatment and storage of radioactive waste. It does not specifically deal with the safety aspects of waste treatment relating to the form or quality of the waste product with regard to its longer-term storage or disposal. These aspects are considered in a number of other safety standards, including GSR Part5 [7], SSG-40 [11] and SSG-41 [12].

1.13. The technical terms utilized in this Safety Guide should be understood in accordance with the description of those terms provided in the IAEA Safety Glossary [13].

STRUCTURE

1.14. Section 2 of the Safety Guide introduces the relevant requirements, such as those in respect of dose limits, the application of the principle of optimization of protection and the setting of design targets. Design approaches for operational states, decommissioning and accident conditions are described in Section 3, while Section 4 deals with the control of sources of radiation and estimation of radiation dose rates in all plant states and in decommissioning. Section 5 deals with design features for operational states. Section 6 covers the design features in accident conditions. Section 7 deals with the design features for decommissioning. Section 8 gives guidance on the radiation monitoring system for all plant states and for decommissioning.

1.15. The Appendix provides recommendations on the application of the optimization principle. Annex I provides information about the sources of radiation during normal operation and decommissioning as well as under accident conditions and Annex II gives examples of zoning that may be used for design purposes.

2. SAFETY OBJECTIVES, DOSE LIMITATION AND OPTIMIZATION OF PROTECTION AND SAFETY

2.1. The radiation risks to workers and the public and to the environment that may arise from activities of nuclear power plants should be assessed and controlled. In para. 2.2 of SF-1 [14] it is stated:

> "2.2. The fundamental safety objective applies for all facilities and activities, and for all stages over the lifetime of a facility or radiation source, including planning, siting, design, manufacturing, construction, commissioning and operation, as well as decommissioning and closure. …"

This Safety Guide has interface with other safety guides that are also aimed to help meeting the relevant safety requirements for the radiation protection of the workers, the public and the environment. A summary of the most relevant safety requirements motivating the recommendations provided in this Safety Guide are provided in the below paras of this section.

SAFETY OBJECTIVES

Radiation protection in design

2.2. In accordance with the principles of radiation protection, provisions are required to be made in the design to comply with para. 2.6 of SSR-2/1 (Rev. 1) [1]:

> "2.6. In order to satisfy the safety principles, it is required to ensure that for all operational states of a nuclear power plant and for any associated activities, doses from exposure to radiation within the installation or exposure due to any planned radioactive release from the installation are kept below the dose limits and kept as low as reasonably achievable. In addition, it is required to take measures for mitigating the radiological consequences of any accidents, if they were to occur."

Furthermore, provisions are required to be made in the design to comply with para. 2.7 of SSR-2/1 (Rev.1) [1]:

> "2.7. To apply the safety principles, it is also required that nuclear power plants be designed and operated so as to keep all sources of radiation under strict technical and administrative control. However, this principle does not preclude limited exposures or the release of authorized amounts of radioactive substances to the environment from nuclear power plants in operational states. Such exposures and radioactive releases are required to be strictly controlled and to be kept as low as reasonably achievable, in compliance with regulatory and operational limits as well as radiation protection requirements."

2.3. In Requirement 5 of SSR-2/1 (Rev. 1) [1] it is stated:

> "The design of a nuclear power plant shall be such as to ensure that radiation doses to workers at the plant and to members of the public do not exceed the dose limits, that they are kept as low as reasonably achievable in operational states for the entire lifetime of the plant, and that they remain below acceptable limits and as low as reasonably achievable in, and following, accident conditions."

2.4. In Requirement 81 "Design for radiation protection" of SSR-2/1 (Rev. 1) [1] it is stated:

> "Provision shall be made for ensuring that doses to operating personnel at the nuclear power plant will be maintained below the dose limits and will be kept as low as reasonably achievable, and that the relevant dose constraints will be taken into consideration."

Safety in design

2.5. In para. 2.8. of SSR-2/1 (Rev. 1) [1] it is stated:

> "2.8. To achieve the highest level of safety that can reasonably be achieved in the design of a nuclear power plant, measures are required to be taken to do the following, consistent with national acceptance criteria and safety objective:
>
> (a) To prevent accidents with harmful consequences resulting from a loss of control over the reactor core or over other sources of radiation, and to mitigate the consequences of any accidents that do occur;
>
> (b) To ensure that for all accidents taken into account in the design of the installation, any radiological consequences would be below the relevant limits and would be kept as low as reasonably achievable;
>
> (c) To ensure that the likelihood of occurrence of an accident with serious radiological consequences is extremely low and that the radiological consequences of such an accident would be mitigated to the fullest extent practicable."

2.6. In accordance with para 4.4. of SSR-2/1 (Rev. 1) [1] acceptable limits should be established for radiation protection:

> "4.4. Acceptable limits for purposes of radiation protection[8] associated with the relevant categories of plant states shall be established, consistent with the regulatory requirements."
>
> "[8] Requirements on radiation protection and safety of radiation sources are established in IAEA Safety Standards Series No. GSR Part 3 [2], Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards."

2.7. In para. 5.31A. of SSR-2/1 (Rev. 1) [1] it is stated:

> "5.31A. The design shall be such that for design extension conditions, protective actions that are limited in terms of lengths of time and areas of application shall be sufficient for the protection of the public, and sufficient time shall be available to take such measures."

2.8. In para. 5.4. of SSR-2/1 (Rev. 1) [1] it is stated:

> "5.4. The design limits shall be specified and shall be consistent with relevant national and international standards and codes, as well as with relevant regulatory requirements."

2.9. In the design of the plant for the minimization of the radiation impact to the site personnel, the public and the environment, Requirement 13 of GSR Part 3 [2] "Safety assessment" should be taken into account:

"Special consideration shall be given at the design stage of a nuclear power plant to the incorporation of features to facilitate radioactive waste management and the future decommissioning and dismantling of the plant."

Furthermore, requirements in para 4.20. of SSR-2/1 (Rev. 1) [1] should be also fulfilled during design of NPPs:

"4.20. In particular, the design shall take due account of:

(a) The choice of materials, so that amounts of radioactive waste will be minimized to the extent practicable and decontamination will be facilitated;

(b) The access capabilities and the means of handling that might be necessary;

(c) The facilities necessary for the management (i. e. segregation, characterization, classification, pre-treatment, treatment and conditioning) and storage of radioactive waste generated in operation, and provision for managing the radioactive waste that will be generated in the decommissioning of the plant."

Safety assessment in the design

2.10. Para. 4.6. of SSR-1 [4] provides requirements for the assessment of the site suitability including an assessment on how characteristics of the site and its environment could influence the transfer of radioactive material released from the nuclear installation to people and to the environment.

"4.6. In the assessment of the suitability of a site for a nuclear installation, the following aspects shall be addressed at an early stage of the site evaluation:

(a) The effects of natural and human induced external events occurring in the region that might affect the site;

(b) The characteristics of the site and its environment that could influence the transfer of radioactive material released from the nuclear installation to people and to the environment;

(c) The population density, population distribution and other characteristics of the external zone, in so far as these could affect the feasibility of planning effective emergency response actions [GSR Part 7], and the need to evaluate the risk to individuals and to the population."

Safety guides SSG-68 [15], SSG-67 [16], SSG-9 (Rev. 1) [17], NS-G-2.13 [18], NS-G-3.2 [19] and GSG-10 [20] provide recommendations for the assessment or reassessment (for safety reviews) of the suitability of a site and also for analysis of secondary and cascading effects of external hazards for designing or assessing the effective radiation protection measures and arrangements.

2.11. Requirement 13 "Safety assessment" of GSR Part 3 [2] states that:

> "The regulatory body shall establish and enforce requirements for safety assessment, and the person or organization responsible for a facility or activity that gives rise to radiation risks shall conduct an appropriate safety assessment of this facility or activity."

Furthermore para. 3.29. of GSR Part 3 [2] states that:

> "3.29. The regulatory body shall establish requirements for persons or organizations responsible for facilities and activities that give rise to radiation risks to conduct an appropriate safety assessment. Prior to the granting of an authorization, the responsible person or organization shall be required to submit a safety assessment, which shall be reviewed and assessed by the regulatory body."

2.12. In the design it should be assessed whether adequate measures are in place to protect people and the environment from harmful effects of ionizing radiation, in accordance with GSR Part 4 (Rev. 1) [6], Requirement 9, "Assessment of the provisions for radiation protection".

2.13. Safety assessment on radiation protection should be performed at different stages or phases, including site evaluation, design, manufacturing, construction, assembly of structures, systems and components (SSC), commissioning, operation, maintenance~~,~~ and decommissioning (or closure) of NPPs. Such assessment should be performed in accordance with the requirements from paras 3.31 - 3.36 of GSR Part 3 [2].

2.14. Continuity of the assessment on radiation protection and consistency about its documentation in the different licensing documents issued or revised at different stages or phases of the plant should be maintained; this also applies to the documents related to environmental impact assessment, safety analysis report and reports about emergency plans.

2.15. The safety assessment on radiation protection should be regularly updated (e. g. as part of the comprehensive evaluation of safety or the periodic safety review of the plant). The corresponding safety assessment documents should take into account the results of the latest deterministic safety analyses and probabilistic safety assessments performed.

2.15A. It should be demonstrated that limits for releases and dose limits for exposed workers specified in regulations or licensing conditions, will not be exceeded in operational states.

2.16. In Requirement 19, para. 5.25 of SSR-2/1 (Rev. 1) [1] it is stated

> "5.25. The design shall be such that for design basis accident conditions, key plant parameters do not exceed the specified design limits. A primary objective shall be to manage all design basis accidents so that they have no, or only minor, radiological consequences, on or off the site, and do not necessitate any off-site protective actions.

In addition, para. 5.26 of [1] indicates, "5.26. The design basis accidents shall be analysed in a conservative manner. …"

In accordance with these requirements, it should be demonstrated in a conservative manner that key plant parameters do not exceed the specified design limits and that all design basis accidents have no or only minor radiological impacts, on or off the site, and do not necessitate any off-site protective actions (e. g. evacuation). Dose limits[2] for workers (including those who are controlling and mitigating design basis accidents) should be considered in the design criteria. Further recommendations are provided in para. 6.14.

2.17. In accordance with Requirement 20 of SSR-2/1 (Rev. 1) [1] the design extension condition (DEC) may be analysed using best estimate assumptions. It should be demonstrated that the identified reasonably practicable provisions prevent severe fuel damage (DEC A) and mitigate severe accidents (DEC B). The potential on-site and off-site radiological consequences resulting from the DEC (given successful accident management measures) should be evaluated to demonstrate the compliance with acceptance criteria reflecting reference levels established by the regulatory body.

2.18. Recommendations of NS-G-3.2 [19] and GSG-10 [20] on prospective radiological impact assessment of the protection of the public and the environment should be taken into consideration during the design stages and plant modifications and kept updated during operation.

Interfaces between safety and security

2.19. In para. 1.36 of GSR Part 3 [2] it is stated:

> "1.36. Safety measures and security measures have in common the aim of protecting human life and health and the environment. In addition, safety measures and security measures must be designed and implemented in an integrated manner, so that security measures do not compromise safety and safety measures do not compromise security."

In addition, Requirement 8 of SSR 2/1 (Rev. 1) [1], "Interfaces of safety with security and safeguards", requests:

> "Safety measures, nuclear security measures and arrangements for the State system of accounting for, and control of, nuclear material for a nuclear power plant shall be designed and implemented in an integrated manner so that they do not compromise one another."

Furthermore, para. 1.37 of GSR Part 3 [2] states that

> "1.37. Security infrastructure and safety infrastructure need to be developed, as far as possible, in a well-coordinated manner. …".

[2] Some Member States apply higher off-site doses as acceptance criteria for the design basis accidents having lower frequencies.

2.19A. IAEA Nuclear Security Series No. 13 (INFCIRC/225/Rev.5) [21] provides recommendations for the physical protection of nuclear material and nuclear facilities. Specifically, in paras 3.28 and 5.13 of [21] it is stated:

> "3.28. For a new nuclear facility, the site selection and design should take physical protection into account as early as possible and also address the interface between physical protection, safety and nuclear material accountancy and control to avoid any conflicts and to ensure that all three elements support each other."
>
> "5.13. The physical protection system against sabotage should be designed as an element of an integrated system to prevent the potential consequences of sabotage by taking into account the robustness of the engineered safety and operational features, and the fire protection, radiation protection and emergency preparedness measures."

Furthermore, IAEA Nuclear Security Series No. 14, [22] provides nuclear security recommendations on radioactive material and associated facilities. Specifically, its paras 3.25 – 3.28 provide guidance on interface of nuclear security with safety.

Radiation protection for emergency response

2.20. Provisions should be designed to ensure the protection and safety for all persons on the site in a nuclear or radiological emergency in line with the requirements stated in paras 5.41 and 5.42 of GSR Part 7 [9]

> "5.42. Arrangements as stated in para. 5.41 shall also include ensuring the provision, for all persons present in the facility and on the site, of:
>
> (a) Suitable assembly points, provided with continuous radiation monitoring;
>
> (b) A sufficient number of suitable escape routes;
>
> (c) Suitable and reliable alarm systems and other means for warning and instructing all persons present under the full range of emergency conditions."

2.21. Radiation protection provisions for emergency response, including monitoring systems, should be designed in compliance with Requirement 67 of SSR-2/1 (Rev. 1) [1], "Emergency response facilities on the site", which states that:

> "The nuclear power plant shall include the necessary emergency response facilities on the site. Their design shall be such that personnel will be able to perform expected tasks for managing an emergency under conditions generated by accidents and hazards. "

Recommendations related to the design of radiation protection provisions for emergency response are provided in Section 6 of this Safety Guide.

2.22. In the design of the means for information exchange and communication to be used between the relevant emergency response facilities of the plant the requirements provided in para. 6.42 of SSR-2/1 (Rev. 1) [1] should be met:

> "6.42. Information about important plant parameters and radiological conditions at the nuclear power plant and in its immediate surroundings shall be provided to the relevant emergency response facilities. Each facility shall be provided with means of communication with, as appropriate, the control room, the supplementary control room and other important locations at the plant, and with on-site and off-site emergency response organizations."

Other recommendations about the information and communication system are provided in Section 6 of this Safety Guide.

2.23. In para. 5.64. of SSR-2/1 (Rev. 1) [1] it is stated:

> "5.64. Escape routes from the nuclear power plant shall meet the relevant national and international requirements for radiation zoning and fire protection, and the relevant national requirements for industrial safety and plant security."

Recommendations related to the design of radiation protection provisions and monitoring systems for emergency response are provided in Section 8 of this Safety Guide.

2.24. Regarding emergency planning zones and emergency planning distances, para. 5.40 of GSR Part 7 [9] requires the following:

> "5.40. Within emergency planning zones and emergency planning distances, arrangements shall be made for the timely monitoring and assessment of contamination, radioactive releases and exposures for the purpose of deciding on or adjusting the protective actions and other response actions that have to be taken or that are being taken. These arrangements shall include the use of pre-established operational criteria in accordance with the protection strategy ..."

Recommendations to meet this requirement are provided in sections 6 and 8 of this Safety Guide.

APPLICATION OF DOSE LIMITS INTO DESIGN

Authorized dose limits and dose constraints for operational states and decommissioning

2.25. The design of the nuclear power plant should be such as to ensure that authorized dose limits and dose constraints[3] for site personnel and the public will not be exceeded over specified periods (e. g. monthly, quarterly, or annually) in operational states (normal operation and anticipated operational

[3] Dose limits for occupational exposure and public exposure are established by the government or the regulatory body. Relevant dose constraints for occupational exposure are established and used by licensees, and those for public exposure are established or approved by the government or regulatory body. For internal exposures, such as those that result from the inhalation and ingestion of radioactive substances, the dose limits apply to the committed dose. See also the IAEA Safety Glossary [13].

occurrences) and decommissioning. In order to comply with the requirements of GSR Part 3 [2], the authorized dose limits and dose constraints should not exceed the values of the dose limits established in GSR Part 3 [2]. For workers who do not enter the designated areas (supervised areas and controlled areas), the authorized dose constraints should be set at the same level as the individual dose limit for members of the public (GSG-7 [10]).

Requirement 12 of GSR Part 3 [2] states that:

> "The government or the regulatory body shall establish dose limits for … public exposure, and registrants and licensees shall apply these limits."

Paragraph 3.120 of GSR Part 3 [2], which relates to responsibilities specific to public exposure, states that:

> "3.120. The government or the regulatory body shall establish or approve constraints on dose and constraints on risk to be used in the optimization of protection and safety for members of the public. …"

Paragraph 3.123 (e) of GSR Part 3 [2] states that:

> "3.123. The regulatory body shall establish or approve operational limits and conditions relating to public exposure, including authorized limits for discharges. These operational limits and conditions: … (e) Shall take into account the results of the prospective assessment for radiological environmental impacts that is undertaken in accordance with requirements of the regulatory body …".

2.26. The authorized annual dose constraints for members of the public apply to the representative person of the population that is an individual receiving a dose that is representative of the doses to the more highly exposed individuals in the population. (See the IAEA Safety Glossary [13], GSR Part 3 [2] and GSG-9 [23]). Studies should be carried out to identify the representative person and critical pathways for the exposure of such person. Discharge limits for specific radionuclides in liquid and gaseous effluents (e. g. annual, quarterly, monthly, daily — the shorter periods permit increased release rates over short time periods and thus increased operational flexibility) should be derived from the application of the authorized dose constraints for representative persons. The discharge limits should ensure that the maximum individual dose for a representative person does not exceed the dose constraint established by the regulatory body.

OPTIMIZATION OF RADIATION PROTECTION AND SAFETY

Application of the optimization principle

2.27. The design should ensure that protection and safety are optimized. Requirement 11 of GSR Part 3 [2] "Optimization of protection and safety" states that:

"The government or the regulatory body shall establish and enforce requirements for the optimization of protection and safety, and registrants and licensees shall ensure that protection and safety is optimized."

2.28. To keep all exposures within authorized dose limits and dose constraints and as low as reasonably achievable, economic and social factors are taken into account:

— The radiation exposure should be reduced by means of radiation protection measures to values such that further expenditure for design, construction and operation would not be warranted by the associated reduction in radiation exposure (economic factors).

— Issues such as reducing major disparities in the occupational doses received by workers of different types who work within the controlled area and avoiding arduous working conditions in radiation areas (social factors) should be taken into account in the design. The types of workers who could potentially receive the highest doses include maintenance and inspection personnel and health physics staff.

2.29. In general, the optimization of radiation protection implies a choice from a set of protective measures, including design options such as shielding, avoidance of materials which can be easily activated, minimization of surfaces which can be easily contaminated, removal of radionuclides from coolants, filtering of air in working areas, remote operation and tooling to minimize radiation exposure time. Feasible options should be identified, criteria for comparison and appropriate values for them should be determined and, finally, the options should be evaluated and compared. Radiological acceptance criteria for normal operation and for anticipated operational occurrences are discussed in paras. 4.8 and 4.9 of SSG-2 (Rev. 1) [24], respectively. Details of different structured approaches to making decisions are given in Appendix.

2.29A. The design of the NPP should be such as to ensure that during operational states the corresponding dose limits and dose constraint for site personnel and for public will not be exceeded. It should be demonstrated that the radiological acceptance criteria for operational states, identified in accordance with the dose limits and dose constraints and reflected in the design limits, are met in the design. Further recommendations on radiological acceptance criteria are provided in SSG-2 (Rev. 1) [24] (see para 4.8 about normal operation and paras 4.9 and 7.23 about anticipated operational occurrences).

2.30. The concept of optimization should also apply to design features whose purpose is to prevent or mitigate the consequences of accidents at the plant that could lead to the exposure of site personnel and/or the public. Conditions that occur more frequently, such as normal operation or anticipated operational occurrences, should have radiological acceptance criteria that are more restrictive than those for less frequent events, such as design basis accidents or design extension conditions (see para. 4.4 of SSG-2 (Rev. 1.) [24]).

2.30A. The design of the NPP should be such as to ensure that the dose limits for site personnel and the reference levels for public will not be exceeded during accident conditions. It should be demonstrated that the corresponding radiological acceptance criteria, identified in accordance with the dose limits and reference levels and reflected in the design limits, are met in the design. Further recommendations on radiological acceptance criteria for accident conditions are provided in SSG-2 (Rev. 1) [24] (e. g. in paras 4.10, 4.11, 7.31, 7.46, 7.58 and 7.60).

2.31. The optimization process should include not only consideration of the protection of the public but also consideration of the protection of workers and all the safety features of the facility or activity, such as those related to the on-site management of radioactive waste.

The safety guides providing recommendations to meet the requirements of GSR Part 3 [2] for protection of the public and for protection of the environment include the following:

— IAEA Safety Standards Series No. GSG-8 [25], Radiation Protection of the Public and the Environment, which provides guidance on the framework for protection of the public and the environment;

— IAEA Safety Standards Series No. GSG-9 [23], Regulatory Control of Radioactive Discharges to the Environment, which provides guidance on application of the principles of radiation protection and the safety objectives associated with the control of discharges and on the process for authorization of discharges;

— IAEA Safety Standards Series No. GSG-10 [20], Prospective Radiological Environmental Impact Assessment for Facilities and Activities, which describes a framework and methodologies for prospective radiological environmental impact assessment.

2.32. The Safety Guide GSG-8 [25] covers the generic application of the requirements given in GSR Part 3 [2] that relate to the protection of the environment and protection of members of the public in planned exposure situations and existing exposure situations. In addition, GSG-8 [25] also covers the generic application of the requirements given in GSR Part 3 [2] and GSR Part 7 [9] in emergency exposure situations and does not deal with the application of the requirements in GSR Part 3 [2] to specific types of facility or activity or in specific exposure situations. For the information on the IAEA more specific publications see para. 3.53 of GSG-8 [25]. The Safety Guide GSG-10 [20] covers the assessment of the exposure of the public only. The wider aspects of optimization of protection and safety are covered in GSG-3 [26] on the predisposal management of radioactive waste. Optimization of the protection of the public in connection with the establishment of radioactive discharge limits for facilities and activities is described in GSG-9 [23]. The result of a radiological environmental impact assessment, as described in this Safety Guide, is a necessary input into the optimization process to be used for establishing discharge limits.

Specific recommendations related to the design measures for protection of workers are provided in sections 5 and 6 of this Safety Guide.

Minimization of radioactive waste

2.33. Radioactive waste arising during operation and decommissioning should be minimised in accordance with Requirement 8 of GSR Part 5 [7] to protect workers, public and the environment and with Requirement 12 of SSR -2/1 (Rev.1) [1].

Requirement 8 of GSR Part 5 [7] states that:

> "All radioactive waste shall be identified and controlled. Radioactive waste arisings shall be kept to the minimum practicable. "

and Requirement 12 of SSR-2/1 (Rev.1) [1] states that:

> "Special consideration shall be given at the design stage of a nuclear power plant to the incorporation of features to facilitate radioactive waste management and the future decommissioning and dismantling of the plant."

Recommendations on how to meet these requirements are provided in Section 5 and Section 7 of this Safety Guide.

2.34. The design for radiation protection should meet the optimization requirements established by the regulatory body for any persons who are exposed as a result of activities in the nuclear power plant, including activities in the predisposal management of radioactive waste, in accordance with para 2.6. of GSR Part 5 [7] which states:

> "2.6. Requirements for radiation protection have to be established at the national level, with due regard to the BSS [2]. In particular, the BSS require radiation protection to be optimized for any persons who are exposed as a result of activities in the predisposal management of radioactive waste, with due regard to dose constraints, and require the exposures of individuals to be kept within specified dose limits."

2.35. Paragraph 4.6. of GSR Part 5 [7] states that:

> "4.6. Measures to control the generation of radioactive waste, in terms of both volume and radioactivity content, have to be considered before the construction of a facility, beginning with the design phase, and throughout the lifetime of the facility, in the selection of the materials used for its construction, and in the control of the materials and the selection of the processes, equipment and procedures used throughout its operation and decommissioning. The control measures are generally applied in the following order: reduce waste generation, reuse items as originally intended, recycle materials and, finally, consider disposal as waste. "

Section 3 provides recommendations to meet this requirement.

2.36. Paragraph 4.7. of GSR Part 5 [7] states that:

> "4.7. Careful planning has to be applied to the siting, design, construction, commissioning, operation, shutdown and decommissioning of facilities in which waste is generated, to keep the volume and the radioactive content of the waste arisings to the minimum practicable."

Section 4 and Section 6 provide recommendations to meet this requirement.

2.37. In sections 5-7 of this Safety Guide recommendations are provided for the design of radiation protection related measures. In many cases these recommendations are related to technological systems designed to limit and retain the activity in the coolant and to limit the activity release to the environment. These technological systems are described in detail in other relevant IAEA safety guides on design of nuclear power plants, including SSG-52 [27], SSG-53 [28], SSG-56 [29], SSG-62 [30], SSG-63 [31] and SSG-39 [32]. Reference paras. of the related safety guides are indicated accordingly in sections 5-7.

Design targets for operational states

2.38. To ensure that a design both reduces doses to levels that are as low as reasonably achievable and represents best practice, design targets should be set for the individual dose and collective dose to workers and for the individual dose to the representative person of the public. The setting of design targets for individual doses to site personnel and members of the public should be consistent with the concept of dose constraints, which is discussed in GSR Part 3 [2]. The design targets should be set at an appropriate fraction of the dose limits[4]. The term 'target' or 'target dose' is used throughout this Safety Guide in respect of both individual and collective doses.

2.39. In order to focus the design efforts on those aspects of the design that contribute most to the collective and individual doses to the workforce, it is useful to set design targets for the collective dose to the groups of workers that are likely to receive the greatest doses, such as maintenance workers and health physics staff. It is also useful to set design targets for the collective dose for each category of work, such as maintenance of the major components, in-service inspection, refuelling and waste management. These, combined with dose assessments at the key stages of the design, can be used to monitor the major contributions to the dose and to identify aspects that contribute more to the dose than was envisaged initially.

2.40. *Deleted*

[4] It should be recognized that design targets are not limits. They are useful design tools in the optimization process. However, provided that any excess can be justified, they may be exceeded. Also, achieving a design target does not, in itself, demonstrate that a design satisfies the optimization principle. A dose should be reduced below a target if this can be done at a cost that is justifiable.

Design targets for accidents

2.41. The adequacy of the design provisions for the protection of the site personnel and public under accident conditions should be judged by means of the comparison of calculated doses with the specified dose criteria (radiological acceptance criteria; see paras. 4.10 and 4.11 of SSG-2 (Rev. 1) [24]) that constitute the design targets for accidents. In general, the higher the probability of the accident condition, the lower the specified design target should be (see also para. 2.30A). The regulatory body may recognize this principle by setting different design targets for accidents with different probabilities of occurrence. In addition, the regulatory body may define design targets by specifying frequency criteria for all accidents in specified dose bands. Dose criteria for design basis accidents should be specified in accordance with para. 2.16 of this Safety Guide and for design extension conditions in accordance with para. 2.17.

Design targets for decommissioning

2.42. Planning for decommissioning begins at the design stage and continues throughout the lifetime of the facility. Paragraph 7.3 of GSR Part 6 [8] states that:

> "7.3. For a new facility, planning for decommissioning shall begin early in the design stage and shall continue through to termination of the authorization for decommissioning."

Recommendations related to specific design features of radiation protection in design for decommissioning are provided in Section 7 of this Safety Guide.

2.43. Paragraph 1.20. of GSR Part 6 [8] states that:

> "1.20. The management of fresh nuclear fuel and the management of spent nuclear fuel and of radioactive waste generated during the operational phase of a facility are not usually considered part of decommissioning. …"

The management of the radioactive waste generated during the decommissioning should be considered in the decommissioning plan.

2.44. Requirement 1 of GSR Part 6 [8] "Optimization of protection and safety in decommissioning" states that:

> "Exposure during decommissioning shall be considered to be a planned exposure situation and the relevant requirements of the Basic Safety Standards shall be applied accordingly during decommissioning."

2.45. The relevant dose limits for the exposure of workers and for the exposure of members of the public should be applied during decommissioning in accordance with GSR Part 3 [2]. In addition, it is stated in para 2.1. of GSR Part 6 [8]:

"… Radiation protection of persons who are exposed as a result of decommissioning actions shall be optimized with due regard to the relevant dose constraints."

Appropriate design targets for decommissioning facilities and processes based on dose constraints should be derived.

2.46. Paragraph 2.2. of GSR Part 6 [8] states that:

"2.2. In addition to provisions to protect against exposure during planned activities, provision shall be made during decommissioning for protection against, and for reduction of, exposure due to an incident. However, if the incident or the particular situation is of such a nature as to warrant remediation or to require confinement of releases of radioactive material under emergency conditions, other IAEA safety standards apply *[*]*)"

[*] GSR Part 3 [2] and GSR Part 7 [9].

2.47. Para 2.3. of GSR Part 6 [8] requests that:

"2.3. National regulations on the protection of the environment and the requirements of Ref. *[*]* addressing protection of the environment shall be complied with during decommissioning, and beyond if a facility is released from regulatory control with restrictions on its future use."

[*] GSR Part 3 [2].

Recommendations to meet this requirement are provided in Section 7 of this Safety Guide.

3. GENERAL ASPECTS OF RADIATION PROTECTION IN DESIGN

SOURCES OF RADIATION

3.1. The magnitudes and locations of the sources of radiation in operational states and during decommissioning should be determined in the design phase. The main sources that cause radiation exposure in operational states and during decommissioning should be taken into account, including the following:

- the reactor core, reactor internals and vessel;
- the reactor coolant and moderator system;
- the steam supply system, feedwater system and turbine generators;
- the waste treatment systems;
- irradiated fuel;
- spent fuel pool;

- the fuel handling and storage system;

- decontamination facilities;

- ventilation systems and

- miscellaneous sources such as sealed sources that are used for non-destructive testing.

Special attention should be given at preventing direct exposure from the largest radiation sources, such as the reactor core, irradiated fuel and spent resins sources during design.

Recommendations on assessing radiation sources under operational states are provided in Section 4 and Section 5.

3.2. The magnitudes, locations, possible transport mechanisms and transport routes of the sources of potential radiation exposure under accident conditions should also be determined in the design phase of the plant. Requirements and recommendations on the safety assessment to be carried out during the development of the design and for the final assessment are given in SSR-2/1 (Rev. 1) [1], GSR Part 4 (Rev. 1) [6] and SSG-2 (Rev. 1) [24].

3.3. The main source of radiation under accident conditions consists of radioactive fission products, activation products and actinides, for which precautionary design measures should be adopted. These are released either from the fuel elements or from the various systems and equipment in which they are normally retained. Recommendations on assessing radiation sources under accident conditions are provided in Section 6. In Annex I examples of methods for assessing radiation sources for selected accidents are described. The presented accidents are selected for illustrative purposes and cover all the major categories of designs for nuclear power plants with light water reactors, CO_2 cooled reactors with UO_2 metal clad fuel, heavy water reactors and reactors with on-load refuelling.

3.4. The source term for a release of radioactive material to the environment should be evaluated for operational states and accident conditions as recommended in paras 2.16-2.19 of SSG-2 (Rev. 1) [24] to demonstrate that the design ensures that national requirements for radiation protection, including restrictions on doses, are met.

DESIGN APPROACH FOR PLANT STATES AND DECOMMISSIONING

Design team and operating experience

3.5. The design teams should have a common understanding of safety and of safety culture, security culture and the significance of radiation risks and hazards for safety. Managers of design teams should advocate a collective commitment to safety by design teams and individuals in accordance with the Requirement 12 (Fostering a culture for safety) of GSR Part 2 [5]:

> "Individuals in the organization, from senior managers downwards, shall foster a strong safety culture. The management system and leadership for safety shall be such as to foster and sustain a strong safety culture."

Concerning security, managers of design teams should also advocate a collective commitment to security culture in accordance with IAEA Technical Guidance No. 28-T, Self-assessment of Nuclear Security Culture in Facilities and Activities [33].

3.6. The design team should be aware of the radiological protection measures that should be incorporated into the design.[5] A key issue is that the experts from relevant operating organizations should be effectively involved in the design of new plants and design modifications to an existing plant, to assist in ensuring that the requirements for radiation protection and waste management are met. Moreover, the applicable operating experience should be transferred to the design organization. In this way the interrelation between design aspects and operational procedures can be properly taken into account.

3.7. The optimization of radiation protection should be carried out at all stages of the lifetime of equipment and installations, from design and construction to operation and decommissioning. A structured approach should be taken to the radiation protection programme and the radioactive waste management programme to ensure the coherent application of the optimization principle. More related recommendations are provided in GSG-7 [10] and SSG-40 [11].

3.8. In order to implement this structured approach, the design organization should have an optimization culture[6] in which the importance of radiation protection is recognized at each stage of the design. An optimization culture is established by ensuring that all participants in a project are aware of the general requirements for ensuring radiation protection and of the direct and indirect impact of their individual activities or functions on the provision of radiation protection for site personnel and members of the public.

3.9. More specifically, an optimization culture should be established on the basis of:

— Knowledge of the practices that result in the occupational exposure of site personnel and members of the public;

— Achieving good feedback of operating experience to the design team;

— Familiarity with the main factors that influence individual and collective doses;

[5] There are various ways of ensuring that persons involved in the design are fully aware of the radiological protection measures that should be incorporated into the design, such as by having experts in radiological protection document the requirements and provide training. It may be appropriate to include an experienced operator in the design team.

[6] An optimization culture may be explained as a system of shared knowledge, common objectives and attitudes that ensures that the management of occupational exposure and the exposure of members of the public benefit from the cooperation of all personnel involved in a project.

— Familiarity with the analytical methods that are available to assist in the optimization of the design;

— Recognition that specialists in radiation protection are to be consulted whenever necessary to ensure that aspects that will have implications for radiation protection are properly evaluated and taken into account in the design.

3.10. Specialists in radiation protection should be closely involved in the design process to provide support with:

— Expertise in all areas that affect the production of radioactive material and its transport in the plant and the dispersion of radionuclides in the environment;

— Ability to evaluate the different sources of radiation in the plant and the resulting doses using the best available analytical methods and data from relevant operating experience;

— Familiarity with the relevant regulations, guidance and best practices;

— Familiarity with maintenance, in-service inspection and other work in high radiation areas that make a major contribution to the radiation exposure of site personnel.

3.11. Due to the importance of chemical parameters in controlling the radioactive sources in the plant, specialists in reactor chemistry should also be involved in the design process. Materials specialists should be involved in controlling the source term due to corrosion products. This also refers to the chemical decontamination processes that are performed during operation and in the decommissioning phase.

Organizational aspects

3.12. The requirement to achieve an adequate level of radiation protection affects a wide range of issues associated with the design. It is necessary therefore to ensure that for all design related decisions that may affect exposures the recommendations of radiation protection specialists have been recorded. However, the design process should be planned so that the implementation of these recommendations is not on the critical path. Means should be provided of ensuring that the designers take into account the required radiation protection measures at every stage of the design process. Such means could include:

— Rules or prescriptions for the layout and design of the plant;

— Design measures to minimize the use of respiratory protection;

— Checklists for use by engineers that can be reviewed by radiation protection specialists.

3.13. The project should be organized to enable the following:

— Radiation protection specialists within the design organization should be consulted at the early stages of the design when options for the major aspects of the design are being evaluated. It may also be appropriate to consult with specialists from external organizations.

— The design should incorporate good engineering practices that operating experience has shown to be effective in reducing exposure; deviations from such practices should be accepted only when a net benefit has been demonstrated.

— Radiation protection specialists should review all decisions that may have a major influence on exposures.

— There should be an appropriate forum for proposing improvements and resolving disputes that may occur between design engineers and radiation protection specialists.

3.14. A systematic and structured management system programme should be applied in the entire design process as recommended by DS513 revision of GS-G-3.1 [34].

3.15. A strong management commitment should be made to ensure that an optimization procedure is effective. In some organizations, this commitment includes the appointment of a manager for optimization who is directly responsible to the senior manager of the design project and thereby is involved in the decision making process.

Design strategy

General approach

3.16. Design targets should be set at the start of the design process, and should include:

— Annual collective dose targets and individual dose targets (e. g. for average and maximum dose) for site personnel;

— Annual individual dose targets for members of the public.

Specific dose targets may be established for operations carried out when the reactor is in operation and for operations carried out during outages. The dose calculations including methods and tools should be subject to review and approval by the regulatory body.

3.17. In practice, these design targets can be addressed independently from each other, although in principle any enhancement of waste treatment systems to reduce the releases of radioactive material to the environment may result in additional work being carried out by site personnel with a consequent increase in their exposures. In providing the best practicable means for reducing releases, the implications for the exposures of site personnel should be taken into account to ensure that there is no undue increase.

3.18. In setting these design targets, account should be taken of experience at relevant plants that have a good operating record in terms of radiation protection, and the targets should be subject of review and approval of the regulatory body. Account should be taken of any differences in the design, operations or policies between these reference plants and the plant under design. Such changes might include the power level, the materials that are used for the primary circuit, the type of fuel, the burnup, the extent of load following, the reactor coolant chemistry, the extent to which the reactor may operate with failed fuel and the extent to which on-load access to the containment is planned for.

3.19. A simple illustration of the use of design targets is given in Fig. 1 for the design of a plant that is based on earlier plant designs. In the initial stages of the design process, design changes are introduced to ensure that the design targets will be achieved. However, achieving the design targets does not ensure that doses will be reduced to levels that are as low as reasonably achievable, and further development of the design may be necessary to ensure that radiation protection is optimized.

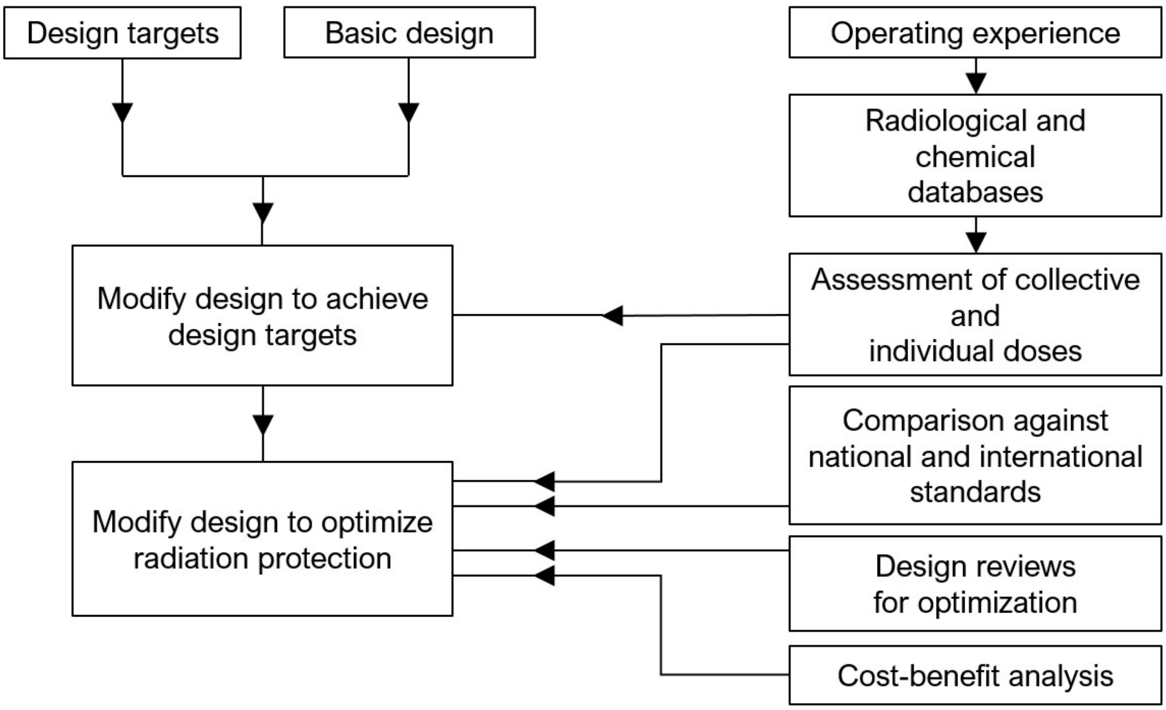

FIG. 1. Strategy for the optimization of radiation protection in the design of a nuclear facility

Radiation protection design for site personnel

3.20. The following procedure should be adopted for developing the design to ensure the radiation protection of site personnel:

(1) The general requirements for the plant should be developed and documented. These should include the principles on which the layout of the plant will be based and restrictions on the use of specific materials in the design of the plant. These documents form part of the management system for the design (see GS-G-3.5 [35]).

(2) A strategy for controlling exposures should be developed so that the most important aspects are considered early in the design and in a logical order. For example, in many reactor

designs, two areas in which there is a major potential for reducing exposures are scheduled and unscheduled maintenance. For example, in some designs of pressurized water reactors, two of the plant items that are important contributors to radiation exposures during maintenance are the steam generators and valves in systems containing radioactive coolant. These should therefore be considered first in the stage of design, and it should be ensured that the design has been proven. This will reduce exposures to levels that are as low as reasonably achievable and will also help to improve the plant availability and therefore the economic performance of the plant.

The second area that should be considered is design features that minimize the production and build-up of radionuclides, since reducing these will reduce radiation and contamination levels throughout the plant, whereas a local solution, such as increasing the shielding and improving ventilation, will have only a local benefit. Subsequently, local plant features should be considered, such as the plant layout, the shielding and the design of systems and components. An example of a simplified strategy for a pressurized water reactor is shown in Fig. 2.

(3) A logical layout for the plant should be developed that is divided into zones based on predicted dose rates and contamination levels, access requirements and specific requirements such as the need to separate safety trains.[7] The dose rates may be calculated by using the source terms that are the basis for the radiation protection aspects of design (see Annex I), or they may be based on operating experience from similar plants provided that any changes in the relevant design and operating parameters are not significant. The zoning should be consistent with national legislation and regulatory requirements. It may be adequate to use the same definition of the zones as will be used when the plant is operating but it is found in many cases that a more specific definition of the zones is necessary for design purposes. Examples are given in Annex II.

(4) The maintenance programme and operational tasks should be defined, preferably on the basis of well established concepts. The number of staff for each task should be based only on the operational requirements and should not be artificially increased, to comply with the regulatory requirements related to the-dose constraints. For tasks for which doses are predicted to be relatively minor, the work can be expressed generically in terms of the number of person-hours that will be spent in each radiation zone. The type of worker who will perform each task is also identified. Types of workers include maintenance personnel,

[7] The term ‘safety train’ refers to a set of plant components that perform a safety function, such as an emergency core cooling pump and its associated equipment and source of water.

in-service inspection personnel, support staff (e. g. scaffolders), decontamination staff and health physics staff.

(5) Collective and individual doses should be evaluated by combining the results of steps 3 and 4. The use of a database is recommended. The maximum use should be made of relevant operating experience, where available, particularly for work that is difficult to predict such as unplanned maintenance.

(6) The proposed procedure is shown in the schematic flow chart of the factors that determine individual and collective doses in Fig. 3. This procedure is repeated at each significant stage of the design, and the level of detail should increase as the design is developed. At each stage, the doses that are evaluated should be compared with the design targets for each type of work.

(7) At each step in Fig. 3, where there are options for the design, optimization studies should be performed. This is particularly important in cases for which it is predicted that the design targets will be exceeded.

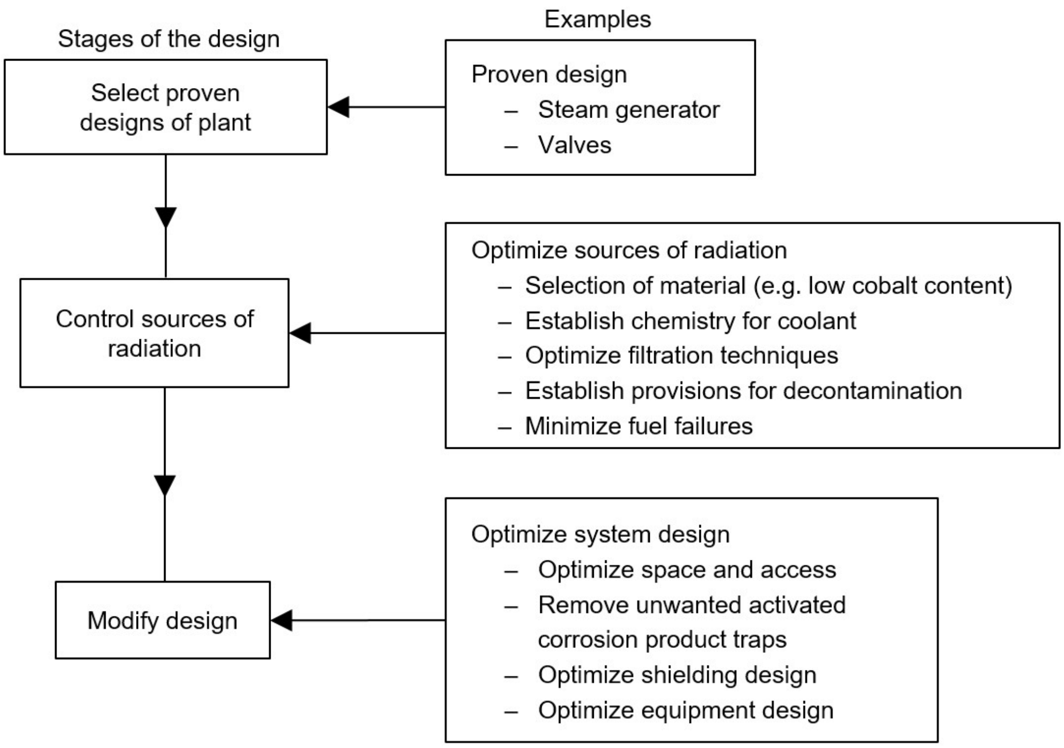

FIG. 2. A simplified strategy for the reduction of exposures in the plant.

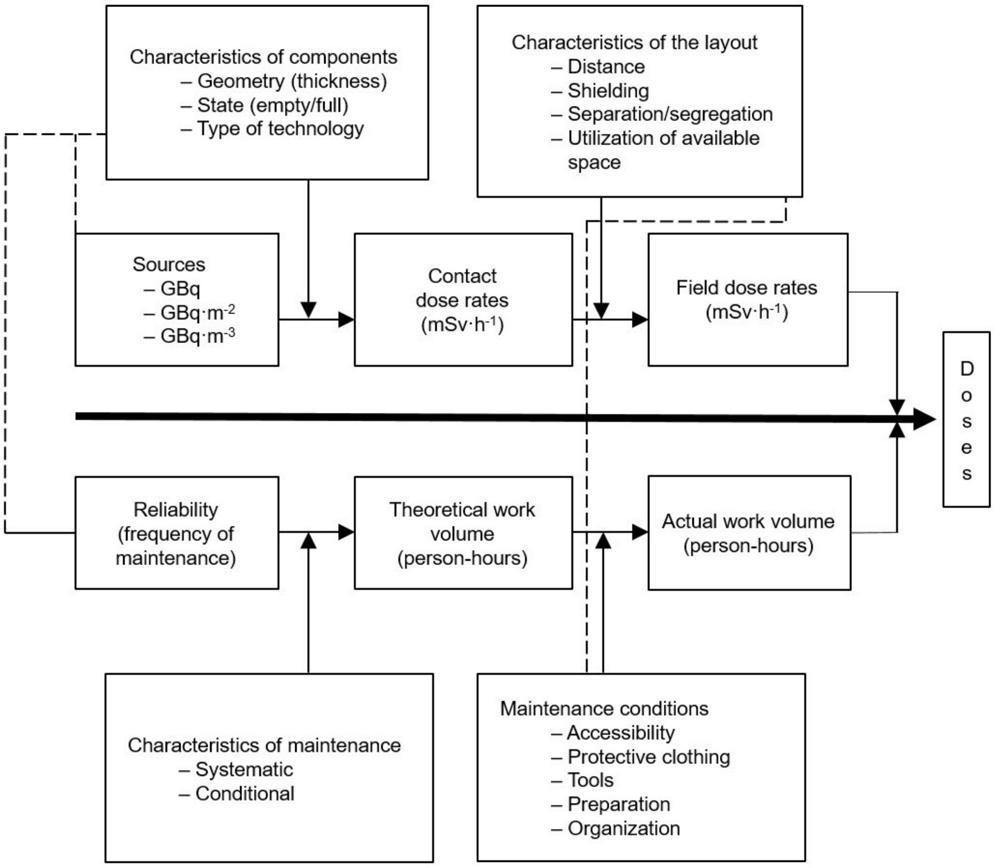

FIG. 3. Schematic flow chart of the origin of doses at a plant (dashed lines indicate the possible impact of certain blocks on others).

3.21. Thus, the procedure is iterative, as is illustrated in Table 1.

TABLE 1. AN EXAMPLE OF THE PRACTICAL IMPLEMENTATION OF THE STRATEGY FOR THE DESIGN PROCESS

Step[a] ⇓	Design targets		Optimization process	Dose rates		Individual and collective doses
Item[b] ⇒	Individual dose target	Collective dose target	Studies to be performed	Zoning	CDR[c]	EWV[d]
Step 1[e]	Average for all workers	Total for facility	Description of advantages/ drawbacks of options	(Not relevant)	(Not relevant)	Estimation of EWV with options
Step 2	Update step 1 value	Update step 1 value	Evaluate main options	Establish approxi-mate zoning	(Not relevant)	Programme definition EWV estimation
Step 3	Definitive value for the average for all workers	Evolution with decisions of step 2	Limited to important points	Evaluate using DST/ RST/AST[f]	Calculate CDRs	Estimation of EWV
Step 4	Values for each craft type[g]	Evolution	Detailed by tasks	Verifi-cation/ precision	Verifi-cation/ precision	Detailed evaluation of EWV

[a] Steps: The design of a complex project for which studies extend for several years is commonly divided into steps. The level of detail in the studies increases with the step number.
[b] Item: The line of the table indicates the main parameters that are to be considered.
[c] CDR: contact dose rate.
[d] EWV: exposed work volume.
[e] The information given in this line of the table is as follows: during step 1, an average dose constraint will be set (all craft/trade types included) as well as a collective dose target, including a margin; the optimization studies will result in a list of advantages and drawbacks of options; no zoning will be performed or contact dose rate calculations made; the exposed work volume will be estimated, with account taken of different options (the work is performed by workers or by robots).
[f] AST: accident source term; DST: design source term; RST: realistic source term.
[g] 'Craft' may be termed 'trade' in some States.

3.22. In pressurized heavy water reactors, for which an important contributor to exposures is the inhalation of airborne tritium, a logical layout of the plant should be developed that is divided into zones on the basis of levels of airborne radionuclides.

3.23. An auditable record should be kept of all the decisions made in the course of the design process and the reasons for those decisions, so that each aspect of the design that affects exposure to radiation is justified. This is part of the management system for the design.

3.24. A preliminary decommissioning plan should be developed to ensure that the design includes the necessary features to reduce and control exposures during decommissioning. In many cases, these features are the same as those necessary for operational states, but some additional special features may be necessary for decommissioning. If these are major, the necessary features for operational states and those for decommissioning should be optimized.

3.25. The design should be such as to facilitate achievement of the targets for occupational doses — both individual doses and collective doses — by adopting some or all of the following measures:

(1) Reduction of dose rates in working areas by:

— Reduction of sources (e. g. by the appropriate selection of materials; reducing surface and airborne contamination; decontamination measures; the control of corrosion, water chemistry, filtration and purification; and the exclusion of foreign material from the primary systems);

— Improvement of shielding;

— Increase in the distance between workers and sources (e. g. by the use of remote handling);

— Improvement of filtered ventilation, especially in pressurized heavy water reactors.

(2) Reduction of occupancy times in radiation fields by:

— Specifying high standards of equipment to ensure very low failure rates;

— Ensuring ease of maintenance and removal of equipment;

— Removing the necessity for some operational tasks by, for example, providing built-in auxiliary equipment and making provision in the design for permanent access;

— Ensuring ease of access and good lighting;

— Optimizing the number of workers and their time in the radiation field by design means.

Design for radiation protection for members of the public

3.26. As indicated in para. 2.38, design targets should be set already in the site evaluation at the start of the design process for annual individual doses to members of the public. Possible developments in the area surrounding the site and likely future population distributions should be taken into account as necessary.

3.27. The design targets should be achieved in the following way:

— Site specific features that affect the doses to members of the public should be identified at an early stage of the design process and taken into account in the design (see GSG-10 [20]). This should include the identification of the representative person and the exposure pathways for the representative person, which should be subject to the approval of the regulatory body.

— One possible approach would be to set targets for radioactive releases for which account is taken of operating experience and the use of best practicable means in the design of the treatment systems for radioactive effluents.

— The resulting doses to the representative person should be evaluated to ensure achievement of the target.

3.28. Activity monitors should be installed at the gates to ensure that no radionuclides leave or unintentionally enter the plant.

Design for radiation protection for the environment

3.28A. The maximum radionuclide concentrations that may be present in relevant local flora and local and migratory fauna, as well as the internal dose that may result from those concentrations, should be assessed through considerations of the exposure pathways for non-human biota. The radiological acceptance criteria established for humans are generally conservative with regard to the protection of other species and, from the radiological point of view, the protection of non-human biota is generally achieved by protecting the human population (see para. 1.33. of GSR Part 3 [2]). Further recommendations on the protection of non-human biota are provided in GSG-8 [25], GSG-9 [23] and GSG-10 [20].

3.29. A radiological environmental impact assessment should be carried out in accordance with NS-G-3.2 [19] and GSG-10 [20] to inform the optimization process being applied to doses to members of the public and to ensure that the design complies with national regulatory requirements and appropriate dose targets.

Design for radiation protection with integration of safety, emergency preparedness and security measures

3.30. The design of radiation protection measures for operational states and decommissioning should take into account the robustness of the engineered safety features and operational features, and also fire protection and emergency preparedness measures in accordance with the Requirement 8 of SSR-2/1 (Rev. 1) [1] (see para. 2.19 of this Safety Guide). Arrangements for an emergency included in the emergency plan should be taken also into consideration during the review and update of the radiation protection design measures for emergency response in the light of experience gained, in accordance with para. 5.4 of SSR-2/2 (Rev. 1) [3]. Recommendations on the measures related to the emergency preparedness are provided in Section 6. See also relevant recommendations of GSG-2 [36].

3.31. To reduce the risk of release of radioactive material in a fire appropriate measures and special arrangements should be made in accordance with SSG-64 [37] and design measures and special arrangements should be implemented for the radiation protection of firefighting personnel and the management of releases to the environment in accordance with para. 5.23 of SSR-2/2 (Rev. 1) [3].

3.32. In accordance with the recommendations indicated in para. 2.19A of this Safety Guide the design of radiation protection measures should consider nuclear security guidance provided in IAEA Nuclear Security Series Implementing Guides NSS No. 10-G (Rev.1) [38], NSS No. 11-G (Rev. 1) [39], NSS No. 19 [40], NSS No. 25-G [41] and NSS No. 27-G [42].

3.33. Measures provided in the design for emergency arrangements and for radiation protection should be appropriate for maintaining safety in the event of an accident; mitigating the consequences of accidents if they do occur; protecting site personnel and the public; protecting the environment in accordance with para 5.2 of SSR-2/2 (Rev. 1) [3]. Appropriate arrangements should be established before the nuclear fuel is first brought into the site and crosses the plant fences. The emergency plan and all emergency arrangements should be completed before the commencement of fuel loading.

Use of operating experience

3.34. The feedback of operating experience in operational states and decommissioning from reactors of similar design should be used to identify the best practices concerning radiation protection and to improve the design for NPPs in accordance with Requirement 24 of SR-2/2 (Rev. 1) [3] and also with recommendations provided in SSG-50 [43].

DESIGN CONSIDERATIONS FOR COMMISSIONING AND OPERATION

Commissioning

3.35. The measures included in the design to provide an optimized level of radiation protection for operational states should be adequate for addressing the requirements for the commissioning phase (in which radiation levels are generally lower because of the lower power levels and the low buildup of radioactive material in the plant's components).

3.36. Measures should be taken during the early commissioning phase to identify any design deficiencies, such as the shielding being inadequate to prevent streaming, so that these can be rectified before the reactor reaches full power operation.

3.37. Test programs to verify the adequacy of shielding should be developed during the design phase. Further recommendations about the commissioning programme are provided in SSG-28 [44].

Outages

3.38. Collective doses at NPPs are mainly received during the outage due to activated corrosion product contamination of the primary circuit, hence specific design features should be considered to ensure that occupational exposures during outages are optimized.

Platforms and lay-down areas

3.39. Provisions for platforms required for safe work should be included in the design as well as storage of scaffolding and temporary shielding materials inside the containment. Enough space for lay-down areas is necessary for maintenance activities to assure easy access and to reduce time spent in the radiation area. Material selection and surface treatment should assure easy decontamination.

Shielding

3.40. Temporary shielding provisions such as structural support or supporting system design features should be considered for outage and maintenance work on primary system components which cannot be shielded permanently. Equipment requiring maintenance during outages should be placed in low dose rate areas where practicable. Shielding should be provided between individual components that constitute substantial radiation sources to help maintenance and inspection personnel servicing other specific components in the area.

Fuel pools and sumps designs

3.41. Fuel pools design should assure easy decontamination of the pools and fuel transfer canals, particularly if this is required for inspection or maintenance of fuel transfer equipment. Design of filtering and cleaning systems should take into account the need to achieve doses that are optimized during maintenance. Specific shielding should be considered in the design for the optimization of the dose rate in the peripheral area of the reactor building during fuel transfer from the reactor building to the spent fuel pool.

3.42. Recommendations of SSG-15 (Rev. 1) [45] regarding the design, commissioning, operation and safety assessment of the different types of spent nuclear fuel storage facilities (wet and dry), considering different types of spent nuclear fuel, should be taken into account. In addition, recommendations of SSG-15 (Rev. 1) [45] related to radiation protection for handling and storage of spent fuel should also be taken into account.

Access to and exit from controlled areas

3.43. Provisions should be considered in the design for efficient access and exit control points and facilities, such as monitoring and registration of personnel accessing the plant-controlled area with personnel protective equipment's. Contamination monitors should also be considered in exit control logistics during outages, periods having increased staffing.

Ease of maintenance

3.44. Where possible, flanged connections should be provided on liquid systems for quick disconnections and access for hydrolysing. Electrical quick disconnections should be used in design to minimise maintenance time. Components should be designed to facilitate draining, flushing, cleaning and decontamination by mechanical or chemical means. Components located in radiation areas should be designed for quick removal and installation (e. g. overhead lift points). Piping, equipment, insulation and shielding should be designed for quick removal and replacement. Ensure valves located inside high radiation areas have sufficient space for maintenance.

Design considerations for initial startup

3.45. Radiation protection infrastructure should be available sufficiently before the planned introduction of radioactive sources or fuel in order to fully establish the radiation protection programme and to ensure that all radiation monitoring equipment is tested and functioning correctly, as recommended in paras 3.33, 3.48, 3.61, 4.28, A-2, A-3 and A-14 of SSG-28 [44].

3.46. Chemistry parameters for initial start-up need to be established, with radiation protection considerations included in the optimisation process as this can have a significant effect on reactor source term later in operation. (See para. 5.19. of SSG-13 [46]).

3.47. During the commissioning phase, surfaces should be preconditioned before and during initial start-up in order to produce a protective layer and to ensure appropriate, passivated surfaces in all systems. The protective layer will reduce the subsequent release of corrosion products into the coolant when the plant is at power and hence will reduce the deposition of radioactive material.

Design considerations for start-up and shut down

3.48. Where oxygenation of the coolant circuit as a result of refuelling would lead to the release of corrosion products into the coolant, such as in a pressurized water reactor, the design should enable a controlled release of activated corrosion products in the primary circuit, while the coolant pumps are still running and the cleanup system still operational, by deliberate oxygenation of the circuit by chemical means.

3.49. Consideration should be given to increasing the capacity of the cleanup system specifically for shutdown, to minimize activated corrosion products in the reactor coolant.

Further information on chemical control during start-up and shut down for different reactor types is given in SSG-13 [46].

3.50. Consideration should be given to the maintenance requirements of systems and components at the design stage to minimise maintenance requirements, particularly for those items in high dose rate areas of the plant.

DESIGN CONSIDERATIONS FOR ACCIDENT CONDITIONS

3.51. The principal design measures that are taken to protect the public against the possible radiological consequences of accidents are required to have the objectives of reducing the probability that accidents will occur (prevention of accidents) and reducing the source term and releases (mitigation of consequences) associated with accidents if they do occur (see SSR-2/1 (Rev. 1) [1]). Accident prevention is not explicitly addressed in this Safety Guide, but reference should be made to the available relevant information (e. g. see TECDOC-1127 [47]; ICRP, Publication 103 (2007) [48]; ICRP, Publication 64 (1993) [49]; para. 3.120 of GSR Part 3 [2]; paras 5.44 and 5.45 of GSG-10 [20]; INSAG-9 [50]; and INSAG-12 [51].

3.52. The design objectives for accident conditions are to limit exposures and radioactive releases to acceptable levels and to optimize:

(1) the risks to the public from possible releases of radioactive material from the nuclear power plant; and

(2) the risks to site personnel from these releases and from direct radiation exposure.

These design objectives should be achieved by means of high quality design and special features, such as safety systems and other safety features, that are incorporated into the design of the plant. Achievement of the design objectives should be confirmed by means of a safety analysis. Deterministic safety analysis and the associated dose assessments, complemented by probabilistic safety assessments, for demonstrating compliance with the radiation acceptance criteria should be based on conservative assumptions for the analysis of design basis accidents and realistic or best estimate assumptions for the analysis of design extension conditions. These issues are discussed in Section 6 of this Safety Guide and in SSR-2/1 (Rev. 1) [1], GSR Part 4 (Rev. 1) [6] and SSG-2 (Rev. 1) [24].

3.53. To achieve the design objectives mentioned, the necessary design provisions and procedures (e. g. for access to the control room, maintenance of essential equipment or process sampling) should be such as to enable the plant operators to manage the situation adequately in an accident. Detailed recommendations on how to protect site personnel under accident conditions are provided in Section 6.

3.54. Practices that are similar to those used for operational states should also be used to ensure that proper plant design is achieved to provide adequate radiation protection for site personnel and the public under accident conditions. A safety culture should be established to ensure that safety matters are given the highest priority and that regulatory requirements on releases of radioactive material under accident conditions are met with adequate margins. A security culture should also be established to ensure that security matters are given due priority in accordance with IAEA Implementing Guide No. 7 [52].

3.55. To achieve the proper design of plant systems and components for radiation protection under accident conditions experts in radiation protection, plant operations, plant design and accident analysis, and regulatory matters should be involved in all stages of design process. There should be continuous interactions among these groups throughout the design process to arrive at a design that provides radiation protection under accident conditions which is acceptable to the regulatory body. The design should also ensure that effective procedures for accident management can be implemented in accordance with the SSG-54 [53].

DESIGN CONSIDERATIONS FOR DECOMMISSIONING

3.56. The design for a nuclear power plant should also ensure that the plant can be safely decommissioned as it is required in Requirement 6 of SSR-2/1 (Rev. 1) [1].

3.57. Long life SSCs that could be radioactive should be designed for the lifetime of the facility, to a reasonable extent, to avoid the necessity to replace the SSCs and lessening the potential for leakage and contamination.

3.58. It should be considered for the plant design throughout the lifetime of the nuclear power plant that the operator is responsible of the adequate maintenance of the documentation to facilitate future decommissioning of the plant, as requested in para. 3.6 (h) of SSR-2/1 (Rev. 1) [1].

3.59. The design should take due account of provision for managing the radioactive waste that will be generated in the decommissioning of the plant as requested in para. 4.8 of SSR-2/1 (Rev. 1) [1].

3.60. The design for a nuclear power plant should also ensure that the generation of radioactive waste and discharges are kept to the minimum practicable in terms of both activity and volume also in decommissioning phase as requested in para 4.20 of SSR-2/1 (Rev. 1) [1].

3.61. The routes for removing large items of plant during decommissioning should be planned at the design stage and the necessary provisions should be incorporated.

3.62. Remote techniques may play a major part in the removal of the most radioactive items during decommissioning. The use of such techniques should be considered at the design stage and it should be ensured in the design that their use is not precluded. It is likely that there will be improvements in remote control techniques over the lifetime of the plant. The best practicable techniques that are available when the work is carried out should be used.

3.63. Monitoring for purposes of radiation protection is required for both plant operation and decommissioning, and the general provisions discussed in Section 7 are also valid for decommissioning. However, in the later stages of decommissioning, some of the initial monitoring equipment may have been removed or become unnecessary or different measures for monitoring may have become necessary by virtue of the decommissioning activities. The design of the monitoring system should therefore be reviewed before each stage of decommissioning begins.

4. CONTROL OF SOURCES OF RADIATION AND ESTIMATION OF RADIATION DOSE RATES

4.1. General description of main sources of radiation is provided in paras 3.1 to 3.3. Practical examples on identification of sources of radiation are provided in Annex I.

ESTIMATING RADIATION DOSE RATES DURING PLANT OPERATION AND DECOMMISSIONING

4.2. Recommendations on estimating radiation doses during operation and decommissioning are provided in this section in accordance with the scope of this Safety Guide, e. g. no recommendations are provided on calculational methods or values of the parameters to be used to evaluate the radiation dose rates expected to occur during operation and decommissioning. Other recommendations about occupational radiation protection considering external exposure and internal exposure as well as skin and lens contamination are provided in GSG-7 [10].

4.3. The first step in any calculation of external dose should be to evaluate the source intensity and its distribution. This may involve making calculations concerning the transport of radionuclides and their redistribution when activated corrosion products or fission products are carried in the reactor coolant and deposited away from the point of origin. The second step is to calculate the fluence rate (flux) at the dose point as a result of radionuclide transport from the source to the dose point and to calculate the radiation dose rate by multiplying the flux by the appropriate conversion factors. Other recommendations on the assessment of external exposure are provided in paras 7.1–7.132 of GSG-7 [10].

4.4. During the design phase, the requirement to provide equipment to assess internal exposures should be considered. The calculation of internal exposure of workers is based on the duration of exposure at the workplace, the atmospheric activity, including activity coming from surface activity of the deposit (use of resuspension factor), the radionuclides involved, the particle size distribution, the breathing rate and dose coefficients factors. Recommendations on the assessment of internal exposure are provided in paras 7.133–7.227 of GSG-7 [10].

SOURCE CATEGORIES FOR NORMAL OPERATION AND DECOMMISSIONING

4.5. The sources of radiation during normal operation and decommissioning should be identified. Ways in which they arise are briefly described in Annex I of this Safety Guide.

4.6. The sources may be grouped into five categories that affect potential exposure in different ways, and which should thus be taken into account in different ways in the design. In general terms, these are:

(a) Those sources for which the design of the shielding will be determined;

(b) Those sources for which shielding is not practicable and that may be major sources of doses to workers during plant operation;

(c) Those sources that are major sources of doses to workers during decommissioning;

(d) Those sources that present special hazards to workers during plant operation, such as small particles containing alpha emitters or with high concentrations of activated cobalt;

(e) Those sources that are important contributors to doses to members of the public during plant operation.

In some cases, one type of source may belong to more than one category.

SOURCES AND PROPAGATION OF RADIATION: SPECIFIC SHIELDING DESIGN

The reactor core and its surroundings

4. 7. The major source of radiation in an operational plant is the reactor core and the surrounding materials that are activated by neutrons that escape from the core.

4. 8. An initial step in evaluating source intensities is to determine the fission rate, the neutron and gamma emission rate, and the spatial and energy distribution of the neutron and gamma flux within the core. This may be achieved by using computer codes in which account is taken of the spatial distribution of materials in the core and changes in fuel composition, the production of actinides and fission product poisons, and changes in control poisons (due to the positions of control rods, the heights of liquid moderators and poison concentrations) with fuel burnup. The neutron and gamma emission rate and neutron and gamma flux distributions that are calculated for the core are used as input data for computer calculations to determine the neutron and gamma flux energy and spatial distributions through the coolant and the structural and shielding materials surrounding the core.

4. 9. The primary sources of radiation should be determined by using the procedures discussed in the references indicated in Annex I.

Reactor components

4. 10. Depending upon the design, many of the components within the reactor vessel are regularly removed and become sources in locations outside the vessel. These include the spent fuel, control rods, neutron sources, in-core instruments and the internals of the reactor.

4. 11. The source terms for all these components that are used for the design basis for the shielding should be based on the maximum activities that could occur over the lifetime of the plant. This is likely to be for the maximum rated fuel assembly and the end of life activity for the other components.

Activity of the coolant

4. 12. When evaluating the source terms due to radioactive material that is released into, transported in and deposited from the primary coolant, the following should be considered:

— Corrosion products;

— Fission products;

— Activation products.

For most types of reactor, corrosion products are the major contributors to radiation levels at the plant during shutdown and thus to the occupational exposure of personnel. In some pressurized water reactors, for example, the activation of 10 g of ^{59}Co and 5 kg of ^{58}Ni in primary circuit components gives rise to 90% of the dose rates and occupational exposure at the plant[8]. Therefore, accurate modelling of the source term for corrosion products is an important factor in optimizing the design. One important activation product is ^{16}N, which is a high energy gamma emitter with a half-life of 7 s and is a major source of radiation when the reactor is at power.

Radiation propagation through shielding

4. 13. Calculations should be carried out for the propagation of radiation (mainly gamma rays) from the sources through simple, single material bulk shielding, or through shields of complicated geometry containing regions of low density (gases and voids) and low attenuation that present preferential transmission paths with scattering surfaces. The complexity of the geometry and the nature of the radiation should determine the type of calculation approach that will be required.

4. 14. In the design of shielding to achieve acceptable dose rates, the calculation for determining attenuation is begun with a design that is estimated on the basis of previous experience. The results should be evaluated in the light of the principle of the optimization of protection with regard to the site personnel and should then be compared with limiting values established for maintaining the integrity of materials, with any radiation effects taken into account. If necessary, the process should be repeated to achieve acceptable radiation levels.

Sources for which shielding is not practicable

4. 15. Some tasks have to be carried out in situations in which the provision of shielding is not practicable. Examples are work in the water chambers of pressurized water reactors steam generators and the removal of insulation from, and the in-service inspection of, the primary circuit pipework of light water reactors. In these cases, the design should be such as to ensure (a) that the work can be carried out as rapidly as practicable and (b) that there is provision for the use of remotely operated equipment, as discussed in paras 5.34 – 5.37.

Sources that dominate decommissioning doses and waste volumes

4.16. The sources of radiation that contribute to doses received during decommissioning are the activation products in the components of the core and the surrounding materials, contamination in the primary and auxiliary circuits, and the accumulation of active material at the plant.

4.17. It should be noted that in a well designed and operated reactor, the major radiation source during decommissioning will be the activation products in and near the core. The important radioisotopes will be those that have a half-life of a few years or more. In many cases, the most important radioisotopes for external exposure that will remain radioactive for dozens of years after shutdown will be ^{60}Co and ^{137}Cs. ^{60}Co arising from impurities in the steel and this will dominate until ^{63}Ni in the steel becomes

[8] This applies for reactors for which a nickel-based alloy is used for steam generator tubing.

important. In this case, the control of impurity levels that is exercised to control the magnitude of this source during operation will also be effective in controlling it during decommissioning. ^{137}Cs as a fission product with long half life will typically stay over the lifetime of the plant and is mainly controlled by quality of fuel elements and filtering of the cooling water. For internal exposure, the most relevant radioisotopes include ^{3}H and ^{90}Sr.

4.18. During decommissioning the magnitude of the source term can affect both the doses to workers and the volume of radioactive waste that is generated. Where appropriate, concrete inside of radiologically controlled areas should be sealed during plant operation to facilitate cleaning and decontamination. Consideration may also be given to sealing concrete to prevent the release of radon gas. The source term in this case may be dominated by radionuclides that are not very important during operation, such as the rare earth isotopes, and control of such impurities may be an important aspect of the design process.

4.19. Considering possible defects in the cladding, the primary and auxiliary circuits may be contaminated by alpha emitters. The amount of irradiated fuel deposited on surfaces may reach a few tens of grams[9]. For such situations, the risk of internal exposure by alpha emitters is a special hazard during maintenance, operation and decommissioning, and the relevant precautions, such as providing breathing protection, should be taken.

Special hazards during operational states and decommissioning

4.20. A special hazard during operational states and decommissioning can be what are generally designated as ‘hot spots’. Hot spots result from the activation of small objects present in the coolant. These objects may be:

— Particles of metal resulting from unusual wear of components and/or fuel assemblies;

— Debris left in the primary circuit or other circuits connected to it;

— Pieces of thick deposits on the fuel.

4.21. The activity concentration for such hot spots will depend on the material and on the activation time. They usually move from circuit to circuit according to the transfer of water. The dose rates generated by these sources are of the order of some tens of mSv/h to a few hundreds of Sv/h on contact. Hot spots are especially relevant for outage and decommissioning.

Sources that are important contributors to doses to members of the public

4.22. As discussed in Annex I, it should be noted that the notion of ‘important contributors to doses’ is relative.

[9] In pressurized water reactors that have been subject to the ‘baffle jetting’ phenomenon, the amount of irradiated fuel may reach a few hundred grams.

4.23. The important contributors to doses to members of the public during operational states and design basis accidents as well as during decommissioning are typically:

— ^{14}C, ^{3}H and ^{85}Kr, because the best practicable means available for their removal by waste treatment systems are not efficient and because their half-lives are long;

— ^{41}Ar is an important contributor, although its half-life is short, because it is released in large volumes of air (in venting of the containment during operation for some pressurized water reactors);

— ^{133}Xe is a weak gamma emitter but it may be of importance when the reactor has been operating with a significant number of defects in the fuel cladding;

— Iodine, caesium and corrosion products.

It should be noted that noble gases, Iodine and other radionuclides with short half life, are only relevant for a limited time span after fuel removal from core or final shut down of the reactor.

4.24. The guidance on the assessment of the dose to the public due to the discharges resulting from the normal operation and against potential exposures are given in GSG-10 [20]. The guidance on the control of discharges and the process for authorization for discharges related to the radiation exposure of the public are described in Section 5 and Annex 1 of GSG-9 [23].

4.25. Consideration should be given to the design features for protecting site personnel from the radiation that results in operational states and during the decommissioning of a nuclear power plant, and the means of implementing the system of dose limitation as described in the SCHEDULE III of GSR Part 3 [2] and in GSG-7 [10]. Recommendations given in SSG-47 [54] about the implementation of the system of dose limitation should also be taken into account during design.

5. SPECIFIC DESIGN FEATURES OF RADIATION PROTECTION IN DESIGN FOR OPERATIONAL STATES

PLANT LAYOUT

5.1. This section is linked to para 2.6 of this Safety Guide and focuses recommendations to meet the design requirements on radiation protection established in SSR-2/1 (Rev. 1) [1] for the structures, systems and components of a nuclear power plant related to the operational states.

As stated in paras 6.72, 6.74 and 6.75 of SSR-2/1 (Rev. 1) [1]:

> "6.72. The plant layout shall be such as to ensure that access of operating personnel to areas with radiation hazards and areas of possible contamination is adequately controlled, and that exposures and contamination are prevented or reduced by this means and by means of ventilation systems. "

"6.74. The plant layout shall be such that the doses received by operating personnel during normal operation, refuelling, maintenance and inspection can be kept as low as reasonably achievable, and due account shall be taken of the necessity for any special equipment to be provided to meet these requirements. "

"6.75. Plant equipment subject to frequent maintenance or manual operation shall be located in areas of low dose rate to reduce the exposure of workers."

5.2. Assessments made for radiation protection design for planned exposure situations should be carried out in accordance with Requirement 9 of GSR Part 4 (Rev. 1) [6][10].

5.3. In the design, a careful assessment should be made of the access requirements for operation, inspection, maintenance, repair and replacement of equipment. The layout of the plant should be designed to facilitate these tasks and to limit the exposure of site personnel.

Classification of areas and zones

5.4. The requirements for the classification of areas as controlled areas and supervised areas are established in Requirement 24 of GSR Part 3 [2]. Each controlled area should have the minimum practicable number of access and exit points for personnel and for materials and equipment.

5.5. Provision should be made for controlling accesses to and exit(s) from the controlled areas and for monitoring persons and equipment leaving the controlled areas. Exit doors should have an interlock with the contamination monitors to avoid uncontrolled exit of contaminated persons or equipment. In addition, means for disabling the interlock during the evacuation should also be provided.

5.6. Designers should consider the requirement for controlled areas to be divided into zones on the basis of the anticipated radiation levels and radioactive contamination levels (i. e. dose rates and activity concentrations for surface or airborne radionuclides; see Annex II) in order to enable the Licensee to apply effective management controls during operation. The greater the radiation or contamination related risks of a zone, the greater is the need to control access to that zone for the purpose of ensuring compliance with individual annual dose limits and taking account of dose constraints.

5.7. In the plant design stage, all rooms should also be classified into planning zones on the basis of their likely dose rates, surface contamination levels, concentrations of airborne radionuclides and occupancy in normal operation and during accident conditions. This approach can assist in layout design by providing a means of easily identifying areas of the plant that may be either suitable or unsuitable for routing high activity systems in accordance with the Requirement 81: Design for radiation protection of SSR-2/1 (Rev.1) [1]. These zones may not necessarily correspond exactly

[10] It has to be noted that that this Section concerns planned exposure situations as defined by GSR Part 3 [2].

with the controlled areas in use during plant operation.

5.8. Consideration should be given to the possibility that it may be necessary during operation or planned maintenance to reclassify certain areas temporarily or permanently. In this regard, particular attention should be paid to the planning of access routes. Under such conditions the zones and the controlled areas should be re-evaluated.

Changing rooms, changing areas and related facilities

5.9. Within the controlled area, changing areas should be provided at selected places to prevent the spread of contamination during normal operation, including maintenance. The facilities included in these areas should correspond to the requirements for access to the potentially more contaminated of the two areas and its anticipated contamination levels.

5.10. Where justified by the possible levels of air contamination, consideration should be given to the provision of permanent changing areas with decontamination facilities for personnel, monitoring instruments and storage areas for protective clothing.

5.11. Within the changing rooms, a physical barrier should be provided to separate clearly the clean area from the potentially contaminated area. The changing rooms should be large enough to meet the needs during periods of maintenance work, including adequate space for storage of personal protective equipment and sufficient exit monitors. Sufficient space should be provided so that cross contamination between personnel is avoided. Allowance should be made for temporary personnel employed as contractors, for example during outage periods.

Control of access and occupancy

5.12. Safety, security and monitoring requirements for access to and exit from controlled areas should be considered together in order to deliver optimised technical solutions, for example by the use of electronic personal dosimeters in accordance with Requirement 24: Arrangements under the radiation protection programme and Paras 3.88 - 3.90, 3.97 of GSR Part 3 [2].

5.13. The access by personnel to areas of high dose rates or high levels of contamination should be controlled by the provision of lockable doors and, where appropriate, the use of interlocks designed in accordance with SSG-39 [32]. Interlocks are provided to ensure that access is only possible when radiation levels are acceptably low and their design should include an alarm that will be activated if they become inoperative. It should be noted that systems provided to protect workers from doses above authorized limits may require classification in accordance with SSG-30 [55].

5.14. The routes for personnel through radiation zones should be minimized to reduce the time spent in transit through these zones.

5.15. To minimize the radiation doses to personnel working in the controlled area and the spread of contamination, the layout of the controlled area should be so designed that personnel do not have to pass through areas of higher radiation zones to gain access to areas of lower radiation zones. The feedback of operating experience with reactors of similar design should be used to provide guidance concerning radiation levels and contamination levels.

5.16. As far as practicable, the design should be such as to limit the possible spread of contamination and to facilitate the erection of temporary containments.

5.17. The design should be such that the occupancy time necessary in radiation areas for the purposes of maintenance, testing and repair should be consistent with the principle of optimization of radiation protection. This can be achieved, for example, by:

(1) Provision of passageways of adequate dimensions for ease of access to plant systems and components. In areas where it is likely that site personnel will have to wear full protective clothing, including masks with portable air supplies or connections to a supply by air hoses, account should be taken of this in deciding on the dimensions of the passageways.

(2) Provision of clear passageways of adequate dimensions to facilitate the removal of plant items to a workshop for decontamination and repair or disposal.

(3) Provision of adequate space in the working areas, to carry out repairs or inspections, for example.

(4) Provision of easy access to high radiation areas such as pressurized water reactors steam generator internals and the valves in the systems that contain primary coolant.

(5) Provision of ‘waiting areas’ in low radiation areas.

(6) Placement of components that are likely to be operated frequently, or to require maintenance or removal, at a convenient height for working.

(7) Provision of ladders, access platforms, crane rails or cranes in areas where it can be foreseen that they will be required to permit the maintenance or removal of plant components. Features to facilitate the installation of temporary shielding should be included in the design.

(8) Use of computer aided design models to optimize aspects of the design that affect working times. Video or photographic records should be made during the construction of the plant to facilitate the planning of work in areas of high radiation levels during operation and thus to shorten working times.

(9) Provision of means for the quick and easy removal of shielding and insulation where this is necessary to perform routine maintenance or inspection.

(10) Provision of special tools and equipment for facilitating work to reduce exposure times.

(11) Provision of remotely controlled equipment.

(12) Provision of a suitable communication system for communication with the site personnel working in radiation areas or contamination areas.

(13) Provision of electrical supplies that are easy to disconnect and reconnect to equipment and provision of sufficient availability of sockets.

Interlock controls that disable access should be considered for areas where dose rates can be temporarily high such as in core instrumentation areas.

OTHER DESIGN CONSIDERATIONS FOR AN EFFECTIVE OPERATIONAL RADIATION PROTECTION PROGRAMME

System Design

5.18. The design of nuclear power plant systems should be based on the feedback of experience gained in reducing radiation exposure at operating reactors.

5.19. The following measures for reducing radiation exposure should be adopted in the system design:

(1) The work space in a zone of high radiation levels around components that require regular maintenance should be shielded from the radiation from other systems;

(2) Non-radioactive components that do not have to be mounted close to active components should be installed outside areas of high radiation levels;

(3) Methods for sampling radioactive liquids with minimal exposure should be provided. Automated methods should be used where reasonably achievable;

(4) Methods for countermeasures (e. g. flushing) to avoid the sedimentation of radioactive sludge in piping and containers should be provided.

(5) For mutual standby systems, if it is necessary to repair one system while the other system is in operation, sufficient shielding should be set between the two systems.

To avoid worker's external exposure, materials containing sealed radioactive sources (e. g. radioactivity measuring devices) that may present a hazard, should preferably be stored in dedicated rooms or areas and not in places where workers are passing through.

5.20. Pipelines containing radioactive fluids should not be located near clean piping and they should be located at a suitable distance from items that need maintenance. Sufficient space for making inspections as well as repairs and modifications should be left between the pipelines and the walls.

5.21. The uncontrolled buildup of particles containing radioactive material should be prevented by means of appropriate design for fluid flow and chemistry control and also by the use of piping with a smooth and even inner surface.

5.22. Pipelines should be so designed that few venting and drainage lines are needed. Drainage should lead to a sump or a closed system. Pipelines should be designed to avoid causing fluid to collect in places.

5.23. In the design of pipelines, welded seams requiring inspection should be avoided to the extent practicable and any such seams should be readily accessible.

5.24. In the design of the coolant circuit and auxiliary circuits, traps where fluid can stagnate and where activated corrosion products can collect should be avoided as far as possible. The total number of joints, and therefore welds, should be kept to a minimum to reduce the number of inspections required.

5.25. Drains should be positioned so that no residual pockets of liquid are left when a circuit is drained. However, the design of a circuit for radioactive liquid should minimize the number of drain points, since high levels of contamination can arise as a result of the stagnant pocket of water in the drain line when the circuit is full and in operation. Provision should be made for the draining and flushing of tanks to reduce radiation sources.

5.26. In direct cycle reactors, the design of the steam drying system should be such as to ensure that the levels of radiation and surface contamination in the turbine building are low when the reactor is shut down (radiation levels are very high in direct cycle reactor turbine buildings during operation due to ^{16}N in the steam phase of the primary coolant).

Component design

5.27. Some general considerations apply in the design of components to take into account the requirements for radiation protection. Many of these considerations are the same as those that apply in system design.

5.28. Components of high reliability that require minimum surveillance, maintenance, testing and calibration should be used to minimize radiation exposure. Remote surveillance should be used where possible.

5.29. The components to be used in areas of high radiation levels should be designed to be easily removable.

5.30. Exposure of site personnel should be reduced by minimizing the possible amount of radioactive material in plant components. Traps and rough surfaces where radioactive particulates could accumulate should be avoided as far as practicable.

5.31. Components and areas of buildings that may become contaminated should be designed for ease of decontamination by either chemical or mechanical means. This should include providing smooth surfaces, avoiding angles and pockets where radioactive material could collect, and providing means of isolation, flushing and drainage for circuits that contain radioactive liquid.

5.32. Components whose maintenance and repair could result in an exposure that is a significant fraction of the relevant annual limit on the collective dose should be well separated.

5.32A. Consideration should be given to the use of seal-less canned reactor coolant pumps to reduce doses due to the maintenance of the seals and to the incidence of losses of coolant resulting from seal failures.

Design for communications infrastructure

5.33. The design should help facilitate the use of wireless devices such as installed and personal monitoring equipment, cameras and remotely operated robots and drones. Electromagnetic compatibility and computer security requirements and signal transmission requirements to areas behind thick biological shielding should be considered in the specification for equipment (see IAEA NSS No. 17-T (Rev. 1) [56], Computer Security Techniques for Nuclear Facilities, Technical Guidance).

Remote techniques

5.34. Remote techniques should be used wherever practicable to minimize the exposure of personnel. Techniques that should be considered include arrangements for remote inspection and for the removal and reinstallation of equipment. These techniques should be considered in the design. Techniques for the inspection and handling of plant items may be only semi remote, in that personnel may still have to enter the controlled area to install equipment on rigs. An example of remote or semi-remote techniques is the provision of equipment for the ultrasonic inspection of welds. Access to the weld may be necessary in order to fit the scanner, but the operator can then move to a low radiation area to operate the equipment.

5.34A. Consideration should be given to incorporating filters and demineralizers within concrete cells, and thus not accessible, together with shielded transport containers to enable relatively high dose rates to accrue on the filters, and thus to minimize radwaste and worker dose. Consideration should also be given to two trains of the coolant clean up filters, to allow continued clean up during oxygenation, while the other filter is removed.

5.35. Criteria for the selection of remotely rather than manually operated equipment such as valves should be established in order to make the 'as low as reasonably achievable' (ALARA) selection process more efficient. This should include consideration of ambient dose rate as well as frequency of use for normal operation.

5.36. Access to high dose rate areas during operation at power for surveillance and maintenance should be avoided wherever practicable. For remote visual inspection, consideration should be given to the use of radiation resistant / tolerant cameras and windows shielded by lead glass or comparable materials.

5.37. Automation aids such as multi stud tensioning devices should be used during refuelling operations. Hands on operations should be avoided wherever possible and occupancy of areas adjacent to fuel routes should be minimised.

Shielding

5.38. Permanent shielding should be part of the design and temporary shielding should be avoided where practicable. For those few locations where permanent shielding cannot be installed, permanent storage locations should be provided for temporary shielding at locations that will minimise handling of shielding.

Design of shielding

5.39. In designing a shield for a specific radiation source, the target dose rate should be set, for which account should be taken of the expected frequency and duration of occupancy of the area. Account should also be taken in setting this target dose rate of the uncertainties associated with the source term and with the analysis made to determine the expected dose rate.

5.40. In establishing specifications for shielding, account should be taken of the buildup of radionuclides over the lifetime of the plant.

5.41. After the potential intensity of the source has been assessed, the process of shielding design should be carried out iteratively, starting with the design of shields without penetrations (e. g. for pipes, cables and access ways). Next, consideration should be given to the necessary penetrations through the shielding, such as those for pipes, cables and access ways, and the provision to be made to maintain the effectiveness of the shielding for the protection of site personnel.

5.42. The choice of materials for a shield should be made on the basis of the nature of the radiation (whether beta and bremsstrahlung, neutrons and gamma rays, or gamma rays only are produced), the shielding properties of materials (e. g. their degree of scattering, absorption, production of secondary radiation, activation), their mechanical and other properties (e.g. stability, compatibility with other materials, structural characteristics, toxicity, disposability, decontaminability), and space and weight limitations.

5.43. Losses in shielding efficiency may occur as a result of environmental conditions. Effects that should be taken into account are those due to the interactions of neutron and gamma rays with the shielding (e. g. the burnup of radionuclides that have a high neutron absorption cross-section, radiolysis and embrittlement), those due to reactions with other materials (e. g. erosion and corrosion by the coolant), and temperature effects (e. g. the removal of hydrogen and/or water from concrete).

5.44. Neutron shielding should be provided for neutron sources such as the reactor core and irradiated fuel. Neutron shielding should also be provided for unirradiated mixed oxide fuel. Additional recommendations about radiation protection aspects in the design of fuel handling facilities are provided in SSG-63 [31] and SSG-70 [57].

5.44A. Neutron transport calculations around containment should be undertaken to eliminate shine paths.

5.45. A combination of materials may be necessary to obtain an optimum design of shielding for the core or for other sources of neutrons. A material, such as water or concrete, containing elements of low atomic number reduces the energies of neutrons for which the cross-sections are below the cross-section threshold for nuclear inelastic scattering of the shielding material(s).

5.46. When neutrons are captured in the shielding, the gamma rays that are emitted as a consequence of the capture should be absorbed. Concrete is commonly used for bulk neutron shielding outside the reactor pressure vessel. In general, the design for neutron shielding should be such that there are no significant levels of neutron radiation in the areas of the plant to which personnel have access.

5.47. For shielding gamma radiation with a broad energy range, shields with the same mass per unit area provide approximately the same attenuation of a gamma ray flux, particularly at higher energies. The use of materials of a high density and high atomic number, such as lead or tungsten should be considered where space is restricted. Otherwise, concrete may be used; its effective density can be increased by the use of special aggregates and additives.

5.48. In relation to the formation of voids during construction, consideration should be given in the design to the application of an appropriate management system programme to facilitate the construction of the shielding in such a way that voids, or low-density areas will be avoided.

5.49. In the design of permanent shielding, account should be taken of relevant external hazards, in particular seismic forces, in accordance with the recommendations provided in SSG-67 [16] and SSG-9 (Rev. 1) [17], and of relevant internal hazards, in accordance with SSG-64 [37].

5.50. In areas where temporary additional shielding may be necessary in operational states of the plant, account should be taken in the design of the weight of the additional shielding and the provision necessary for transporting and installing it.

5.51. Where reactor coolant is used for shielding purposes (e. g. sufficient water coverage of spent fuel in spent fuel pools), or assumptions made about the shielding effect of the reactor coolant on occupational exposure, automatic sensors and controls should exist for ensuring that levels of the liquid stay within permitted ranges.

5.52. The provision for shielding that is incorporated into the design to protect site personnel during plant operation from direct or scattered radiation should also be designed to ensure adequate protection of the public during plant operation. The design should consider adequate shielding to prevent sky shine as well as ground shine.

Penetrations through the shielding

5.53. Penetrations through the shielding introduce pathways by which neutrons and gamma radiation can propagate preferentially. Whether the primary source is a source of neutrons or of gamma radiation, the following basic means of controlling dose rates due to penetrations are the same and should be applied:

(1) Minimizing the area and number of straight-through paths containing material of very low density (e. g. gases, including air);

(2) Placing the penetrations in areas of low occupancy or no personnel access in the shine path e. g. at height.

(3) Providing shielding plugs;

(4) Providing zigzag or curved pathways to guarantee no line of sight though shields. In this case, width or density of the shielding may be increased near the penetration to compensate for the loss of material due to the penetration;

(5) Filling the gaps with grouting or other compensatory shielding material.

5.54. An optimization study should be carried out to select the approach to shielding penetrations that will reduce occupational exposure to levels that are ALARA, as recommended by GSG-7 [10].

5.55. In some cases, depending on the intensity and location of the source with respect to the penetration, no additional shielding features may be necessary. In other cases, plugs or labyrinths of complex design should be incorporated and computer based shielding calculations may be made to justify the design. Labyrinth structures should be used to avoid duct streaming, noting that streaming may occur when shielding materials are used in combination, for example streaming of gamma radiation through low atomic number materials.

5.56. Points for access by personnel to areas of high radiation levels are a particular case of shielding penetrations for which the dimensions of the penetration are large compared with the thickness of the shielding. In determining the provision to be made for shielding access ways, account should be taken of the magnitude of the source and the limiting dose rate value outside the area containing the source. A labyrinth or wall shield should generally be used such that only a minor amount of scattered radiation can reach the entrance to the area.

Ventilation

5.57. A dedicated active ventilation system should be provided for maintaining appropriate clean conditions in work spaces within the controlled area as described in ISO 26802: 2010 [58]. General and specific considerations for such active ventilation systems are provided in SSG-62 [30].

5.58. For the purposes of radiation protection, the primary objective of providing a ventilation system should be to control the contamination of the working environment by airborne radionuclides and to reduce the need to wear respiratory protection.

5.59. Both the spread of contamination and the amount of releases to the environment should be limited by providing features such as air cleaning filters and by maintaining appropriate pressure differentials.

5.60. The efficiency of filter systems in gas cleanup equipment should be established in the design basis. To ensure that efficiency remains above the design limit, the design should allow for suitable periodic tests and /or ongoing measurements such as sampling the air from upstream and downstream of the filter system. Pressure differentials on filter systems should be monitored as well.

5.61. Elements (e. g. filters, fans) of cleanup equipment should have adequate redundancy in compliance with the safety classification of the relevant system, to ensure their reliability during maintenance and replacement of filter media.

5.62. Arrangements and instrumentation for safe replacement and transportation of filter waste from cleanup equipment should be provided. These arrangements include for example using of filter holders providing easy detachment of spent filters and transportation of spent filters in containers ensuring necessary shielding.

5.63. In addition to radiological hazards, non-radiological hazards posed by the leakage of primary coolant such as combustion in the case of liquid sodium[11] or asphyxiation in the case of CO_2 should be taken into account in the design basis of ventilation systems.

5.64. In relation to radiation protection, the ventilation system should also provide suitably conditioned air to ensure the comfort of personnel, and maintain appropriate environmental conditions required to ensure reliability of plant equipment.

5.65. In designing a ventilation system to control airborne contamination, account should be taken of the following:

— Mechanisms of thermal and mechanical mixing;

— The limited effectiveness of dilution in reducing airborne contamination;

— Exhausting of the air from areas of potential contamination at points near the source of the contamination;

— The use of exhaust rates that are commensurate with the potential for contamination in the area;

[11] Possible combustion of leaked sodium coolant as well as design based means for its suppression should be taken into account in the design basis of systems venting rooms, where sodium containing equipment are located.

— The need to ensure that the exhaust air discharge point is not close to an intake point of the ventilation system.

5.66. The airflow in the ventilation system should be such that the pressure in a region of lower airborne contamination is higher than the pressure in a region of potentially higher contamination. Thus the airflow in the ventilation system should be directed from regions of lower airborne contamination to regions of higher contamination and air should be extracted from the latter. The airflow should be such as to minimize the resuspension of contamination. The pressure of rooms located in controlled areas should be maintained below atmospheric pressure to prevent radioactive releases into the atmosphere in operational states.

5.67. Portable ventilation systems (fans, filters and tents) should also be used in areas where airborne contamination may arise during maintenance, and provision should be made for sufficient space in which to operate such systems.

Decontamination

5.68. The need for decontamination should be considered at the design stage. If it is considered that a worthwhile reduction in radiation exposure would result, the necessary provision for decontamination facilities should be made. Both routine and non-routine decontamination should be considered. Decontamination processes should be optimized using automation (e. g. use of robotic) where this is reasonably achievable[12].

5.69. When decontamination facilities are being planned, all components that are expected to come into contact with coolant or radioactive waste material should be considered as possible items for decontamination.

5.70. Special consideration should be given to rooms where leaks or spills of contaminated liquid might occur. These areas should be designed to allow easy decontamination (e. g. by means of a special coating on floors) and control of the spread of contamination. Adequate bunding and sloping of these rooms should be arranged to limit the contaminated areas and for the quick draining and collection of spilled liquids.

5.71. The system of active floor drains should be extended to all rooms where there are systems that contain radioactive fluids. The rooms should be so designed that the floor channels and slopes are capable of draining design basis leaks in a controlled manner to systems intended for active fluids. The system of active floor drains should be designed to avoid flooding in the event of clogged sumps or insufficient suction. The effects of changes in room temperature and pressure should be considered in the design of the system of active floor drains. The sumps or the rooms should be provided with liquid

[12] Related insights are provided in IAEA TECDOC 1946, Decontamination Approaches during Outages in Nuclear Power Plants - Experiences and Lessons Learned [59].

level detectors that actuate a high level alarm. Recommendations on the design of alarms are provided in Section 8.

5.72. The floor drain system should include filtration to prevent an excessive amount of particulates entering the subsequent water treatment systems.

5.73. There should be an adequate tank volume so that any temporary transfers of radioactive water do not burden systems that are intended for other purposes. The tank volume should also be sufficient to ensure that any releases of liquid radioactive effluent to the environment will remain small.

5.74. The coatings and/or the lining of fuel storage pool and fuel handling pools, as well as the equipment used in these areas, will become contaminated. When the water level in such pools is lowered, surfaces may dry out, and the dispersal of material on the surfaces into the air may cause a hazard due to airborne radioactive material. Systems should be provided for decontaminating such surfaces before they dry out. Systems should also be provided for decontaminating, before they dry out, fuel transport flasks and components that may have to be removed from the pools for repair.

5.75. Provision should be made for periodic on-line chemical decontamination of the active system circuits, including the installation of filters or ion exchange columns for the purposes of such decontamination. Further information on reactor coolant clean-up is given in Annex 1.

5.76. Decontamination facilities should be provided for removing radioactive material from the surfaces of casks and packages (e. g. transport containers for irradiated fuel elements or waste packages) before shipment, from components that may need to be repaired and from tools and equipment.

5.77. Provision should be made for the decontamination of personnel and of reusable protective clothing.

5.78. Drains from the decontamination facilities should connect to the treatment systems for radioactive effluent.

Waste treatment systems

5.79. The equipment in treatment systems[13] for solid, liquid and gaseous radioactive waste may contain radioactive material in high concentrations, and radiation protection from this material should be provided for site personnel. An estimate should be made of the expected radionuclide content in treated waste, and of the consequent maximum radiation level that can arise in each area of the waste treatment system. Consideration should be given to the sources that give rise to the highest radiation levels (such as ion exchange resins, discarded radioactive components and filter waste). In the assessment of the sources and the estimation of radiation levels, account should be taken of the changes in the activity

[13] Requirements and recommendations on the management of radioactive waste before it is sent to a repository are established in IAEA Safety Requirements and Safety Guides GSR Part 5 [7], SSG-40 [11] and SSG-41 [12].

concentration of waste that can occur as a result of treatment, particularly increases in the activity concentration (e. g. for incinerator ash or compressed waste).

5.80. The design should be such as to minimize the deposition of resins and evaporation concentrates in the piping and components of the waste treatment system, as well as their crystallization and deposition in tanks.

5.81. The design of waste treatment systems should incorporate features to reduce the likelihood of leaks. Special attention should be paid to preventing the leakage of resin and concentrates from the tanks. Features should be incorporated to ensure that any leaks are promptly detected. In the tank rooms, either each tank should be surrounded by a bund wall that could retain a volume of fluid of the capacity of the tank, or the walls of each room should be readily decontaminable up to the height that would be flooded if the leak were not isolated. In order to prevent the leakage of radioactive liquid waste, bolted-flange tanks should be avoided where possible.

5.82. The design should provide features to minimize exposure during replacement of filters and ion exchange resins. In particular, the design should be such that it is possible to carry out by remote control reverse flow flushing, washing, regeneration and change of resins.

Storage of spent fuel and radioactive waste at the plant

5.83. Facilities should be provided for the safe storage of the radioactive waste that arises at the plant, with account taken of its form (solid, liquid, gas or a mixture), its radionuclide content and its nature in terms of the extent to which it has been processed. The safe storage of waste will depend in part on the design, construction, operation and maintenance of the facility concerned. The design features of facilities should be such that the radioactive waste can be received, handled, stored and retrieved without causing undue occupational or public exposure or environmental effects. Further recommendations on this subject are provided in SSG-40 [11].

5.84. The design of storage facilities for spent fuel and radioactive waste should incorporate the following functions:

(a) Maintaining the confinement of stored materials;

(b) Maintaining subcriticality (in spent fuel storage facilities);

(c) Providing for radiation protection (by means of shielding and contamination control);

(d) Providing for the removal of heat (from spent fuel);

(e) Providing for ventilation, as necessary;

(f) Allowing the retrieval of the spent fuel (or ‘irradiated fuel’) and radioactive waste for transport off the site.

5.85. The storage facility should provide protection for the waste to prevent degradation that could pose problems for operational safety during its storage or upon its retrieval. It should be ensured that the shielding and confinement functions of the storage facility, including the containers, are fulfilled throughout the facility lifetime. This should be achieved by means of design features, the selection of appropriate materials, and ageing management programme (see para. 5.16 of SSG-48 [60]) and maintenance and repair (see SSG-74 [61]) or replacement, with account taken of the following:

(a) Chemical stability against corrosion caused by processes acting within the waste and/or external conditions;

(b) Protection against radiation damage, especially stability under conditions of the degradation of organic materials and damage to electronic devices;

(c) Resistance to impacts caused by operational loads or due to incidents and accidents;

(d) Resistance to thermal effects, if applicable.

(e) If the waste contains nuclides emitting high-energy beta rays (e. g. ^{90}Y), bremsstrahlung should be taken into account in vessels with higher atomic numbers.

5.86. Consideration should be given to the possibility of changes in the stored waste, which could lead to:

(a) Generation of hazardous gases caused by chemical and radiolytic effects (for example, the generation of hydrogen gas caused by radiolysis) and the buildup of overpressure;

(b) Generation of combustible or corrosive substances;

(c) Acceleration of the corrosion of metals (in particular, mild steel).

5.87. The possibility of accidents should be taken into account in the design of storage facilities. The resulting features can differ from, and should be complementary to, those designed for normal operation.

5.88. In addition to radiological hazards, non-radiological hazards (for example, fire or explosion), which may contribute to radiologically significant consequences, should also be considered in the design of storage facilities in accordance with SSG-64 [37].

5.89. Where appropriate, equipment should be provided with suitable interlocks or physical limitations to prevent dangerous or incompatible operations. Such interlocks or limitations should prevent undesirable movement (for example, the movement of waste that gives rise to high dose rates into an area occupied by site personnel or vice versa).

5.90. The need for remote handling should be considered in cases where the waste container gives rise to high dose rates or where there is a risk that radioactive aerosols or gases could be released to the working environment.

5.91. Any remote handling devices should be designed to provide means for their maintenance and repair, for example, by the provision of a shielded service room, to keep occupational radiation exposures as low as reasonably achievable.

PROTECTION OF THE PUBLIC DURING PLANT OPERATION

Discharge criteria

5.92. To protect the public from radiological consequences due to the operation of the plant, plant operators are required to ensure that doses to members of the public arising from radioactive substances in the effluents and from direct radiation due to the plant do not exceed the prescribed limits, and that the optimization principle is applied (GSG-8 [25] and GSG-9 [23]). This is requested by the Requirement 30 (Responsibilities of relevant parties specific to public exposure), Requirement 31 (Radioactive waste and discharges) and Requirement 32 (Monitoring and reporting), of GSR Part 3 [2]. The design should be such that regulatory limits for discharges will not be exceed and as low as reasonably achievable. This is commonly done by specifying discharge limits for the most significant radionuclides, as described in para. 3.37 of this Safety Guide.

5.93. Whenever possible, the discharge limits may be set on the basis of operating experience. However, a careful analysis should be made of the operating experience so as to take into account possible differences in the design of similar units, such as in the types of alloys in contact with the primary coolant. Such differences are likely to influence the nature and activity of the discharges. In the case of some radionuclides, such as ^{14}C and ^{3}H, practicable techniques for their removal are not readily available. However, in making use of operating experience in setting discharge limits for these radionuclides, account should be taken of the variations in production rates for reactors of similar designs. In addition, when radioactive discharges are very low, the monitoring process used may have a strong influence on the interpretation of the operating experience.

5.94. Three types of effluents should be considered: liquids (mainly aqueous), gases from process systems and ventilation air.

5.95. Minimisation of the frequency and impact of events, that have the potential to increase public dose should be considered.

5.96. Public doses due to all exposure pathways, including direct shine, from the plant should be calculated based on conservative assumptions about public occupancy around the plant. Requirements for site evaluation of nuclear plants vary from country to country and so relevant assumptions should be made. Studies should be carried out to ensure that shielding is optimised to ensure offsite exposure is ALARA.

Waste streams source reduction

5.97. The design measures that are taken to control the sources of radioactive material in the plant so as to protect the site personnel will also affect the activity of the waste streams and discharges. However, some radionuclides should be given greater consideration in terms of protecting the public than in terms of protecting site personnel. The isotopes of iodine, for which an operating limit should be specified, are an example. If this operating limit is exceeded for a specified period, the reactor should be brought into an appropriate state to prevent unacceptable public radiation exposures. In practice, such limits are usually determined by the requirement to limit the consequences of postulated events such as fuel failure or a steam generator tube rupture, rather than the release limits for operational states, for which the removal of iodine from waste streams can be achieved by means of the waste management systems. The basis for this derivation should be clearly established, with consideration given to the capacity of the waste treatment system and the authorized discharge limits as well as to remaining within the design basis for accidents and to operational radiation protection.

Effluent treatment system

5.98. The flows and the activity concentrations of liquid and gaseous effluents need to be monitored and controlled to ensure that the regulatory discharge limits are not exceeded (SSR-2/1 (Rev. 1) [1]). Liquid and gaseous treatment facilities that are based on the best practicable means should be provided, as discussed in the following subsections. Safety guides GSG-10 [20] and GSG-9 [23] provide recommendations and information on the calculation of the exposure of the public resulting from radioactive discharges.

Liquid treatment systems

5.99. The major sources of contaminated water that require treatment include: primary coolant that is discharged for operational reasons; floor drains that collect water that has leaked from the active liquid systems and fluids from the decontamination of the plant and fuel flasks; water that is used to backflush filters and ion exchangers; leaks of secondary coolant; laundries and changing room showers; and chemistry laboratories. The foregoing are essentially aqueous in nature and the guidance that follows is given on this basis. Where non-aqueous liquid waste is generated in sufficient volumes, the provision of a separate waste treatment system to deal with it should be considered. Further guidance on the treatment of aqueous and non-aqueous liquid waste is provided in SSG-40 [11].

5.100. Proven methods of treating the radioactive waste water to reduce radioactive contamination use mechanical filtration, ion exchange, centrifuges, distillation or chemical precipitation. The different treatment processes in the liquid waste treatment system should be connected so as to give the operator sufficient flexibility to deal with liquids of different origins and unusual compositions, and to re-treat water if the authorized low activity for discharge is not attained after the initial treatment. In the case of direct cycle reactors such as boiling water reactors, which generally produce larger volumes of radioactive water resulting from leakage from the turbine circuit, water that is of low chemical and solid content is recycled to the primary circuit after suitable treatment. The same recycling is a good practice for non-aerated primary coolant in pressurized water reactors but, in practice, the discharge of primary coolant may be necessary to control the levels of airborne tritium in the plant. In addition, radioactive water may be present in the secondary (turbine) circuit of a pressurized water reactor as a result of operating with some primary circuit to secondary circuit leakage in the steam generator. In this case, treatment of the water from the secondary circuit may be necessary to reduce the activity before the water is discharged. N-16 monitoring equipment installed in the secondary coolant system is effective in detecting the leakage of cooling water from the primary system to the secondary system in pressurized water reactors.

5.101. For water that cannot be recycled into the plant, provision should be made to reduce its radioactive contamination to such levels that the design target doses and discharge limits discussed in Section 2 are met. If necessary, reduction of the radionuclide content of the water can be achieved by means of several passages of the water through the liquid waste treatment system.

5.102. Consideration should be given to the amount of solid waste that is produced by the liquid waste management systems. The volumes of liquid that require treatment should be reduced as low as reasonably achievable by the careful design of the circuits that contain radioactive water to prevent leakage and by minimizing the potential for the plant to require decontamination. The treatment should be appropriate for the level and type of contamination in the water to achieve the required decontamination factors in a way that minimizes the doses to the site personnel and the production of solid waste. This should be achieved by segregating the waste from different sources into waste streams. Each waste stream should contain all the waste with similar characteristics in terms of its chemical and particulate content so that the optimum treatment can be applied to each stream. Account should also be taken in the design of the acceptance criteria for both the anticipated storage and the final disposal of the solid waste that will be produced. For example, this may limit the use of organic materials in demineralizers.

Gas treatment systems

5.103. All discharges of radionuclides to the atmosphere should be reduced by the best practicable means and are required to be subject to the applicable authorized limits, including dose constraints and optimization requirements (see Section 2). A system for the management of gaseous waste should be provided to comply with this requirement.

5.104. The management system for gaseous waste should be designed to collect all the radioactive gas that is produced in the plant and to provide the necessary treatment before it is discharged to the environment. In the case of noble gases, the discharge of radioactive gas should be delayed where there is a potential for the gas to contain short lived radionuclides such as ^{133}Xe. This is commonly done using delay tanks or pipes or carbon delay beds. The removal of long lived noble gases, such as ^{85}Kr, is often not justified but, if necessary, it can be achieved by using cryogenic devices of an appropriate design and choice of material.

5.105. The isotopes of iodine, which usually have the greatest radiological impacts, are commonly removed by means of charcoal filters. Means should be provided for testing these filters using the most penetrating form of iodine to ensure their efficiency over the lifetime of the plant. Special attention should be paid to the behaviour of iodine due to its different physical and chemical forms. Detailed information is provided in Annex-I (e. g. see paras I-121 and I-122).

5.106. Particulate material from both the management system for gaseous waste and the ventilation systems should be removed using filters. It is a good practice to ensure that all gas discharged from the plant that may be radioactive passes through high efficiency filters.

5.107. All radioactive gaseous effluents discharged to the atmosphere should be released from elevated points, with the topography of the site taken into account. The level of elevation required should be justified in the optimization process, with consideration given to accident conditions. (See DS529, revision of NS-G-3.2 [19] and GSG-10 [20]). Different measurements should be provided to monitor the selected radionuclides released via stack (see paras 8.28 – 8.32).

5.108. Exhausting of the air from areas of potential contamination at points near the source of the contamination should be avoided.

Radiological support facilities

5.109. The plant design should include the auxiliary facilities that are necessary for effective radiological control in the operation and maintenance of the nuclear power plant and for responding to emergencies. In particular, auxiliary facilities are necessary for limiting the spread of contamination within the controlled area and preventing the spread of contamination outside the controlled area, for carrying out adequate monitoring of the workplace and individual monitoring, for providing the workers with the required protective equipment, and for managing other health physics operations. These auxiliary facilities should include the following:

(1) A health physics operations office, including storage and testing facilities for radiological instruments and protective equipment, facilities (including space, power and gas supplies) to carry out sample measurements, computer equipment for display of remote monitoring data.
(2) A changing room for protective clothing;
(3) A personnel decontamination facility including provision for shower and hair wash;
(4) An equipment decontamination facility;
(5) Laundry facilities for contaminated clothing, where these services are not provided by an external provider;

(6) A first aid room;

(7) A radiochemistry laboratory (for the preparation of samples and the measurement of activity). The requirements for this facility will be in accordance with the reactor chemistry analysis requirements;

(8) A storage area for contaminated items and tools;

(9) A workshop for maintenance of contaminated equipment;

(10) A store for radiation sources;

(11) Facilities for the management, and storage of waste;

(12) A dosimetry laboratory or dosimetry control if there is an external service provider;

(13) A data recording and storage system for creating relevant databases e.g. for dosimetry records, or instrument control and updating them with the appropriate records as required;

(14) An alternative or remote health physics control centre;

(15) Assembly areas at the plant for use during a plant emergency;

(16) Emergency response facilities;

(17) An identified sheltering area for the plant personnel.

5.110. The following equipment should be provided and should be available before the plant begins to operate:

(1) Protective clothing and boots;

(2) Protective equipment for the respiratory tract;

(3) Air samplers and equipment for measuring airborne activity concentrations;

(4) Portable dose rate meters with an audible alarm at variable settings and devices for monitoring personnel contamination and surface contamination;

(5) Portable shielding, signs, ropes, stands and remote handling tools;

(6) Communication equipment;

(7) Meteorological instruments;

(8) Equipment for monitoring individuals for intakes of radionuclides;

(9) Temporary containers for solid radioactive waste and special containers for radioactive liquids;

(10) Emergency equipment (including additional protective clothing, self-powered air samplers and emergency vehicles);

(11) First aid equipment;

(12) Equipment for sampling and analysis around waste storage areas, such as borehole monitoring equipment for underground storage facilities for radioactive waste;

(13) Dosimetry for monitoring individuals' external exposure;

(14) Personal radiation dosimeters;

(15) Dosimeters adapted to the type of radiation and tele dosimeter systems.

6.SPECIFIC DESIGN FEATURES OF RADIATION PROTECTION IN DESIGN FOR ACCIDENT CONDITIONS

PLANT LAYOUT

6.1. In accordance with para 2.10, in the early stage of the site selection, and during the subsequent site arrangements, the radiation protection aspects of design of the NPP should be taken in consideration. Those radiation protection aspects should include the suitability assessment of the site for accident conditions, in accordance with Requirement 4, para. 4.6 of SSR-1 [4].

The aspects to be addressed in the suitability assessment should include the following:

(a) The effects of events potentially happening in nuclear installations located in the area surrounding the site assessed and having radiological consequences on it (See also SSG-68 [15]);

(b) Site characteristics having relevance in the transfer of radioactive material potentially released from the nuclear installation to the public and/or to the environment. This includes site characteristics relevant for the design of the escape routes aiming to minimize the risk of the site personnel and the population during evacuation (see NS-G-3.2 [19]);

(c) Density and distribution of the population (see NS-G-3.2 [19]), relevant to evaluate the risk for individuals and for the population, and other characteristics of the area surrounding the nuclear installation that could affect the feasibility of planning effective emergency response actions SSR-1 [4].

(d) In case of multi-unit sites, the effects of events having radiological consequences outside of one of the units and affecting the other units (see SSG-79 [62]).

6.2. The design characteristics of the necessary emergency response facilities[14] should be considered from the design stage of the nuclear power plant. Such design characteristics should include the radiation protection means necessary for the personnel working in emergency response facilities, in accordance with Requirement 67 and para. 6.42 of SSR-2/1 (Rev. 1) [1] (see paras 2.21 – 2.22 of this publication) and with paras 5.25, 5.41 and 5.42 of GSR Part 7 [9] (see para 2.20 of this Safety Guide).

6.3. In accordance with para. 5.42 of GSR Part 7 [9], the design should provide for radiation protection of all persons present in the facility and on the site with suitable assembly points and sheltering places, equipped with continuous radiation monitoring. Recommendations for such monitoring are provided in Section 8.

[14] See 'emergency response facility or location' in the Safety Glossary [13].

6.4. From the early stage of the site selection and during the design phase of the site, sufficient escape routes should be designed for all persons expected to be present in the facility and on the site, in accordance with para. 5.42 of GSR Part 7 [9] and considering the recommendations from para. 6.1 of this Safety Guide. For the efficient communication in the escape routes, the design of a suitable and reliable alarm system should be considered, for warning and instructing all persons present in the facility and on the site and seeking to minimize the risk of radiation hazard.

6.5. Safe routes to places where workers have to perform response functions in accident conditions should be considered in the design. Access to the necessary rooms of the nuclear power plant (including those accommodating relevant systems) and other arrangements (e. g. zoning, shielding, ventilation and sheltering) should be ensured in those safe routes, aiming to keep internal and external exposures of site personnel within acceptable levels according to the design targets. Safe routes should avoid areas having potential for hazardous conditions, including fire, chemical hazards, anoxia and/or high temperature. In addition, safe routes should also avoid areas having potential for high dose rates due to the presence of atmospheric contamination or pipes containing highly contaminated water (e. g. areas where the post-accident sampling system is located – see design recommendations for them in paras 4.67 and 4.68 from SSG-62 [30]).

OTHER DESIGN CONSIDERATIONS FOR AN EFFECTIVE RADIATION PROTECTION PROGRAMME

6.6. The possible radiological consequences of design basis accidents and design extension conditions, including severe accidents, should be determined to demonstrate compliance with design targets related to protection of site personnel, the public and the environment, in accordance with the Requirement 5 of SSR-2/1 (Rev.1) [1] (see para 2.3 of this Safety Guide).

Protection of site personnel under accident conditions

6.8. To complete the design for the protection of site personnel from radiation that arises from accident conditions a proper assessment should be made of the magnitudes and locations and the possible transport mechanisms and exposure pathways for the radiation sources that will be present in and after accident conditions. All potential accident scenarios including severe accidents should be considered in this assessment.

6.9. The design should be such that the operating organization can ensure the safety of all persons involved in emergency response on the site in the event of radiological emergency, in compliance with international requirements for emergency preparedness of GSR Part 7 [9] (e. g. from paras 5.43, 5.52 and 5.53 of GSR Part 7 [9].

6.10. An analysis should be made of the areas of the nuclear power plant in which it is necessary to maintain habitability for the purpose of taking both accident management measures and emergency preparedness measures. Areas to which access is expected to be required in emergencies include the control room, the supplementary control room and other emergency response facilities and locations, rooms where emergency systems, including emergency power systems, are located (see SSG-34 [63]), (or spaces adjacent to such rooms), areas where manual actuation may be required, on-site sampling facilities (e. g. for the containment and for the stack) and the laboratories. The design should provide for adequate respiratory protection readily available and properly maintained for access to these areas, as necessary. For this purpose, plant operating instructions for actions for accident management and emergency preparedness should be developed. Design modifications should be based on the findings of the habitability assessments, in accordance with related requirements of GSR Part 7 [9] and with recommendations of GS-G-2.1 [64]; see also insights from SR Series No. 32 [65].

6.11. The design of radiation protection in accident conditions (including severe accidents), potentially combined with hazards, should be based on the habitability and accessibility assessments performed. These assessments should take into account calculations of radioactive releases, dispersion of radionuclides and dose rates (considering the radionuclides making the major contribution to the dose rates). The duration of the emergency response should also be considered in the assessments.

6.12. In multi-unit sites, to ensure habitability and accessibility in accident conditions the design should take due account of the potential for specific hazards giving rise in one unit and impacting on several or even all units of the site simultaneously, in accordance with Requirement 17 of SSR-2/1 (Rev.1) [1]. The sharing of information between the operating organizations of neighbouring units is to be considered, as recommended in para 2.72. of SSG-54 [53].

6.13. The postulated hazardous conditions[15] in which emergency workers may be required to perform response functions on or off the site should be identified. Arrangements should be made for taking all practicable measures to provide radiation protection for emergency workers for the range of radiological conditions in combination with other potentially hazardous conditions in which they may have to perform response functions on or off the site. These arrangements should include:

- arrangements to assess continually and to record the doses received by emergency workers;
- procedures to ensure that doses received and contamination are controlled in accordance with established guidance and in compliance with international standards (GSR Part 7 [9]);
- arrangements for the provision of appropriate specialized protective equipment (which depends on the severity of the hazard), procedures and training for emergency response in the postulated hazardous conditions;
- arrangements for the provision and secure storage of sufficient amount of consumables (e. g. respiratory protection and pollution protective clothing) that will be necessary during the postulated event.

[15] Beside nuclear and radiological risk it also includes internal and external hazards, such as fires, releases of hazardous chemicals, anoxia risk, storms or earthquakes.

6.14. Provisions should be made for shielding the radiation, in addition to those provisions required during operation, to ensure that personnel can have access to and can occupy the relevant working places so as to operate and maintain essential equipment[16] without exceeding established dose limits as specified in paras 4.12 - 4.19 of GSR Part 3 [2] and paras 5.49 - 5.61 of GSR Part 7 [9][17]. This includes access to equipment in cases where maintenance or repair may be necessary after an accident. In general, provision should be made to render direct intervention by operators superfluous by installing automatic or remotely controlled equipment (e. g. remotely controlled valves).

6.15. Consideration should be given in anticipation to movements of the source material (e. g. handling of fuel assemblies), to a decrease in the effectiveness of the shielding (e. g. due to concrete erosion), losses of shielding efficiency and scattered radiation including sky shine radiation, all of which may have a major impact on radiation levels after an accident.

6.16. Provision should also be made to minimize the airborne radioactive contamination in areas to which access will be required for ensuring the safety of the plant or the site personnel, such as the reactor building, the fuel storage area, the plant control room, supplementary control room and other emergency response facilities and locations (i. e. technical support centre, operational support centre and emergency centre). Such provision may be achieved by operation of the ventilation system in a recirculation mode. In this case heat removal would have to be provided by cooling the air in a recirculation system. An appropriate fraction of the circulation air should be filtered (e. g. using high efficiency particulate air filters and iodine filters) and/or a mobile local exhaust system should be prepared if the inward leakage of contaminated air may be expected to be too high to permit occupancy of the room without the use of respiratory protection. The releases from the plant can be limited by means of secondary containment or, exceptionally, by venting to the atmosphere, through filters. Requirements for control room habitability in particular should be addressed, in terms of maintaining overpressure, air cooling and oxygen supply (e. g. compressed air bottles); such habitability should also be kept under conditions of releases of gaseous chemicals. Reliable isolation (dampers) and habitability should be ensured in case of loss of power supply. Recommendations regarding ventilation systems for the main control room, supplementary control room and other emergency response facilities are provided in paras 4.160 – 4.167 of SSG-62 [30]); additional insights can be found in GS-G-2.1 [64].

[16] Essential equipment here means equipment that should continue to be operable to prevent the escalation of an accident or further radioactive releases (e. g. pumps in water cooled reactors or gas circulators in gas-cooled reactors, which are required to maintain core cooling), and equipment that is required for monitoring the state of the plant after an accident.

[17] In the event of an emergency, radiation dose limits for normal operation may be exceeded. Use should then be made of dose levels given in para. 4.15 of GSG-7 [10] and other conditions as established in Section 4 of GSG-7 [10] for interventions in emergencies.

6.17. Consideration should be given to the requirements and the means for sampling of gases and liquids after an accident (e. g. remote sampling), and provisions for shielding should be made as necessary to enable such samples to be taken and tested without undue radiation exposures of site personnel.

6.18. Consideration should be given to the provision of safe locations for monitoring vehicles equipped with air dose rate measurement, air concentration measurement, radionuclide analysis, GPS and adequate filtration.

6.19. Provision should be made for alerting and assembling site personnel and for — at least provisionally — sheltering site personnel not involved in emergency response until its evacuation. Multi-system communication (e. g. with provision of satellite phone) should be secured between the control room, supplementary control room, other emergency response facilities and locations and assembly points for personnel.

6.20. The ready identification of rooms clearly marked signs and the removal of any obstacles to the free movement of site personnel in passageways should be ensured for the protection of personnel, mainly by decreasing the duration of exposures during safety related actions under accident conditions. These factors should be taken into consideration and dealt with appropriately at the design stage.

6.21. In addition, areas should be identified within the plant in which radiation exposures are expected to remain low in accidents. These areas may be used in evacuating site personnel and monitoring them for contamination (SR Series No. 32 [65]). Dose recording devices (including spare devices) for individual monitoring should also be stored here.

6.22. In an emergency, arrangements should be made to ensure that relevant information related to the protection of the workers is recorded and retained for use during the emergency, in evaluations conducted following the emergency and for the long term health monitoring and follow-up of emergency workers who may potentially be affected.

DESIGN PROVISIONS FOR THE PROTECTION OF THE PUBLIC UNDER ACCIDENT CONDITIONS

6.23. Provisions for shielding should be incorporated into the design to protect the public under accident conditions from direct or scattered radiation (including sky-shine). In addition, site boundary monitors should be properly placed to allow for the monitoring of the spread of radioactive plumes, based on topographical and meteorological data (see paras 8.33 – 8.36).

6.24. Compliance with acceptance criteria for accident conditions should be assessed by means of safety analyses in accordance with Requirement 14 of GSR Part 4 (Rev. 1) [6] and Requirement 42, paras 5.71 – 5.74 of SSR-2/1 (Rev. 1) [1]. In cases where the preliminary safety analysis shows that the acceptance criteria are not met, additional protective features should be incorporated into the design or operational measures should be developed to meet the acceptance criteria, i. e. the acceptable limits for doses to the public and for releases to the environment.

6.25. Generally, the releases that are evaluated for accident conditions are releases to the atmosphere, since an accidental release of radioactive material directly to the aquatic environment is usually unlikely. However, this should be verified for each design or each plant, and consideration should be given, for instance, to the contamination of groundwater by direct leakages. It should be postulated that releases to the atmosphere may end up in the surface water through rain and/or wind, or due to washing out of radioactive materials inside the plant, increasing the content of fission products in the aquatic environment.

6.26. It should be taken into account that the dispersion of radionuclides into the atmosphere in an accident mainly depends on the composition of the source term, the kinetic of the release, the release point and the meteorological conditions. It is the usual design practice to assume that an unfavourable meteorological situation prevails during and after the accident (see the recommendations provided in paras of NS-G-3.2 [19] and GSG-10 [20]). The assumptions to be used for the assessment of the consequences of the dispersion should be agreed by the regulatory body on the basis of regional and on-site meteorological and environmental conditions; further recommendations are provided in Section 5 of GSG-9 [23]. A methodology for the calculation of doses to the public should be developed in accordance with the requirements of the regulatory body and it should be carefully validated, as recommended in GSG-10 [20]; the methodology should include the preparation of a list of radionuclides making a major contribution to the doses. International guidance exists for the definition of a representative person (see IAEA Safety Glossary [13]). Design targets are usually set so that no banning of food is assumed, at least for design basis accidents; and thus, for these situations, the consumption of food that has been produced within the potentially affected area is used as an input to the dose calculation for the representative persons.

6.27. In design basis accidents, conservative assumptions should be made with regard to the source terms (see details in SSG-2 (Rev. 1) [24] and in Annex I of this Safety Guide), duration of the exposure, the meteorological conditions, living habits and shielding-of and occupancy by the public at the time of the accident, to demonstrate compliance with the radiological acceptance criteria for doses to the public.

6.28. Within the off-site areas where protective actions are planned in the event of a severe emergency (e.g. the precautionary action zone[18] and the urgent protective action planning zone[19]), arrangements should be made for promptly assessing any radioactive contamination, releases of radioactive material and doses for the purpose of determining or modifying urgent protective actions following a release of

[18] Precautionary action zone (PAZ): See 'emergency planning zone' at the IAEA Safety Glossary [13].

[19] Urgent protective action planning zone (UPZ). See 'emergency planning zone' at the IAEA Safety Glossary [13].

radioactive material (see the international safety requirements for emergency response in GSR Part 7 [9] and the requirements 44, 45 and Schedule IV of GSR Part 3 [2]).

6.29. For design extension conditions, specific analysis should be performed to demonstrate compliance with national regulatory requirements concerning both the short term and the long term consequences of an accident. The source term is usually evaluated by using best estimate methods, in contrast to the conservative assumptions that are made for design basis accidents. In addition, a probabilistic dispersion code may be used to evaluate the risk to the representative persons.

6.30. Design measures that may be used to achieve reductions in radiological consequences for the public of radioactive releases in accident conditions include:

(1) Ensuring that event sequences and accident scenarios potentially leading to early radioactive releases or large radioactive releases are 'practically eliminated';

(2) Providing design means to minimize the scope of fuel damage and protect the barriers against releases of fission products from the fuel;

(3) Achieving leak tightness, isolation and by-pass prevention of the containment;

(4) Filtering the exhaust air or using charcoal delay beds in order to reduce the releases of airborne radioactive material, with due account taken of the fact that some pathways for accidental releases may bypass the filtered exhaust system[20];

(5) Achieving a high decontamination factor for the filters by using best practices in the design, the filter material and the filter depth, for example, or by providing dehumidifiers before the filter;

(6) Providing shielding in places where radioactive material released to the containment or to a building would otherwise cause radiation exposure above the limits set for the accident analysis owing to direct or scattered radiation (including sky shine and ground shine);

(7) Providing means of sealing the containment building or reducing the flow volume of exhaust air to provide for decay time within the building;

(8) Reducing both mass and activity of radioactive material released by decreasing the discharge velocity of fluids or the closure time of valves;

[20] For boiling water reactors, in case of releasing the main steam from the pressure vessel to the atmosphere via the main steam relief safety valve, in the early stages of an accident, a significant amount of volatile radioactive materials may be released by the water contained in the suppression chamber.

(9) Ensuring the effectiveness of the spray system in trapping iodine by adding appropriate chemicals (e. g. hydrazine hydrate) or by adding chemicals in the reactor sump[21] (e. g. sodium tetraborate);

(10) Defining an exclusion zone at the design stage to which public access is prevented.

In addition, several types of safety related design measures (which may be based on probabilistic safety analyses) should be taken, including:

(1) Developing or upgrading safety systems and safety features for design extension conditions to minimize equipment malfunctions and operator errors, to prevent and/or mitigate severe accidents;

(2) Ensuring that power is available for essential equipment, instrumentation, including health physics instruments, and protection systems.

6.31. In an emergency, arrangements should be made to ensure that relevant information related to the protection of the public is recorded and retained for use during the emergency, in evaluations conducted following the emergency and for the long term health monitoring and follow-up of members of the public who may potentially be affected.

6.32. For design extension conditions, analysis should be performed to demonstrate the scope and duration of the emergency countermeasures to be implemented; the demonstration should be based on considerations related to established reference levels. Use of emergency countermeasures (e. g. sheltering, iodine prophylaxis and relocation of people) may be considered in safety demonstration of design. Such consideration should be limited in area and time, and in accordance with national regulations. Dose reduction factors can be applied provided that clear instructions in emergency plans are available and that sufficient time and other conditions allow to ensure, with high level of confidence, that those countermeasures can be implemented.

6.33. Certain items important to safety should be qualified for the environmental conditions to which they will be exposed either during operational states or in accident conditions, as recommended in SSG-69 [66]. Environmental conditions to be considered in the qualification should include effects of radiation and potential synergistic effects (for example radiation and temperature).

[21] In the case of spray systems, care should be taken with regard to the control of tritium in the containment.

7. SPECIFIC FEATURES OF RADIATION PROTECTION IN DESIGN FOR DECOMMISSIONING

7.1 Radiation protection aspects of the design for decommissioning discussed in this section consider decommissioning actions in accordance with para. 1.3. of SSG-47 [54],

> "Decommissioning actions that include the decontamination, dismantling and removal of structures, systems and components (SSCs), including management of the resulting radioactive waste and radiation protection of workers carrying out the decommissioning, as well as the conduct of characterization surveys to support decommissioning."

The management of spent nuclear fuel and of radioactive waste generated during the operational phase of a facility should not be considered part of decommissioning. If operational radioactive waste or nuclear fuel is present in the facility after its permanent shutdown, and in case such removal is not possible during the period of transition between permanent shutdown and decommissioning, such material should be removed prior to the conduct of decommissioning actions and accordingly the final decommissioning plan should be adjusted, as stated in para 8.10. of GSR Part 6 [8].

7.2. Decommissioning actions, included in the initial, updated and the final decommissioning plan, should be designed to achieve the progressive and systematic reduction in radiological hazards during decommissioning. The design for these actions should include a proper planning and assessment, in accordance with national regulatory requirements, to ensure, protection of workers and the public and protection of the environment to demonstrate that the decommissioned facility achieves the planned end state. Summary of decommissioning plans should be included in the safety analysis report of different plant stages and submitted to the regulatory authority when applying for different licences (construction, commissioning, operation and decommissioning licences). The recommendations provided in the following paras of this section are applicable for the initial design stage and for the revision of the design before decommissioning. The decommissioning should be taken into consideration at the design phase of the plant in order to reduce the production of radioactive waste and the dose received by workers at the time of the decommissioning. Considerations for decommissioning during design and construction are also discussed in paras 7.5- 7.9. of SSG-47 [54].

7.3. In this section, consideration is also given to the design features for protecting site personnel from the radiation that results during the decommissioning of a nuclear power plant, and the means of implementing the system of dose limitation as described in Schedule III of GSR Part 3 [2] and in GSR Part 6 [8]. Guidance given in SSG-47 [54] is also taken into account here. Tasks are different in different stages of the decommissioning and are conducted in work environment that is continuously changing, therefore the design and other provisions for the radiation protection should consider that. Prior to decommissioning, an assessment of radiological hazards and imposed risks should be performed, with continuous re-assessment throughout the execution of decommissioning activities. In this assessment the preparation for decommissioning should also be included. The results of the assessment should be presented in the safety analysis report.

PLANT LAYOUT

7.4. The paras 5.1 and 5.2 of this Safety Guide provide recommendations for the plant layout for operational states. In the design of the plant layout, a careful assessment should be made of the access requirements also for decommissioning of equipment. The design of the layout of the plant should take into account to limit the exposure of site personnel during decommissioning. Worker dose reduction can be also achieved with adequate layout design to give enough space for cutting and segmenting operations.

7.5. Before the start of decommissioning the plant layout should be reviewed to assess whether there are needs to initiate radiation protection related changes in the layout (for example providing enough space for cutting and segmenting activities) requested by the final decommissioning plan.

7.6. For detailed planning, information should be available about the radiological situation, the arrangements of SSCs, the course of dismantling and planned radiation protection measures. The dose estimation should not only cover dismantling but also preparatory work, clean-up after dismantling and activities related to waste handling. Layouts, schemes and pictures defining and illustrating the local situation should be made available. In addition, it should be considered essential the careful planning and knowledge of the available decontamination and dismantling techniques to ensure that radiation exposure of workers is kept ALARA. Feasible dismantling techniques should be compared to select the optimum option based on individual and collective dose.

Classification of areas and zones

7.7. The requirements for the classification of areas as controlled areas or supervised areas for decommissioning are also established in the Requirement 24, paras 3.88 - 3.92 of GSR Part 3 [2]. Each controlled area should have a minimum practicable number of access and exit points for personnel as well as for materials and equipment necessary for the decommissioning.

7.8. Provisions should be designed and regularly reviewed for controlling access and the exit(s) from the controlled areas and for monitoring persons and equipment leaving the controlled areas for decommissioning needs as well. The design should consider each decommissioning phase. Before the decommissioning starts the design should be reviewed identifying and re-evaluating, and if it is necessary modifying, the access and exit points for personnel and for materials and equipment and zones for the decommissioning.

7.9. Recommendations for the classification of areas and zones for operational states are provided in paras 5.4 – 5.8 of this Safety Guide. Decommissioning tasks should also be considered in the identification and classification of areas and zones during the early stage of the design.

7.10. During the design of the plant consideration should be given to the minimization of the number and size of contaminated areas to facilitate decontamination during decommissioning to minimize the radioactive waste and the radiation exposure to workers in accordance with the recommendations of para. 7.6 (a) of SSG-47 [54].

7.11. Consideration should be given to the possibility that it may be necessary during decommissioning phases to reclassify certain areas temporarily or permanently. In this regard, particular attention should be paid to the planning of access routes. Under such conditions the zones and the controlled areas should be re-evaluated.

Changing rooms, changing areas and related facilities

7.12. Paragraphs 5.9 - 5.11 of this Safety Guide provide recommendations on radiation protection aspects for the design of changing rooms (including physical barriers separating clearly the clean area from the potentially contaminated area) ensuring that these changing areas and related facilities at NPP are suitable for the start of the decommissioning as well. The attention should be paid to consider the different phases of decommissioning to design new facilities or new arrangements for the decommissioning if necessary.

Control of access and occupancy

7.13. Paragraph 5.12 provide recommendations for the access to and exit from controlled areas for the operational states. These recommendations should be considered for the decommissioning as well. Before the decommissioning starts the design of these points should be re-assessed, including their classification, to ensure that the systems provided to protect workers from doses above authorized limits are adequate.

7.14. The recommendations provided in para. 5.13 of this Safety Guide are also applicable to decommissioning. Before the decommissioning starts the design of the applicable systems (see para 5.13) these systems should be re-assessed, including their safety classification, and systems provided to protect workers from doses above authorized limits should be modified.

7.15. The routes for personnel through radiation zones and contamination zones should be minimized during decommissioning as well to reduce the time spent in transit through these zones.

7.16. Systems provided to protect workers from doses above authorized limits during decommissioning may require safety classification in accordance with SSG-30 [55].

7.17. To minimize the radiation doses to personnel working in the controlled area during decommissioning and the spread of contamination, the layout of the controlled area should be so designed or revised that the personnel do not have to pass through areas of higher radiation zones to gain access to areas of lower radiation zones. The feedback of operating experience in decommissioning with reactors of similar design including experience in decommissioning after accidents should be used to provide guidance concerning radiation levels and contamination levels.

7.18. As far as practicable, the revised design for the decommissioning should be such as to limit the possible spread of contamination during decommissioning and to facilitate the erection of temporary containments needed for decontamination activities.

7.19. The design or the revised design for decommissioning should be such that the occupancy time necessary in radiation areas and contamination areas also for the purposes of decommissioning should be consistent with the principle of optimization of radiation protection. This can be achieved, for example, by:

(1) Provision of passageways of adequate dimensions for ease of access to plant systems and components. In areas where it is likely that site personnel will have to wear full protective clothing, including masks with portable air supplies or connections to a supply by air hoses, account should be taken of this in deciding on the dimensions of the passageways, noting that there may be an increase in the number of areas with these requirements during decommissioning.

(2) Provision of clear passageways of adequate dimensions to facilitate the removal of plant items to a workshop for decontamination or disposal. The routes for removing large items of plant during decommissioning should be planned at the design stage and the necessary provisions should be incorporated.

(3) Facilitation of access to SSCs, including compartmentalization of processes (e. g. through incorporation of hatches and large doors) in accordance with para 7.6.(b) of SSG-47 [54].

(4) Provision of adequate space in the working areas, to carry out decommissioning, for example.

(5) Provision of ladders, access platforms, crane rails or cranes in areas where it can be foreseen that they will be required to permit the removal of plant components during decommissioning. Consideration should be given to the requirements for access by robotic demolition machines as well as human access. Features to facilitate the installation of temporary shielding should be included in the design.

(6) Use of computer aided design models to optimize decommissioning aspects of the design that affect working times during decommissioning. Video or photographic records should be made during the construction of the plant to facilitate the planning of work in areas of high radiation levels during decommissioning and thus to shorten working times.

(7) Provision of means for the quick and easy removal of shielding and insulation during decommissioning.

(8) Provision of special tools and equipment for facilitating work during decommissioning to reduce exposure times.

(9) Provision of remote-controlled equipment;

(10) Provision of a suitable communication system for communication during decommissioning with the site personnel working in radiation areas or contamination areas.

OTHER DESIGN CONSIDERATIONS FOR AN EFFECTIVE RADIATION PROTECTION PROGRAMME FOR DECOMMISSIONING

System design

7.20. The design of nuclear power plant systems should take into account the feedback from experience gained in reducing radiation exposure at decommissioned plants.

7.21. The following measures for reducing radiation exposure during decommissioning should be adopted in the system design in accordance with paras 7.6. (d), (f), (g) and (n) of SSG-47 [54]:

(1) The use of modular construction in order to facilitate the dismantling of SSCs;

(2) Facilitation of the removal and/or decontamination of material or equipment, including by means of built-in decontamination mechanisms, such as protective coverings and liners in process cells and areas where liquids might be present;

(3) For materials that may be exposed to neutron radiation or materials in contact with reactor coolant, use of materials that are resistant to activation, that are resistant to degradation by chemicals and that have sufficient wear resistance to minimize the spread of activated corrosion products;

(4) Consideration of provisions for the installation of 'test coupons' to facilitate the radiological characterization of SSCs;

(5) Provision of countermeasures (e. g. flushing) to avoid the sedimentation of radioactive sludge in piping and containers used during decommissioning;

(6) Waste management concept, especially concerning treatment of radioactive material towards clearance or disposal, and options for logistics;

(7) Water supply and drainage systems.

7.22. Pipelines containing radioactive fluids should not be located near clean piping. Sufficient space for decommissioning should be left between the pipelines and the walls.

7.23. In the design of pipelines, welded seams should be readily accessible. The minimization of underground piping and of embedded pipes in the building structures should be considered in accordance with the para 7.6 (c) of SSG-47 [54].

7.24. Provisions used for the draining and flushing of tanks to reduce radiation sources for operational states (see para 5.25) should be assessed and if necessary, with additional provisions for decommissioning.

Component design

7.25. Recommendations provided in paras 5.27 – 5.32 of this Safety Guide should also be applied to component design in Section 7.

7.26. The components to be used in areas of high radiation levels should be designed to be easily decontaminated by chemical or mechanical means after the operation and easily removable during decommissioning (see para 5.29).

Auxiliary radiological support facilities

7.27. The plant design should include the auxiliary facilities that are necessary for effective radiological control in decommissioning of the nuclear power plant and for responding to emergencies. In particular, auxiliary facilities are necessary for limiting the spread of contamination within the controlled area and preventing the spread of contamination outside the controlled area, for carrying out adequate monitoring of the workplace and individual monitoring, for providing the workers with the required protective equipment, and for managing other health physics operations during decommissioning as well. These radiological support facilities should include the characteristics (1) to (14) listed in para 5.109; items (15) to (17) from that para. are not applicable here.

7.28. During the design stage, identification and reservation of locations for new facilities that might support decommissioning (i. e. new waste management facilities) should be considered (see para. 7.6.(m) of SSG-47 [54]).

7.29. The following equipment should be provided and should be available before the decommissioning of the plant begins:

(1) Protective clothing (boots);

(2) Protective equipment for the respiratory tract;

(3) Air samplers and equipment for measuring airborne activity concentrations;

(4) Portable dose rate meters with an audible alarm at variable settings and devices for monitoring personnel contamination and surface contamination;

(5) Portable shielding, signs, ropes, stands and remote handling tools;

(6) Communication equipment;

(7) Meteorological instruments;

(8) Equipment for monitoring individuals for intakes of radionuclides;

(9) Temporary containers for solid radioactive waste and special containers for radioactive liquids;

(10) Emergency equipment (including additional protective clothing, self-powered air samplers and emergency vehicles);

(11) First aid equipment;

(12) Equipment for sampling and analysis around waste storage areas, such as borehole monitoring equipment for underground storage facilities for radioactive waste.

Remote techniques during decommissioning

7.30. Remote techniques (see para. 5.33) may play a major part in the removal of the most radioactive items during decommissioning. The use of such techniques should be considered at the design stage and it should be ensured in the design that their use is not precluded. It is likely that there will be improvements in remote control techniques over the lifetime of the plant and between initial stage and later stages of decommissioning. The best practicable techniques that are available when the work is carried out should be used.

Decontamination during decommissioning

7.31. As indicated in para. 5.68 the need for decontamination should be considered at the design stage (see also para 5.76). The decontamination facilities and activities included in the design for operational states should be assessed in decommissioning plan. If necessary new facilities should be designed for decommissioning.

7.32. When decontamination facilities are being planned, all components that are expected to come into contact with contaminated waste material should be considered possible items for decontamination during decommissioning as well.

7.33. Special consideration should be given to rooms where leaks or spills of contaminated liquid might occur during decommissioning. These areas should be designed to allow easy decontamination (e. g. by means of a special coating on floors) and control of the spread of contamination. Adequate bunding and sloping of these rooms should be arranged to limit the contaminated areas and for the quick draining and collection of spilled liquids during decommissioning as well.

7.34. The system of active floor drains should be extended to all rooms where there are systems that contain radioactive fluids during decommissioning. The sumps or the rooms should be provided with liquid level detectors that actuate a high-level alarm, as recommended for the operational states in para. 5.71.

7.35. The floor drain system should include filtration to prevent an excessive amount of particulates entering the subsequent water treatment systems during decommissioning.

7.36. There should be an adequate tank volume for storage of contaminated water do not burden systems that are intended for other purposes during decommissioning. The tank volume should also be sufficient to ensure that any releases of liquid radioactive effluent resulting from decontamination during decommissioning to the environment will remain small.

7.37. The coatings of fuel storage pools and fuel handling pools, as well as the equipment used in these areas, will become contaminated. When the water level in such pools is lowered, surfaces may dry out, and this may cause a hazard due to airborne radioactive material during decommissioning as well. Systems should be provided for decontaminating such surfaces before they dry out before decommissioning starts. Systems should also be provided for decontaminating, before they dry out, fuel transport flasks and components that may have to be removed from the pools for decommissioning.

7.38. Decontamination facilities during decommissioning should be provided for removing radioactive material from the surfaces of components and tools and equipment that will be removed from the service.

7.39. Provision should be made for the decontamination of personnel and of reusable protective clothing for decommissioning.

7.40. Drains from the decontamination facilities should connect to the treatment systems for radioactive effluent during decommissioning.

Ventilation during decommissioning

7.41. A dedicated active ventilation system should be provided for maintaining appropriate clean conditions in workspaces within the controlled area. Specific circumstances that arise during decommissioning (e. g. modification of the ventilation system might be necessary during decommissioning) should be taken into account at the design phase. The availability of the relevant ventilation systems should be assessed in the final decommissioning plan in accordance with the Requirement 12 of GSR Part 6 [8]. Plans for removing components of the ventilation system during decommissioning should be also considered.

7.42. For the purposes of radiation protection, the primary objective of providing a ventilation system should be to control the contamination of the working environment by airborne radionuclides and to reduce the need to wear respiratory protection. It should be also considered that dismantling activities might result in elevated activity concentrations of airborne radionuclides (see para 8.21 of SSG-47 [54]).

7.43. Both the spread of contamination and the amount of releases to the environment should be limited by providing features such as air cleaning filters and by maintaining appropriate pressure differentials. It should be also considered that decommissioning actions might result in elevated discharges for a limited period of time and might also lead to changes in pressure conditions (see para. 8.21 of SSG-47 [54]).

7.44. In relation to radiation protection, the ventilation system should also provide suitably conditioned air to ensure the comfort of personnel during the decommissioning.

7.45. In designing a ventilation system to control airborne contamination, account should be taken of the following:

— Mechanisms of thermal and mechanical mixing;

— The limited effectiveness of dilution in reducing airborne contamination;

— Exhausting of the air from areas of potential contamination at points near the source of the contamination;

— The use of exhaust rates that are commensurate with the potential for contamination in the area;

— The need to ensure that the exhaust air discharge point is not close to an intake point of the ventilation system.

Activities planned for decommissioning should be also considered in designing and reassessing relevant ventilation systems.

7.46. The airflow in the ventilation system used during decommissioning should be such that the airflow in the ventilation system should be directed from regions of lower airborne contamination to regions of higher contamination and air should be extracted from the latter as to minimize the resuspension of contamination (see para 7.44) Particular sources during decommissioning with specific location and characteristic, and potential changes in pressure conditions due to activities during decommissioning should be also considered in the design.

7.47. Portable ventilation systems (fans, filters and tents) should also be used in areas where airborne contamination may arise during decommissioning, and provision should be made for sufficient space in which to operate such systems.

Waste treatment systems[22] used during decommissioning

7.48. The equipment in treatment systems for solid, liquid and gaseous radioactive waste may contain radioactive material in high concentrations, and radiation protection from this material should be provided for site personnel. Before the decommissioning starts for the preparation of the final decommissioning plan an estimate should be made of the radionuclide content in treated waste, and of the consequent maximum radiation level that can arise in each area of the waste treatment system. Consideration should be given, in the decommissioning plan to the sources that give rise to the highest radiation levels (such as ion exchange resins, discarded radioactive components and filter waste).

[22] Requirements and recommendations on the management of radioactive waste before it is sent to a repository are established in IAEA Safety Requirements and Safety Guides GSR Part 5 [7], SSG-41 [13] and SSG-40 [11].

7.49. During the finalization of the decommissioning plan, it should be assessed in which extent the waste treatment systems will be used. If there are planned changes in the conditions of operational states for the decommissioning phase their design should be reassessed taking also in consideration recommendations provided in para 5.81. Other safety impacts (e. g. fire hazards) causing radiation protection consequences should be incorporated in the assessment and the design should be modified accordingly.

7.50. The design or the modified design for decommissioning should be such that during decommissioning it is possible to carry out by remote control reverse flow flushing, washing, regeneration and change of resins for the final removal.

7.51. The use of hazardous substances in the construction of SSCs that could result in mixed hazardous and radioactive waste during decommissioning should be avoided (see para. 7.6.(k) of SSG-47 [54]).

Storage of radioactive waste at the plant during decommissioning

7.52. As stated in para 5.83, facilities should be provided for the safe storage of the radioactive waste that arises at the plant, with account taken of its form (solid, liquid, gas or a mixture), its radionuclide content and its nature in terms of the extent to which it has been processed. The design features of facilities should be such that the radioactive waste can be received, handled, stored and retrieved without causing undue occupational or public exposure or environmental effects during decommissioning as well. Further recommendations on this subject are provided in SSG-40 [11].

7.53. The design of storage facilities for radioactive waste should incorporate the following functions during decommissioning:

(a) Maintaining the confinement of stored materials;

(b) Providing for radiation protection (by means of shielding and contamination control);

(c) Providing for ventilation, as necessary;

(d) Allowing the retrieval of the waste for transport off the site;

(e) Further design features against incidents and accidents (see paras 6.54 and 6.55).

7.54. The storage facility should provide protection for the waste to prevent degradation that could pose problems for safety during its storage or upon its retrieval during decommissioning. It should be ensured that the shielding and confinement functions of the storage facility, including the containers, are fulfilled throughout the facility lifetime including decommissioning. This should be achieved by means of design features, the selection of appropriate materials, and maintenance and repair or replacement, with account taken of the recommendations listed in para 5.85.

7.55. Consideration should be given to the possibility of changes in the stored waste during decommissioning, which could lead to:

(a) Generation of hazardous gases caused by chemical effects and the buildup of overpressure;

(b) Generation of combustible or corrosive substances;

7.56. The possibility of accidents during decommissioning should be taken into account in the design of storage facilities. The resulting features can differ from, and should be complementary to, those designed for normal use.

7.57. In addition to radiological hazards, non-radiological hazards (for example, fire or explosion), which may contribute to radiologically significant consequences, should also be considered in the design of storage facilities for decommissioning stage.

7.58. Where appropriate, equipment should be provided with suitable interlocks or physical limitations to prevent dangerous or incompatible operations during decommissioning. Such interlocks or limitations should prevent undesirable movement (for example, the movement of waste that gives rise to high dose rates into an area occupied by site personnel or vice versa).

7.59. The need for remote handling during decommissioning should be considered in cases where the waste container gives rise to high dose rates or where there is a risk that radioactive aerosols or gases could be released to the working environment.

7.60. Any remote handling devices should be designed to provide means for their maintenance and repair, for example, by the provision of a shielded service room, to keep occupational radiation exposures as low as reasonably achievable during decommissioning.

Protection of the public during plant operation and decommissioning

Discharge criteria during decommissioning

7.61. To protect the public from radiological consequences due to the decommissioning of the plant, plant operators are required to ensure that doses to members of the public arising from radioactive substances in the effluents and from direct radiation due to the plant do not exceed the prescribed limits, and that the optimization principle is applied (see requirements 30, 31 and 32 of GSR Part 3 [2] and also the safety guides GSG-8 [25], GSG-9 [23].

7.62. Three types of effluents should be considered during decommissioning: liquids (mainly aqueous), gases from process systems and ventilation air.

Waste stream source reduction during decommissioning

7.63. The design measures that are taken to control the sources of radioactive material in the plant during decommissioning so as to protect the site personnel will also affect the activity of the waste streams and discharges. However, some radionuclides should be given greater consideration in terms of protecting the public than in terms of protecting site personnel (see para 5.97).

Effluent treatment systems during decommissioning

7.64. The flows and the activity concentrations of liquid and gaseous effluents need to be monitored and controlled to ensure that the authorized discharge limits are not exceeded during decommissioning (SSR-2/1 (Rev. 1) [1]). Liquid and gaseous treatment facilities that are based on the best practicable means should be provided during decommissioning. References GSG-10 [20] and GSG-9 [23]. provide recommendations and information on the calculation of the exposure of the public resulting from radioactive discharges.

Liquid treatment systems during decommissioning

7.65. The major sources of contaminated water that require treatment during decommissioning include: primary coolant; floor drains that collect water that has leaked from the active liquid systems and fluids from the decontamination of the plant and fuel flasks; water that is used to backflush filters and ion exchangers; leaks of secondary coolant; laundries and changing room showers; and chemistry laboratories. The foregoing sources are essentially aqueous in nature and the guidance that follows is given on this basis. Where non-aqueous liquid waste is generated in sufficient volumes, the provision of a separate waste treatment system to deal with it should be considered. Further guidance on the treatment of aqueous and non-aqueous liquid waste is provided in SSG-40 [11].

7.66. Proven methods of treating the radioactive waste water to reduce radioactive contamination use mechanical filtration, ion exchange, centrifuges, distillation or chemical precipitation. The different treatment processes in the liquid waste treatment system should be connected so as to give the operator sufficient flexibility to deal with liquids of different origins and unusual compositions, and to re-treat water if the authorized low activity for discharge is not attained after the initial treatment. In the case of pressurized water reactors radioactive water may be present in the secondary (turbine) circuit as a result of operating with some primary circuit to secondary circuit leakage in the steam generator. In this case, treatment of the water from the secondary circuit may be necessary to reduce the activity before the water is discharged during decommissioning.

7.67. Consideration should be given to the amount of solid waste that is produced by the liquid waste management systems during decommissioning. The volumes of liquid that require treatment should be reduced as low as reasonably achievable by the careful design of the circuits that contain radioactive water to prevent leakage and by minimizing the potential for the plant to require decontamination during decommissioning. The treatment should be appropriate for the level and type of contamination in the water to achieve the required decontamination factors in a way that minimizes the doses to the site personnel and the production of solid waste. This should be achieved by segregating the waste from different sources into waste streams. Each waste stream should contain all the waste with similar characteristics in terms of its chemical and particulate content so that the optimum treatment can be applied to each stream. Account should also be taken in the design of the acceptance criteria for both the anticipated storage and the final disposal of the solid waste that will be produced.

Gas treatment systems during decommissioning

7.68. All discharges of radionuclides to the atmosphere should be reduced by the best practicable means and are required to be subject to the applicable authorized limits, including dose constraints and optimization requirements. A system for the management of gaseous waste should be provided to comply with these requirements during decommissioning.

7.69. The management system for gaseous waste should be designed to collect all the radioactive gas that is produced in the plant and to provide the necessary treatment before it is discharged to the environment during decommissioning (see para 5.104).

7.70. Particulate material from both the management system for gaseous waste and the ventilation systems should be removed using filters. It is a good practice to ensure that all gas discharged from the plant that may be radioactive passes through high efficiency filters during decommissioning.

7.71. All radioactive gaseous effluents discharged to the atmosphere should be released from elevated points, with the topography of the site taken into account during decommissioning as well.

Shielding for decommissioning

Design of shielding for decommissioning

7.72. In reviewing the initial design of shields and designing a shield for a specific radiation source for decommissioning, the target dose rate should be set, for which account should be taken of the expected frequency and duration of occupancy of the area during decommissioning. This is especially important for decommissioning after accident. Account should also be taken in setting this target dose rate of the uncertainties associated with the source term and with the analysis made to determine the expected dose rate.

7.73. In establishing specifications for shielding for decommissioning, account should be taken of the buildup of radionuclides over the lifetime of the plant. Decontamination activities should be implemented as part of the decommissioning process to reduce radionuclides that have built up over the lifetime of the plant.

7.74. After the potential intensity of the source has been assessed, the process of shielding design for decommissioning should be carried out iteratively, starting with the design of shields without penetrations. Next, consideration should be given to the necessary penetrations through the shielding, such as those for temporary pipes, cables and access ways, and the provision to be made to maintain the effectiveness of the shielding for the protection of site personnel.

7.75. The choice of materials for a shield should be made on the basis of the nature of the radiation, the shielding properties of materials, their mechanical and other properties, and space and weight limitations (see para. 5.42).

7.76. Losses in shielding efficiency may occur as a result of environmental or other conditions. During decommissioning a relevant effect is related to different conditions of systems compared to the operating conditions, e. g. water filled (and shielded) systems are emptied for dismantling. Effects that should be taken into account are those due to the interactions of gamma rays with the shielding, those due to reactions with other materials (e. g. erosion and corrosion by the coolant), and temperature effects (e. g. the removal of hydrogen and/or water from concrete). Thus, the effective efficiency of the shielding should be considered at the time of the decommissioning.

7.77. In the design of temporary shielding designed for decommissioning, account should be taken of relevant external hazards, in particular seismic forces and of relevant internal hazards.

7.78. In areas where temporary additional shielding may be necessary in specific decommissioning tasks, account should be taken in the design of the weight of the additional shielding and the provision necessary for transporting and installing it.

Penetrations through the shielding during decommissioning

7.79. Penetrations through the shielding introduce pathways by which gamma radiation can propagate preferentially. The basic means of controlling dose rates due to penetrations, point for access by personnel or equipment provided in paras 5.53 - 5.56 are applicable for decommissioning as well.

Shielding / Barriers during decommissioning

7.80. The provision for shielding that is incorporated into the design to protect site personnel during dismantling phases and to protect the public under normal and accident conditions from direct or scattered radiation should also be designed to ensure adequate protection of the public during plant decommissioning. In this respect it may be necessary to consider ‘sky shine’, particularly if buildings have roofs of light construction, and to restrict public access to the site by providing barriers such as fences. Notably temporally waste storage units generally located at the periphery of the NPP site should be designed as having a limited impact to personnel and public doses.

8. DESIGN FOR RADIATION MONITORING FOR OPERATIONAL STATES, ACCIDENT CONDITIONS AND DECOMMISSIONING

GENERAL

8.1. For the effective implementation of the provision in the plant design for the radiation protection of site personnel, the public and the environment[23], a well planned radiation monitoring programme is required (GSR Part 3 [2], paras 3.37 and 3.38). The requirements for the operational aspects of such a radiation monitoring programme are established in requirements 19 – 33 and paras 3.68 - 3.144 of GSR Part 3 [2] and in Requirement 82 of SSR-2/1 (Rev. 1) [1].

8.2. Elements of the radiation monitoring programmes should include at least the monitoring of the following, as appropriate:

(i) External and internal exposures of workers;

(ii) Discharges;

(iii) Background radioactivity in the environment;

(iv) Other parameters important for the assessment of public exposure (e.g. environmental and meteorological conditions at the site).

8.3. The monitoring plan should include the type of parameters to be monitored, the type of data to be collected, the methodology for data collection (including the location and frequency of data collection), the required resolution and precision of any measurements, data backup requirements, as well as requirements for data processing and analysis.

8.4. Radiation monitoring systems should be designed to be calibrated, tested, maintained, repaired or replaced, inspected and monitored as required to ensure their capability of performing their functions in all conditions specified in their design basis in accordance with the Requirement 29 of SSR-2/1 (Rev. 1) [1].

8.5. In compliance with Requirement 30 and with paras 5.48 - 5.50 of SSR-2/1 (Rev. 1) [1] it should be verified that radiation monitoring systems are capable of performing their intended functions when necessary, and in the prevailing environmental conditions, throughout their design life, with due account taken of plant conditions during maintenance and testing.

8.6. Monitoring for purposes of radiation protection is required for both plant operation and decommissioning, and the general provisions discussed here are required for both. However, in the later stages of decommissioning, some of the initial monitoring equipment may have been removed or become unnecessary or different measures for monitoring may have become necessary by virtue of the

[23] In this publication "protection of the environment" is also included in radiation protection in accordance with GSG-10 [20].

decommissioning activities. The design of the monitoring system should therefore be reviewed before each stage of decommissioning begins.

8.7. Installed and portable equipment for radiation measurement is used to ensure the protection of the personnel in the plant, the environment and the public from radiation that is produced during both plant operation and decommissioning. This is achieved by monitoring ambient conditions in the workplace and off the site and by monitoring personnel for contamination at fixed points of access and egress between different zones.

8.8. Radiation dose rates, radiation doses and radioactive material in systems and rooms at the plant and releases of radioactive material should be monitored by this instrumentation. Air monitoring systems should be provided to detect radioactive material in the air of the rooms and in the ventilation systems. Radiation measurements should be made on process streams to monitor the transport of radioactive material in liquid and gas systems inside the plant. Radiation measurements of releases should be made to monitor both liquid and gaseous radioactive effluents from the plant. Some of the radiation measuring systems and equipment may also provide information that is relevant to the operation of other systems.

8.9. Equipment for performing these monitoring tasks should be provided in the design of a nuclear power plant. The rationale and the design basis for the measurement channels, their measuring ranges and detector locations should be documented. These systems are subject to national regulatory requirements, and should be designed, implemented, calibrated, tested, maintained, repaired or replaced, inspected and monitored in compliance with relevant national and international codes and standards. Items important to safety should be redundant to ensure that monitoring is always possible and should have reliable and diverse data communication system with a backup power supply that can operate independently for a period consistent with the safety analyses performed in the event of a power outage (SSG-54 [53] para. 2.115). In special cases, it may be necessary to use two or more measuring channels to cover the specified range of measurement. In these cases, the measuring ranges should overlap sufficiently. Basic information on the electrotechnical and radiation measuring requirements for the design of instrumentation and devices is given in the standards of the International Electrotechnical Commission (IEC) and the International Organization for Standardization (ISO).

8.10. In compliance with Requirement 19 of GSR Part 4 (Rev. 1) [6] data on operational safety performance, including radiation doses and the generation of radioactive waste and effluents should be collected and assessed. The scope of the data to be collected for facilities and activities should be in accordance with the graded approach. Data on operating experience should be used, as appropriate, to update the safety assessment.

8.11. In the selection of radiation monitoring devices the following characteristics, at the minimum, should be considered:

(1) Range of dose rate or activity concentration;

(2) Physical quantities and units displayed and archived,

(3) Device sensitivity and accuracy;

(4) Uncertainty of measurement results;

(5) Radionuclides to be monitored;

(6) Provision of threshold alarms;

(7) Power supply and backup power supply;

(8) Environmental conditions;

(9) Provision for testing, calibration and easy maintenance;

(10) Provision for functioning in all plant states, including accident conditions;

(11) Response to overload conditions;

(12) Failure mode indication;

(13) Potential for interference with or corruption of monitored data due to other radionuclides present in the area, for example in the case of monitoring for neutrons and tritium;

(14) Provision for saving the results in searchable database.

8.12. Recommendations regarding instrumentation and control systems, including general recommendations and design guidelines, provided in SSG-39 [32] should be taken into account.

8.13. Measurement systems should be designed to maintain their operability under specified environmental conditions. The range of conditions of temperature, pressure, humidity, vibration and ambient radiation fields at least should be specified. Regarding the radiation field dose, dose rate and at least gamma and neutron radiation should be taken into consideration.

8.14. Continuity of power for the monitoring of the key plant parameters and for the completion of short term actions necessary for safety should be maintained in the event of loss of the AC (alternating current) power sources (SSR-2/1 (Rev. 1) [1], para 6.44D).

8.15. A system that gives relevant data on measured radiation values at the plant should be provided in the main control room, the health physics room, at some local control points and in the plant's computer information system. Alarm signals should be provided to the extent that is justified on the basis of the design goals of the radiation measuring systems. All data relevant for radiation protection, together with the time of the measurement and the coordinates of the measurement location should be available in real time in the emergency response facility.

8.16. In compliance with Requirement 6 and para. 3.4 of GSR Part 6 [8] and paras 2.11 and 2.14 of SSG-47 [54] the initial decommissioning plan prepared during the plant design stage should demonstrate that decommissioning can be performed in a safe manner. Accordingly, decommissioning plan and its supporting documents should indicate that the national regulatory requirements for protection of the workers, the environment and the public will be ensured during decommissioning. Responsibilities and measures for radiation monitoring, discharge control and radiological surveys during decommissioning and after its completion should be indicated in the decommissioning plan. It should be also considered that decommissioning actions might result in elevated discharges for a limited period of time.

8.17. In compliance with para 9.3 of GSR Part 6 [8], if the approved decommissioning end state is release from regulatory control with restrictions on the future use of the remaining structures, appropriate controls and programmes for monitoring and surveillance should be established and maintained for the optimization of protection and safety, and protection of the environment. It should be considered, weather any action is necessary during the design phase.

AREA MONITORING SYSTEMS WITHIN THE PLANT

8.18. The type and frequency of workplace monitoring should be sufficient to enable the evaluation of the radiological conditions in all workplaces (para. 3.97 of GSR Part 3 [2]).

8.19. Area monitoring includes the measurement of dose rates, activity concentration in air and surface contamination. Passive detectors (e. g. thermo luminescent dosimeters) might be installed for backup and retrospective evaluation of radiation environment.

8.20. In the controlled areas, continuously operating fixed instruments with a local alarm and an unambiguous readout should be installed so as to give information on radiation dose rates and airborne contamination in selected areas. Further recommendations on how to ensure the habitability of the main control room in the event of radioactive contamination of the site are provided in para 4.86 of SSG-62 [30]. For monitoring special maintenance operations that last only a short time, and especially for monitoring in areas where high dose rates may vary, complementary portable dose rate meters should also be provided, with alarms to notify if preset values are exceeded. When designing audible alarm systems, the likely noise level in the relevant areas should be taken into account, application of visual signal might be advantageous in such environment.

8.21. For monitoring of special maintenance operations and especially in case of operations with possible contaminations surface contamination meters should be provided. Tools and methods for sample collection, sample preparation and laboratory measurement of the samples should be developed and available.

8.22. External radiation monitoring systems should be installed in:

— The reactor containment;

— The rooms that are adjacent to the upper part (refuelling area) of the containment;

— The spent fuel storage facility;

— The fuel handling machine;

— The treatment and storage facilities for radioactive waste;

— The decontamination facilities;

— The transport routes for fuel and waste;

— Areas for the handling and storage of fresh mixed oxide fuel or fresh fuel containing reprocessed uranium;

— Emergency response facilities.

8.23. Permanently installed monitors for detecting radioactive contamination in air should be provided at selected locations in a nuclear power plant. The activity concentration in air should be determined, at least for those accessible rooms of the controlled area where airborne radioactive material may be present in amounts that could influence the radiation doses to workers. Monitors should also be located at the ventilation ducts for exhaust air from the following areas:

1. The containment;
2. The fuel storage facilities;
3. The auxiliary building;
4. The radioactive waste building.

8.24. In selecting these air monitors, the physical form (i. e. in gaseous or particulate form) in which airborne contamination is present as well as the chemical forms of certain radionuclides (e. g. radioactive iodine) should be taken into account. Measurements of air contamination should be conducted in a way that makes the sampling as representative as practicable.

8.25. Provision should also be made for the monitoring of air and surface contamination at the entries to and exits from areas where radiation work is to be carried out.

INDIVIDUAL MONITORING

8.26. Facilities should be provided for monitoring for exposure and contamination of operating personnel. Equipment for monitoring individual doses to workers should include the means necessary to measure, evaluate and record the doses received from external and internal sources. Contributions of alpha, beta, gamma and neutron radiation should be taken into consideration. These equipment and systems are subject to national regulatory requirements. The general operational aspects of individual monitoring of doses (external radiation dosimeters, methods of assessment of internal dose, etc.) are considered in GSG-7 [10].

8.27. At entrances to controlled areas equipment for individual monitoring should be provided. At exits from controlled areas the following equipment should be provided (para 3.90 of GSR Part 3 [2]):

- Equipment for monitoring for contamination of skin and clothing;
- Equipment for monitoring for contamination of any objects or material being removed from the area;
- Washing or showering facilities and other personal decontamination facilities.

MONITORING OF DISCHARGES

8.28. Equipment is required to be provided to monitor and record all discharges of radioactive liquid and gaseous effluents to the environment (SSR-2/1 (Rev. 1) [1]). In addition, equipment should be provided to monitor systems that may contribute to large fractions of the overall releases of the plant. Design specific aspects should be taken into consideration. Monitoring of the following systems should be provided where applicable:

— Plant off-gas system;

— Vent header of radioactive waste tanks;

— Building ventilation with potential radioactive contamination;

— Liquid effluents of radioactive waste system;

— Condenser cooling water discharge.

8.29. Monitoring programmes should be designed to demonstrate that discharges are in compliance with the limits and in order to check the assumptions used to evaluate doses to the representative person, in accordance with para. 5.75 of GSG-9 [23] and with RS-G-1.8 [67]. Monitoring of the source involves measuring activity concentrations or dose rates at the discharge point or within the facility.

8.30. In compliance with para 5.77 of GSG-9 [23], monitoring programmes should be developed and conducted in line with a graded approach.

8.31. The equipment for effluent monitoring should be capable of determining the total activity and the nuclide composition of the discharge. This may be done by on-line measurements and laboratory analysis. Guidance on monitoring of effluents is provided in GSG-9 [23].

8.32. The regulatory body can make provision for independent monitoring. The characteristics of independent monitoring and the resources devoted to independent monitoring should be based on a graded approach and should incorporate best practices and scientifically sound analytical methods. Such monitoring may be undertaken by the regulatory body or on behalf of the regulatory body by another organization that is independent of the operating organization.

ENVIRONMENTAL MONITORING

8.33. In para. 6.84 of SSR-2/1 (Rev. 1) [1] it is stated:

> "6.84. Arrangements should be made to assess exposures and other radiological impacts, if any, in the vicinity of the plant by environmental monitoring of dose rates or activity concentrations, with particular reference to:
>
> (a) Exposure pathways to people, including the food chain;
>
> (b) Radiological impacts, if any, on the local environment;
>
> (c) The possible buildup, and accumulation in the environment, of radioactive substances;
>
> (d) The possibility of any unauthorized routes for radioactive releases."

8.34. Monitoring programmes should be designed involving the measurement of radionuclide concentrations in environmental media and of doses or dose rates in the environment. Methods and tools should be appropriate for measurements in normal operational states, in design basis accident conditions and also in design extension conditions.

8.35. Environmental monitoring together with effluent monitoring should provide sufficient information to determine whether the levels of public exposures comply with the dose limits and to demonstrate that protection and safety is optimized.

8.36. Environmental monitoring should be designed to verify the results of discharge monitoring, and to confirm predictions of radionuclide transfer in the environment.

8.37. The environmental monitoring programme should be established during site characterization (see recommendations provided in NS-G-3.2 [19] and GSG-10 [20]). Further details about environmental monitoring including requirements and operational aspects are considered in RS-G-1.8 [67].

PROCESS MONITORING

8.38. In compliance with Requirement 71 of SSR-2/1 (Rev. 1) [1] process sampling systems and post-accident sampling systems should be provided for determining, in a timely manner, the concentration of specified radionuclides in fluid process systems, and in gas and liquid samples taken from systems or from the environment, in all operational states and in accident conditions at the nuclear power plant. Nuclear power plants should be fitted with installed radiation measuring systems for monitoring activity concentrations for process fluids and gases. The purpose of these measurements is to detect fuel failures and the leakage of radioactive material from or into a process system.

8.39. Installed radiation measuring equipment should be used for monitoring activity concentrations in the primary circuit water and secondary circuit of pressurized water reactors and of the primary coolant and main steam lines of boiling water reactors. Design of process radiation monitoring should depend on the type of reactor and design of the plant. Large leaks, which may necessitate rapid action, may be detected by means of radiation monitoring of either the main secondary steam lines (response to ^{16}N) or the main condenser air exhaust lines (response to fission products).

8.40. Treatment systems for radioactive gases as well as treatment systems for liquid and solid waste should be fitted with suitable systems for process radiation monitoring.

8.41. Appropriate means should be provided to allow monitoring of the activity in fluid systems that have a potential for significant radioactive contamination. In addition, means should be provided for the collection of process samples for more detailed analysis in on-site radiochemical laboratories. Since they are hard to measure and have high risk in case of internal contamination, special attention should be paid to waste with alpha decay nuclides generated during radiochemical processing.

8.42. Auxiliary systems that may also become contaminated are:

- Storage, cooling and cleanup systems for irradiated fuel;
- Sumps connected to drain systems for radioactive liquids;
- Ventilation ducts for radioactive discharges;
- Circuits or systems separated by only one barrier from radioactive circuits (e.g. which may become contaminated owing to leaks in heat exchangers).

Equipment should be provided for regular sampling to determine the radionuclide content of these systems.

8.43. Fuel elements are removed from the reactor core after a specified burnup or if they have unacceptable defects. A monitoring system should be incorporated into the reactor design to detect defects in fuel elements. This system may operate by measuring the activity of those fission products that are the most significant for the detection of unacceptable defects in fuel elements in the bulk coolant or in the bulk off-gas during operation of the plant. A monitoring system should be capable of identifying specific fuel elements or channels containing elements that have unacceptable defects. This may be done either on-line or under shutdown conditions.

8.44. In compliance with para 6.68A of SSR-2/1 (Rev. 1) [1] the design should also include monitoring and controlling the activity in water and in air for operational states and means for monitoring the activity in water and in air for accident conditions that are of relevance for the spent fuel pool.

RADIATION MONITORING UNDER ACCIDENT CONDITIONS

8.45. The radiation monitoring systems at a nuclear power plant should include provisions that are relevant to postulated accidents and, to the extent that is necessary and practicable, they should also be operable during severe accidents. Provision should be made for having portable monitoring instrumentation (for monitoring dose rates and surface and airborne contamination) with ranges that are appropriate for severe accident conditions. The aim should be to enable the operator to have a quick and reliable way of assessing radiation levels throughout the plant and in its vicinity and, consequently, to take any action that may be necessary under such accident conditions.

8.46. Further recommendations on emergency response are given in GSR Part 7 [9]. Recommendations in respect of the organization for planning and conducting the emergency response following an accident, and the monitoring that is necessary to ensure that access can be gained, where required, after an accident at a nuclear power plant are established in GSR Part 7 [9] and SR Series No. 32 [65]. Special attention should be paid to the occupancy of the main control room and the necessary emergency response measures on the site.

8.47. In accordance with international recommendations in GSG-7 [10], arrangements should be made to assess promptly: exposures and releases of radioactive material; and radiological conditions on and off the site. This should include acquiring the information needed in support of mitigatory action by the operating organizations, emergency classification, urgent protective actions on the site, the protection of workers and recommendations for urgent protective actions to be taken off the site. These arrangements should also include providing access to instruments displaying or measuring those parameters that can readily be measured or observed in the event of a nuclear or radiological emergency and which form the basis for classifying events. The response of instrumentation or systems at the facility should be adequate for the full range of emergencies, including severe accidents, as agreed with the regulatory body.

8.48. Means should be provided so that the operating organizations is aware of the performance of the radiation monitoring systems under the environmental conditions that occur as a result of an accident. The most onerous design requirements are associated with the radiation measurement systems that are within or close to the reactor containment.

8.49. A proper assessment should be made of all the possible areas for concentrations of radioactive material within the plant and the releases that may occur as a result of accidents, including the nuclide composition of the releases and the expected environmental contamination, to ensure that the design of the instrumentation is adequate to achieve its purpose, which includes ensuring that it covers the necessary range. This is particularly true for severe accidents where the radiation fields within the containment and in the gases that may be discharged from it may reach ambient levels of external radiation giving rise to dose rates of up to 10^6 Gy/h and activity concentrations of iodides and aerosols of up to 10^{15} Bq/m^3.

8.50. The operability of measurement systems should be maintained under specified environmental conditions following accidents. The operational ranges of temperature, pressure, humidity, vibration and ambient radiation fields at least should be specified.

8.51. Airborne iodine and particulate radioactive material should be measured by passing air samples through combined particulate and iodine filters, on which gamma ray spectroscopy can then be carried out either with mobile equipment or with equipment in a laboratory that is operable under accident conditions. Provision should be made in advance for the transport of mobile monitoring equipment.

8.52. For design basis accidents, the emergency power supply to the continuous radiation monitoring systems should comply with the single failure criterion.

8.53. The radiation measurement data under accident conditions should be available in the main control room, in the supplementary control room and other areas where information is needed for operation or for managing an accident. Suitable communications systems should be provided to enable information and instructions to be transmitted between different locations and to provide external communication with such other organizations as may be required. Provision should be made for the direct transfer of relevant data to the emergency response centre.

8.54. Following an accident, there should be a means of taking representative samples from both the gas and the water within the reactor containment for laboratory measurements. The sampling equipment should be designed to withstand not only design basis accident conditions but also design extension conditions. The laboratory should have arrangements for the safe handling and analysis of such 'hot' samples.

8.55. An automatic external radiation measuring network should be installed close to the site. This type of measuring system provides the operating organizations and the emergency response organization with real-time data on environmental radiation levels. Such data on environmental radiation levels are useful in the early phase of a release from a plant in making decisions on which emergency measures should be implemented and in determining the source term for radioactive releases outside the containment.

Appendix

APPLICATION OF THE OPTIMIZATION PRINCIPLE

A.1. Optimization should be applied in order to keep the risk of incurring exposures, the number of people exposed, and the magnitude of individual doses as low as reasonably achievable, taking economic and societal factors into account.

A.2. Optimization techniques should only be applied below any limits or constraints established by the government or regulatory body on risk or dose that they consider to be tolerable for a new nuclear power plant. Optimization arguments should not be used to justify levels of risk or dose above any limits or constraints set.

A.3. There is no level for risk or dose below which optimization is not required, however if it has been demonstrated that further reduction in risk or dose could not be achieved at reasonable cost, then further optimization should not be necessary.

A.4. The fundamental role of optimization in the design of a nuclear power plant and its components is to ensure that a structured approach is taken to making decisions on engineering provisions for controlling radiation-doses and risk. This is frequently a matter of judgement. The optimization should achieve an overall balance of safety, with account taken of the need for radiation risk or dose reduction, the impact on health, safety and wellbeing of staff, the impact on the public and the environment, security, the need to ensure reliable energy production and the costs involved. A qualitative approach based on the utilization of the best available and proven technology may be sufficient for making decisions on the optimum level of protection that can be achieved. At the design stage of a plant, or for a major modification or decommissioning, where a large expenditure is involved, the use of a more structured approach is appropriate (SSG-47 [54]) and decision aiding techniques should be used including identification of the key high dose activities which should be identified and prioritised for optimization.

A.5. For types of reactor for which significant operating experience is available, many of the criteria and input parameters that are required in such a decision making process ~~can~~ should be quantified. This is because:

- A considerable amount of data have been obtained from operating plants on parameters that are relevant to the exposure of site personnel and members of the public;
- Progress has been made in understanding the phenomena that determine the production and transport of radioactive material within the plant;
- Specialized computer software has been developed to make predictions for situations where the quality of the data is poor or where significant features of the design have been changed.

A.6. If such a database is available, a differential cost–benefit analysis or other appropriate methods (e. g. SR Series No. 21 [68] and ICRP 101b [69]) should be used. In some cases, it may not be possible to quantify all the factors involved or to express them in comparable units. It may also be difficult to balance individual and collective doses, and to take into account the implications for occupational doses of further reductions in public dose as well as the broader social factors that such a reduction might entail. For these situations, the use of more sophisticated qualitative decision aiding techniques such as multicriteria analysis may be useful. In these analyses, the options should be evaluated against several attributes. One such methodology is described in RCEP 12 Report [70].

A.7. If a differential cost–benefit analysis is performed, a monetary value for the averted dose needs to be established, which may or may not be approved by the regulatory body. Different values are used in different States [71, 72].

A.8. Where a monetary value of averted dose is used in the control of occupational exposure, a baseline value for the monetary value of dose and an increase in this value as the individual dose approaches the dose limit may be applied. This approach is consistent with the aim of avoiding major disparities in the doses that are received by personnel of different types who work in the controlled area. This relationship reflects the aversion to both such disparities and to the risk itself and ensures that the major effort is focused on those workers who may receive the highest doses.

A.9. The results of all such analyses are only a tool for use in the decision making process and do not provide the decision itself. There should be a major contribution from expert judgement. For example, an analysis may not be able to justify, on economic grounds, the provision of remote equipment to eliminate the need for personnel to enter areas with high radiation levels or contamination levels, but the decision may be taken to provide such equipment on social grounds. The level of sophistication with which these analyses are performed needs to reflect the magnitude of the dose that is under consideration.

A.10. Evolutionary designs, which have been designed taking account of experience of earlier ones, should show how the evolution has maintained or improved the design from a safety perspective. However, there are safety benefits in standardisation as a wider pool of experience will inevitably provide better feedback for future improvement in safety and this should also be considered.

A.11. In optimizing the design, it should be recognized that radiation is only one of several types of hazards that will be experienced by site personnel. Measures to reduce radiation exposure should not increase the total hazard [73].

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

[2] EUROPEAN COMMISSION, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards, IAEA Safety Standards Series No. GSR Part 3, IAEA, Vienna (2014).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Commissioning and Operation and IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), IAEA, Vienna (2016).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSR-1, IAEA, Vienna (2019).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), IAEA, Vienna (2016).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste, IAEA Safety Standards Series No. GSR Part 5, IAEA, Vienna (2009).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Facilities, IAEA Safety Standards Series No. GSR Part 6, IAEA, Vienna. (2014).

[9] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL CIVIL AVIATION ORGANIZATION, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, INTERPOL, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, PREPARATORY COMMISSION FOR THE COMPREHENSIVE NUCLEAR-TEST-BAN TREATY ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, UNITED NATIONS OFFICE FOR THE COORDINATION OF HUMANITARIAN AFFAIRS, WORLD HEALTH ORGANIZATION, WORLD METEOROLOGICAL ORGANIZATION, Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSR Part 7, IAEA, Vienna (2015).

[10] INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, Occupational Radiation Protection, IAEA Safety Standards Series No. GSG-7, IAEA, Vienna (2018).

[11] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors, IAEA Safety Standards Series No. SSG-40, IAEA, Vienna (2016).

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Predisposal Management of Radioactive Waste from Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. SSG-41 (2016).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Glossary: Terminology Used in Nuclear Safety and Radiation Protection, 2018 Revision, IAEA, Vienna.

[14] EUROPEAN ATOMIC ENERGY COMMUNITY, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1, IAEA, Vienna (2006).

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Nuclear Installations Against External Events Excluding Earthquakes, IAEA Safety Standards Series No. SSG-68, IAEA, Vienna (2021).

[16] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Design for Nuclear Installations, IAEA Safety Standards Series No. SSG-67, IAEA, Vienna (2021).

[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Hazards in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-9 (Rev. 1), IAEA, Vienna (2022).

[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of Seismic Safety for Existing Nuclear Installations, IAEA Safety Standards Series No. NS-G-2.13, IAEA, Vienna (2009). *(Revision: DS522).*

[19] INTERNATIONAL ATOMIC ENERGY AGENCY, Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear power Plants, IAEA Safety Standards Series No. NS-G-3.2, IAEA, Vienna (2002). *(Revision: DS529).*

[20] INTERNATIONAL ATOMIC ENERGY AGENCY, UNITED NATIONS ENVIRONMENT PROGRAMME, Prospective Radiological Environmental Impact Assessment for Facilities and Activities, IAEA, IAEA Safety Standards Series No. GSG-10 (2018).

[21] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5), IAEA Nuclear Security Series No. 13, IAEA, Vienna (2011).

[22] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Security Recommendations on Radioactive Material and Associated Facilities, IAEA Nuclear Security Series No. 14, IAEA, Vienna (2011).

[23] INTERNATIONAL ATOMIC ENERGY AGENCY, UNITED NATIONS ENVIRONMENT PROGRAMME, Regulatory Control of Radioactive Discharges to the Environment IAEA Safety Standards Series No. GSG-9, IAEA, Vienna (2018).

[24] INTERNATIONAL ATOMIC ENERGY AGENCY, Deterministic Safety Analysis for Nuclear Power Plants IAEA, IAEA Safety Standards Series No. SSG-2 (Rev. 1), Vienna (2019).

[25] INTERNATIONAL ATOMIC ENERGY AGENCY, UNITED NATIONS ENVIRONMENT PROGRAMME, Radiation Protection of the Public and the Environment, IAEA Safety Standards Series No. GSG-8, IAEA, Vienna (2018).

[26] INTERNATIONAL ATOMIC ENERGY AGENCY, UNITED NATIONS ENVIRONMENT PROGRAMME, Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste, IAEA Safety Standards Series No. GSG-3, IAEA, Vienna (2013).

[27] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of the Reactor Core for Nuclear Power Plants, IAEA Safety Standards Series No., SSG-52, IAEA, Vienna (2019).

[28] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of the Reactor Containment and Associated Systems for Nuclear Power Plants, IAEA Safety Standards Series No SSG-53, IAEA, Vienna (2019).

[29] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-56, IAEA, Vienna (2020).

[30] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Auxiliary Systems and Supporting Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-62, IAEA, Vienna (2020).

[31] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Fuel Handling and Storage Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-63, IAEA, Vienna (2020). *(Supersedes NS-G-1.4).*

[32] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Instrumentation and Control Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-39, IAEA, Vienna (2016).

[33] INTERNATIONAL ATOMIC ENERGY AGENCY, Self-assessment of Nuclear Security Culture in Facilities and Activities, IAEA Nuclear Security Series No. 28-T, IAEA, Vienna (2017).

[34] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, IAEA Safety Standards Series GS-G-3.1 IAEA, Vienna (2006).

[35] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Nuclear Installations, IAEA Safety Standards Series No. GS-G-3.5, IAEA, Vienna (2009).

[36] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, PAN AMERICAN HEALTH ORGANIZATION AND WORLD HEALTH ORGANIZATION, Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSG-2, IAEA, Vienna (2011).

[37] INTERNATIONAL ATOMIC ENERGY AGENCY, Protection against Internal Hazards in the Design of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-64, IAEA, Vienna (2021).

[38] INTERNATIONAL ATOMIC ENERGY AGENCY, National Nuclear Security Threat Assessment, Design Basis Threats and Representative Threat Statements, IAEA Nuclear Security Series No. 10-G (Rev. 1), IAEA, Vienna (2021).

[39] INTERNATIONAL ATOMIC ENERGY AGENCY, Security of Radioactive Material in Use and Storage and of Associated Materials, IAEA Nuclear Security Series No. 11-G (Rev. 1), IAEA, Vienna (2019).

[40] INTERNATIONAL ATOMIC ENERGY AGENCY, Establishing the Nuclear Security Infrastructure for a Nuclear Power Programme, IAEA Nuclear Security Series No. 19, IAEA, Vienna (2013).

[41] INTERNATIONAL ATOMIC ENERGY AGENCY, Use of Nuclear Material Accounting and Control for Nuclear Security Purposes at Facilities, IAEA Nuclear Security Series No. 25-G, IAEA, Vienna (2015).

[42] INTERNATIONAL ATOMIC ENERGY AGENCY, Physical Protection of Nuclear Material and Nuclear Facilities (Implementation of INFCIRC/225/Revision 5), IAEA Nuclear Security Series No. 27-G, IAEA, Vienna (2018).

[43] INTERNATIONAL ATOMIC ENERGY AGENCY, Operating Experience Feedback for Nuclear Installations, IAEA Safety Standards Series No. SSG-50, IAEA, Vienna (2018).

[44] INTERNATIONAL ATOMIC ENERGY AGENCY, Commissioning for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-28, IAEA, Vienna (2014).

[45] INTERNATIONAL ATOMIC ENERGY AGENCY, Storage of Spent Nuclear Fuel, IAEA Safety Standards Series No. SSG-15 (Rev. 1), IAEA, Vienna (2020) *(Supersedes SSG-15).*

[46] INTERNATIONAL ATOMIC ENERGY AGENCY, Chemistry Programme for Water Cooled Nuclear Power Plants IAEA Safety Standards Series No. SSG-13, IAEA, Vienna (2011).

[47] INTERNATIONAL ATOMIC ENERGY AGENCY, A Simplified Approach to Estimating Reference Source Terms for LWR Designs, IAEA-TECDOC-1127, IAEA, Vienna (1999).

[48] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, The 2007 Recommendations of the International Commission on Radiological Protection, Publication 103, Elsevier, Oxford (2007).

[49] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Protection from Potential Exposure — A Conceptual Framework, Publication 64, Pergamon Press, Oxford (1993).

[50] INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Potential Exposure in Nuclear Safety: A Report by the International Nuclear Safety Advisory Group, INSAG-9, IAEA, Vienna (1995).

[51] INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP, Basic Safety Principles for Nuclear Power Plants, 75-INSAG-3 Rev. 1: A Report by the International Nuclear Safety Advisory Group, INSAG-12, IAEA, Vienna (1999).

[52] INTERNATIONAL ATOMIC ENERGY AGENCY, Nuclear Security Culture, IAEA Nuclear Security Series No. 7, IAEA, Vienna (2008).

[53] INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Management Programmes for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-54, IAEA, Vienna (2016).

[54] INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. SSG-47, IAEA, Vienna (2018).

[55] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Classification of Structures, Systems and Components in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-30, IAEA, Vienna (2014).

[56] INTERNATIONAL ATOMIC ENERGY AGENCY, Computer Security Techniques for Nuclear Facilities, IAEA Nuclear Security Series No. 17-T (Rev. 1), IAEA, Vienna (2021).

[57] INTERNATIONAL ATOMIC ENERGY AGENCY, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-70, IAEA, Vienna (in preparation).

[58] ISO 26802: 2010 Nuclear Facilities. Criteria for the design and operation of containment and ventilation systems for nuclear reactors (2010).

[58] INTERNATIONAL ATOMIC ENERGY AGENCY, Decontamination Approaches during Outages in Nuclear Power Plants - Experiences and Lessons Learned, IAEA-TECDOC-1946, IAEA, Vienna (2021).

[60] INTERNATIONAL ATOMIC ENERGY AGENCY, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants, IAEA Safety Standards Series No. SSG-48, IAEA, Vienna (2018).

[61] INTERNATIONAL ATOMIC ENERGY AGENCY, Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-74, IAEA, Vienna (in preparation).

[62] INTERNATIONAL ATOMIC ENERGY AGENCY, Hazards Associated with External Human Induced Events in Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. SSG-79, IAEA, Vienna (in preparation).

[63] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Electrical Power Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-34, IAEA, Vienna (2016).

[64] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR OFFICE, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS OFFICE FOR THE COORDINATION OF HUMANITARIAN AFFAIRS AND WORLD HEALTH ORGANIZATION, Arrangements for Preparedness for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GS-G-2.1, IAEA, Vienna (2007). *(Revision: DS504)*.

[65] INTERNATIONAL ATOMIC ENERGY AGENCY, Implementation of Accident Management Programmes in Nuclear Power Plants, IAEA Safety Report Series No. 32 IAEA, Vienna, (2004).

[66] INTERNATIONAL ATOMIC ENERGY AGENCY, Equipment Qualification for Nuclear Installations, IAEA Safety Standards Series No. SSG-69, IAEA, Vienna (2021).

[67] INTERNATIONAL ATOMIC ENERGY AGENCY, Environmental and Source Monitoring for Purposes of Radiation Protection, IAEA Safety Standards Series No. RS-G-1.8, IAEA, Vienna (2005).

[68] INTERNATIONAL ATOMIC ENERGY AGENCY, optimization of radiation protection in the control of occupational exposure, Safety Reports Series No. 21, IAEA, Vienna (2002).

[69] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, (ICRP), 2006. The Optimisation of Radiological Protection - Broadening the Process. ICRP Publication 101b. Ann. ICRP 36 (3).

[70] ROYAL COMMISSION ON ENVIRONMENTAL POLLUTION, Best Practicable Environment Option, 12th Report, HMSO, London (1988).

[71] LOCHARD, J., LEFAURE, C., SCHIEBER, C., SCHNEIDER, T., A model for the determination of monetary values of the man-sievert, J. Radiol. Prot. 16 3 (1996) 201–204.

[72] LEFAURE, C., Monetary Values of the Person-Sievert — From Concept to Practice: The Findings of an International Survey, Rep. CEPN-R-254, Centre d'Étude sur l'Évaluation de la Protection dans le Domaine Nucléaire, Fontenayaux-Roses (1998).

[73] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, General Principles for the Radiation Protection of Workers, Publication 75, Pergamon Press, Oxford and New York (1997).

Annex I

SOURCES OF RADIATION AND SOURCE TERMS IN DIFFERENT PLANT STATES AND THEIR MINIMIZATION

General

I-1. One of the early tasks for the design~~er~~ is to identify all sources of radiation and potential releases of radioactive materials (source term) because they can affect radiation levels throughout the plant and in its surroundings. Radiation effects of the sources and the source terms under different conditions need to be assessed. Any practicable means need to be employed by which the amount and activity of sources or the releases of radioactive material can be reduced without excessive cost or reduction of the reliability of components.

I–2. Steps for assessment of radiation effects in addition to quantification of sources of radiation include identification and characterization of mechanisms and pathways for releases of radioactive materials within and outside of the plant. It needs to be recognized that radiation sources are located in many places (reactor core, reactor coolant system, fuel stores, systems for treatment and storage of radioactive wastes, etc), the spectrum of radionuclides is very large, and the mechanisms resulting in the potential source term to the environment vary according to specific conditions. All these steps are strongly determined by the plant design.

I-3. In the context of radiation sources, it is important to understand that a major source in a given operational regime or plant state may become a minor one in a different operational regime or plant state. Similarly, the importance may vary with the issue that is being addressed. Some isotopes that are of minor importance for dose rate considerations during operation become of major importance during decommissioning. Also, even when dealing with reactors of the same type, changes in the design may have a strong influence on the relative importance of different sources.

I–4. This Annex supports the use of the Safety Guide by providing examples of different items important for radiation protection of the personnel and the public for different plant states and for various designs. Different sections of this Annex include:

- Overview of potential sources of radiation in the plant,
- Sources of radiation and source term during normal operation and decommissioning,
- Source term under design basis accidents,
- Source term under design extension conditions,
- Methods of calculation.

Anticipated operational occurrences are not discussed in this Annex as a special category of plant states, since phenomenologically they are similar to design basis accidents.

OVERVIEW OF POTENTIAL SOURCES OF RADIATION IN THE PLANT

Fission products in the nuclear fuel in the core

I–5. The main source of radioactive material in the nuclear power plant are fission products and actinides produced in the nuclear fuel in the core. The majority of fission products and actinides are unstable nuclei and emit radiation. The inventory of fission products and other radionuclides in the reactor fuel and core depends on several factors, in particular on:

- Quantity of fissile material,
- Fuel power and burn-up: for isotopes with long half-life (years) inventory increases with burn-up, for isotopes with short half-life, inventory depends mainly on reactor power, after reaching certain stable value there is no further increase,
- Neutron flux distribution in the core, operational power history (including transients), fuel management,
- Decay time after reactor shutdown.

The most typical fission products from U-235 fission have mass numbers around 90-100 in one group, and 130-140 in another group. The number of radioisotopes produced by fission is very large, so that in radiological analysis more than 1000 isotopes are sometimes considered. Experience from the analysis of radiological consequences indicates that consideration of about 30-40 radionuclides which are either a major contributor to early phase dose or are likely to be released is sufficient to obtain reasonable prediction.

Particular attention should be paid to the use of nuclide inventory software for accidents. In general, the libraries of radionuclide analysis software are optimised for normal operation or for measurements in the natural environment. Therefore, it is essential to use a library appropriate for nuclides released during an accident.

Fission products and actinides in the reactor coolant

I–6. Fission products that are released from fuel with defective cladding are a source of radiation in the reactor coolant. The activity of this source depends on a large number of parameters: the number and size of cladding defects, the local power in the vicinity of the defect, the burnup of the fuel and others. In modern reactors, the occurrence of fuel cladding defects is extremely rare. Furthermore, the main cause of cladding defects (~80%), which is interaction with small migrating objects (debris), is considerably reduced when a filtering grid is installed in the lower part of the fuel assembly.

I–7. Fission products also enter the coolant from residual surface contamination of the cladding by uranium (the efficiency of the cleaning in the manufacturing process is not absolute) and also from the uranium content of the cladding (a few ppm). A limit for uranium contamination (‘tramp uranium’) therefore needs to be specified.

I-8. Defects in the fuel cladding may result in the release of fission products to the coolant, which can add significantly to the activity of the coolant and contamination of the cooling circuit. Defective fuel elements need to be removed as soon as possible after a failure occurs to reduce the exposure of site personnel from this source. Where refuelling is not on-load, means are provided for detecting failed cladding and appropriate limits are set for the coolant activity and for the plant shutting down within a prescribed time interval if these are exceeded.

I–9. In addition, for Pressurized water reactors, a spiking phenomenon is observed for fission products during the shutdown and other reactor transients. The effect of spiking is further enhanced in the case that the process is associated with increase of fuel temperature (e. g. in reactivity accidents and steam line breaks). The fission products that are accumulated in all the spaces in the fuel element (in fractures in the fuel pellets, in the gap between the fuel pellets and the cladding, and in the expansion chamber) may be released to the coolant when the pressure is decreased. Water can enter the fuel and wash out the fission products. Thus, the release is not limited to gases and volatile species. The release depends mainly on the characteristics of the cladding defects.

Corrosion products in the reactor coolant

I-10. Corrosion products contained in the coolant are activated as a result of temporary deposition in the core and during the normal passage of the coolant through the core. They are deposited in other parts of the primary circuit. This source needs to be minimized by the following means: (a) reducing the corrosion and erosion rate by the proper selection of materials and the control of the coolant chemistry; (b) selection of materials to minimize the concentration of nuclides (particularly of cobalt in steel) that are known from experience to become major sources of radiation; (c) providing removal systems (such as particulate filters and ion exchange resins); (d) minimizing the concentration in feedwater of nuclides that can be activated in the core ; (e) providing surface treatments to reduce adherence of corrosion products to pipes and components.

I-11. The presence of materials with a high cobalt content (which produces the activation product ^{60}Co), such as Stellite, needs to be reduced by applying the optimization principle. This is particularly important for components within the reactor core. Stellite is used for valve facings, elements of the control rod drive mechanism, reactor vessel internal components and bearings, in the primary coolant circuit and chemical control circuits, in the turbine systems of boiling water reactors and in directly connected circuits because of its hardness. In the case of direct cycle reactors, the use of materials with high cobalt content needs to be minimized in components of the feedwater system that are situated after the condensate purification plant. For direct cycle, light water cooled, pressure tube reactors, for which the pressure tube and fuel cladding are made of zirconium or zirconium alloys of high purity and low activation cross-section, another important source of corrosion products (crud) is the feedwater circuit following the condensate polishing plant. Special attention requires the choice of heater material for the feedwater, with due consideration to the possible installation of filters in the feedwater or core coolant return circuit close to the core inlet.

I-12. Special attention is needed for the selection of materials and to the coolant chemistry, which also make an important contribution to the reliability of the steam supply system for the nuclear plant. The compatibility of materials and coolant, which is of the utmost importance to minimizing the amount of maintenance, repair and statutory inspection necessary for primary circuit components, needs to be given careful consideration. Only those materials are to be used that have been shown to be compatible with the coolant under the conditions (of temperature of coolant and material and coolant composition) that will prevail in the reactor. A specific concern is the possible occurrence of intergranular stress corrosion cracking.

I-13. Hard facing materials with lower cobalt content have been used to replace Stellite as a dose reduction measure in some primary circuit components of Pressurized water reactors such as Konvoi and EPR. Stellite and potential replacement materials do not have identical physical properties so selection of hard facing materials for primary circuit components needs to be carefully optimised including consideration of required material properties as well as the consequences to occupational exposure of the material selected.

I-14. The cobalt content of stainless steel and nickel-based alloys in contact with reactor primary coolant and/or under neutron fluid should be as low as is reasonably achievable to achieve occupational exposures as low as is reasonably achievable. The maximum values of cobalt content and stainless steel and nickel-based alloys in contact with reactor primary coolant and/or under neutron flux should be specified and strictly controlled.

I-15. Other materials such as silver (for example in control rods and seals) and antimony (used in seals and pump bearings) that produce activation products contributing significantly to occupational exposure need also be reduced or eliminated by applying optimisation processes.

I-16. Primary circuit chemistry needs to be optimised to reduce corrosion and release from components (especially steam generator tubing) in contact with the primary coolant and hence reduce the levels of corrosion products to levels that are ALARA.

I-17. In water cooled reactors, corrosion products are removed by treating the water with ion exchange resins to remove soluble species and by the installation of particulate filters. Their capacities need to be adequate to cope with the enhanced release of corrosion products (‘crud bursts’) and fission products (‘spiking’) that occurs during the start-up and cooldown stages.

I-18. Systems to remove corrosion products, both radioactive and non-radioactive, are to be provided for the primary coolant for all types of reactors, whether the coolant is liquid or gas.

Activation products in the reactor coolant

I–19. If the coolant contains oxygen (such as in light water reactors, heavy water reactors and CO_2 cooled reactors), a major source of radiation during power operation will be ^{16}N, which is formed by the interaction of fast neutrons with ^{16}O as the coolant passes through the reactor core. Nitrogen-16 is a strong gamma emitter with gamma ray energies of 6 and 7 MeV. Since the half-life of ^{16}N is short (7.1 s), the significance of this isotope will be reduced where the transport time between the core and a component in the coolant system is long compared with the half-life. In this case, other activation products of the coolant such as ^{41}Ar (gas-cooled reactors), ^{19}O and ^{18}F (water cooled reactors) may be the most important contributors to the radiation levels. In a pressurized water reactor, where the time for the coolant to traverse one loop is of the same order of magnitude as the half-life of ^{16}N, this isotope is a dominant contributor to the dose rate around the primary circuit during operation. Thus, ^{16}N activity has to be taken into account in the design of the shielding, especially if reactor building access is planned during power operation. The let-down water pipe needs to be sufficiently long to ensure decay of ^{16}N prior to the pipe exiting to the auxiliary building where operator access may be permitted during operation.

I–20. In water cooled reactors, and particularly in heavy water reactors, tritium is an important source of internal radiation exposure. In light water reactors, tritium as tritiated water is an important source in liquid and gaseous effluents released to the environment since there is currently no cost-effective method for removing it from waste streams. In addition, in the case of heavy water moderated reactors, photoneutrons are emitted from the interaction of gamma rays with deuterium.

I–21. In the case of light water reactors, activation products will be produced mainly in the materials of the structure of the fuel assemblies, in the cladding of the fuel pins, in the pressure vessel internal structures, in the control rods, in the primary and secondary neutron sources pins, in the pressure vessel itself, in the water and its impurities, and in the primary shield. In the case of gas-cooled reactors, the activation products will be mainly in the fuel cladding and the shield material within the pressure vessel (i. e. between the reactor core and the heat exchangers and above and below the core), in the restraint tank and to some extent in the heat exchangers themselves. In heavy water reactors of the pressure tube type, activation products are found mainly in fuel cladding, pressure tubes, calandria tubes, control tubes, the calandria tank and the shield tanks. Boron/lithium free chemistry can significantly reduce the tritium source term.

I–22. For fast breeder reactors with sodium as a coolant, where the coolant pumps and steam generators are inside the vessel, the secondary coolant and the structural materials of the components become activated. The most important radionuclides are ^{22}Na, ^{24}Na, ^{54}Mn, ^{58}Co, ^{60}Co and ^{59}Fe.

Activity in spent fuel pool water

I-23. Water in the fuel storage pool is maintained at a low activity level by means of a clean-up system consisting of particulate filters and ion exchange resins. Where modifications are made to the fuel storage pool of a reactor in which there have been major fuel failures, the design provides a means for containing any radioactive material that might otherwise leak into the pool water by bottling the fuel or some equivalent handling.

Filtration and purification systems

I–24. In the cleanup systems of water cooled and moderated reactors (such as light water reactors and heavy water reactors), there will be an accumulation of radioactive material in filters and ion exchange resins. This will consist of fission products such as iodine and caesium that have escaped to the coolant through fuel cladding defects,[24] and of radioactive corrosion products that are transported by the coolant or moderator. Filters and ion exchange resins and, more generally, all components in which an accumulation of radioactive products occurs, will generate very high activities that require shielding. Radioactive noble gases may be formed in these filters by the decay of iodine isotopes. In heavy water reactors, photoneutrons are produced in the heavy water by the photons from ^{16}N. This source is significant in determining the shielding requirements of the coolant circuit external to the core. In gas-cooled reactors, the gas treatment system will accumulate activated corrosion products such as ^{58}Co and ^{60}Co, and fission products such as iodine and caesium, and it will become an important source of radiation.

Secondary coolant system

I–25. In pressurized water reactors and pressurized heavy water reactors, the steam and turbine system is separated from the radioactive systems by a material barrier (the heat exchanger tubes). Thus, in these reactors radioactive material can only reach the steam and turbine system if leaks occur between the primary and secondary circuits. Provided that the leak rates are monitored (e. g. by measurement of the activity of the water or of ^{16}N in the secondary circuit) and kept to such a level that the activity in the secondary system is low, protective measures against direct and scattered radiation from this system are not necessary. Thus, the maximum tolerable leakage rate between the primary and secondary circuits needs to be kept very low.

I–26. In direct cycle plants, an additional source of secondary system contamination that needs to be considered is leakage from equipment for concentrating radioactive waste that involves steam heating. One such source of contamination is through tube leaks that allow contaminated waste to enter the condensed heating steam. Contaminated condensed water from such steam may then be introduced into the secondary system.

[24] In reactors with on-load refuelling and a detection capability for failed fuel, the release of fission products to the coolant can be kept low.

I–27. In fast breeder reactors, the secondary sodium coolant may become activated to ^{22}Na and ^{24}Na. This can give rise to dose rates in parts of the buildings outside the containment if the delay for the sodium transport from the steam generator to these areas is not long compared with the half-lives of ^{22}Na and ^{24}Na.

Liquid waste treatment system

I–28. The liquid waste treatment system collects liquid waste and purifies it to such levels that it can be either reused in the plant, released in accordance with the relevant authorization, or disposed of safely in storage.

I–29. The composition of liquid wastes (i. e. activity concentration and solid and chemical content) varies according to their origin. It is general practice to segregate and treat liquid wastes according to their expected compositions. The liquids in the liquid waste treatment system therefore have a wide range of activity concentration. The segregation of liquid wastes could be made in accordance with the following categories:

— High purity (e. g. leakage wastes from the primary circuit of Pressurized water reactors during power operation);

— High chemical content (e. g. decontamination agents);

— High solid content (e. g. liquid wastes from floor drains);

— Detergent containing liquid wastes (e. g. liquid wastes from laundry drains and personnel showers);

— Oil containing liquid wastes (e. g. in gas-cooled reactors, liquid wastes from floor drains from the area of the lubricating oil tank for the circulator);

— Very high tritium content liquid wastes (for pressurized heavy water reactors).

I–30. The mixing of a small volume of effluent with a high activity concentration with a large volume of effluent with a low activity concentration is to be avoided.

I–31. In light water reactors, before treatment some of the liquid wastes may have a radionuclide content as high as that of the reactor coolant, with the exception of short lived nuclides, which will have decayed, and gases, which will have been evolved as a result of depressurization. Concentrations of up to a few 10^{10} Bq/ m^3 may be found in such untreated liquids. Thus, since the liquid waste treatment system processes active liquids, radioactive material will accumulate in parts of the system such as filters, ion exchangers and evaporators.

I–32. In most cases, the accumulated radionuclide content will consist of activated material such as ^{60}Co, ^{58}Co, ^{51}Cr, ^{54}Mn and ^{59}Fe (depending upon the composition and corrosion rates of the material used in the primary circuit). Fission products such as isotopes of iodine, caesium and strontium may be important if failure of fuel cladding occurs.

Gas treatment systems

Off gas system

I–33. A number of radioactive gases with relatively short half-lives (such as ^{16}N, ^{19}O, ^{13}N) are formed in water cooled reactors by activation of the coolant. Fission gases are also released to the coolant through fuel cladding defects. Where necessary, these gases are removed from the coolant by a special off gas system. In the special case of direct cycle boiling water reactors, these gases will only stay in the coolant for a short period of time before they are removed by the off-gas system. However, in indirect cycle systems such as Pressurized water reactors , the removal of fission gases may be necessary only before shutdown of the plant, when it will be essential to reduce the activity in systems that may have to be opened during shutdown.[25] In the case of defective fuel being present in the core and a high degassing rate (e.g. in a boiling water reactor), activity concentrations of the order of 5×10^{11} Bq/m^3 may be found in the high activity part (head end) of the system. An appreciable fraction of the radioactive material will, in this case, consist of short lived isotopes (e.g. with a half-life of less than 1 h). In cases where the average stay time of the gas in the primary circuits is long (as may be the case in a pressurized water reactor that is operated at a low degassing rate), isotopes with long half-lives will constitute the most significant fraction.

I–34. Components such as holdup tanks, holdup pipes, charcoal delay beds or cryogenic devices are provided in the off-gas system to delay the release to the environment of the extracted gases for a time that is sufficient to allow for a large fraction of the radionuclides to decay.

I–35. Of major importance in the design of an off-gas system is the formation of radiolytic gas in a direct cycle boiling water reactor and the existence of high hydrogen concentrations in the primary coolant of a pressurized water reactors. For pressurized heavy water reactors, large amounts of hydrogen could build up in the cover gas of the moderator and to some extent in the primary circuit. This may lead to the formation of combustible gas mixtures in those parts of the plant where air may enter the system. A recombiner needs to be provided to avoid the formation of such combustible mixtures. The reduction of the gas volume by the recombiner will also increase the delay time of a given system by a factor of about 10. Other solutions are possible, such as the strict separation obtained by physical means and by the application of appropriate procedures for aerated and hydrogenated gaseous effluents.

Solid waste

I–36. Apart from the fuel, the following constitute the major solid radioactive wastes in terms of activity and volume that arise during operation:

[25] In such plants, gases are usually removed by the purification system.

(1) Components and structures that become activated or contaminated and have to be removed (e. g. control rods, neutron source assemblies, defective pumps, flux measuring assemblies, structures or parts thereof);

(2) Irradiated components of the fuel assembly from gas-cooled reactors (in these reactors the assemblies are dismantled at the nuclear power plant);

(3) Ion exchange resins, filter material, filter coating material, catalysts, desiccants and similar;

(4) Concentrates from evaporators, precipitates;

(5) Contaminated tools;

(6) Contaminated clothing, towels, plastic sheet, paper and similar.

I–37. The total volume of unprocessed waste that arises per year of operation from a 1000 MW(e) nuclear power plant may be as high as a few tens up to few hundred cubic metres, the major part being low level waste. The activity concentration of the waste varies over a wide range, with a small percentage having a maximum activity concentration of the order of 5×10^{16} Bq/m^3 for activated components and 5×10^{14} Bq/m^3 for ion exchange resins and pre-coat filter material. In most cases, long lived activation products such as ^{60}Co and, when fuel cladding defects have occurred, long lived fission products (particularly ^{134}Cs and ^{137}Cs) are the major radioactive sources.

I–38. Solid waste needs to be carefully managed to allow its volume to be minimized. However, reducing releases to the environment to very low levels will result in an increase in the volume of solid waste.

Irradiated fuel

I–39. Irradiated fuel has a very high radionuclide content owing to the fission products and transuranics that accumulate in it. For on-load refuelling systems, delayed neutrons that are emitted from the fuel while it is in the refuelling system also have to be taken into account. An additional source of radiation arises from activation of the materials that are used to construct the fuel assemblies or stringers.

I–40. For wet (pool) fuel storage and handling systems, water cleanup systems with particulate filtration and ion exchange need to be provided. They are usually combined with heat removal systems. The radioactive content of the water is removed by filters and ion exchange resins, which themselves become sources of radiation. Contamination of the handling, cleanup and heat removal systems also gives rise to additional sources.

I–41. In several plants, a dry fuel handling system is used, with initial dry fuel storage of fuel assemblies prior to dismantling, followed by pool storage of the fuel elements. The fuel handling system and dry fuel store become contaminated owing to radioactive corrosion products that flake from the fuel elements. Some components from the dismantled fuel assemblies are stored in a vault at the nuclear power plant.

Storage of fresh fuel

I–42. Where fuel is manufactured from fresh uranium, the activity of fresh (unirradiated) fuel is low[26]. Since most of the radiation emitted by the fuel is not penetrating, it will be largely absorbed by the fuel cladding. Thus, the external exposure is of minor significance.

I–43. However, in the case of mixed oxide fuel, the new fuel may be radioactive as a result of the recycled plutonium that it contains. In some fuels, recycled uranium may be used. In this case, the new fuel will be a significant source of both neutrons and gamma and it will need to be shielded and contained at all times until it is inserted into the reactor. The magnitude of the neutron source term will depend upon the time that has elapsed since the plutonium was created, since actinides that emit neutrons will be produced as the plutonium decays.

I–44. In the case of ^{232}Th–^{233}U fuel, the new fuel may be highly radioactive owing to the presence of ^{232}U progeny. It will need to be shielded and contained at all times until it is inserted into the reactor.

Decontamination facilities

I–45. The radioactive material in the waste solutions consists mainly of ~~the~~ corrosion products containing radionuclides such as ^{60}Co, ^{58}Co, ^{51}Cr, ^{59}Fe, ^{54}Mn. This material arises from the decontamination of components, of contaminated areas, of reusable protective clothing and possibly also of personnel (see paras 3.62 and 7.30; 5.68-5.76 and 7.31-7.38 of this Safety Guide) in the facilities that are provided to remove radioactive contamination from surfaces. Whereas the activity concentrations in the waste arising from the decontamination of personnel and of clothing are low, concentrations may be medium or high in solutions arising from the decontamination of components before major repair work.

Miscellaneous sources

I–46. There are also other sources of radiation at nuclear power plants, such as neutron startup sources, corrosion samples, in-core and ex-core detectors, calibration sources for instruments and sources that are used for radiographic inspections.

SOURCES OF RADIATION AND SOURCE TERM DURING NORMAL OPERATION AND DECOMMISSIONING

I–47. Radiological effects of normal operation and decommissioning considers impacts on man or on non-human biota that are attributable to direct radiation and the releases of radioactive materials from the plant. The sources of radiation include sources on the site and also sources of radiation that are to

[26] The term 'fresh fuel' means new or unirradiated fuel, even though the fuel may have been fabricated from fissionable materials recovered by reprocessing previously irradiated fuel.

be or have been discharged off the site from the operation of the facilities and which could have a radiological impact on persons off the site. During normal operation, gaseous and liquid discharges from the plant are released in a controlled way.

I–48. In assessment of radiological impact, all external exposure including direct ionizing radiation from the buildings and facilities and from transportation, from the liquid discharges and from the plume of radioactive gases and aerosols released from the facility, external exposure from radioactive fall-out (deposition) on the ground and on skin, hair or clothing, radionuclides resuspended into the air, and internal exposure from inhalation of radionuclides are to be addressed.

I-49. Fission products and actinides are produced as a result of the fission process. The most significant isotopes in terms of doses to site personnel and members of the public are usually the isotopes of the noble gases iodine and caesium, but others such as strontium and the isotopes of plutonium may also be important. When the reactor is at power, the fuel elements emit neutrons and gamma rays as a result of the fission process and the decay of fission products. Gamma rays are also emitted as a result of neutron capture in the core and the surrounding material. Other forms of radiation such as beta particles and positrons are emitted from the core and the vessel region during power operation, but these are not important for the purposes of radiation protection because of the limited penetration range of these charged particles.

I-50. Fission products such as ^{131}I, ^{134}Cs and ^{137}Cs make a low contribution to dose rates around the reactor coolant circuits because both the source term and the deposition rate are low. However, this contribution to dose rates may increase significantly in situations where components such as heat exchangers and valves are opened or entered for maintenance and repair.

I–51. If a reactor continues to operate with significant fuel cladding defects, a non-negligible mass of fuel (a few grams to a few tens of grams) may be released to the coolant. In this situation, the alpha activity of the water and of the deposits may not be negligible (the alpha emitters, mainly in particulate form, deposit quickly). Together with fission and corrosion products, it is an important potential source of internal exposure when circuits and components are opened for maintenance and repair. It is also a potentially important source during decommissioning.

I–52. The main contributors to dose rates during maintenance and repair are activated corrosion products, such as ^{60}Co, ^{58}Co, ^{110m}Ag, ^{124}Sb, ^{54}Mn, ^{59}Fe and ^{51}Cr. These are present as deposits on all the components and pipes of the primary coolant circuit and the circuits that are connected to it.

I–53. In the case of Pressurized water reactors with nickel based materials in the steam generators, important phenomena occur during the period when the reactor is brought from operation at power to a cold shutdown state, namely involving major changes in the physical (temperature, pressure) and chemical (from reducing to oxidizing conditions, pH) conditions. The solubility of deposited oxides of corrosion products and metallic species increases considerably. A large amount of the activated corrosion products deposited on the fuel are released to the coolant and the activity concentration of the water may be increased by two or three orders of magnitude. The release rate is not constant, and it decreases when the temperature is decreased from hot conditions to 80°C. The release increases sharply when peroxide water is injected, and a spike is observed (especially for ^{58}Co). After the spike, the evolution of the activity concentration of water is determined by the purification constant (i. e. the ratio of the purification flow rate to the mass of water). The dissolution of deposits outside the core is generally negligible. No decontamination of these components (primary pipes, steam generator, pumps) is therefore observed. The corrosion products of high activity that are removed after the spike occurred accumulate mainly on the ion exchangers of the chemical and volumetric control system. The activity may be equal to the total activity accumulated on the fuel during the operational period. These phenomena are greatly influenced by the design (mainly, the composition of the alloy in the steam generator tubes, which may be nickel or iron based). During this period of primary coolant purification, the contribution of radioactive material in the water to dose rates around the reactor coolant system, chemical and volume control system and residual heat removal systems is not negligible in comparison with the contribution of the deposits. During purification, recontamination of the out-of-flux surfaces can be observed (^{58}Co in steam generators, ^{110m}Ag on cold parts of the circuits, on the Non-Regenerative Heat Exchanger for example). After purification, the main contributors to the dose rates are the activated corrosion product deposits.

I–54. The neutrons and gamma rays that are emitted by the core represent a very intense source. The residual neutron flux outside the primary shielding is a source of activation of structural materials. It can therefore induce a build-up of supplementary sources with associated dose rates during shutdown periods and will be a major source of radiation during the decommissioning of the plant.

I–55. Other sources (including ^{41}Ar, airborne contamination by ^{3}H and volatile fission products and rare gases) need to be considered when access to the reactor building is authorized during the operation of the reactor. In a pressurized water reactor, the activation of ^{40}Ar contained in the air is a source of ^{41}Ar which is a gamma ray emitter. The ventilation of the reactor cavity results (in some designs) in the ^{41}Ar contamination being transferred to the whole free volume of the reactor building above the operating deck. Although the corresponding dose rate (external exposure) is low, it may not be negligible when the individual dose rate target is less than 10 μSv/h or less. Tritium is also an important possible source of airborne contamination in heavy water reactors and in the fuel building of a light water reactor. Argon-41 is also produced in the CO_2 coolant of gas-cooled reactors and in the systems of heavy water reactors that contain helium gas, such as the liquid zone control system and the moderator cover gas system. In pressurized water reactors, if the coolant temperature is not properly controlled its reduction can result in significant iodine plate-out on reactor coolant piping. Such iodine can be released as airborne contamination during shutdown, when the primary circuit is drained down. Iodine ‘hideout’ can become airborne and result in significant airborne contamination levels in the containment. This is the case even with low levels of fuel failure.

I–56. After shutdown of the plant, the main radiation source in the vicinity of the vessel is the gamma radiation from the fission products and activation products created in the vessel, in the metallic parts of the insulation and in any material that has been exposed for a sufficiently long time to the neutron flux. For some designs of heavy water reactor, neutrons produced by subcritical multiplication of the photoneutron source give rise to a significant power level accompanied by gamma radiation for a short period of time (about 24 h).

I–57. In cases where there is a separate oxygen containing fluid moderator system (such as in a pressure tube reactor), the isotope that is the major source of radiation during reactor operation will be ^{16}N. After shutdown, the radiation levels around the primary coolant system will be due mainly to activated corrosion products. The tritium present in the water coolant or moderator contributes to the radiation hazard only if it is released from the system and becomes airborne. This hazard has to be taken into account in the design of light water reactors also since operation with a limited leakage of primary coolant is tolerated.

I–58. In direct cycle water reactors, ^{16}N, which is carried over to the steam phase, will be the major source of radiation during power operation. The sky shine effect needs to be checked for carefully for buildings with potentially light structures, such as the roof of the turbine building or of the pressure suppression containment. Downstream from the condenser, ^{19}O also needs to be considered a major source of radiation. In the event of fuel pin failures, an additional source of radiation will be volatile fission products, mainly the noble gases, and volatile fission products such as iodine and caesium. During power operation, this source will be of minor importance compared with ^{16}N, but after reactor shutdown these isotopes and their progeny (e. g. ^{140}Ba) will be the major radiation source in this system. Another source may be non-volatile corrosion products that are carried over with water droplets in steam.

I–59. Carbon-14 is produced in light water reactors and heavy water reactors by (n, α) reactions with the ^{17}O present in the oxide fuel and moderator, by (n, p) reactions with the ^{14}N present in impurities in the fuel and by ternary fission. Because of the large moderator mass, ^{14}C is produced mainly from ^{17}O reactions in the moderator in heavy water reactors. This may be the main source term for this nuclide and a contributor to the global long-term collective dose commitment. However, in some heavy water reactor systems the contribution of ^{14}C to the total collective dose is relatively small because ^{14}C is effectively removed from the moderator by the purification system.

I–60. The coolant of some gas cooled reactors contains tritium, ^{35}S in the form of carbonyl sulphide and ^{14}C. The ^{35}S is produced mainly from the chlorine impurity in the graphite moderator, the tritium from the lithium impurity in the graphite and ^{14}C from the nitrogen impurity in the coolant and moderator.

Because these are pure beta emitters, they present a health hazard only if inhalation or ingestion of the isotopes is possible.

I–61. For fast breeder reactors with sodium coolant, the dominant sources are ^{22}Na and ^{24}Na. Sodium vapours may rise into primary components that may penetrate the cover plate shield of the reactor vessel. If these components penetrate the shield, considerable shielding is required to yield acceptable dose rates on the operating floor. Tritium that is generated in the fuel by ternary fission is released to the primary coolant through the stainless steel cladding of the fuel (the principal mechanism is diffusion). The Na coolant may be covered by an inert gas such as argon. The activation of the cover gas gives rise to ^{39}Ar and ^{41}Ar, which may leak into the reactor building.

I–62. In pressurized water reactors and pressurized heavy water reactors, provision needs to be made for cleaning the fluid circuits and for waste disposal from the secondary side in case primary to secondary leaks do occur. The leakage of primary coolant to the secondary circuit can also be detected by monitoring tritium in the feedwater. The presence of radioactivity in the feedwater can lead to the uncontrolled release of radioactive material to the environment through feedwater leaks as well as the venting of steam.

I–63. Increasing the delay time will reduce the content of short-lived isotopes in the effluent but will not significantly alter the content of isotopes with half-lives longer than the delay time. However, the increase of the delay time to 30 days considerably reduces the release of the rare gas effluents, particularly ^{133}Xe. In this case, the most important radionuclides that are released are ^{85}Kr and ^{14}C.

I–64. The ventilation of buildings may be a source of gaseous release and, to a less extent aerosols. The main isotopes are ^{3}H (from evaporation of the pools) and ^{41}Ar.

I–65. In some cases, it is not possible to prevent the dilution of radioactive gases with inactive gases such as air before they are processed. Examples of this are:

— The calandria vault gas (in pressure tube reactors);

— The cover gases of containers in which liquids with some content of volatile substances are stored (e. g. storage tanks for collected reactor coolant leakage water in light water reactors and storage tanks or some other equipment in the liquid waste treatment system). In some cases, gases are formed by decay, e. g. the decay of iodine to xenon;

— Coolant gas leaking into sections that contain air in gas cooled reactors;

— Air which has entered the pressure vessel of an light water reactor after it has been depressurized and the water level has been lowered prior to opening the vessel.

I–66. Vents for these gases need to be so located that the radioactive material they contain are kept away from the plant operators. In the case of AGRs and the calandria vault gas of pressure tube reactors, the radioactive material is mostly ^{41}Ar. In the case of light water reactors, fission product gases usually dominate. In pressure tube reactors, the same is true for process vents that are in direct contact with coolant (e. g. in storage tanks).

I–67. During the handling and storage of irradiated fuel, some radionuclides are released to the surrounding coolant. Radioactive corrosion products may go into solution or be released as particles while the fuel is being transported or stored in water, or if part of the fuel route is dry, and particularly if the cladding is oxidized, activated material may flake from the surface of fuel assemblies as a result of thermal or mechanical shock. In addition, defective fuel may release fission products, of which isotopes of noble gases, iodine, caesium and strontium are the most significant.

SOURCE TERM UNDER DESIGN BASIS ACCIDENTS

General

I–68. The main source of radiation in a nuclear power plant under accident conditions for which precautionary design measures are adopted consists of radioactive fission products. These are released either from the fuel elements or from the various systems and equipment in which they are normally retained. Examples of accidents in which there may be a release of fission products from the fuel elements are loss of coolant accidents (as a result of station blackout with loss of coolant through the valves or due to a pipe rupture) and reactivity accidents in which the fuel cladding may fail due to overpressurization or overheating of the cladding material. Another example of an accident in which fission products may be released from the fuel rods is an accident in handling spent fuel, which may result in a mechanical failure of the fuel cladding from the impact of a fuel element that is dropped. The most volatile radionuclides usually dominate the accident source term (the release to or from the reactor containment). Recommendations and guidance on the assessment of accidents are presented in Ref. [I–1].

I–69. Account needs to be taken of the possibility of radioactive material accumulating on and being released from air filters or components of the liquid waste treatment system after accidents. In comparison with the radiation emanating from fission products and actinides, activation products are usually of minor importance.

I–70. In the following subsections, examples of methods for determining radiation sources are described for selected design basis accidents. The scenarios are selected for illustrative purposes only and to cover all the major categories of design of nuclear power plants. Not all accident scenarios leading to radioactive releases are discussed here in the same detail. A generalized approach to evaluating the source term from severe accidents is given in Ref. [I–2].

Loss of coolant accidents in light water reactors

I–71. The number of fuel cladding failures that may be expected as a consequence of any of the potential range of loss of coolant accidents up to a double ended rupture of a main coolant pipe and the fraction of each fission product released from the failed fuel need to be calculated conservatively. The subsequent release of fission products from the coolant to the containment and their behaviour in the containment (e. g. plateout, deposition by dousing or spraying and iodine reactions within the building) need to be assessed, following the general rules for safety analysis [I-1]. For the purposes of this assessment, it needs to be assumed that the reactor core has operated for a sufficiently extended period so that the maximum fission product inventory is present in the core at the time of the accident. The leak rate of the containment as a function of time after the accident needs to be determined (e. g. on the basis of the leak rate at design pressure and the time dependent pressure after the accident). The time to achieve containment isolation, occurring as a result of the high pressure in the containment, needs to be taken into account in the analysis. A method for evaluating the release to the environment as a result of a loss of coolant accident at a pressurized water reactor is given in Ref. [I–3].

I–72. An alternative conservative approach to a mechanistic determination of the scope of the core damage in analysis of loss of coolant accidents is the practice, in some States, of specifying the fractions of core inventory of fission products that are assumed to reach the containment atmosphere after the accident. This fraction is specified differently for different groups of chemical elements but will usually be independent of the design measures taken against accidents of such types. Thus, these fractions are set as an assumed upper limit irrespective of the performance characteristics of the emergency core cooling system, Refs. [I-4 to I-6].

I–73. The behaviour of the radioactive material in their pathways from the containment depends upon the design of the plant. In the designs with single shell containment, the radioactive material may reach the atmosphere directly. In the designs with double containment, only small part of the leakages from the primary containment is released to the atmosphere (so called secondary containment by-pass), while most is confined by a secondary containment. with a delayed release to the environment preferably through filters. In other designs the~~y~~ leakages escape to a surrounding building, from which they are released with delay either directly or via a stack through filters.

Break of a steam line in a boiling water reactor

I-74. The break of a main steam line in a boiling water reactor may have more severe consequences than the break of a coolant recirculation pipe in a boiling water reactor or the break of a primary circulation loop in a pressurized water reactor Refs. [I-4 to I-6]. The reason is the location of the break, which is inside the containment for a pressurized water reactor but may be outside the containment in case of a boiling water reactor. The severity of the accident depends upon the diameter of the broken pipe and the characteristics of the plant safety systems.

I–75. If the location of the steam line break is within the containment, the sequence of events is similar to that for loss of coolant accidents in Pressurized water reactors, possibly with a certain fraction of the fuel cladding damage. The conservative fission product inventory for full power operating conditions has to be assumed. The design analysis for the potential radioactive release has to consider the time needed for containment isolation to take place and the effectiveness of the coolant purification system.

I-76. If the location of the steam line break is outside the containment and the main steam line isolation valves near the containment immediately close to isolate the reactor, only a fraction of the radioactive material present in the steam under operating conditions would be released. Condensation of steam in the building in which the break occurs and the plateout of substances other than noble gases, will result in a certain reduction in the amount of the radionuclides that are available for release to the atmosphere. Usually, the release of coolant into a building, other than the containment will cause such an overpressure that radioactive material will escape from the building either through predetermined release points (usually in the roof) or through doors or other weak structures which will be breached by the overpressure. Mixing of the steam with the air in the building may be assumed if the possible pipe break and the escape points from the building are not closely located. After the relief of overpressure, release to the outside will not be through uncontrolled release points but via the stack through the ventilation system and filters.

I-77. In some plants, leakage control systems have been added between the main steam isolation valves to limit the escape of radioactive material by this path.

I-78. The possibility of direct releases from the building after the relief of overpressure needs to be considered if the overpressure relief openings will not close and the underpressure of the building relative to the atmosphere.

Break of a steam line in a pressurized water reactor

I-79. Initially, the break of a steam line in a pressurized water reactor will release only insignificant quantities of radionuclides that may be present in the secondary system during normal operation. There is always some limited design leak rate between the primary and secondary side of the steam generators, so that certain amount of primary coolant may be released to the environment following a steam line break.

I-80. As a consequence of the steam line break, the integrity of the steam generator tubes, which depends on the pressure difference between the primary and the secondary sides, needs to be assessed. If the structural integrity of the steam generator tube cannot be assured, the amount of primary water that could enter the secondary side needs to be estimated. After the shutdown of the reactor, the radionuclide content of the leaking water may increase with time owing to the effects of fission product spiking.

I-81. Depending upon the design of the steam generator, the primary water that leaks into the secondary side may mix with the inventory of the secondary coolant in the steam generator. The steam produced shortly after the accident, which will escape through the broken steam line, will have a higher than normal moisture content because of the depressurization. Subdivision of fission products between the liquid phase and steam phase of the coolant (partitioning) needs to be conservatively taken into account.

I-82. Even without induced damage of steam generator tubes, a double ended break of the steam line could lead to significant radioactive releases to the atmosphere owing to releases of steam from the broken steam line, if the break cannot be isolated from the steam generator. With iodine spiking occurring in the primary coolant (potentially further enhanced by fuel temperature rise) and with the maximum primary to secondary leakage according the limits and conditions, the activity concentration of the escaping steam could be significant. This potential would be even greater if failure of the fuel cladding occurs. The significance of the release for this event is due to:

(1) the high activity concentration for the leakage;

(2) the break being not fully isolable;

(3) the dry-out of the affected steam generator which results in no partitioning of radioactive material within the steam generator.

I-83. After shutdown, the production of steam will depend on the decay heat. The moisture content of the steam will be low because of the low steam flow, and the high efficiency of the steam separators and driers. Thus, the steam, which may be released by pressure relief valves, will have relatively low concentrations of water soluble substances such as iodine and caesium. The release of radioactive material is expected to be minimized by the isolation of the defective steam generator and other safety actions that will depend on the design.

Steam generator tube rupture in a pressurized water reactor

I–84. The rupture of a steam generator tube in a pressurized water reactor can potentially lead to releases of radioactive material to the atmosphere and it may be radiologically dominant design basis accident. These releases could be significant because, even if iodine spiking does not occur before the initiation of this event, it will occur during the course of the transient. Actual accidents of steam generator tube rupture have occurred in several operating nuclear power plants.

I–85. The design basis steam generator tube rupture is postulated to be a double ended break in usually one tube but could also be more steam generator tubes. The breach of this primary to secondary side barrier initiates the release of reactor coolant into the secondary side. Subsequent to a reactor trip, the actuation of the steam pressure relief valves on the secondary side would release contaminated steam to the atmosphere. A potential for radioactive releases exists even if the steam generator tubes are not uncovered because of direct carry-over of the primary coolant into the steam line. The sources of radiation during this event are the radioactive fission products that are present in the primary to secondary break flow. Maximizing the break flow therefore maximizes the amount of radioactive fission products that are available for release to the atmosphere through the secondary side pressure relief valves.

I-86. After a reactor trip, the magnitude of the decay heat and the operator actions to isolate the affected steam generator and depressurize the primary circuit determine the magnitude of radioactive releases.

In some new designs, the actions to isolate the affected steam generator are automated. In existing designs, the release of radioactive material to the atmosphere will be terminated when the pressures of the primary and secondary circuits have equalized. The operator will cool down the plant using the intact steam generator(s). However, in case of failure to isolate the secondary side of the damaged steam generator, the releases to the atmosphere can continue until complete cooldown of the unit.

I–87. The nature of the transient depends on the automatic safety systems and the time at which the operator starts to take effective action. The time assumed for this varies. A design value of between 10 and 30 min to first operator action is recommended in Ref. [I–1]. A method for determining the radioactive release following the rupture of a steam generator tube is given in Ref. [I–7].

Fuel handling accidents

I–88. In a design basis analysis of the effects of a postulated fuel handling accident, such as the dropping of spent fuel during its transfer from the vessel to the storage pool, the first step is to determine the radioactive inventory of the fuel at the time of the accident. Assumptions about the details of the history of fuel irradiation need to be chosen so as to lead to conservative (i. e. high) estimate of the activity.

I–89. The minimum time that elapses between the shutdown of a plant and the beginning of fuel handling operations needs to be used to determine the maximum source term inventory in the fuel rods at the start of refuelling operations. The number of fuel rods that may become defective as a result of the impact needs to be determined either by means of conservative estimate or by the evaluation of actual occurrences with similar fuel elements or in experiments. The fraction of the noble gas inventory that is released depends on the volume of free space within the fuel rod. There is no general consensus as to which is the predominant mechanism of release of iodine to the pool water from rods with cracked cladding. Iodine may be mainly leached out by water penetrating the defective fuel rod, or the main release may be of ‘gaseous’ iodine, which is assumed to be present in the free space within the fuel rod.

I–90. The usual conservative approach is to neglect the solubility of noble gases in the pool water. However, a significant fraction of the iodine and caesium will be retained in the pool water. The release of iodine into the atmosphere above the pool may best be described in terms of a partitioning coefficient (the ratio of the activity concentrations (Bq/kg) in steam and in water). For that part of the iodine present in organic compounds such as methyl iodine no solubility in water is conservatively assumed in many States.

I–91. To determine the amounts of various radioactive species that are released to the atmosphere of the plant, it is necessary to take into account other features and parameters such as the partitioning, the elapsed time until shutdown of the ventilation system, and the design and effectiveness of the system that extracts the air immediately above the pool (this may involve an air sweeping system at the pool surface).

I–92. To simplify the evaluation, the fraction of iodine released from the fuel that is expected to enter the room atmosphere above the fuel storage pool may be conservatively specified as a global figure for certain reactor designs. This fraction is the inverse of the 'decontamination factor' which is also sometimes used.

I–93. In addition to noble gases and iodine, up to a few per cent of the caesium inventory may be slowly leached out by water that penetrates the defective rods. This caesium will be in ionic form in the water, and its transfer to the air above the pool water may be neglected.

I–94. The amounts of noble gases and iodine released to the environment will be controlled by the ventilation rate (forced or natural) and by the type of pool air sweeping system used, if such a system is available. The reduction effected by filters in the concentration of iodine in the exhaust air will be taken into account by means of an appropriate decontamination factor based on the filter design. The release may be terminated by the isolation of the appropriate part of the plant, especially if the storage pool is situated within a containment. If this isolation is done by operator action, a usual time delay can be assumed (e. g. between 10 and 30 min), Ref. [I–1].

Accidents in auxiliary systems with radioactive material

I–95. Examples of accidents that may occur in auxiliary systems are pipe breaks in the auxiliary systems containing gaseous or liquid radioactive waste, the ignition of filters or absorbers, explosions in storage tanks, spilling of liquid radioactive wastes, and fires in radioactive waste systems. The consequences will depend upon the design features of the systems concerned, for which there are significant differences in different reactor designs. For this reason, the assumptions to be chosen for the purposes of accident analysis need to be made on a case by case basis.

I–96. One important type of accident is that caused by a crack in the pipework of the residual heat removal system provided that the system is connected directly to the reactor coolant system. The system comes into operation following a reactor shutdown or a break in operation of the chemical and volume control system when the reactor is at power. In both cases, the important contribution to the source term is the fission product spike that will occur as a result of the shutdown or that may have occurred before the break.

I–97. The analyses of such faults require that the leak rate from the affected pipe, the transport of radioactive gases through the release pathway and the active ventilation system (if adequately classified), the behaviour of iodine and the efficiency of the filtration system under the accident conditions ~~all~~ are to be determined conservatively.

I–98. A method for analysing accidents of this type is described in Ref. [I-8] and supporting references.

Accidents in heavy water reactors

I–99. Reactors using heavy water (deuterium oxide) as a moderator, a coolant or both have the potential for the same type of accidental release of radioactive material as the corresponding light water reactors described above. For a pressure tube reactor, the analyses for loss of coolant accidents need to include ruptures of the pressure tubes as well as header or pipe breaks. Note that rupture of a pressure tube in combination with a header or pipe break is not required or considered in the design basis accidents. Accidents involving failure of steam generator tubes or heat exchanger tubes also need to be analysed.

I–100. The heavy water in the operating plant contains tritium, which is the activation product of deuterium. The tritium is in the oxide form (i. e. water) and is not normally an important factor in the potential radioactive hazard to the public following an accident. However, the presence of tritium needs to be taken into account for the protection of site personnel during and following certain accidents.

Accidents in reactors with on-load refuelling

I–101. For reactors with on-load refuelling capabilities, the possibility of accidents resulting from faults in the refuelling operation, either while the fuelling machine is connected to the reactor core or while the spent fuel is being transferred to the fuel storage pool, needs to be considered. The severity of the consequences is equal to or less than that for a small loss of coolant, depending on the location of the fault and the time elapsed after removal of the fuel from the reactor core.

Other accidents

I–102. Areas of the nuclear power plant in which other postulated initiating events resulting in releases of radioactive material to the environment may occur include:

(1) Spent fuel handling areas (i. e. fuelling machines, the dry fuel store, the fuel dismantling cell, the fuel storage pool and the loading bay for fuel transport flasks);

(2) The active effluent treatment plant;

(3) The treatment and cooling plant for fuel pool water;

(4) The coolant treatment plant;

(5) The store for solid radioactive waste;

(6) The fuel element debris vault;

(7) Ventilation filters.

SOURCE TERM UNDER DESIGN EXTENSION CONDITIONS

General

I-103. Accidents considered in the design that are either more severe than design basis accidents or that involve additional failures such that they have a very low probability of occurrence are classified as design extension conditions. Two separate categories of design extension conditions are identified: design extension conditions without significant fuel degradation; and design extension conditions progressing to core melting (i. e. severe accidents). The possible severity of the consequences of such accidents is characterized by the design of the plant and the nature of the failures and operator errors. In such cases, safety systems may fail to perform their required safety functions owing to the failures and errors, potentially leading to significant core damage. The potential exists for large releases of radioactive material to the environment in particular during a severe accident.

I–104. In accordance with general rules for deterministic safety analysis, the source term in case of design extension conditions without significant fuel degradation in accordance with the general rules for safety analysis, Ref. [I-1]. can be determined using a best-estimate approach. This means that plant operating parameters, characteristics of plant systems and various assumptions used in the analysis can be selected realistically, or at least less conservatively than in case of design basis accidents. The phenomena taking place in design extension conditions without significant fuel degradation do not differ from design basis accidents. Further text is therefore devoted to issues associated with severe accidents, in particular severe accidents taking place in the reactor core (and thus in the containment) of light water reactors. Severe accidents associated with containment by-pass would result in large radioactive releases and thus need to be practically eliminated.

I-105. Because of the significant core damage during severe accidents, such accidents are analysed in detail to determine their possible radiological consequences, which may have a significant impact on public health and safety. Such analyses can quantify the type and magnitude of the radiological source terms for the inventory of radionuclides that is available for release to the environment. For accidents with fuel melting, in addition to gap inventory a significant part of the total core fission product inventory will be released. The following groups of radioisotopes of fission products are to be considered: noble gases (Xe, Kr), halogens (I, Br), alkali metals (Cs, Rb), tellurium group (Te, Sb, Se), strontium Sr, noble metals (Ru, Rh, Pd, Mo, Tc, Co), cerium group (Ce, Pu, Np), lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am), and barium Ba.

I–106. Recommendations and guidance on the issues associated with severe accidents and for performing severe accident analyses and on quantifying the source term are provided in Ref. [I-1] and in Refs [I-2, I-9, I-10, I-11 and I-12].

DETERMINATION OF SOURCE TERMS FOR PLANT OPERATION AND DECOMMISSIONING

Phases of release of radionuclides from the fuel

I–107. Before the accident, a major part of the fission products is captured in the fuel, from where they are released in several subsequent phases of the accident progression. The phases of a severe accident can be distinguished as follows:

- Coolant activity release. During this phase, (relatively) small amounts of radioactivity in the coolant itself is released from the reactor coolant system.

- Gap activity release. This phase involves the release of radioactivity from the gap between the fuel pellets and cladding, including in particular noble gases, iodine and caesium.
- In-vessel release. During this phase associated with gradual melting and slumping of core materials practically all the noble gases and significant fractions of the volatile nuclides are released from the fuel.
- Ex-vessel release. If failure of the reactor pressure vessel occurs, the volatile radionuclides not released during the in-vessel phase are released from the molten corium into containment. In existing designs, the molten core interacts with the reactor cavity concrete and the less volatile nuclides are likely to be released into containment. The presence of water in the reactor cavity overlying core debris can significantly reduce the ex-vessel releases either by cooling the core debris, or by scrubbing the releases and retaining a large fraction in the water.
- Late in-vessel release. Simultaneously with ex-vessel release, late in-vessel releases of some of the volatile nuclides, which had deposited in the reactor coolant system during the in-vessel phase, will occur and be released into containment.

Fission product release from molten fuel

I–108. The releases from molten fuel depend on many parameters such as temperature, oxidizing-reducing conditions, interactions with structural materials, burnup, fuel material, state of the fuel (solid or liquid). There are two possible methods to determine the fractions of fission products released from molten fuel, both derived from extensive experimental data: simulation of the release rate by computer codes, or direct specification of total fraction of individual groups of fission products.

I–109. Fractions of fission products released from the molten fuel are strongly determined by volatility. The most volatile fission products are naturally noble gases, and then iodine, tellurium and caesium. In the very early stages of a core melt accident release, iodine and tellurium exceed caesium by several tens of times. Frequently used release fractions are generalized sets of experimental data included in the document NUREG-1465, Ref. [I-12]. According to the document, in all phases of severe accidents, all noble gases, about 75 % of halogens and alkali metals, about 30 % of tellurium group, 12 % of barium and strontium, and about 0.5% of other fission products are released. However, there are large uncertainties in these values and other sources show rather different values.

I–110. For a comprehensive consideration of radiological consequences, it is necessary to mention that in addition to fission products, other radionuclides present in the primary coolant are also released, including products of activation of the coolant or additives, and corrosion products However, in comparison with activity of fission products, activity of these additional radionuclides in case of a severe accident is not significant and can be neglected.

Behaviour of fission products in the reactor coolant system

I–111. It is essential that chemical affinities of fission products, their redistribution within the fuel release kinetics in various atmospheres, and potential for retention in the reactor coolant system are understood, to determine the release into the containment. Behaviour of iodine, caesium, ruthenium and tellurium require special attention. These are volatile species and thus they could be almost completely released from the fuel.

I–112. Since the number of moles of caesium in the core is typically much larger than that of iodine, most of the iodine combines with caesium and is released from the reactor coolant system as CsI. The majority of the caesium (around 90%) is expected to be released from the reactor coolant system as CsOH. The caesium source term might be attenuated in the reactor coolant system by any reaction of both CsI and CsOH with boric acid, because these reactions produce less volatile caesium borates. In a similar way, Cs may interact with Mo, leading to less volatile caesium molybdate, but potentially increasing the gaseous iodine fraction. Additionally, CsOH also interacts with steel, diffusing into the inner chromium oxide layer, providing further potential attenuation.

I–113. Ruthenium is considered separately from other refractory materials, due to its distinct oxidation processes. Ruthenium is low volatility under reducing and low oxidizing conditions but is highly and rapidly released under high oxidizing conditions, with potential release fractions measured in tens of percent. Tellurium is released from the fuel at about the same rate as noble gases, iodine and caesium but its chemical affinity for metallic zircaloy, and the fact that its release occurs during the oxidation of the cladding, will delay transport to the containment, compared to iodine and caesium.

I–114. The vapour species released from the fuel will condense in the cooler primary circuit conditions, either to form new aerosol particles, or to condense on already existing aerosol particles or on structural surfaces. A major part of iodine, caesium and the less volatile radionuclides released from the melting fuel will behave primarily as aerosols. A number of physical-chemical processes occur on the way which affect the retention of aerosols in the reactor coolant system and thus the source term to the environment. Aerosol deposition will occur due to several processes such as diffusion, thermophoresis, diffusiophoresis, electrophoresis, sedimentation and inertial impaction.

I–115. Particles deposited on the surfaces of the primary circuit may resuspend back to the gas stream, decreasing retention of radionuclides in the circuit. When aerosols are deposited on a dry, uncooled surface, energy generated by the fission product decay may be capable of reheating the deposited fission products to sufficiently high temperature such that re-vaporization may occur. The significance of the re-vaporization process depends on the level of fission products retained in the vessel prior to vessel rupture, the temperature of the reactor coolant system piping and the ability of the damaged vessel to release the fission products from the reactor coolant system. Resuspension is also caused by sudden increase in the gas flow rate.

I–116. Several other phenomena can affect the release of fission products into containment. If the reactor coolant system is at high pressure at the time of failure of the reactor pressure vessel, quantities of molten core materials could be ejected from the reactor pressure vessel into the containment at high velocities which is associated with a significant amount of radioactive material added to the containment atmosphere, primarily in the form of aerosols- high pressure melt ejection.

I–117. Another phenomenon is a possible steam explosion either within the reactor pressure vessel (in-vessel steam explosion) or within a flooded cavity (ex-vessel steam explosion). This could lead to fine fragmentation of some portion of the molten core debris with an increase in the amount of airborne radionuclides.

I–118. For simplicity, a conservative assumption can be made that fission products released from the fuel are transported instantaneously and directly to the containment. Although the immediate release of fission products into the containment is an acceptable conservative assumption, it may be considered that retention of fission product aerosols in the reactor coolant system can have important reducing effects on the source term to the environment.

Behaviour of fission products in containment

I–119. Behaviour of fission products in the containment atmosphere depends on their physical and chemical forms, the pathways that the various species enter the containment, the time after release, the additional sources of fission products (resuspension, revaporization or corium-concrete interaction) and the natural processes or active engineered safeguard features (containment sprays, ventilation) to remove fission products from the containment atmosphere.

I–120. Noble gases cannot be scrubbed from the containment atmosphere and are relatively insoluble in water. Thus, the only mechanism for changing the quantity of these gases is via the decay chain. Several processes leading to production and removal of aerosols from the atmosphere take place in the containment. Aerosol removal will be influenced by the natural depletion processes associated with sedimentation and diffusiophoresis, and to a lesser degree by thermophoresis and will be significantly affected by the operation of containment sprays and passage through the water pools. Radioactive aerosols removed by various processes are replaced in the containment atmosphere by other radioactive aerosols produced as a result of the ongoing degraded core phenomena (in-vessel and ex-vessel).

I–121. Behaviour of iodine requires special attention due to its different physical and chemical forms. The majority (more than 90%) of the iodine entering the containment will be in the form of CsI. CsI will exist primarily in the form of aerosols for typical containment temperatures and will ultimately settle out from the containment atmosphere via natural deposition processes. Active fission product removal processes, such as sprays will provide rapid removal of fission products from the containment atmosphere. Once the iodine enters the containment, however, additional reactions can occur that will influence its ultimate importance to public dose. Once removed from the containment atmosphere in particulate form (as CsI), the iodine will dissolve in containment water pools or plate out on wet surfaces in its ionic form [I-13]. Subsequently, iodine behaviour within the containment will depend upon time and the acidity (pH) of the water. Because of the presence of other dissolved fission products, radiolysis is expected to occur within the cavity and in sump water lowering the pH of the respective water pools. Without adequate pH control, the dissolved iodine will slowly be converted to elemental iodine and be re-released into the containment atmosphere as elemental iodine.

I–122. Organic iodine will also be produced slowly over time from reactions of the elemental iodine with carbon bearing compounds. When pH control is available and the pH is maintained at a value greater than 7, very little (less than 1%) of the dissolved iodine will be converted to elemental iodine. Elemental iodine and organic iodine compounds are gaseous and will be transported from the containment in much the same manner as noble gases. Elemental iodine is soluble and can be removed from the containment atmosphere via the operation of sprays. Thus, for several delayed containment overpressure sequences, a significant quantity of CsOH can revaporize and be available for release.

I–123. The deposition and resuspension of aerosols in the containment depends on the physical and chemical properties of the aerosols and the driving hydrodynamic and thermal-hydraulic conditions. Deposition by gravitation is the most important retention mechanism of aerosols in the containment in the absence of sprays, and growth of particles is an important uncertainty affecting the aerosol removal rate. The most important parameters are atmospheric relative humidity and temperature gradients at structure surfaces. A substantial part of the airborne radioactive material in a severe accident is hygroscopic, or soluble, even in superheated conditions. This leads to faster particle growth, and thus also faster deposition inside the containment.

I–124. Several processes are considered to bring the aerosols into contact with each other and with structural surfaces: Brownian motion and turbulent diffusion can move small particles across gas flow streamlines to come into contact. Larger particles, unable to respond to rapid variations in flow streamlines because of their inertia, can impact on other particles. Small particles can be swept out of the path of larger particles undergoing gravity settling and become agglomerated with the larger aerosols.

I–125. If available, containment sprays may be used also in severe accidents as a mitigation system to reduce the containment pressure and thus to prevent overpressure failure, and to enhance fission product depletion. The spray nozzles generate water droplets with the diameter typically about 1 mm. When falling, the droplets will partly agglomerate. There will be also evaporation and/or condensation phenomena depending on the relative humidity in the containment. As a result, the droplet size distribution will change during the fall.

I–126. Hydrogen combustion also influences the transport phenomena and the convective flow patterns set up in the containment. Resuspension of deposits is another consequence of such energetic phenomena within the containment. Hydrogen explosion can cause severe damage to the reactor building and release all the volatile fission products into the environment. Major hydrogen explosions, potentially leading to loss of containment integrity, belongs to conditions to be practically eliminated, Ref. [I-1].

I–127. The processes associated with radionuclides behaviour has been described in a number of documents, for example in [I-11], [I-12], [I-13], [I-14], [I-15], [I-16], or in the most recent state of the art OECD/NEA report [I-17].

Releases of fission products from containment

I–128. The releases to the environment are controlled by a containment leak rate, typically expressed as a pressure dependent value. The most usual approach is consideration of an equivalent hole size corresponding to the containment leak rate or a vent pathway size for vented containment sequences. It is appropriate to take into account that local concentrations of fission products in different containment sub-volumes to be considered as points of release, may be significantly different.

I–129. In more sophisticated calculations it can be taken into account that the releases from the containment are not realized through an ideal nozzle with given diameter but rather through the microscopic cracks in the steel liner and in the concrete containment wall, with significant retention capacity for the fission products having prevailing form of aerosols. The documents Refs [I-17] and [I-18] suggested a decontamination factor 15 for iodine and 100 for caesium in dry conditions, and almost complete retention of aerosols in wet conditions.

I–130. At least 3 different groups of radionuclides with significantly different behaviour are to be considered separately in the source term: 1) noble gases; 2) various forms of iodine; 3) all other fission products, typically existing as aerosols. For noble gases there is neither physical nor chemical change of their forms. Since the iodine represents the substantial contribution to consequences, consideration of its various forms is essential. The iodine exists in three different forms: iodine particulate (mostly as CsI), iodine elemental and as HI, and iodine organic.

I–131. Iodine forms behave differently during their transport in the environment and also have different health effects. Elemental iodine (I_2), hydrogen iodide (HI), and organic iodides (methyl iodide CH_3I, etc) are volatile species which behave similarly to gases and are more harmful to human health. When a radioactive plume containing these elements passes through the sky with noble gases, it causes an increase in the external radiation dose rate, which decreases with the movement of the plume. However, if it rains during the passage of the plume, the iodine and caesium contained in the plume will be deposited on the ground surface. This is the main cause of serious environmental contamination in the area around the accident reactor. These forms also have higher deposition rate than particulate forms. A higher share of elemental and organic forms leads to higher doses in the NPP vicinity, and correspondingly to smaller doses at longer distance from the plant. A difficult problem is that iodine changes its form along the process from its release from fuel to the release from the containment, resulting in late uncertainties. Consequently, there are different considerations for the iodine forms to be considered in the release to the environment, depending on the scope and objectives of the analysis.

I–132. The radionuclides scrubbed from the discharge or settled from the containment atmosphere, which are typically collected in water pools in the lower part of the containment, are also a potential source of water borne releases to the environment. Further, if ex-vessel core concrete interaction cannot be controlled, there is a likelihood of radionuclide releases following basemat failure. However, unless special conditions exist (like containment basemat failure resulting in opening a direct pathway to the environment), such pathways are usually not considered in determination of public doses in broader surroundings of the plant.

I–133. The release of fission products to the environment in case of basemat melt-through or basemat penetration (i. e. loss of containment integrity) requires special attention, including the potential release to the ground, transfer into the groundwater and subsequent transport to surface waters.

Containment filtered venting

I–134. Containment venting systems may be implemented to prevent containment failure in the case of a severe accident inducing a slow containment over-pressurisation. Timing for activation of the containment venting system can be predicted. It has to be taken into account the possibility of hydrogen generation, evacuation of off-site neighbours and containment design pressure resistance. Basically, filter venting should be used to minimise environmental contamination. In the case of boiling water reactors, filter venting can be divided into dry-well venting and wet-well venting, and wet-well venting via a suppression chamber can significantly reduce the release of soluble caesium. In addition, it has been confirmed that iodine dissolved in the suppression chamber water, in the form of I^- ions, is converted to volatile (I_2) and (CH_3I) by chemical reactions and is released into the environment with wet-well venting.

I–135. Different means of fission product filtration to limit the environmental source term can be used, including venturi scrubbers, gravel scrubbers, gravel bed or sand bed filters, multi-venturi scrubbers, metal fibre filters, jet and packing filters, and others. The retention factors of the filters depend on a number of factors such as filter loading, humidity and the decay heat generation of aerosols within the filter. In general, the retention of radionuclides in aerosol form is expected to be good. Noble gases are not captured by particle filters. Retention of iodine, in particular, requires charcoal filters. The Advanced Containment Experiments (ACE) Project measured the performance of eight potential containment-venting filtration devices, Ref. [I-19].

METHODS OF CALCULATION

I–136. Methods of performing the calculations to determine the primary sources of radiation and the data required can be found, for example, in Ref. [I-20]. Suitable computer codes for implementing the methods, where required, are generally available from the Radiation Safety Information Computational Center, Oak Ridge National Laboratories, Tennessee, USA, Ref. [I-21], and from the OECD/NEA Data Bank Computer Program Services, Ref. [I-22].

I–137. A detailed description of the methods used to calculate the fluence from the radiation sources and the data to be used is given in Ref. [I-20], which contains extensive bibliographies. Where computer codes are required to apply the method, suitable codes are generally available through the Radiation Safety Information Computational Center, Oak Ridge National Laboratories, Tennessee, USA [I-21], or the OECD/NEA Data Bank Computer Program Services [I-22].

I–138. Recommendations to perform safety analysis for all plant states (operational states and accident conditions) and for all aspects of plant states, including their radiological consequences are provided in SSG-2 (Rev. 1), Ref. [I-1]. Details of how to assess the radiation exposure of the public due to releases of radioactive material to the environment are given in Refs [I-23, I-24].

Source terms for corrosion products

I-139. The corrosion of steels and alloys that are in contact with the primary coolant leads to the in situ growth of an oxide layer and the release of ions into the coolant. The driving force for this mechanism is the concentration gradient between the bulk of the coolant and pores in the oxide layer.

I-140. The phenomena and the relationships that need to be modelled are illustrated in Fig. I–1. In principle, the behaviour of corrosion products can be modelled by methods that range from hand calculations to the use of complex software that includes analytical and phenomenological models.

I-141. In the case of light water reactors, parameters relating to the solubility in water of oxides at the temperature and pH of the coolant are very important parameters that determine the behaviour of corrosion products in the primary coolant. More specific details of the relevant parameters for coolant activity in Pressurized water reactors are given below:

— In the case of Pressurized water reactors, parameters relating to the solubility in water of unsaturated nickel, nickel oxide, nickel ferrites and cobalt ferrites at a coolant temperature range of 280 °C–340 °C and a pH range of 6.5–7.4 at 300°C are very important for determining the behaviour of corrosion products in the primary coolant.

— The models that are used to describe the behaviour of corrosion products need to have the capability of modelling a large interacting 'water–metal' system for which the following parameters are typical:

— Area in contact with the primary coolant: ~30 000 m^2;

— Mass of coolant: 200–300 t;

— Velocity of coolant: 0.1–15 $m\cdot s^{-1}$;

— Duration of one circuit (reactor → steam generator → reactor): ~10 s including ~1 s in flux;

— Variety of alloys: Zircaloy® 4/Inconel® 600, Inconel® 690, Incoloy® 800/ Inconel® 718/hard facing materials (Stellite®)/stainless steel.

— The order of magnitude of the mass of precursors of radioactive species (essentially ^{58}Ni (n, p) ^{58}Co and ^{59}Co (n, g) ^{60}Co) is:

— Release (average): 1 mg/dm2 /month

— Cycle duration: 12-18 months

— Area excluding Zircaloy® (~ no release): 17000 m2

—^{59}Co level (impurity): $\sim 5 \cdot 10^{-4}$ g·g^{-1}

—^{58}Ni in nickel based alloys (Inconel® 600, 690): $\sim 3 \cdot 10^{-1}$ g·g^{-1}

— Therefore, the input to the reactor coolant during a ten month cycle is ~10 g·cycle^{-1} of ^{59}Co and ~5 kg·cycle^{-1} of ^{58}Ni;

— Wear of hard facing materials (of parts in the internal structures of the core, pump bearings, valves, control rod drive mechanisms, etc.) is in addition to the figure for ^{59}Co;

— As a result, approximately 10 g of ^{59}Co and 5 kg of ^{58}Ni are the origin of the ^{60}Co and ^{58}Co deposits, respectively, that are responsible for 90% of the dose rates and occupational exposures. This applies for reactors in which a nickel-based alloy is used for steam generator tubing.

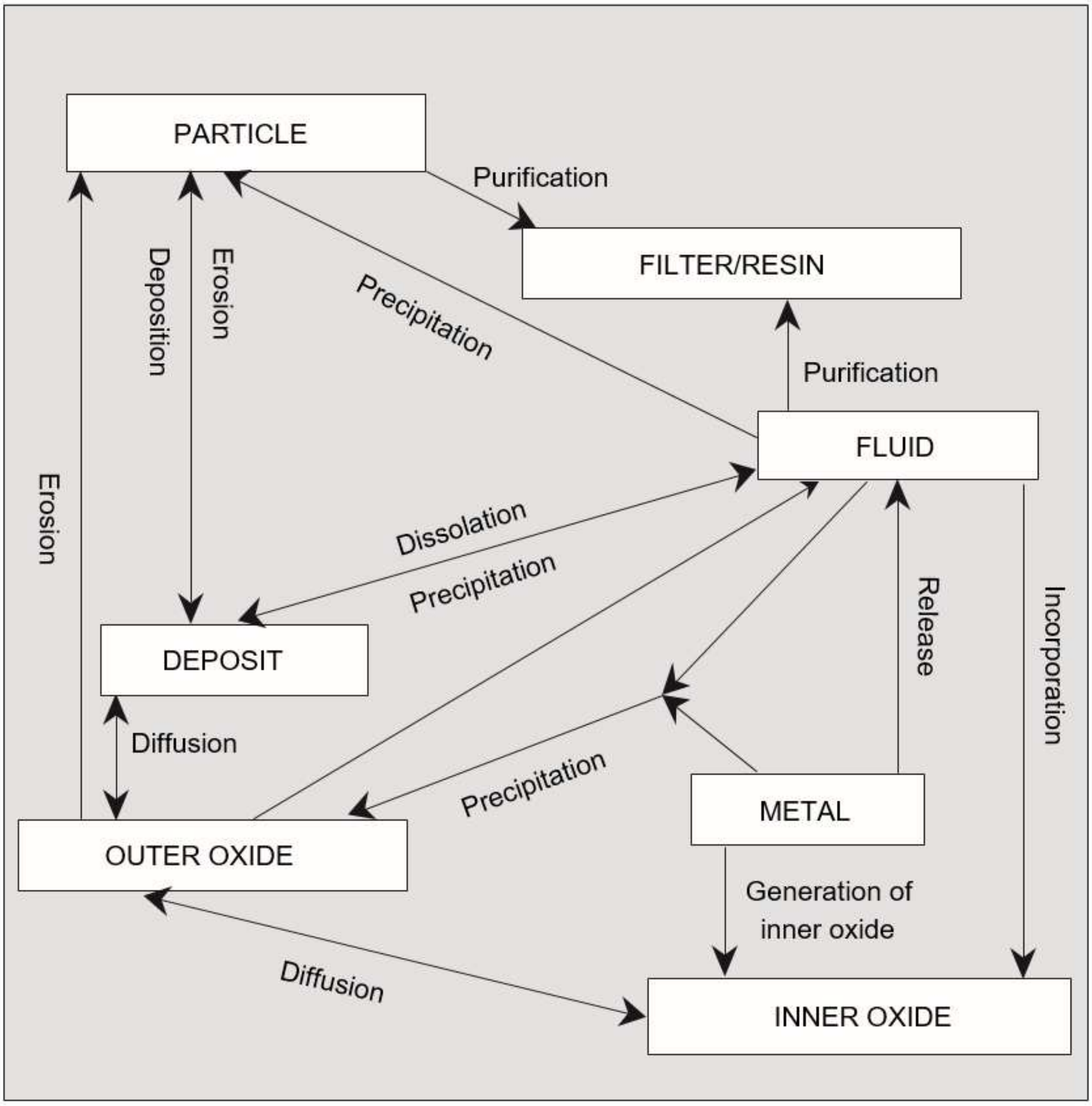

FIG. I-1. Flow chart of phenomena required to be considered in modelling the behaviour of corrosion products.

I–142. In the case of fast reactors, the secondary coolant circuit enters the neutron flux and it is necessary to evaluate the source terms that are due to corrosion of the secondary circuit. Some of the important phenomena that affect the source term for corrosion products are the following:

— Ionic species can precipitate and agglomerate to particles.

— These particles circulate in the fluid and are likely to form deposits either within the reactor core or on out-of-flux surfaces. By this process they become activated during circulation or after they have been deposited on in-core surfaces.

— Ions and particles can be removed from the primary coolant by the coolant purification system. The effectiveness of this process depends on the flow rate and on the decontamination factors of the filters and the ion exchange columns of the coolant purification system. If any of these factors are too low, the purification system will be ineffective.

Because the primary circuit is an almost closed and non-isothermal system, the above processes compete with reverse processes: for example, particles and deposits may dissolve.

I-143. The models that are used need to be appropriate for the properties of the system that is being addressed.

I-144. Examples of other factors that need to be modelled include the following:

— When the concentration of oxides in the primary coolant is very low (in a pressurized water reactor it is typically a few 10^{-9} $g \cdot g^{-1}$);

— When the release of elements from alloys is not proportional to their composition;

— When chemical conditions vary throughout the fuel cycle within a specified range;

— When bulk coolant and surface temperatures need to be taken into account;

— When wear by friction is significant.

I–145. The phenomena associated with the behaviour of corrosion products are so complex that the accuracy of both hand calculations and calculations made using computer codes that are based on analytical models is poor. However, the results of calculations made with codes in which the physical and chemical phenomena are taken into account are much more accurate. They do not give accurate results in absolute terms, but they correctly predict the relationships between the important design parameters and the source term. They are therefore a very important aid for optimizing the levels of the sources of ^{58}Co and ^{60}Co.

I–146. Owing to the complex nature of the phenomena involved, another essential input to evaluating the source term due to corrosion products is operating experience at relevant plants. The relevance of an operating plant will depend on how all the relevant factors at that plant compare with those for the plant that is being designed. These factors include the materials of the coolant circuit and their impurities, the coolant chemistry, the shutdown procedures and all the other factors that have been mentioned. Collecting the most accurate operating experience involves making regular measurements at exactly the same locations throughout the lifetime of the plant, including during transients such as reactor shutdowns.

I–147. To optimize the levels of sources of radiation for a plant that is being designed, it is also necessary to know the nature and composition of the radioactive material that is deposited on components at relevant operating plants. This is best achieved by using a collimated gamma spectrometer. The large changes that occur in the physical and chemical conditions of the coolant when going from operation at power to cold shutdown are the cause of a significant dissolution of corrosion products deposited on the fuel elements. The extent of the corresponding spiking of the coolant activity is a function of a large number of parameters. The spiking value is not predictable. However, for a given reactor type, a variation band can be indicated. Because deposits of corrosion products vary from one fuel cycle to another in the same plant, it is necessary to ensure that the operating data that are used are converted into values that are sufficiently bounding for design purposes.

I–148. For evaluating the source terms for the purposes of modifying or decommissioning a plant, there is no substitute for the results of the latest 100 measurements that have been made at the same plant at all the relevant dose points.

Source terms for fission products

I–149. The usual approach for determining source terms for fission products is:

— To calculate the inventory of fission products in the fuel — several well-known computer codes are available to perform this evaluation (such as the ORIGEN, FISPIN, APOLLO and THALES2 codes in the USA, in the United Kingdom, France and Japan respectively);

— To determine the amounts of radionuclides, and the corresponding activity, that are available in all the gas gaps in the fuel elements;

— To determine the total activity of the radionuclides that will be released to the coolant through cladding defects.

— To determine the transport of the radionuclides in the reactor coolant system (primary, secondary and auxiliary systems, as applicable) and the containment, up to the determination of the source term to the environment.

Description of all steps in performing deterministic safety analyses for various reactor designs, i. e. the selection of initiating events and acceptance criteria, selection of computer codes and modelling assumptions, preparation of input data and presentation of the results of calculations is covered in Ref. [I-25]. Specific document devoted to deterministic safety analysis of severe accidents is Ref. [I-11].

I–150. Most of the codes currently used for severe accident analysis are based on fixed chemical forms. For iodine and caesium, which are important fission products for source term evaluation, caesium iodide (CsI) and caesium hydroxide (CsOH) are usually assumed to be in particulate (aerosol) form in the main transfer pathways. In-core experiments and analyses have shown that different chemical forms of iodine and caesium compounds can be formed in the reactor cooling system [I-26].

I–151. In particular, a significant fraction of gaseous iodine has been found to migrate from the reactor cooling system to the simulated containment when boron carbide (B4C) control material is included in the core. In addition, monitoring after Fukushima Daiichi accident showed a significant amount of gaseous iodine along with particulate iodine [I-27].

I–152. Historically, the release of radionuclides to the coolant is represented by coefficients whose values were derived from early experiments and depend on the element being considered. In this case, some very important parameters such as the local power and temperature and the 'size' of the defect are not taken into account. The agreement with operating experience is generally poor. However, in calculating the source terms due to fission products, the corresponding uncertainties in the activity of fission products in the coolant are compensated for by assuming a much larger proportion of defective fuel pins (for light water reactors, this is typically 0.25% of the total number of fuel pins in the core) than is found in operating reactors. The corresponding source term for fission products is used for the design of shielding at locations where radioactive material accumulates, such as at filters and ion exchangers.

I–153. More accurate results are obtained with modern codes for fission product releases by including the dependence of the release coefficient (s–1) on the half-life of each isotope and by taking into account the parameters that were omitted in the earlier approach. In this case, the agreement with operating experience is good, and the predictions made on the basis of such codes can be used as a significantly less conservative basis for the design of shielding.

I–154. This improvement is important for the optimization of shielding because the difference between the two approaches can lead to source terms that differ by a factor of 3 to 10 depending on the isotope. For a point source emitting a 1 MeV gamma ray, a reduction in the source term by a factor of 5 would lead to a reduction in the thickness of a concrete shield of approximately 20 cm.

I–155. An alternative method is to use reasonably bounding values that are derived from operating experience at relevant plants. The factors that determine the relevance of other plants that are operating include the design of the fuel elements and the rating and burnup of the fuel.

I–156. During power transients, fission products are released to the coolant in a short time period through the cladding defects. This release is the cause of a spike in the activity of the coolant. The magnitude and period of the release are difficult to predict, but reasonably bounding values can be derived from operating experience. In Ref. [I-7], a correlation of the release and the duration with the pre-transient coolant activity is reported.

I–157. In the case of the modification or decommissioning of a plant, there is no substitute for recent measurements that have been made on the same plant.

REFERENCES TO ANNEX 1

[I-1] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Specific Safety Guide SSG-2 (Rev. 1), Deterministic safety analysis for nuclear power plants, IAEA, Vienna (2019).

[I-2] INTERNATIONAL ATOMIC ENERGY AGENCY, A Simplified Approach to Estimating Reference Source Terms for LWR Designs, IAEA-TECDOC- 1127, IAEA, Vienna (1999).

[I-3] DUTTON, L.M.C., et al., Realistic Methods for Calculating the Releases and Consequences of a Large LOCA, Rep. EUR-14179-EN, EURATOM, Luxembourg (1991).

[I-4] NUCLEAR REGULATORY COMMISSION, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, Regulatory Guide 1.4**,** NRC, Washington, DC (1974).

[I-5] NUCLEAR REGULATORY COMMISSION, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors, Regulatory Guide 1.3, NR**C,** Washington, DC (1984).

[I-6] BUNDESMINISTERIUM DES INNEREN, Accident Calculation Bases for the Guidelines for the Assessment of the Design of Nuclear Power Plants with PWRs against Accidents Pursuant to § 28, para. (3) of the Radiation Protection Ordinance of October 18, 1983 (BAnz 1983, Nr. 245a), Amended 29 June 1994 (BAnz. 1994, Nr. 222a), Bundesamt für Strahlenschutz, Salzgitter (1994).

[I-7] DUTTON, L.M.C., et al., Methods for Calculating the Release of Radioactivity following Steam Generator Tube Rupture Faults, Rep. EUR-15615-EU, EURATOM, Luxembourg (1994).

[I-8] CAPITAO, J.A., Synthesis of Auxiliary Building Faults Benchmark, European Commission Technical Note No. I.97.103, EC, Luxembourg (1997)..

[I-9] Bal Raj Seghal, Nuclear Safety in Light Water Reactors: Severe Accident Phenomenology, Elsevier, First Edition 2012.

[I-10] SARNET Lecture Notes on Nuclear Reactor Severe Accident Phenomenology, Edited by Bal Raj Seghal and Pascal Piluso, SARNET-SPREAD-D-118, September 2008.

[I-11] Approaches and tools for severe accident analysis for nuclear power plants, Safety Report Series No. 56, IAEA, Vienna (2008).

[I-12] SOFFER, L., BURSON, S.B., FERRELL, C.M., LEE, R.Y., RIDGLEY, J.N., Accident Source Terms for Light-Water Nuclear Power Plants, Rep. NUREG- 1465, Nuclear Regulatory Commission, Washington, DC (1995).

[I-13] OECD NEA/CSNI/R(2007)1 State-of-the Art Report on Iodine Chemistry, OECD NEA/CSNI/R(2007)1, February 2007.

[I-14] Passive ALWR Source Term (Chapter 4.9) DOE/ID 10321, February 1991.

[I-15] POWERS, D. et al, 'A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments', NUREG/CR-6189, SAND94-0407, July 1996.

[I-16] POWERS D. A., BURSON S. B., "A Simplified Model of Aerosol Removal by Containment Sprays", NUREG/CR-5966, June 1993..

[I-17] OECD NEA/CSNI/R(2009)5 State-of-the Art Report on Nuclear Aerosols, December 2009.

[I-18] F. Parozzi et al, Investigations on Aerosol Transport in Containment Cracks, paper at International Conference Nuclear Energy for New Europe 2005, Bled, Slovenia, September 5-8, 2005.

[I-19] Status Report on Filtered Containment Venting, Nuclear Safety, NEA/CSNI/R(2014)7, 7 July 2014.

[I-20] JAEGER, R.G., et al. (Eds), Engineering Compendium on Radiation Shielding, 3 Vols, Springer Verlag, Berlin (1968, 1970, 1975).

[I-21] OAK RIDGE NATIONAL LABORATORIES, Radiation Safety Information Computational Center, Oak Ridge, TN (web site: http://www-rsicc.ornl.gov/ rsicc.html).

[I-22] OECD NUCLEAR ENERGY AGENCY, Data Bank Computer Program Services (web site: http://www.nea.fr/html/dbprog/cpsabs_j.html).

[I-23] INTERNATIONAL ATOMIC ENERGY AGENCY, Generic Models for Use in Assessing the Impact of Discharges of Radioactive Substances to the Environment. Safety Reports Series No. 19, IAEA, Vienna (2000).

[I-24] BUNDESMINISTERIUM FÜR UMWELT, NATURSCHUTZ UND REAK- TORSICHERHEIT, Determination of the Radiation Exposure from Discharge of Radioactive Substances from Nuclear Plants or Facilities, General Administra- tive Provision Regarding § 45 Radiation Protection Ordinance of 21 February 1990 (BAnz 1990, Nr. 64a), Bundesamt für Strahlenschutz, Salzgitter (1990).

[I-25] INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Analysis for Nuclear Power Plants, Safety Reports Series no. 23, IAEA, Vienna (2002).

[I-26] N. Girault et al. Understanding of Iodine Chemistry in RCS of Nuclear Reactors, Nucl. Eng. Des., 239, 1162-1170, 2009. - *PHEBUS-FP programme in France*.

[I-27] J. Ishikawa, K. Kawaguchi and Y. Maruyama, Analysis for Iodine Release from Unit 3 of Fukushima Dai-ichi Nuclear Power Plant with Consideration of Water Phase Iodine Chemistry," J. Nucl. Sci. Technol. DOI : 10.1080/00223131.2014.951417, September.

Annex II

EXAMPLES OF ZONING FOR DESIGN PURPOSES

II.1. A good example of radiation zoning that may be used for design purposes is shown below (Table II–1) [II–1].

II.-2. A good example of zoning that addresses radiation, surface contamination and airborne contamination is given by the classification of zones within the controlled area in Swedish nuclear power plants (Table II–2) [II–2].

TABLE II–1. EXAMPLE OF RADIATION ZONING THAT MAY BE USED FOR DESIGN PURPOSES

Access requirement	Design dose equivalent rate (µSv/h)	
	Mean	Maximum
Uncontrolled areas on-site	—	1
Continuous (> 10 person-hours per week)	1	5
1–10 person-hours per week	10	50
< 1 person-hours per week	100	500
1–10 person-hours per year	1000	10 000
< 1 person-hours per year	10 000	[a]

[a] Dose rates in excess of 10 mSv/h are acceptable provided that the exposure time is correspondingly short.

TABLE II–2. CLASSIFICATION OF ZONES WITHIN THE CONTROLLED AREA IN SWEDISH NUCLEAR POWER PLANTS FOR RADIATION, SURFACE CONTAMINATION AND AIRBORNE CONTAMINATION

Zone identification	Blue zone	Yellow zone	Red zone
Radiation zones	< 25 µSv/h	25–1000 µSv/h	> 1000 µSv/h
Surface contamination zones	For total β < 40 kBq/m^2	40–1000 kBq/m^2	> 1000 kBq/m^2
	For total α < 4 kBq/m^2	4–100 kBq/m^2	> 100 kBq/m^2
Zones for airborne contamination	1 DAC [a]	1–10 DAC	> 10 DAC

[a] DAC: derived air concentration.

REFERENCES TO ANNEX II

[II–1] NUCLEAR ELECTRIC, Preconstruction Safety Report for Sizewell B Barnwood, Gloucester (1996).

[II–2] FORSMARK NUCLEAR POWER PLANT, Radiation Protection Instructions (2003), Instruction F-I-201, Forsmark Kraftgrupp AB, Östhammar (2003).

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