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International Atomic Energy Agency

IAEA SAFETY STANDARDS

No. SSG-90

for protecting people and the environment

Radiation Protection Aspects of Design for Nuclear Power Plants

SPECIFIC SAFETY GUIDE

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

The publications by means of which the IAEA establishes standards are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety. The publication categories in the series are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides**.

Information on the IAEA's safety standards programme is available at the IAEA Internet site

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The site provides the texts in English of published and draft safety standards. The texts of safety standards issued in Arabic, Chinese, French, Russian and Spanish, the IAEA Safety Glossary and a status report for safety standards under development are also available. For further information, please contact the IAEA at: Vienna International Centre, PO Box 100, 1400 Vienna, Austria.

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by email to Official.Mail@iaea.org.

RELATED PUBLICATIONS

The IAEA provides for the application of the standards and, under the terms of Articles III and VIII.C of its Statute, makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

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Other safety related IAEA publications are issued as **Emergency Preparedness and Response** publications, **Radiological Assessment Reports**, the International Nuclear Safety Group's **INSAG Reports**, **Technical Reports** and **TECDOCs**. The IAEA also issues reports on radiological accidents, training manuals and practical manuals, and other special safety related publications.

Security related publications are issued in the **IAEA Nuclear Security Series**.

The **IAEA Nuclear Energy Series** comprises informational publications to encourage and assist research on, and the development and practical application of, nuclear energy for peaceful purposes. It includes reports and guides on the status of and advances in technology, and on experience, good practices and practical examples in the areas of nuclear power, the nuclear fuel cycle, radioactive waste management and decommissioning.

RADIATION PROTECTION ASPECTS OF DESIGN FOR NUCLEAR POWER PLANTS

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

IAEA SAFETY STANDARDS SERIES No. SSG-90

RADIATION PROTECTION ASPECTS OF DESIGN FOR NUCLEAR POWER PLANTS

SPECIFIC SAFETY GUIDE

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FOREWORD

by Rafael Mariano Grossi
Director General

The IAEA's Statute authorizes it to "establish…standards of safety for protection of health and minimization of danger to life and property". These are standards that the IAEA must apply to its own operations, and that States can apply through their national regulations.

The IAEA started its safety standards programme in 1958 and there have been many developments since. As Director General, I am committed to ensuring that the IAEA maintains and improves upon this integrated, comprehensive and consistent set of up to date, user friendly and fit for purpose safety standards of high quality. Their proper application in the use of nuclear science and technology should offer a high level of protection for people and the environment across the world and provide the confidence necessary to allow for the ongoing use of nuclear technology for the benefit of all.

Safety is a national responsibility underpinned by a number of international conventions. The IAEA safety standards form a basis for these legal instruments and serve as a global reference to help parties meet their obligations. While safety standards are not legally binding on Member States, they are widely applied. They have become an indispensable reference point and a common denominator for the vast majority of Member States that have adopted these standards for use in national regulations to enhance safety in nuclear power generation, research reactors and fuel cycle facilities as well as in nuclear applications in medicine, industry, agriculture and research.

The IAEA safety standards are based on the practical experience of its Member States and produced through international consensus. The involvement of the members of the Safety Standards Committees, the Nuclear Security Guidance Committee and the Commission on Safety Standards is particularly important, and I am grateful to all those who contribute their knowledge and expertise to this endeavour.

The IAEA also uses these safety standards when it assists Member States through its review missions and advisory services. This helps Member States in the application of the standards and enables valuable experience and insight to be shared. Feedback from these missions and services, and lessons identified from events and experience in the use and application of the safety standards, are taken into account during their periodic revision.

I believe the IAEA safety standards and their application make an invaluable contribution to ensuring a high level of safety in the use of nuclear technology. I encourage all Member States to promote and apply these standards, and to work with the IAEA to uphold their quality now and in the future.

THE IAEA SAFETY STANDARDS

BACKGROUND

Radioactivity is a natural phenomenon and natural sources of radiation are features of the environment. Radiation and radioactive substances have many beneficial applications, ranging from power generation to uses in medicine, industry and agriculture. The radiation risks to workers and the public and to the environment that may arise from these applications have to be assessed and, if necessary, controlled.

Activities such as the medical uses of radiation, the operation of nuclear installations, the production, transport and use of radioactive material, and the management of radioactive waste must therefore be subject to standards of safety.

Regulating safety is a national responsibility. However, radiation risks may transcend national borders, and international cooperation serves to promote and enhance safety globally by exchanging experience and by improving capabilities to control hazards, to prevent accidents, to respond to emergencies and to mitigate any harmful consequences.

States have an obligation of diligence and duty of care, and are expected to fulfil their national and international undertakings and obligations.

International safety standards provide support for States in meeting their obligations under general principles of international law, such as those relating to environmental protection. International safety standards also promote and assure confidence in safety and facilitate international commerce and trade.

A global nuclear safety regime is in place and is being continuously improved. IAEA safety standards, which support the implementation of binding international instruments and national safety infrastructures, are a cornerstone of this global regime. The IAEA safety standards constitute a useful tool for contracting parties to assess their performance under these international conventions.

THE IAEA SAFETY STANDARDS

The status of the IAEA safety standards derives from the IAEA's Statute, which authorizes the IAEA to establish or adopt, in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and with the specialized agencies concerned, standards of safety for protection of health and minimization of danger to life and property, and to provide for their application.

With a view to ensuring the protection of people and the environment from harmful effects of ionizing radiation, the IAEA safety standards establish fundamental safety principles, requirements and measures to control the radiation exposure of people and the release of radioactive material to the environment, to restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation, and to mitigate the consequences of such events if they were to occur. The standards apply to facilities and activities that give rise to radiation risks, including nuclear installations, the use of radiation and radioactive sources, the transport of radioactive material and the management of radioactive waste.

Safety measures and security measures[1] have in common the aim of protecting human life and health and the environment. 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.

The IAEA safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. They are issued in the IAEA Safety Standards Series, which has three categories (see Fig. 1).

Safety Fundamentals

Safety Fundamentals present the fundamental safety objective and principles of protection and safety, and provide the basis for the safety requirements.

Safety Requirements

An integrated and consistent set of Safety Requirements establishes the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements are governed by the objective and principles of the Safety Fundamentals. If the requirements are not met, measures must be taken to reach or restore the required level of safety. The format and style of the requirements facilitate their use for the establishment, in a harmonized manner, of a national regulatory framework. Requirements, including numbered 'overarching' requirements, are expressed as 'shall' statements. Many requirements are not addressed to a specific party, the implication being that the appropriate parties are responsible for fulfilling them.

Safety Guides

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it

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[1] See also publications issued in the IAEA Nuclear Security Series.

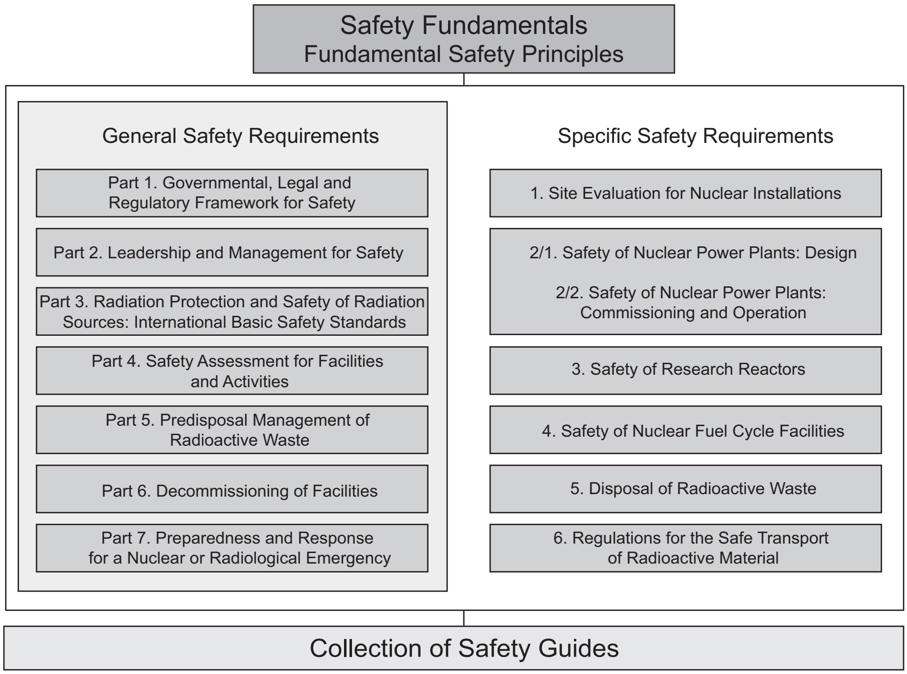

FIG. 1. The long term structure of the IAEA Safety Standards Series.

is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as ‘should’ statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

The principal users of safety standards in IAEA Member States are regulatory bodies and other relevant national authorities. The IAEA safety standards are also used by co-sponsoring organizations and by many organizations that design, construct and operate nuclear facilities, as well as organizations involved in the use of radiation and radioactive sources.

The IAEA safety standards are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes and to protective actions to reduce existing radiation risks. They can be used by States as a reference for their national regulations in respect of facilities and activities.

The IAEA's Statute makes the safety standards binding on the IAEA in relation to its own operations and also on States in relation to IAEA assisted operations.

The IAEA safety standards also form the basis for the IAEA's safety review services, and they are used by the IAEA in support of competence building, including the development of educational curricula and training courses.

International conventions contain requirements similar to those in the IAEA safety standards and make them binding on contracting parties. The IAEA safety standards, supplemented by international conventions, industry standards and detailed national requirements, establish a consistent basis for protecting people and the environment. There will also be some special aspects of safety that need to be assessed at the national level. For example, many of the IAEA safety standards, in particular those addressing aspects of safety in planning or design, are intended to apply primarily to new facilities and activities. The requirements established in the IAEA safety standards might not be fully met at some existing facilities that were built to earlier standards. The way in which IAEA safety standards are to be applied to such facilities is a decision for individual States.

The scientific considerations underlying the IAEA safety standards provide an objective basis for decisions concerning safety; however, decision makers must also make informed judgements and must determine how best to balance the benefits of an action or an activity against the associated radiation risks and any other detrimental impacts to which it gives rise.

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

The preparation and review of the safety standards involves the IAEA Secretariat and five Safety Standards Committees, for emergency preparedness and response (EPReSC) (as of 2016), nuclear safety (NUSSC), radiation safety (RASSC), the safety of radioactive waste (WASSC) and the safe transport of radioactive material (TRANSSC), and a Commission on Safety Standards (CSS) which oversees the IAEA safety standards programme (see Fig. 2).

All IAEA Member States may nominate experts for the Safety Standards Committees and may provide comments on draft standards. The membership of the Commission on Safety Standards is appointed by the Director General and includes senior governmental officials having responsibility for establishing national standards.

A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards.

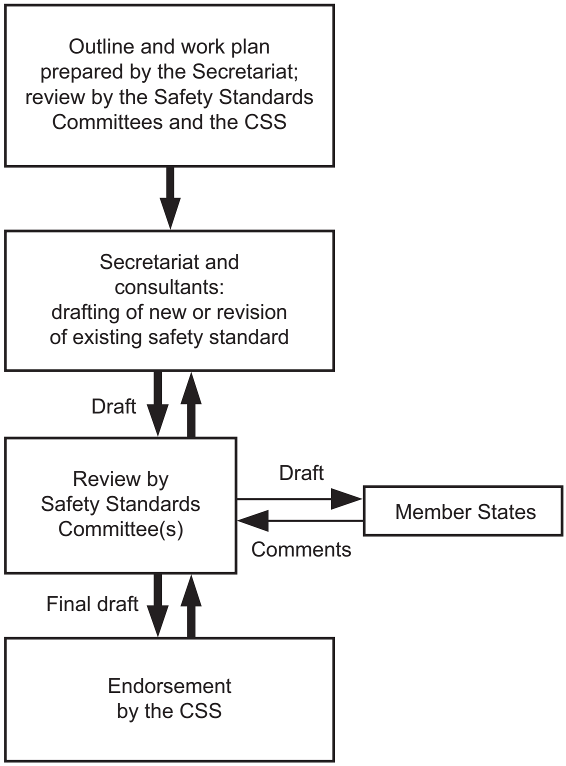

FIG. 2. The process for developing a new safety standard or revising an existing standard.

It articulates the mandate of the IAEA, the vision for the future application of the safety standards, policies and strategies, and corresponding functions and responsibilities.

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

The findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP), are taken into account in developing the IAEA safety standards. Some safety standards are developed in cooperation with other bodies in the United Nations system or other specialized agencies, including the Food and Agriculture Organization of the United Nations, the United Nations Environment Programme, the International Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization.

INTERPRETATION OF THE TEXT

Safety related terms are to be understood as they appear in the IAEA Nuclear Safety and Security Glossary (see https://www.iaea.org/resources/publications/iaea-nuclear-safety-and-security-glossary). Otherwise, words are used with the spellings and meanings assigned to them in the latest edition of The Concise Oxford Dictionary. For Safety Guides, the English version of the text is the authoritative version.

The background and context of each standard in the IAEA Safety Standards Series and its objective, scope and structure are explained in Section 1, Introduction, of each publication.

Material for which there is no appropriate place in the body text (e.g. material that is subsidiary to or separate from the body text, is included in support of statements in the body text, or describes methods of calculation, procedures or limits and conditions) may be presented in appendices or annexes.

An appendix, if included, is considered to form an integral part of the safety standard. Material in an appendix has the same status as the body text, and the IAEA assumes authorship of it. Annexes and footnotes to the main text, if included, are used to provide practical examples or additional information or explanation. Annexes and footnotes are not integral parts of the main text. Annex material published by the IAEA is not necessarily issued under its authorship; material under other authorship may be presented in annexes to the safety standards. Extraneous material presented in annexes is excerpted and adapted as necessary to be generally useful.

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. Design requirements for the safety of nuclear power plants, including radiation protection, are established in IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [1].

1.2. Detailed requirements on radiation protection (for all facilities and activities) are established in IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [2].

1.3. Effective radiation protection is a combination of good design, high quality construction and proper operation. Recommendations that address the radiation protection aspects of operation are provided in IAEA Safety Standards Series Nos GSG-7, Occupational Radiation Protection [3], and SSG-40, Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors [4].

1.4. Other requirements of the IAEA safety standards are relevant to radiation protection aspects of design for nuclear power plants, including the following:

- IAEA Safety Standards Series No. SSR-2/2 (Rev. 1), Safety of Nuclear Power Plants: Commissioning and Operation [5];
- IAEA Safety Standards Series No. SSR-1, Site Evaluation for Nuclear Installations [6];
- IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [7];
- IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [8];
- IAEA Safety Standards Series No. GSR Part 5, Predisposal Management of Radioactive Waste [9];
- IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [10];
- IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [11].

1.5. This Safety Guide is a revision of IAEA Safety Standard Series No. NS-G-1.13, Radiation Protection Aspects of Design for Nuclear Power Plants, which it supersedes.[1]

OBJECTIVE

1.6. This Safety Guide provides recommendations on how to meet the requirements of SSR-2/1 (Rev. 1) [1], in particular Requirements 5, 12, 19, 81 and 82. 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 radiation hazards for operational states, accident conditions and decommissioning.

1.7. The purpose of this Safety Guide is to provide recommendations for ensuring radiation protection in the design of new nuclear power plants, in design modifications to operating plants and in checking of the adequacy of the design at different stages in the lifetime of operating plants (e.g. as part of the comprehensive evaluation of safety or the periodic safety review of the plant).

1.8. This Safety Guide is for use by organizations responsible for designing, manufacturing and constructing nuclear power plants, by operating organizations and contractors, including plant operators who are involved in planning, managing and implementing the design and design modification of nuclear power plants, and by regulatory bodies and technical support organizations (referred to as 'providers of technical services' in GSG-7 [3]).

SCOPE

1.9. This Safety Guide:

(a) Describes the safety requirements that are applicable to radiation protection, including those relating to the principles of dose limitation and optimization, as a basis for the measures to be implemented in the design of nuclear power plants;
(b) Provides recommendations for the measures to be taken in the design for the protection of site personnel, the public and the environment;

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[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection Aspects of Design for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.13, IAEA, Vienna (2005).

(c) Outlines the methodologies used to calculate on-site and off-site radiological conditions and to verify that the design provides an adequate level of radiation protection.

1.10. In addition to providing recommendations on measures to protect site personnel, members of the public and the environment when the plant is in operational states and during decommissioning, this Safety Guide also provides recommendations on radiation protection aspects of design for accident conditions.[2]

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 (e.g. district heating, desalination). For other reactor types, including future plant systems in which limited operating experience has been gained, some recommendations may need judgement applied or may not be appropriate.

1.12. This Safety Guide addresses radiation protection design aspects of the handling, treatment and on-site 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 Part 5 [9], SSG-40 [4] and IAEA Safety Standards Series No. SSG-41, Predisposal Management of Radioactive Waste from Nuclear Fuel Cycle Facilities [12].

1.13. The terms used in this Safety Guide should be understood in accordance with the description of those terms that appear in the IAEA Nuclear Safety and Security Glossary [13].

STRUCTURE

1.14. Section 2 of this 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. Recommendations on design approaches for operational states, decommissioning and accident conditions are provided in Section 3, while Section 4 provides recommendations on the control of sources of radiation and estimation of radiation dose rates in all

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[2] This Safety Guide does not address the design measures that are necessary to reduce the probability of occurrence and to prevent the development of accidents. These aspects are considered in SSR-2/1 (Rev. 1) [1] and in other Safety Guides.

plant states and in decommissioning. Section 5 provides recommendations on design features for radiation protection in operational states, and Section 6 provides recommendations on design features for radiation protection in accident conditions. Section 7 provides recommendations on design features for radiation protection in decommissioning. Section 8 provides recommendations on radiation monitoring systems 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

SAFETY OBJECTIVES

2.1. IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [14] states that "**The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation**". Furthermore, para. 2.2 of SF-1 [14] states:

> "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."

2.2. This Safety Guide interfaces with other Safety Guides that are also intended to help meet the relevant safety requirements that underpin the fundamental safety objective. The following section provides a summary of the most relevant safety requirements and recommendations that underlie the recommendations in this Safety Guide in the areas of radiation protection in design; safety in design; safety assessment in design phases; interfaces between safety and security; and radiation protection for emergency response.

Radiation protection in design

2.3. In accordance with SSR-2/1 (Rev. 1) [1], provisions for radiation protection are required to be made in the design of a nuclear power plant. Paragraph 2.6 of SSR-2/1 (Rev. 1) [1] states:

> "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."

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

> "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."

2.5. Requirement 5 of SSR-2/1 (Rev. 1) [1] states:

> "**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.**"

2.6. Requirement 81 of SSR-2/1 (Rev. 1) [1] states:

> "**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

2.7. Paragraph 2.8 of SSR-2/1 (Rev. 1) [1] states (reference omitted):

"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 objectives:

(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."

2.8. Paragraph 4.4 of SSR-2/1 (Rev. 1) [1] states that (footnote omitted) "Acceptable limits for purposes of radiation protection associated with the relevant categories of plant states shall be established, consistent with the regulatory requirements."

2.9. Paragraph 5.31A of SSR-2/1 (Rev. 1) [1] states:

"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."

2.10. Paragraph 5.4 of SSR-2/1 (Rev. 1) [1] states that "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."

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

"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."

2.12. Paragraph 4.20 of SSR-2/1 (Rev. 1) [1] states:

"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 design phases

2.13. SSR-1 [6] establishes requirements for the assessment of site suitability, including an assessment of 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. Paragraph 4.6 of SSR-1 [6] states:

"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 action [11], and the need to evaluate the risk to individuals and to the population."

2.14. IAEA Safety Standards Series Nos SSG-68, Design of Nuclear Installations Against External Events Excluding Earthquakes [15], SSG-67, Seismic Design for Nuclear Installations [16], SSG-9 (Rev. 1), Seismic Hazards in Site Evaluation for Nuclear Installations [17], SSG-89, Evaluation of Seismic Safety for Existing Nuclear Installations [18], NS-G-3.2, Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants [19] and GSG-10, Prospective Radiological Environmental Impact Assessment for Facilities and Activities [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.

2.15. Requirement 13 of GSR Part 3 [2] states:

> "**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.**"

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

> "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."

2.17. In the design of a nuclear power plant, the assessment of whether adequate measures are in place to protect people and the environment from harmful effects of ionizing radiation is required to be undertaken in accordance with Requirement 9 of GSR Part 4 (Rev. 1) [8].

2.18. Safety assessment of the provisions for radiation protection should be performed at different stages, including siting, design, manufacturing, construction, assembly, commissioning, operation, maintenance, final shutdown, defuelling and decommissioning of a nuclear power plant. Such assessment is required to be performed in accordance with Requirement 13 of GSR Part 3 [2], with particular attention paid to the requirements established in paras 3.29–3.36 of GSR Part 3 [2]. Safety assessment in the design process is also required to be carried out in accordance with Requirement 10 of SSR-2/1 (Rev. 1) [1].

2.19. The safety assessment of the provisions for radiation protection should continue throughout the various stages in the lifetime of a nuclear power plant, and the documentation of the assessment in the authorization documents issued or revised at each stage should be kept consistent; this also applies to the documentation related to environmental impact assessment, the safety analysis report and emergency plans.

2.20. The safety assessment on radiation protection should be regularly updated consistent with national requirements (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.

2.21. Paragraph 5.71 of SSR-2/1 (Rev. 1) [1] states:

> "It shall be demonstrated that the nuclear power plant as designed is capable of complying with authorized limits on discharges with regard to radioactive releases and with the dose limits in all operational states, and is capable of meeting acceptable limits for accident conditions."

2.22. Paragraph 5.25 of SSR-2/1 (Rev. 1) [1] states:

> "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 SSR-2/1 (Rev. 1) [1] states that: "The design basis accidents shall be analysed in a conservative manner."

2.23. Acceptable limits for workers (including those who control and mitigate design basis accidents) should be considered in the design criteria. Further recommendations are provided in Section 6.

2.24. In accordance with Requirement 20 of SSR-2/1 (Rev. 1) [1], design extension conditions could be analysed using best estimate assumptions. Recommendations related to the safety analyses for design extension conditions are provided in IAEA Safety Standards Series No. SSG-88, Design Extension Conditions and the Concept of Practical Elimination in the Design of Nuclear Power Plants [21]. The potential on-site and off-site radiological consequences resulting from design extension conditions should be evaluated to demonstrate compliance with the acceptance criteria reflecting reference levels established by the regulatory body. Further recommendations are provided in Section 6.

2.25. The recommendations provided in 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 during plant modifications and kept updated during operation.

Interfaces between safety and security

2.26. Paragraph 1.36 of GSR Part 3 [2] states:

> "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."

2.27. In addition, Requirement 8 of SSR 2/1 (Rev. 1) [1] states:

> "**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 "Security infrastructure and safety infrastructure need to be developed, as far as possible, in a well coordinated manner." Therefore, organizations involved in the design should be made aware of the commonalities and differences between safety and security so as to be able to factor both into the design, thus developing synergies between safety and security.

2.28. IAEA Nuclear Security Series No. 13, Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Revision 5) [22], provides recommendations for the physical protection of nuclear material and nuclear facilities. Specifically, para. 3.28 of IAEA Nuclear Security Series No. 13 [22] states:

> "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.”

In addition, para. 5.13 of NSS No. 13 [22] states:

> “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.”

2.29. Paragraphs 3.25–3.28 of IAEA Nuclear Security Series No. 14, Nuclear Security Recommendations on Radioactive Material and Associated Facilities [23], provide recommendations on the interfaces of nuclear security with safety for radioactive material, associated facilities and associated activities.

Radiation protection for emergency response

2.30. Provisions to ensure the protection and safety of all persons on the site in a nuclear or radiological emergency are required in accordance with the requirements established in GSR Part 7 [11]. Recommendations related to the design of radiation protection provisions for emergency response are provided in Section 6 of this Safety Guide.

2.31. Paragraph 5.42 of GSR Part 7 [11] states:

> “Arrangements … 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.”

2.32. The radiation protection provisions in the design should also include measures to address simultaneous design extension conditions at multiple nuclear power plant units (see para. 6.9).

2.33. Radiation protection provisions for emergency response, including monitoring systems, are required to be designed in compliance with Requirement 67 of SSR-2/1 (Rev. 1) [1], which states:

> "**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 and monitoring systems for emergency response are provided in Sections 6 and 8 of this Safety Guide.

2.34. Paragraph 6.42 of SSR-2/1 (Rev. 1) [1] states:

> "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."

Recommendations about such information and communication systems are provided in Section 6 of this Safety Guide.

2.35. Paragraph 5.64 of SSR-2/1 (Rev. 1) [1] states:

> "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."

2.36. Paragraph 5.40 of GSR Part 7 [11] states:

> "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 on how to implement these arrangements are provided in Sections 6 and 8 of this Safety Guide.

APPLICATION OF DOSE LIMITS IN DESIGN

Authorized dose limits and dose constraints for operational states and decommissioning

2.37. The design of the nuclear power plant is required 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. annually) in operational states and during decommissioning (see para. 5.71 of SSR-2/1 (Rev. 1) [1]). Paragraphs 3.26–3.28 of GSR Part 3 [2] establish requirements for regulatory bodies to use in establishing dose limits and dose constraints. Recommendations on the application of dose constraints and dose limits are provided in paras 3.28–3.33 and paras 3.34–3.48, respectively, of GSG-7 [3].

2.38. In accordance with para. 3.78 of GSR Part 3 [2], employers, registrants and licensees are required to ensure that workers exposed to radiation from sources within a practice that are not required by or directly related to their work have the same level of protection against such exposure as members of the public.

2.39. The authorized annual dose constraints for members of the public apply to the representative person of the population, who is an individual receiving a dose that is representative of the doses to the more highly exposed individuals in the population [13]. Studies should be carried out to identify the representative person and the critical pathways for the exposure of such a person. Discharge limits for specific radionuclides in liquid and gaseous effluents (e.g. calculated on an annual, quarterly, monthly or daily basis, with the shorter periods permitting increased release rates over short time periods and thus increasing operational flexibility) should be derived from the application of the dose constraints for representative persons. The discharge limits should ensure that the maximum individual dose

[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 for planned exposure situations, 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.

for a representative person does not exceed the dose constraint established by the regulatory body.

OPTIMIZATION OF PROTECTION AND SAFETY

Application of the optimization principle

2.40. Requirement 11 of GSR Part 3 [2] 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.**"

2.41. As well as keeping exposures below dose limits, exposures are to be kept as low as reasonably achievable, taking into account economic and social factors, as follows:

(a) Radiation exposure should be taken into account early in the design process and should be reduced by means of radiation protection measures such that further expenditure on design, construction, operation and decommissioning would not be warranted by the associated reduction in radiation exposure.
(b) Measures 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 should be taken into account in the design.

2.42. In general, the optimization of protection implies a choice from a set of protection measures, including design options such as shielding, maintenance of the integrity of systems containing radioactive materials, avoidance of materials that can be easily activated, minimization of surfaces that can be easily contaminated, removal of radionuclides from coolants, filtering of air in working areas, and remote operation and use of tools to minimize radiation exposure time. These measures should be considered in the hierarchy of control measures such that passive and engineered solutions are considered before active and procedural controls. Feasible options should be identified, appropriate criteria for their comparison should be determined, and the options should be evaluated and compared. Recommendations on radiological acceptance criteria for normal operation and anticipated operational occurrences are provided in paras 4.8, 4.9 and 7.23 of IAEA Safety Standards Series No. SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [24]. Recommendations on different structured approaches to making decisions are given in the Appendix to this Safety Guide.

2.43. 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]). Further recommendations on radiological acceptance criteria for accident conditions are provided in paras 4.10, 4.11, 7.31, 7.46, 7.58 and 7.60 of SSG-2 (Rev. 1) [24].

2.44. The optimization process should include consideration of the protection not only of the public but also of workers and items important to safety in the plant, including those related to the on-site management of radioactive waste.

2.45. Other Safety Guides providing recommendations to meet the requirements of GSR Part 3 [2] for optimization of protection and safety include the following:

(a) GSG-7 [3], which addresses the radiation protection of workers;
(b) IAEA Safety Standards Series No. GSG-8, Radiation Protection of the Public and the Environment [25];
(c) IAEA Safety Standards Series No. GSG-3, Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste [26], which addresses wider aspects of optimization of protection and safety for predisposal management facilities;
(d) IAEA Safety Standards Series No. GSG-9, Regulatory Control of Radioactive Discharges to the Environment [27], which provides recommendations on the application of the principles of radiation protection and the safety objectives associated with the control of discharges and on the process for the authorization of discharges;
(e) GSG-10 [20], which describes a framework and methodologies for prospective radiological environmental impact assessment, covering the assessment of public exposure only.

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

Minimization of radioactive waste

2.47. Radioactive waste arising during operation and decommissioning is required to be minimized. Requirement 8 of GSR Part 5 [9] states that "**All radioactive waste shall be identified and controlled. Radioactive waste arisings shall be kept to the minimum practicable.**" 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 in the design are provided in Sections 5 and 7 of this Safety Guide.

2.48. 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 for the predisposal management of radioactive waste, in accordance with para 2.6. of GSR Part 5 [9]. Paragraph 4.6 of GSR Part 5 [9] states:

> "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 of this Safety Guide provides recommendations on how to meet this requirement.

2.49. Paragraph 4.7 of GSR Part 5 [9] states:

> "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."

Sections 5 and 7 provide recommendations on how to meet this requirement.

Design targets for operational states

2.50. Sections 5–7 of this Safety Guide provide recommendations 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. Recommendations for these related technological systems are provided in detail in other relevant Safety Guides on design of nuclear power plants, including IAEA Safety Standards Series Nos SSG-52, Design of the Reactor Core for Nuclear Power Plants [28]; SSG-53, Design of the Reactor Containment and Associated Systems for Nuclear Power Plants [29]; SSG-56, Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants [30]; SSG-62, Design of Auxiliary Systems and Supporting Systems for Nuclear Power Plants [31]; SSG-63, Design of Fuel Handling and Storage Systems for Nuclear Power Plants [32]; and SSG-39, Design of Instrumentation and Control Systems for Nuclear Power Plants [33]. References to the relevant requirements provided in these Safety Guides are indicated accordingly in Sections 5–7 of this Safety Guide.

2.51. To ensure that a design 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. 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, as required by GSR Part 3 [2]. The design targets should be set at an appropriate fraction of the dose limits.[4] A dose should be reduced to below a target if this can be done at a cost that is justifiable. The terms ‘target’ and ‘target dose’ are used throughout this Safety Guide in respect of both individual and collective doses.

2.52. In order to focus the design efforts on those aspects of the design that contribute most to the collective and individual doses to workers, it is useful to set design targets for the collective dose to the groups of workers who are likely to receive the highest doses, such as maintenance workers and radiation protection officers. It is also useful to set design targets for the collective dose for each category of work, such as maintenance of major components, in-service inspection, refuelling and waste management. These, combined with dose

[4] Design targets do not represent limits; they are instead useful design tools in the optimization process. Provided that any excess can be justified, design targets may be exceeded. Achieving a design target does not, in itself, demonstrate that a design satisfies the optimization principle.

assessments at key stages of the design (see GSG-7 [3]), 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.

Design targets for accidents

2.53. The adequacy of the design provisions for the protection of site personnel and the public under accident conditions should be judged by comparing the calculated doses with the specified 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 frequency of the accident conditions, the lower the specified design target should be, as indicated in para. 2.43. The regulatory body may recognize this principle by setting different design targets for accidents with different frequencies of occurrence. In addition, the regulatory body may define design targets by specifying frequency criteria for all accidents in specified dose bands.

Design targets for decommissioning

2.54. Planning for decommissioning begins at the design stage and continues throughout the lifetime of the facility. Paragraph 7.3 of GSR Part 6 [10] states that "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 radiation protection in design for decommissioning are provided in Section 7 of this Safety Guide.

2.55. Appropriate design targets for decommissioning facilities and for decommissioning actions based on dose constraints should be derived, taking into account Requirement 1 of GSR Part 6 [10], which states that **"Exposure during decommissioning shall be considered to be a planned exposure situation and the relevant requirements of [GSR Part 3 [2]] shall be applied accordingly during decommissioning."**

2.56. Requirement 14 of GSR Part 6 [10] states that "**Radioactive waste shall be managed for all waste streams in decommissioning.**" The management of the radioactive waste generated during decommissioning is required to be considered in the decommissioning plan (see para. 7.4 of GSR Part 6 [10]) and should be supported by the design of the facility. Recommendations on meeting these requirements are provided in Section 7 of this Safety Guide.

2.57. The relevant dose limits for workers and for members of the public are required to be applied during decommissioning in accordance with Requirement 11 of GSR Part 3 [2]. In addition, para. 2.1 of GSR Part 6 [10] states that "Radiation protection of persons who are exposed as a result of decommissioning actions shall be optimized with due regard to the relevant dose constraints."

2.58. Paragraph 2.2 of GSR Part 6 [10] states (references omitted):

> "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."

Recommendations for corresponding design considerations are provided in Section 7 of this Safety Guide.

2.59. Paragraph 2.3 of GSR Part 6 [10] states:

> "National regulations on the protection of the environment and the requirements of [GSR Part 3 [2]] 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."

Recommendations for the design to meet these requirements are provided in Section 7 of this Safety Guide.

3. GENERAL ASPECTS OF RADIATION PROTECTION IN THE DESIGN OF NUCLEAR POWER PLANTS

SOURCES OF RADIATION IN NUCLEAR POWER PLANTS

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

(a) The reactor core (including fuel and non-fuel components), reactor internals and vessel, and the surrounding materials that are activated;

(b) The reactor coolant and associated systems such as the moderator, volume control and reactor water cleanup systems;
(c) The steam supply system, feedwater system and turbine generators (depending on the design);
(d) The waste treatment systems, spent resins and storage systems;
(e) Irradiated fuel, including connected cooling and cleaning systems;
(f) The fuel handling and storage system;
(g) Decontamination facilities;
(h) Ventilation systems;
(i) Miscellaneous sources such as sealed sources that are used for non-destructive testing.

During the design stage, special attention should be given to preventing direct exposure from the radiation sources that produce the highest radiation levels, such as the reactor core, irradiated fuel and spent resins. Recommendations on assessing radiation sources under operational states are provided in Sections 4 and 5. The main sources of radiation during the decommissioning of a plant are contaminated materials from areas that have been in contact with radioactive substances (mainly residues on surfaces and inside tanks and pipelines) as well as materials close to the core that have been activated (affecting the entire respective material volume). These contaminated materials should be described individually in the radiological characterization of the decommissioning plan, as recommended in IAEA Safety Standards Series No. SSG-47, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities [34]. Recommendations on assessing radiation sources during decommissioning are provided in Section 7.

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 stage of the nuclear power plant. Requirements on the safety assessment to be performed during the development of the design and on the final assessment are established in SSR-2/1 (Rev. 1) [1] and GSR Part 4 (Rev. 1) [8], and associated recommendations are provided in SSG-2 (Rev. 1) [24].

3.3. The main sources of radiation under accident conditions are fission products, activation products and actinides, for which precautionary design measures should be adopted. These products and actinides are released either from the fuel elements or from the various systems and equipment in which the products and actinides are normally retained. Recommendations on assessing radiation sources under accident conditions are provided in Section 6. Examples of methods for assessing radiation sources for selected accidents are described in Annex I. The presented accidents have been 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 are met. Examples of sources of radiation and source terms for various designs of nuclear power plants are described in Annex I.

DESIGN APPROACH FOR PLANT STATES AND DECOMMISSIONING

Design teams and operating experience

3.5. The design teams for a nuclear power plant should have a common understanding of safety and of safety culture, security culture and the significance of radiation risks and hazards. Managers of design teams should advocate a collective commitment to safety by the team as a whole and by individuals within the team. Requirement 12 of GSR Part 2 [7] states:

> "**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.**"

Managers of design teams should also advocate a collective commitment to security culture in accordance with IAEA Nuclear Security Series No. 7, Nuclear Security Culture [35].

3.6. The design teams should be aware of the radiation protection measures that need to be incorporated into the design.[5] Experts from relevant operating organizations should be effectively involved in the design of new plants and design

[5] There are various ways of ensuring that persons involved in the design are fully aware of these radiation protection measures, such as by having specialists in radiation protection document the measures and provide training on them. It may be appropriate to include an experienced operator in the design team.

modifications to an existing plant in order to help ensure that the requirements for radiation protection and waste management are met. Moreover, 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 protection is required to be carried out at all stages of the lifetime of equipment and installations, from design and construction to operation and decommissioning (see Requirement 5 of SSR-2/1 (Rev. 1) [1]). 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. Further recommendations to establish and maintain the radiation protection programme are provided in GSG-7 [3] and SSG-40 [4]. Guidance on optimization and decision making in radiation protection is provided in Ref. [36].

3.8. In order to implement a structured approach to radiation protection, the design organization should exhibit a strong safety culture by ensuring that the importance of radiation protection is recognized at each stage of the design.

3.9. 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 effects of their individual activities or functions on the provision of radiation protection for site personnel, members of the public and the environment. More specifically, an optimization culture should be established on the basis of the following:

(a) Knowledge of the practices that result in the exposure of site personnel and members of the public;
(b) Feedback of operating experience to the design team;
(c) Familiarity with the main factors that influence individual doses and collective dose;
(d) Familiarity with the analytical methods that are available to assist in the optimization of the design;
(e) 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 by means of the following:

(a) Expertise in all areas that affect the production of radioactive material and its transport and accumulation in the plant and the dispersion of radionuclides in the environment;
(b) The evaluation of the different sources of radiation in the plant and the resulting doses using the best available analytical methods and data from relevant operating experience;
(c) Familiarity with the relevant regulations, guidance and best practices;
(d) Familiarity with maintenance, in-service inspection and other work in areas in which there are high radiation levels and that make a major contribution to the radiation exposure of site personnel.

3.11. Owing 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 accumulation of radionuclides in corrosion products. This also refers to the chemical decontamination processes that are performed during operation and in the decommissioning stage. Recommendations on chemistry programmes for water cooled nuclear power plants are provided in IAEA Safety Standards Series No. SSG-13 (Rev. 1), Chemistry Programme for Water Cooled Nuclear Power Plants [37].

Organizational aspects

3.12. Achieving an adequate level of radiation protection is relevant to a wide range of aspects associated with the design. It is therefore necessary to ensure that, for all design related decisions that might affect exposures, the recommendations of radiation protection specialists have been taken into consideration. Moreover, the design process should be planned so that the implementation of these recommendations is feasible. Means should be provided for ensuring that the designers take into account the necessary radiation protection measures at every stage of the design process. Such means could include the following:

(a) Rules for the layout of the plant;
(b) Design measures to minimize the use of personal protective equipment;

(c) Checklists[6] that can be used by engineers and reviewed by radiation protection officers.

3.13. The nuclear power plant design project should be organized to allow for the following:

(a) Radiation protection officers within the design organization should be consulted at the early stages of the design when the options for the major aspects of the design are being evaluated. It may also be appropriate to consult with specialists from external organizations.
(b) 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.
(c) Radiation protection officers should review all decisions that might have a major influence on exposures.
(d) There should be an appropriate ongoing forum for proposing improvements and resolving disputes that might occur between design engineers and radiation protection specialists.

3.14. Recommendations on the application of a systematic and structured management system in the entire design process are provided in IAEA Safety Standards Series No. GS-G-3.1, Application of the Management System for Facilities and Activities [39].

3.15. A strong management commitment should be made to ensure that the optimization process in the operating organization is effective. In some organizations, this commitment includes the appointment of a manager for optimization who reports directly to the senior manager of the design project and thereby is involved in the decision making process. Further recommendations are provided in paras 2.9, 2.10 and 3.8–3.33 of GSG-7 [3] and in the Appendix of this Safety Guide.

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[6] An example checklist for optimization of protection in the design of a nuclear power plant is given in Ref. [38].

Design strategy

General approach

3.16. Design targets should be set at the start of the design process for a nuclear power plant and should include dose constraints for workers and for members of the public, including the following:

(a) Annual collective dose targets and individual dose targets (e.g. for average and maximum effective doses) for site personnel;
(b) Annual individual dose targets for members of the public.

Specific dose targets should also be established for activities conducted when the reactor is in operation and during outages. The dose calculations, including the methods and tools used for performing them, should be subject to review and approval by the regulatory body.

3.17. In practice, design targets can be addressed independently from each other, although in principle any enhancement of waste treatment systems to reduce releases of radioactive material to the environment may result in additional work for site personnel with a consequent increase in their exposures. In establishing 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 unjustified increase.

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

3.19. A simple illustration of the use of design targets is given in Fig. 1 for the design of a plant on the basis of 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 in itself 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.

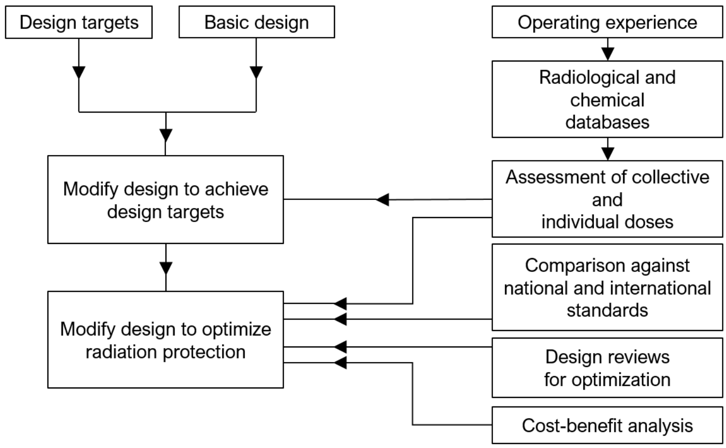

FIG. 1. Strategy for the optimization of radiation protection in the design of a nuclear power plant.

Radiation protection design for site personnel

3.20. The following procedure should be followed for developing the design of a nuclear power plant to ensure the radiation protection of site personnel:

(1) The general criteria for the plant design should be developed and documented. These criteria 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. The documents produced will form part of the management system for the design (see IAEA Safety Standards Series No. GS-G-3.5, The Management System for Nuclear Installations [40]).
(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.[7] Consequently, these aspects should be considered first in the design stage, and it should be ensured that the design has been proven. The aim should be to reduce exposures to levels that are as low as reasonably achievable, which will also help to improve the availability and therefore the economic

[7] For example: (a) In many reactor designs, two areas in which there is a major potential for reducing exposures are scheduled and unscheduled maintenance activities; (b) 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.

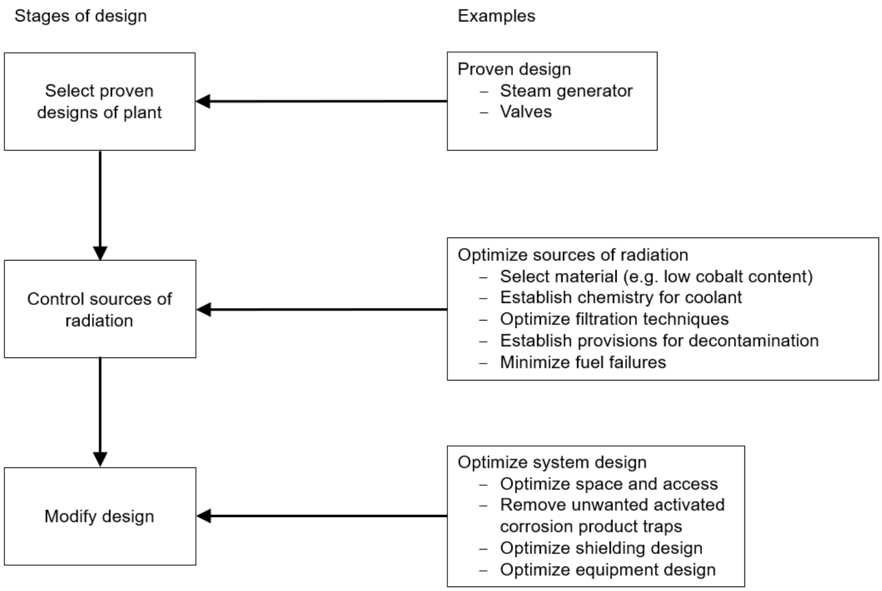

FIG. 2. A simplified strategy for the reduction of exposures in a pressurized water reactor.

performance of the plant. Design features that minimize the production and buildup of radionuclides should also be considered; such features will reduce radiation and contamination levels throughout the plant, whereas a local solution (e.g. increasing the shielding, improving ventilation) will have only a local benefit. Local plant features such as the plant layout, the shielding and the design of systems and components, should subsequently be considered. An example of a simplified strategy for reducing exposures in a pressurized water reactor is shown in Fig. 2.

(3) A logical layout for the plant should be developed, dividing the plant into zones on the basis of predicted dose rates and contamination levels (see para. 6.73 of SSR-2 (Rev. 1)), access requirements and specific issues such as the need to separate safety trains[8]. The dose rates may be calculated using the source terms that form 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 differences in the design and operating parameters are

[8] 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.

not significant. The zoning should be consistent with national legislation and regulatory requirements. It may be adequate to use the same zones that will be used when the plant is operating, but a more specific definition of the zones is often necessary for design purposes (see the examples 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 and security requirements and should not be artificially increased, to comply with the regulatory requirements or 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 should also be identified, for example maintenance personnel, in-service inspection personnel, technical support personnel (e.g. scaffolders, insulation workers), decontamination staff and radiation protection officers.

(5) Collective doses and individual doses should be evaluated by combining the results of steps 3 and 4. The use of a database on occupational exposure is recommended. 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 Fig. 3, which presents a schematic flow chart of the factors that determine individual and collective doses. This procedure should be 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. The factors that determine individual and collective doses are shown in Fig. 3. At each step in Fig. 3 involving design options, optimization studies should be performed. This is particularly important in cases for which it is predicted that the design targets will be exceeded.

(7) The hierarchy of control measures should be considered carefully. Hazard elimination is the best solution, followed by hazard reduction if elimination is not possible. The next step should be to isolate the hazard and to use engineered controls to reduce the risk to workers, with administrative controls and personal protective equipment at the bottom of the hierarchy.

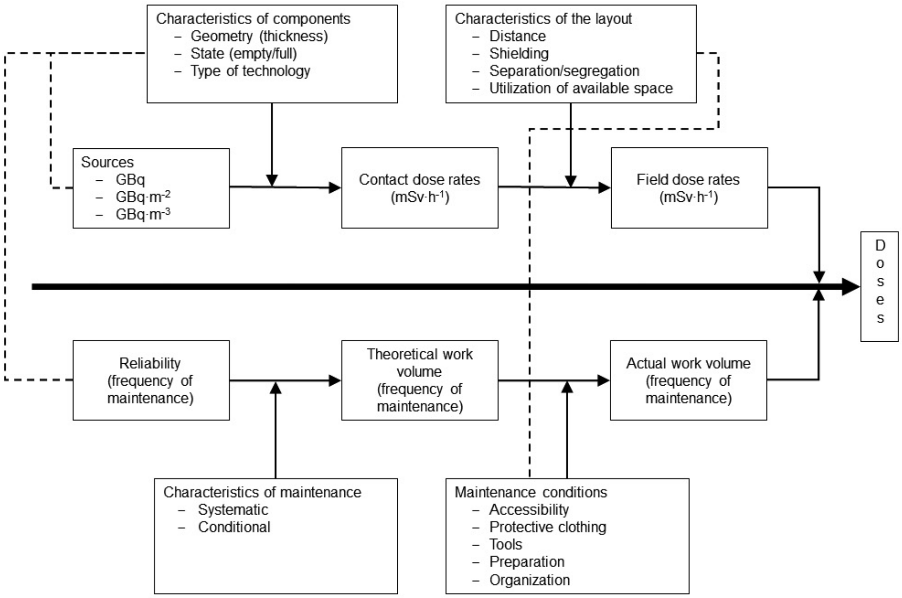

FIG. 3. Schematic flow chart of the origin of doses at a nuclear power plant (dashed lines indicate possible interactions).

3.21. Table 1 gives an example of the practical implementation of a strategy for the design process. In this strategy, the design is divided into four steps representing an increasing level of detail. The main parameters to be considered for each step are the individual and collective dose targets, the studies to be performed, the zoning, the contact dose rates and the exposed work volume. In Step 1, for example, an average individual dose target is set, as well as a collective dose target, including a margin. The studies to be performed as part of the optimization process result in a list of advantages and drawbacks for each option. No zoning is performed and no contact dose rate calculations are made. The exposed work volume is estimated, with account taken of different options (e.g. whether the work is performed by workers or by robots).

3.22. An important contributor to exposures of workers is the inhalation of airborne contamination such as tritium in pressurized heavy water reactors. A logical layout of the nuclear power plant should be developed, dividing the plant into zones on the basis of levels of airborne radionuclides.

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

	Design targets		Optimization process	Dose rates		Individual and collective doses
Item / Step	Individual dose target	Collective dose target	Studies to be performed	Zoning	Contact dose rate	Exposed work volume
Step 1	Average for all workers	Total for facility	Description of advantages/ drawbacks of options	(Not relevant)	(Not relevant)	Estimation of exposed work volume with options
Step 2	Update Step 1 value	Update Step 1 value	Evaluate main options	Establish approximate zoning	(Not relevant)	Programme definition of exposed work volume estimation
Step 3	Definitive value for the average for all workers	Evaluation with decisions of Step 2	Limited to important points	Evaluate using design source term/realistic source term/accident source term	Calculate contact dose rates	Estimation of exposed work volume
Step 4	Values for each task type	Evolution	Detailed by tasks	Verification/ precision	Verification/ precision	Detailed evaluation of exposed work volume

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 design that affects exposure to radiation is justified. This should be 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 will be the same as those necessary for operational states, but some additional features may be necessary for decommissioning. If these additional features are major, the necessary features for operational states and for decommissioning should be optimized. More recommendations on radiation protection design aspects for decommissioning are provided in Section 7.

3.25. The design should be such as to facilitate achievement of the targets for occupational dose — both individual doses and collective doses — by adopting some or all of the following measures using the hierarchy of preventive measures described in para. 3.93 of GSR Part 3 [2]:

(a) Reduction of dose rates in working areas by the following:
    (i) Reducing sources (e.g. by the appropriate selection of materials; reduction of surface and airborne contamination; decontamination measures; control of corrosion, water chemistry, filtration and purification; and the exclusion of foreign material from the primary systems);
    (ii) Improving shielding;
    (iii) Increasing the distance between workers and sources (e.g. through remote handling);
    (iv) Controlling the direction of air flow containing radionuclides and improving filtered ventilation, especially in pressurized heavy water reactors.
(b) Reduction of occupancy times in radiation fields by the following:
    (i) Specifying high standards of equipment to ensure very low failure rates;
    (ii) Ensuring ease of maintenance and of equipment removal;
    (iii) Removing the necessity for certain operational tasks (e.g. by providing built-in auxiliary equipment or making provision in the design for permanent access);
    (iv) Ensuring ease of access and good lighting;
    (v) Optimizing the number of workers and their time in a radiation field by design means;
    (vi) Making provisions for remote or automated inspection devices;

(vii) Isolating sources and important passages.

Design for radiation protection for members of the public

3.26. The design targets for annual individual doses to members of the public, mentioned in para. 2.39, should already be set in the site evaluation at the start of the design process. 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:

(a) Site specific features that affect 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, which should be subject to the approval of the regulatory body. One possible approach would be to set targets for radioactive releases in which account is taken of operating experience and the use of best practicable means in the design of the treatment systems for radioactive effluents.
(b) The resulting doses to the representative person should be evaluated to ensure achievement of the target.

3.28. All potential effluent pathways should be monitored. Provisions should also be made to ensure that no radionuclides unintentionally leave or enter the plant, for example by the use of radioactivity monitors at the gates. See also paras 3.20 and 3.21 of GSG-9 [26].

Design for radiation protection of the environment

3.29. The maximum radionuclide concentrations that might be present in relevant local flora and local and migratory fauna, as well as the internal dose that might result from those concentrations, should be assessed through consideration 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 a radiological point of view, the protection of non-human biota and the environment is generally achieved by protecting the human population (see para. 1.33 of GSR Part 3 [2]). However, if there is a need to demonstrate that the environment is being protected against the effects of radionuclides, the environmental assessment should address this need. Further recommendations on the protection of non-human biota are provided in GSG-8 [25], GSG-9 [26] and GSG-10 [20].

3.30. A radiological environmental impact assessment should be carried out in accordance with the recommendations provided in GSG-10 [20] and Sections 4, 5 and 6 of NS-G-3.2 [19] 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

3.31. The design of radiation protection measures for operational states and decommissioning should take into account the robustness of items important to safety and operational features (e.g. equipment storage and access, contamination control), as well as fire protection and emergency preparedness measures. The arrangements in the emergency plan should also be taken 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) [5]. Recommendations on measures related to emergency preparedness are provided in IAEA Safety Standards Series No. GSG-2, Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency [41], and in Section 6 of this Safety Guide.

3.32. Recommendations on reducing the risk of a release of radioactive material due to a fire are provided in IAEA Safety Standards Series No. SSG-64, Protection Against Internal Hazards in the Design of Nuclear Power Plants [42]. Design measures are required to be established 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) [5].

3.33. The nuclear security guidance provided in IAEA Nuclear Security Series No. 10-G (Rev. 1), National Nuclear Security Threat Assessment, Design Basis Threats and Representative Threat Statements [43], No. 11-G (Rev. 1), Security of Radioactive Material in Use and Storage and of Associated Materials [44], No. 19, Establishing the Nuclear Security Infrastructure for a Nuclear Power Programme [45], No. 25-G, Use of Nuclear Material Accounting and Control for Nuclear Security Purposes at Facilities [46], and No. 27-G, Physical Protection of Nuclear Material and Nuclear Facilities (Implementation of INFCIRC/225/Revision 5) [47], should be taken into consideration in the design of radiation protection measures.

3.34. Measures provided in the design for emergency arrangements and for radiation protection should be appropriate for maintaining safety in the event of an accident, for mitigating the consequences of accidents if they do occur, for protecting site personnel and the public, and for protecting the environment in accordance with para. 5.2 of SSR-2/2 (Rev. 1) [5]. The emergency plan and all emergency arrangements should be completed and implemented before the commencement of fuel loading.

Use of operating experience

3.35. The feedback of operating experience in operational states and decommissioning from reactors of similar design is required to be used to identify best practices concerning radiation protection and to improve the design of nuclear power plants (see Requirement 24 of SR-2/2 (Rev. 1) [5]. Associated recommendations are provided in IAEA Safety Standards Series No. SSG-50, Operating Experience Feedback for Nuclear Installations [48]). The feedback and experience from accident conditions should also be used to identify good practices.

DESIGN CONSIDERATIONS FOR THE COMMISSIONING AND OPERATION OF NUCLEAR POWER PLANTS

Design considerations for commissioning

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

3.37. The measures included in the design to provide an optimized level of protection and safety for operational states should be adequate to address the requirements for the commissioning stage (see IAEA Safety Standards Series No. SSG-28, Commissioning for Nuclear Power Plants [49]), even if in the commissioning stage the radiation levels are generally lower because of the lower power levels and the lower buildup of radioactive material in the plant's components.

3.38. Test programmes to verify the adequacy of shielding and engineered features designed to restrict occupational exposure should be developed during the design stage and implemented during commissioning. These programmes should aim to demonstrate compliance with operational limits and conditions for safe operation (see also Requirement 28 of SSR-2/1 (Rev. 1) [1]). Further recommendations on the commissioning programme for nuclear power plants are provided in SSG-28 [49].

Design considerations for initial startup

3.39. Radiation protection infrastructure should be designed and available for a sufficient amount of time before the planned introduction of radioactive sources and nuclear fuel in order to fully establish the radiation protection programme and to ensure that all radiation monitoring equipment is tested and functioning correctly (see paras 3.33, 3.48, 3.61, 4.28, A–2, A–3 and A–14 of SSG-28 [49]).

3.40. Chemistry parameters for initial startup and for continued operation, including radiation protection considerations, should be specified in the design, as these parameters can have a significant effect on the reactor source term later in operation.

3.41. Surfaces should be specified in the design and preconditioned before and/or during initial startup in order to produce a protective layer and to ensure appropriate, passivated surfaces in all systems. Therefore, 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 (see para. 5.19 of SSG-13 (Rev. 1) [37]).

Design considerations for operation

3.42. Section 5 of this Safety Guide provides recommendations on radiation protection design aspects for nuclear power plant operation.

Outages

3.43. In the context of this Safety Guide, an 'outage' means a period during normal operation of a nuclear power plant when the reactor is shut down for maintenance, testing or refuelling. During an outage, exposures are received mainly due to activated corrosion product contamination of the primary circuit. Specific design features should therefore be considered to ensure that occupational exposures during outages are optimized.

Platforms and lay-down areas

3.44. The design should include provisions for platforms that allow for work to be performed safely and to store scaffolding and temporary shielding materials inside controlled areas. The design should also provide enough space for lay-down areas during maintenance activities to ensure easy access and to reduce time spent in areas with higher radiation levels. Material selection and surface treatment should be used to ensure easy decontamination.

Shielding

3.45. Temporary shielding provisions or supporting system design features should be considered for outage and maintenance work on primary system components that cannot be shielded permanently. Equipment needing 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 reduce the exposure of maintenance and inspection personnel servicing other components in the area. Shielding calculations for radiation levels at various locations should be performed, including for plant operation at full power as well as for specific preventative maintenance during plant outages.

Fuel storage and handling

3.46. The design of fuel pools and fuel transfer channels should include provisions for shielding and easy decontamination, particularly if this is necessary for the inspection or maintenance of fuel transfer equipment. The design of filtering and cleaning systems should take into account the need to ensure that occupational doses are optimized during maintenance.

3.47. The relevant recommendations on radiation protection aspects in the design and handling of nuclear fuel storage provided in SSG-63 [32] and IAEA Safety Standards Series No. SSG-15 (Rev. 1), Storage of Spent Nuclear Fuel [50], should be taken into account. Further recommendations are given in paras 5.93–5.103 of this Safety Guide.

Access to and exit from controlled areas

3.48. The design should include provisions for efficient access and exit control points and related facilities, such as for monitoring, for contamination and for registration of personnel accessing controlled areas using personal protective equipment. The provision of additional contamination monitors during outages should be considered.

Ease of maintenance

3.49. Where possible, without compromising nuclear safety and security, flanged connections should be provided on liquid systems for quick disconnection and easy access for cleanup; it should be noted, however, that flanges may increase the risk of leakages of active fluid. Electrical quick disconnects should be used in design to minimize 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. Valves located inside areas with high radiation levels should have sufficient space for maintenance (e.g. to accommodate temporary shielding).

Design considerations for startup and shutdown

3.50. Where oxygenation of the coolant circuit 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 is still operational, by deliberate oxygenation of the circuit by chemical means. Recommendations on design considerations for cleaning and avoiding corrosion products during startup and shutdown are provided in SSG-56 [30].

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

3.52. Further recommendations on chemical control during startup and shutdown for different reactor types are provided in SSG-13 (Rev. 1) [37].

3.53. Consideration should be given to the maintenance of systems and components at the design stage to minimize the maintenance needs, in particular for those items in high dose rate areas of the plant.

DESIGN CONSIDERATIONS FOR ACCIDENT CONDITIONS

3.54. 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 para. 2.8 of SSR-2/1 (Rev. 1) [1]). Accident prevention is not explicitly addressed in this Safety Guide; recommendations are provided in paras 5.44 and 5.45 of GSG-10 [20] and further information is provided in Refs [51–55].

3.55. The design objectives for accident conditions are to limit radioactive releases to acceptable levels and to optimize the following:

(a) The risks to the public from possible releases of radioactive material from the nuclear power plant;
(b) 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 safety features for design extension conditions that are incorporated into the design of the nuclear power plant. Achievement of the design objectives is required to be confirmed by means of a safety analysis (see Requirement 42 of SSR-2/1 (Rev. 1) [1]). 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. Further recommendations are provided in Section 6 of this Safety Guide.

3.56. To achieve the design objectives specified in paras 3.54 and 3.55, 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 operating personnel to manage the situation adequately in an accident (see Section 6).

3.57. Design approaches that are similar to those used for operational states should be used to ensure that the nuclear power plant design provides adequate radiation protection for site personnel and the public under accident conditions. A safety culture is required to be established (see Requirement 12 of GSR Part 2 [7]) 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.

3.58. To achieve the proper design of plant systems and components for radiation protection under accident conditions, experts in radiation protection, plant operations, plant design, accident analysis and regulatory matters should be involved in all stages of the design process. There should be continuous interaction among these groups throughout the design process to achieve a design that provides for radiation protection under accident conditions and is acceptable to the regulatory body. The design should facilitate the implementation of effective procedures for accident management in accordance with IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants [56].

DESIGN CONSIDERATIONS FOR THE DECOMMISSIONING OF NUCLEAR POWER PLANTS

3.59. The design of a nuclear power plant is required to ensure that the plant can be safely decommissioned (see Requirement 6 of SSR-2/1 (Rev. 1) [1]).

3.60. Long life structures, systems and components that could be radioactive should, to a reasonable extent, be designed for the lifetime of the facility in order to avoid the necessity to replace them and to lessen the potential for leakage and contamination.

3.61. Throughout the lifetime of the nuclear power plant, the operating organization is responsible for the adequate maintenance of documentation to facilitate future decommissioning (see para. 3.6(h) of SSR-2/1 (Rev. 1) [1]). Paragraph 4.8 of SSR-2/1 (Rev. 1) [1] states:

> "The design shall be such as to ensure that the generation of radioactive waste and discharges are kept to the minimum practicable in terms of both activity and volume, by means of appropriate design measures and operational and decommissioning practices."

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

3.63. Remote techniques may play a major part in the removal of highly radioactive items during decommissioning. The use of such techniques should be considered at the design stage to ensure 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 conducted should be used.

3.64. In accordance with SSR-2/2 (Rev. 1) [5] and GSR Part 6 [10], monitoring for purposes of radiation protection is required for both plant operation and decommissioning, and the general provisions discussed in Section 8 of this Safety Guide 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 IN NUCLEAR POWER PLANTS

ESTIMATING RADIATION DOSE RATES DURING PLANT OPERATION AND DECOMMISSIONING

4.1. During the design stage of a nuclear power plant, the need for equipment to assess radiation exposures should be considered. This assessment should be based on estimations of radiation doses during commissioning, operation and decommissioning, recommendations for which are provided in this section. Further recommendations on occupational radiation protection (external exposure and internal exposure) are provided in GSG-7 [3].

4.2. The first step in any calculation of external dose should be to evaluate the nature and magnitude of the radiation source and its intensity and 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 a defined point and then to calculate the radiation dose rate by multiplying the flux by the appropriate conversion factors. Recommendations on the assessment of external exposure are provided in paras 7.1–7.132 of GSG-7 [3].

4.3. At the design stage, the estimation of internal exposure of workers is based on the likely duration of exposure in the workplace, the radionuclides and the levels of airborne activity (including activity resuspended from surfaces, using resuspension factors), the particle size distribution, the breathing rate and dose conversion factors. Recommendations on the assessment of internal exposure are provided in paras 7.133–7.227 of GSG-7 [3].

SOURCE CATEGORIES FOR NORMAL OPERATION AND FOR THE DECOMMISSIONING OF NUCLEAR POWER PLANTS

4.4. The sources of radiation during normal operation and decommissioning of a nuclear power plant should be identified at the design stage. The ways in which they arise are briefly described in Annex I of this Safety Guide. The sources may be grouped into categories according to how they might affect exposures in different ways that should be considered in the design. In general terms, these categories are as follows:

(a) Sources for which the design of the shielding will be determined;
(b) Sources for which shielding is not practicable and that may be major sources of doses to workers during plant operation;
(c) Sources that are major sources of doses to workers during decommissioning;
(d) Sources that present special hazards to workers during plant operation, such as particles containing alpha emitters or high concentrations of activated cobalt;
(e) Sources that are important contributors to doses to members of the public during plant operation;
(f) Sources that are important contributors to doses to members of the public during plant decommissioning.

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

Sources for which shielding is needed

The reactor core and its surroundings

4.5. The main sources of radiation in an operational nuclear power plant are the reactor core and the surrounding materials that are activated by neutrons that escape from the core.

4.6. 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 is achieved using computer codes that take into account 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 through the structural and shielding materials surrounding the core to determine corresponding design criteria.

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

Reactor components

4.8. Depending on the design of the nuclear power plant, several components within the reactor vessel are regularly removed and become sources in locations outside the vessel. These include the fuel elements, control rods, neutron sources, burnable poison rods, in-core instrumentation and the internals of the reactor.

4.9. The source terms for all of these components, which 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 apply to the maximum rated fuel assembly and the end-of-life activity for the other components.

Activity of the coolant

4.10. 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:

(a) Corrosion products;
(b) Fission products;
(c) Activation products.

4.11. For most types of reactors, corrosion products are the main contributors to radiation levels in 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.[9] 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 seven seconds and is a major source of radiation when the reactor is at power.

Radiation transmission through shielding

4.12. Calculations should be made for the transmission of radiation (mainly gamma rays) from each source 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 needed.

4.13. In the design of shielding to achieve acceptable dose rates, the calculation for determining attenuation should start with a design that is estimated on the basis of previous experience. The results should be evaluated to ensure the optimization of protection with regard to 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

4.14. Some tasks have to be performed in situations in which it is not practicable to provide shielding. Examples are work in the water chambers of pressurized water reactor 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 5.41–5.44.

Sources that dominate decommissioning doses and waste volumes at nuclear power plants

4.15. Sources of radiation that contribute to doses received during decommissioning of nuclear power plants will include activation products in the core components

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

and the surrounding materials, contamination in the primary and auxiliary circuits, and the accumulation of active material at the plant.

4.16. In a well designed and properly operated reactor, the main 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 significant sources of exposure for many years after shutdown will be ^{60}Co and ^{137}Cs. Cobalt-60 arises from impurities in materials, and this will dominate until ^{63}Ni becomes the dominant radioisotope. 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. Caesium-137, as a fission product with a long half-life, will typically remain over the lifetime of the plant and be controlled mainly by the quality of the fuel elements and filtering of the cooling water. For internal exposure, the most relevant radioisotopes include ^{3}H and ^{90}Sr.

4.17. 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, the concrete inside controlled areas should be coated before plant operation in order to facilitate cleaning and decontamination, provided that the concrete parts are accessible for such cleaning and decontamination.

4.18. Considering possible defects in the fuel cladding, the primary and auxiliary circuits may be contaminated with alpha emitters. The amount of irradiated fuel deposited on surfaces may reach a few tens of grams.[10] For such situations, the risk of internal exposure by alpha emitters is a special hazard during maintenance, operation and decommissioning, and relevant precautions, such as providing respiratory protective equipment, should be taken.

Sources that present special hazards to workers during plant operation and decommissioning

4.19. A special hazard during plant operation and decommissioning can be dose rate ‘hot spots’. Such hot spots result from the activation of small objects present in the coolant. These objects may include the following:

(a) Particles of metal resulting from excessive wear of components or fuel assemblies;

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

(b) Debris left in the primary circuit or other circuits connected to it;
(c) Pieces of crud deposits on the fuel.

The activity concentration of such hot spots will depend on the material and the activation time. They usually move from circuit to circuit carried by the transfer of water. The contact dose rates generated by these sources range from a few tens of mSv/h up to a few hundred Sv/h. Hot spots are especially relevant during outages and decommissioning.

4.20. The recommendations provided in SSG-47 [34] on the application of dose limits should be taken into account during the design of the plant.

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

4.21. The important contributors to doses[11] to members of the public during operational states and design basis accidents are typically as follows:

(a) Carbon-14, ^{3}H and ^{85}Kr, because the best practicable means available for their removal by waste treatment systems are not efficient and because they have long half-lives;
(b) Argon-41, which is an important contributor even though it has a short half-life, because it is released in large volumes of air (e.g. in venting of the containment during operation for some pressurized water reactors);
(c) Xenon-133, which 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;
(d) Iodine, caesium and corrosion products.

Noble gases, radioiodine and other radionuclides with a short half-life are relevant only for a limited time span after fuel removal from the core or final shutdown of the reactor.

4.22. During decommissioning, the radionuclide distribution in expected environmental releases is different from that in releases during normal operation and in potential accident releases (e.g. activation products in materials surrounding the core). This also affects preparations to mitigate environmental releases during decommissioning. Further recommendations are provided in SSG-47 [34].

[11] The notion of ‘important contributors to doses’ is relative (see Annex I).

4.23. Recommendations on the assessment of doses to the public due to discharges are provided in GSG-10 [20]. Recommendations on the control of discharges and the process for authorization for discharges related to the radiation exposure of the public are provided in GSG-9 [26].

5. SPECIFIC DESIGN FEATURES FOR RADIATION PROTECTION IN NUCLEAR POWER PLANTS DURING OPERATIONAL STATES

PLANT LAYOUT

5.1. Paragraph 6.72 of SSR-2/1 (Rev. 1) [1] states:

> "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."

5.2. Paragraph 6.74 of SSR-2/1 (Rev. 1) [1] states:

> "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."

5.3. Paragraph 6.75 of SSR-2/1 (Rev. 1) [1] states that "Plant equipment subject to frequent maintenance or manual operation shall be located in areas of low dose rate to reduce the exposure of workers."

5.4. Assessments made for radiation protection design for planned exposure situations are required to be carried out in accordance with Requirement 9 of GSR Part 4 (Rev. 1) [8].

5.5. In the design of a nuclear power plant, 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. Training facilities that include realistic mock-ups of important plant features should be considered during the design stage.

5.6. Foreign material exclusion is vital to the safe and reliable operation of nuclear power plants (see para. 7.11 of SSR-2/2 (Rev. 1) [5]). The entry of foreign material into primary or secondary plant systems can cause equipment damage leading to high radiation levels and avoidable occupational exposure. Steps should be taken at the design stage to aid foreign material exclusion, including provision for temporary covers during maintenance, and equipment for system inspection following maintenance.

Classification of areas and zones

5.7. 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 entrance and exit points for personnel and for materials and equipment, but have sufficient emergency exit points for personnel.

5.8. Provisions are required to be made for controlling access to and exit from the controlled areas and for monitoring persons and equipment leaving the controlled areas (see para. 3.90 of GSR Part 3 [2]). 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 an evacuation should be provided.

5.9. Designers should consider the need for controlled areas within a nuclear power plant to be divided into zones on the basis of the anticipated radiation levels and contamination levels (i.e. dose rates and activity concentrations for surface or airborne radionuclides; see Annex II) in order to enable the operating organization to apply effective management controls during operation. The greater the radiation or contamination related risks of a zone, the greater 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.

5.10. In the plant design stage, all rooms should also be classified into planning zones on the basis of their likely dose rates, surface and airborne contamination levels, and occupancy in normal operation, anticipated operational occurrences and during accident conditions. This approach can assist in plant layout design by providing a means of easily identifying areas of the plant that may be either suitable or unsuitable for the installation of plant equipment that could constitute a radiation source in accordance with Requirement 81 of SSR-2/1 (Rev. 1) [1]. These planning zones may not necessarily correspond exactly with the controlled areas that are in use during plant operation.

5.11. It may be necessary during operation or planned maintenance to reclassify certain areas temporarily or permanently. This possibility should be considered in the planning of access routes. If areas are reclassified, the zones and the controlled areas should be re-evaluated.

Changing areas and related facilities

5.12. Within controlled areas, changing areas are required to be provided where they are necessary to prevent the spread of contamination during normal operation, including maintenance (see para. 3.90 of GSR Part 3 [2]). The facilities in changing areas, including washing or showering facilities and other personal decontamination facilities at the exit of controlled areas, should be commensurate with the need to control access to areas of higher contamination. Where justified by the possible contamination levels, consideration should be given to the provision of permanent changing areas with decontamination facilities for personnel, and with monitoring instruments and storage areas for protective clothing.

5.13. Within changing areas, a physical barrier should be provided to clearly separate the clean area from the potentially contaminated area. The changing areas 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. Temporary personnel employed as contractors, for example during outage periods, should also be taken into consideration.

Control of access and occupancy

5.14. Safety, security and monitoring requirements for access to and exit from controlled areas should be considered together in order to deliver optimized technical solutions.

5.15. Access by personnel to areas of high dose rates or high levels of contamination should be controlled by the provision of doors that are lockable and, where appropriate, equipped with additional interlocks. Interlocks should be provided to ensure that access is possible only when radiation levels are acceptably low, and their design should include an alarm that will be activated if they become inoperative. Structures, systems and components provided to protect workers from undue exposures may need classification in accordance with IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants [57].

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

5.17. To limit the spread of contamination and radiation doses to personnel working in controlled areas, the layout of a controlled area should be designed such that personnel do not have to pass through higher radiation zones to gain access to 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.

5.18. As far as reasonably practicable, the design should be such as to limit the possible spread of contamination and to facilitate the implementation of temporary confinements.

5.19. 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 protection. This can be achieved, for example, by the following:

(a) 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.
(b) Provision of clear passageways of adequate dimensions to facilitate the removal of plant items to a workshop for decontamination and repair or disposal.
(c) Provision of adequate space in the working areas to perform tasks such as repairs or inspections.
(d) Provision of access to high radiation areas such as pressurized water reactor steam generators, and valves in systems that contain primary coolant.
(e) Provision of ‘waiting areas’ in low radiation areas.
(f) Placement of components that are likely to be operated frequently or need maintenance or removal at a convenient height for working.

(g) Remote or automated operation for components that will be operated frequently.
(h) Provision of ladders, access platforms, crane rails or cranes in areas where their use is foreseen for the maintenance or removal of plant components. Features to facilitate the installation of temporary shielding should be included in the design.
(i) Use of computer aided design models to optimize design aspects 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.
(j) Provision of means for the quick and easy removal of shielding and insulation where this is necessary in order to perform routine maintenance or inspection.
(k) Provision of special tools and equipment to facilitate work and thus reduce exposure times.
(l) Provision of remotely controlled equipment.
(m) Provision of a suitable system for communication with the site personnel working in radiation areas or contamination areas.
(n) Provision of electrical supplies that are easy to disconnect from and reconnect to equipment, and sufficient availability of sockets.
(o) Control of local areas during transfer of high activity radiation sources so as not to affect other simultaneous operations.

5.20. 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 RADIATION PROTECTION PROGRAMME AT A NUCLEAR POWER PLANT

System design

5.21. The design of nuclear power plant systems should be based on the feedback of experience gained in reducing radiation exposure at operating reactors, which will form the basis for establishing and maintaining a radiation protection programme in accordance with Requirement 24 of GSR Part 3 [2] and the recommendations provided in GSG-7 [3].

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

(a) The workspace around components that need regular maintenance in areas of high radiation levels should be shielded from the radiation from other systems. Where possible, components that need regular maintenance should not be placed in such areas.
(b) Non-radioactive components that do not have to be mounted close to active components should be installed outside areas of high radiation levels.
(c) Methods for sampling radioactive liquids that involve minimal exposure should be provided. Automated methods should be used where reasonably achievable.
(d) Methods for dose reduction (e.g. flushing) to avoid the sedimentation of radioactive sludge in piping and containers should be provided.
(e) For mutual standby systems, if it is necessary to maintain one system while the other system is in operation, sufficient shielding should be installed between the two systems.

5.23. To restrict the external exposure of site personnel, materials containing sealed radioactive sources (e.g. radioactivity measuring devices) that might present a hazard should preferably be stored in dedicated rooms or areas and not in places where workers are passing through.

5.24. 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 performing inspections as well as repairs and modifications should be left between the pipelines and the walls, and embedded pipework is to be avoided.

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

5.26. Pipelines should be so designed that few venting and drainage lines are needed. Drainage should lead to a sump or a closed system.

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

5.28. 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 needed.

5.29. Consideration should be given to having two trains of coolant clean up filters or a single train system with multiple filters in parallel, to allow continued clean up during oxygenation while the other filter is removed.

5.30. 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 contamination.

5.31. In boiling water 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 the turbine building during operation due to ^{16}N in the steam phase of the reactor coolant).

5.32. Recommendations on the design of alarms are provided in paras 4.143–4.176 of IAEA Safety Standards Series No. SSG-51, Human Factors Engineering in the Design of Nuclear Power Plants [58], and in Section 8 of this Safety Guide. See also para. 5.66 of SSR-2/1 (Rev. 1) [1].

Component design

5.33. Some general considerations apply to the design of components in order to take into account the requirements for radiation protection. Many of these considerations are the same as those that apply in system design (see paras 5.21–5.32).

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

5.35. The components installed in areas of high radiation levels should be designed to be easily removable.

5.36. 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.

5.37. 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, using decontaminable paints for wall surfaces and providing means of isolation, flushing and drainage for circuits that contain radioactive liquid.

5.38. Components whose maintenance and repair could result in an exposure that is a significant fraction of the dose targets for the plant should be well separated.

5.39. 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 eliminate losses of coolant resulting from seal failures.

Design for communication infrastructure

5.40. 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 Nuclear Security Series No. 17-T (Rev. 1), Computer Security Techniques for Nuclear Facilities [59]).

Remote techniques

5.41. 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 remote removal and reinstallation of equipment. These techniques should be considered in the design. Some techniques for the inspection and handling of plant items might be only semi-remote, meaning that personnel might still have to enter the controlled area to install equipment on rigs. An example of such a technique is the provision of equipment for the ultrasonic inspection of welds. Access to the weld might be necessary in order to fit the scanner, but the operator can then move to a low radiation area to operate the equipment.

5.42. Criteria for the selection of remotely rather than manually operated equipment such as valves should be established in order to make the optimization process more efficient. This should include consideration of the dose rate as well as frequency of use for normal operation.

5.43. 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 or radiation tolerant cameras and windows shielded by lead glass or comparable materials.

5.44. Automation aids such as multistud 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 minimized.

Shielding

5.45. Permanent shielding is required to be part of the design (see Requirement 4 of SSR-2/1 (Rev. 1) [1]), 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 minimize handling of the shielding. Permanent frameworks for shielding should also be provided to reduce the installation time for temporary shielding.

Design of shielding

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

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

5.48. After the potential radiation output of the source has been assessed, the process of shielding design should be carried out iteratively, starting with the design of shielding without penetrations (e.g. to shield pipes, cables and access ways). Next, consideration should be given to shielding that needs penetrations (e.g. for pipes, cables or access ways to pass through) and the provisions to be made to maintain the effectiveness of the shielding for the protection of site personnel.

5.49. The choice of shielding materials should be made on the basis of the nature of the radiation (e.g. beta and bremsstrahlung, neutrons and gamma rays, gamma rays only), the shielding properties of the materials (e.g. degree of scattering, absorption, production of secondary radiation, activation), the mechanical and other properties of the materials (e.g. chemical and thermal properties, mechanical stability, compatibility with other materials, structural characteristics, toxicity, disposability, ease of decontamination, radiation resistance) and space, installation and weight limitations.

5.50. Losses in shielding efficiency might occur as a result of environmental conditions. Effects that should be taken into account are those caused by the interactions of neutrons and gamma rays with the shielding (e.g. the depletion of materials that have a high neutron absorption cross-section, radiolysis and embrittlement), those resulting from 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).

5.51. 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 on radiation protection aspects in the design of fuel handling facilities are provided in SSG-63 [32].

5.52. Neutron transport calculations related to containment should be undertaken to eliminate leakage of neutron radiation (e.g. duct streaming paths from penetration).

5.53. A combination of materials may be necessary to obtain an optimal shielding design for the reactor core or for other neutron sources. Materials containing elements of a low atomic number (e.g. water, concrete) reduce the energies of neutrons for which the cross-sections are below the cross-section threshold for nuclear inelastic scattering of the shielding material(s).

5.54. When neutrons are captured in the shielding, the gamma rays that are emitted as a consequence of the capture need also to 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.

5.55. 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 (e.g. lead, 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.

5.56. With regard 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 and low density areas will be avoided.

5.57. In the design of permanent shielding, account should be taken of relevant external hazards, in particular seismically induced 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 [42].

5.58. 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 provisions necessary for transporting and installing it.

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

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

5.61. Consideration should be given to incorporating filters and demineralizers within concrete cells (i.e. to ensure that they are not accessible), together with shielded transport containers inside the plant to enable relatively high dose rates to accrue on the filters, in order to minimize radioactive waste while still ensuring that occupational exposures are optimized.

Penetrations through shielding

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

(a) Minimizing the area and number of straight-through paths containing material of very low density (e.g. gases, including air).
(b) Placing the penetrations in areas of low occupancy or no personnel access in the shine path (e.g. at height).
(c) Providing shielding plugs.
(d) Providing zigzag or curved pathways to guarantee no line of sight through shields. In this case, the width or density of the shielding may be increased near the penetration to compensate for the loss of material due to the penetration.
(e) Filling the gaps with grouting or other compensatory shielding material.

5.63. An optimization study should be performed to select the approach to shielding penetrations that will ensure that occupational exposures are as low as reasonably achievable.

5.64. Depending on the intensity and location of the source with respect to the penetration, no additional shielding features may be necessary. In some 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, but streaming may occur when shielding materials are used in combination, for example streaming of gamma radiation through materials containing elements of a low atomic number.

5.65. Personnel access points to areas with 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 provisions 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

5.66. A dedicated active ventilation system should be provided to maintain appropriate clean conditions in workspaces within the controlled area, as described in ISO 26802: 2010 [60]. Further recommendations are provided in SSG-62 [31].

5.67. The primary radiation protection function of a ventilation system should be to control and reduce the airborne radionuclides in the working environment to reduce the need to wear respiratory protection (see Requirement 73 of SSR-2/1 (Rev. 1) [1]).

5.68. The spread of contamination and the amount of radionuclides released to the environment should be limited by providing features such as air cleaning filters and by maintaining appropriate pressure differentials between rooms, across filters, and between plant systems and the environment.

5.69. The efficiency of filter systems in gas cleanup equipment within the design specification should be established in the design basis. To ensure that efficiency remains above the design limit, the design is required to allow for suitable periodic tests and/or ongoing measurements such as sampling the air from upstream and downstream of the filter system (see para. 6.63 of SSR-2/1 (Rev. 1) [1]). Pressure differentials on filter systems should also be monitored.

5.70. 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 the replacement of filter media.

5.71. Arrangements and instrumentation for the safe replacement and transportation of filter waste from cleanup equipment should be provided. These arrangements should include, for example, the use of filter holders that provide easy detachment of spent filters and the transportation of spent filters in containers that provide the necessary shielding.

5.72. In addition to radiological hazards, non-radiological hazards posed by the leakage of primary coolant (e.g. combustion in the case of liquid sodium[12], asphyxiation in the case of CO_2) should be taken into account in the design basis of ventilation systems.

5.73. The ventilation system is also required to provide suitably conditioned air to ensure the comfort of personnel and to maintain appropriate environmental conditions to ensure the reliability of plant equipment (see Requirement 73 of SSR-2/1 (Rev. 1) [1]).

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

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

(a) Mechanisms of thermal and mechanical mixing;
(b) The limited effectiveness of dilution in reducing airborne contamination;
(c) Exhausting of the air from areas of potential contamination at points near the source of the contamination;
(d) The use of exhaust rates that are commensurate with the potential for contamination in the area;
(e) The need to ensure that the exhaust air discharge point is not close to an intake point of the ventilation system;
(f) The need to avoid extracting potentially contaminated air in the vicinity of likely worker positions.

5.75. The airflow in the ventilation system should be such that the pressure in a region of lower airborne contamination levels is higher than the pressure in a region of potentially higher contamination levels. 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.

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

Decontamination

5.77. The need for decontamination is required to be considered at the design stage (see para. 6.76 of SSR-2/1 (Rev. 1) [1]). If it is considered that a worthwhile reduction in radiation exposure would result, the necessary provisions 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 equipment) where this is reasonably achievable (see also Ref. [61]).

5.78. 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.

5.79. Special consideration should be given to areas where leaks or spills of contaminated liquid might occur. These areas should be designed to allow easy decontamination (e.g. a special coating on the floors, easily decontaminated paint on the walls) and control the spread of contamination. Adequate bunding and sloping of these rooms should be arranged to limit the contaminated areas and ensure the quick drainage and collection of spilled liquids.

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

5.81. 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 cleanup is given in para. I–25 of Annex I.

5.82. 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.

5.83. Provision is required to be made for the decontamination of personnel (see para. 6.76 of SSR-2/1 (Rev. 1) [1]), and this should include the decontamination of reusable protective clothing.

5.84. Drains from decontamination facilities should be connected to the treatment systems for radioactive effluents.

Floor drain systems

5.85. 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 level detectors that actuate a level alarm as necessary. Further recommendations on equipment and floor drainage systems are provided in paras 4.269–4.285 of SSG-62 [31].

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

5.87. There should be sufficient tank volume so that any temporary transfers of radioactive water through the floor drain system (including transfers associated with infrequent events such as primary circuit decontamination) do not burden systems that are intended for other purposes. It is required to be ensured that any releases of liquid radioactive effluent to the environment will remain within authorized limits (see Requirement 34 of SSR-2/1 (Rev. 1) [1]). Further recommendations are provided in SSG-62 [31].

Waste treatment systems

5.88. Requirements on the management of radioactive waste before it is sent to a repository are established in GSR Part 5 [9], and related recommendations are provided in SSG-40 [4] and SSG-41 [12].

5.89. The equipment in treatment systems for solid, liquid and gaseous radioactive waste at a nuclear power plant may contain radioactive material in high concentrations, and radiation protection 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 levels that could arise in each area of the waste treatment system. Consideration should be given to the sources that give rise to the highest radiation levels (e.g. ion exchange resins, discarded radioactive components, filter waste). In the assessment of the sources and the estimation of radiation levels, account should be taken of the changes in the activity 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).

5.90. The design should be such as to minimize the sedimentation of radioactive sludge or the deposition of resins and evaporation concentrates in the piping and components of the waste treatment system, as well as the crystallization and deposition of such materials in tanks.

5.91. 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 can retain a volume of fluid equal to at least the capacity of the tank, or the walls of each room should be able to be readily decontaminated 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.

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

Storage of spent fuel and radioactive waste at a nuclear power plant

5.93. Facilities are required to be provided for the storage of radioactive waste that arises at the plant (see para. 6.59 of SSR-2/1 (Rev. 1) [1]), with account taken of its form (solid, liquid, gas or a mixture), its radionuclide content and the extent to which it has been processed. The storage of waste will depend in part on the design, construction, operation and maintenance of the facility concerned. The design 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 are provided in paras 6.73–6.83 of SSG-40 [4] and paras 4.199–4.232 of SSG-62 [31].

5.94. Facilities are required to be provided for the safe storage of spent fuel (see Requirement 80 of SSR-2/1 (Rev. 1) [1]). Recommendations on radiation protection aspects in the design of a spent fuel storage facility are provided in paras 3.45–3.48 and 3.107–3.112 of SSG-63 [32].

5.95. The design of storage facilities for radioactive waste at a nuclear power plant should incorporate the following functions (see also paras 6.73–6.83 of SSG-40 [4]):

(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) Providing for inspection and/or monitoring of the waste packages and storage facility, as necessary;
(e) Allowing for maintenance and repair of waste packages;
(f) Allowing for retrieval of radioactive waste for transport off the site;
(g) Allowing for expansion of the storage capacity, as appropriate;
(h) Allowing for movement of waste inside the storage facility;
(i) Considering eventual decommissioning.

5.96. Paragraph 6.66 of SSR-2/1 (Rev. 1) [1] states:

> "The fuel handling and storage systems for irradiated and non-irradiated fuel shall be designed:
>
> (a) To prevent criticality by a specified margin, by physical means or by means of physical processes, and preferably by use of geometrically safe configurations, even under conditions of optimum moderation;
> (b) To permit inspection of the fuel;
> (c) To permit maintenance, periodic inspection and testing of components important to safety;
> (d) To prevent damage to the fuel;
> (e) To prevent the dropping of fuel in transit;
> (f) To provide for the identification of individual fuel assemblies;
> (g) To provide proper means for meeting the relevant requirements for radiation protection;
> (h) To ensure that adequate operating procedures and a system of accounting for, and control of, nuclear fuel can be implemented to prevent any loss of, or loss of control over, nuclear fuel."

5.97. Paragraph 6.67 of SSR-2/1 (Rev. 1) [1] states:

> "In addition, the fuel handling and storage systems for irradiated fuel shall be designed:
>
> (a) To permit adequate removal of heat from the fuel in operational states and in accident conditions;
> (b) To prevent the dropping of spent fuel in transit;
> (c) To avoid causing unacceptable handling stresses on fuel elements or fuel assemblies;
> (d) To prevent the potentially damaging dropping of heavy objects such as spent fuel casks, cranes or other objects onto the fuel;

(e) To permit safe keeping of suspect or damaged fuel elements or fuel assemblies;
(f) To control levels of soluble absorber if this is used for criticality safety;
(g) To facilitate maintenance and future decommissioning of fuel handling and storage facilities;
(h) To facilitate decontamination of fuel handling and storage areas and equipment when necessary;
(i) To accommodate, with adequate margins, all the fuel removed from the reactor in accordance with the strategy for core management that is foreseen and the amount of fuel in the full reactor core;
(j) To facilitate the removal of fuel from storage and its preparation for off-site transport."

5.98. Storage facilities should provide protection for waste and spent fuel 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, the ageing management programme (see para. 5.16 of IAEA Safety Standards Series No. SSG-48, Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants [62]), by maintenance and repair (see IAEA Safety Standards Series No. SSG-74, Maintenance, Testing, Surveillance and Inspection in Nuclear Power Plants [63]) and/or by replacement, with account taken of the following:

(a) Chemical stability against corrosion caused by processes acting within the waste and/or caused by 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, incidents or accidents;
(d) Resistance to thermal effects, if applicable;
(e) If the waste contains nuclides emitting high-energy beta rays (e.g. ^{90}Y), bremsstrahlung in vessels made of materials containing elements with a high atomic number.

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

(a) Generation of hazardous gases caused by chemical and radiolytic effects (e.g. 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).

5.100. The possibility of accidents should be taken into account in the design of storage facilities. Safety systems, and safety features for design extension conditions designed for this purpose may differ from, but should be complementary to, the features designed for normal operation.

5.101. In addition to radiological hazards, non-radiological hazards (e.g. fire or explosion), which might contribute to radiologically significant consequences, should also be considered in the design of storage facilities, in accordance with the recommendations provided in SSG-64 [42].

5.102. Where appropriate, equipment should be provided with suitable interlocks or physical limitations to prevent dangerous or incompatible operations, such as the incorrect placement of waste, the accidental release of loads or the application of incorrect forces in lifting and handling operations, as such operations will give rise to high dose rates. Additionally, interlocks could be used for the purposes of control of access and occupancy (see para. 5.15).

5.103. The need for remote handling should be considered in cases where waste containers give rise to high dose rates or where there is a risk that radioactive aerosols or gases could be released into the working environment. The design of remote handling devices should include means for their maintenance and repair, for example, the provision of a shielded service room to keep occupational exposures as low as reasonably achievable. The design of remote handling devices should also incorporate means to recover them and return them to a stable and safe state in the event of a malfunction or breakdown.

PROTECTION OF THE PUBLIC DURING PLANT OPERATION

Discharge criteria

5.104. The operating organization is required to ensure that doses to members of the public arising from radioactive discharges and from direct radiation in operational states do not exceed the dose limits and that the optimization principle is applied (see Requirement 5 of SSR-2/1 (Rev. 1) [1] and related recommendations in GSG-8 [25]). GSG-10 [20] and GSG-9 [26] provide recommendations on the calculation of the exposure of the public resulting from radioactive discharges.

5.105. The design is required to demonstrate that authorized limits for discharges will not be exceeded (see para. 5.71 of SSR-2/1 (Rev. 1) [1]). This is commonly done by specifying discharge limits for the most significant radionuclides, as described in para. 2.39 of this Safety Guide. Whenever possible, the discharge limits may be set on the basis of operating experience. Any modifications to discharge limits should be accepted by the regulatory body and should be reflected in the safety analysis report. When radioactive discharges are very low, the monitoring process used might have a strong influence on the interpretation of the operating experience in support of setting discharge limits. 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. Novel designs and new techniques should be considered in order to limit the production of ^{14}C and ^{3}H. 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.

5.106. Minimization of the frequency and impact of events that have the potential to increase public doses from discharges should be considered. Public doses due to all exposure pathways, including direct radiation, from the plant should be calculated on the basis of conservative assumptions about public occupancy around the plant. Studies should be performed to ensure that shielding is optimized to ensure that the exposure of persons outside the site is as low as reasonably achievable.

Waste stream source reduction

5.107. The design measures taken to control radioactive material in the plant (i.e. to protect site personnel) might also affect the activity of 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 operational limit should be specified, are an example. If this operational limit is exceeded for a specified period, the reactor should be brought to an appropriate state to prevent unacceptable public exposures. In practice, such limits are usually determined by the need 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

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

5.109. The flows and the activity concentrations of liquid and gaseous effluents are required to be monitored and controlled to ensure that the authorized discharge limits are not exceeded (see para. 6.81 of SSR-2/1 (Rev. 1) [1]). Liquid and gaseous waste treatment facilities are required to be provided (see Requirements 78 and 79 of SSR-2/1 (Rev. 1) [1]). These should be based on available best practices.

Liquid waste treatment systems

5.110. The major sources of contaminated water that need 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. These sources produce effluents that are essentially aqueous in nature, and the recommendations that follow are given on this basis. Where non-aqueous liquid waste (e.g. spent oil or organic solvents) is generated in sufficient volumes, the provision of a separate waste treatment system should be considered. Further recommendations on the treatment of aqueous and non-aqueous liquid waste are provided in SSG-40 [4].

5.111. Proven methods of treating radioactive wastewater 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 sufficient flexibility to deal with liquids of different origins and unusual compositions, and to enable retreatment if the authorized limits for discharge are 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 with low chemical and solid content is recycled to the primary circuit after suitable treatment. The same recycling is 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 leakage from the primary circuit to the secondary circuit in the steam generator. In this case, it may be necessary to treat the water from the secondary circuit to reduce the activity before the water is discharged. Nitrogen-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.

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

5.113. Consideration should be given to the amount of solid waste that is produced by the liquid waste treatment systems. The volumes of liquid that need treatment should be reduced to as low as reasonably achievable by the careful design of the circuits that contain radioactive water to prevent leakage and to minimize the potential for the plant to need decontamination. The treatment should be appropriate for the level and type of contamination in the water to achieve the necessary decontamination factors in a way that optimizes doses to site personnel and minimizes the production of solid waste. This should be achieved by segregating waste from different sources into waste streams. Each waste stream should contain waste with similar characteristics in terms of its chemical and particulate content so that each stream can be treated optimally. 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. This may result in certain limits, for example on the use of organic materials in demineralizers.

Gaseous waste treatment systems

5.114. Discharges of radionuclides to the atmosphere are required to be reduced to as low as reasonably achievable (see para. 6.61 of SSR-2/1 (Rev. 1) [1]). The treatment and control of gaseous waste should be based on available best practices.

5.115. The system for the treatment and control of 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 (e.g. ^{85}Kr) is often not justified but, if necessary, it can be achieved by using cryogenic devices of an appropriate design and material.

5.116. The isotopes of iodine, which usually have the greatest radiological impact, 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. Further information is provided in Annex I, in particular paras I–118 and I–119.

5.117. Particulate material from the system for gaseous waste treatment and control and from the ventilation systems should be removed using filters. It is good practice to ensure that all gas discharged from the plant that might be radioactive passes through high efficiency filters.

5.118. 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 needed should be determined in the optimization process, with consideration given to accident conditions (see also NS-G-3.2 [19] and GSG-10 [20]). Different types of measurement should be provided to monitor radionuclides released via the stack (see paras 8.27–8.31).

Radiological support facilities

5.119. The plant design is expected to 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 performing adequate monitoring of the workplace and individual monitoring, for providing the workers with the necessary protective equipment, and for managing other health physics operations. These auxiliary facilities should include the following:

(a) A health physics operations office, including storage and testing facilities for radiological instruments and protective equipment, facilities (with sufficient space, and power and gas supplies) to take sample measurements, and computer equipment for the display of remote monitoring data;
(b) A changing room for changing into and out of protective clothing;

(c) A personnel decontamination facility including provisions for showering and hair washing;
(d) An equipment decontamination facility;
(e) Laundry facilities for contaminated clothing, where these services are not provided by an external provider;
(f) A first aid room;
(g) A radiochemistry laboratory for the preparation of samples and the measurement of activity, the specifications of which are in accordance with the reactor chemistry analysis needs;
(h) A storage area for contaminated items and tools;
(i) A workshop for maintenance of contaminated equipment;
(j) A secure storage location for radiation sources;
(k) Facilities for the management and storage of waste;
(l) An effluent storage tank area;
(m) A dosimetry laboratory, or a dosimetry control point if this service is provided by an external provider;
(n) A data recording and storage system for creating relevant databases (e.g. for dosimetry records or instrument control) and for updating them as necessary;
(o) An alternative or remote health physics control centre;
(p) Assembly areas at the plant for use during an emergency;
(q) Emergency response facilities;
(r) An identified sheltering area for plant personnel.

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

(a) Protective clothing and boots;
(b) Respiratory protective equipment;
(c) Air samplers and other equipment for measuring airborne activity concentrations;
(d) Portable dose rate meters with an audible alarm at variable settings, and devices for monitoring personnel contamination and surface contamination;
(e) Portable shielding, signs, ropes, stands and remote handling tools;
(f) Communication equipment;
(g) Meteorology instruments;
(h) Equipment for monitoring individuals for intakes of radionuclides;
(i) Temporary containers for solid radioactive waste and special containers for radioactive liquids;
(j) Emergency equipment (including extendable high dose rate monitors, additional protective clothing, self-powered air samplers and emergency vehicles);

(k) First aid equipment;
(l) Equipment for sampling and analysis around waste storage areas, such as borehole monitoring equipment for underground storage facilities for radioactive waste;
(m) Personal dosimeters for monitoring individual external exposure.

6. SPECIFIC DESIGN FEATURES FOR RADIATION PROTECTION IN ACCIDENT CONDITIONS

PLANT LAYOUT

6.1. At an early stage of site selection, and during the subsequent site arrangements, the radiation protection aspects of the design of a nuclear power plant should be taken in consideration. This includes the suitability assessment of the site for accident conditions, as required by para. 4.6 of SSR-1 [6]. The radiation protection aspects that should be addressed in the suitability assessment include the following:

(a) The effects of events that might happen in nuclear installations located in the area surrounding the site (see also SSG-68 [15]);
(b) Site characteristics that are relevant in the transfer of radioactive material potentially released from the nuclear power plant, including site characteristics relevant to the design of the escape routes aiming to minimize the risk to site personnel and the public during evacuation (see NS-G-3.2 [19]);
(c) Density, distribution and habits of the population (see NS-G-3.2 [19]) that are relevant to evaluation of the risk for individuals and for the population as a whole, and other characteristics of the area surrounding the nuclear installation that could affect the feasibility of planning effective emergency response actions (see SSR-1 [6]);
(d) At multiple unit sites, the effects of events in one nuclear power plant unit on one or more other units (see IAEA Safety Standards Series No. SSG-79, Hazards Associated with Human Induced External Events in Site Evaluation for Nuclear Power Plants [64]).

6.2. The design characteristics of the necessary emergency response facilities[13] should be considered during the design stage of the nuclear power plant. Such design characteristics include the means necessary for the radiation protection of personnel working in emergency response facilities provided in accordance with Requirement 67 and para. 6.42 of SSR-2/1 (Rev. 1) [1], and paras 5.25, 5.41 and 5.42 of GSR Part 7 [11].

6.3. Paragraph 5.42 of GSR Part 7 [11] states:

> "Arrangements…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."

Recommendations on such radiation monitoring are provided in Section 8 of this Safety Guide.

6.4. 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, sheltering) should be provided in the safe routes, with the aim of keeping internal and external exposures of site personnel within acceptable levels according to the design targets. Safe routes should avoid areas with the potential for hazardous conditions, including fire, chemical hazards, anoxia and/or high temperature. Safe routes should also avoid areas with the potential for high dose rates due to the presence of airborne contamination or sources of external radiation generated by the accident. Recommendations on design to reduce the exposure of personnel from process and post-accident sampling systems are provided in paras 4.67 and 4.68 of SSG-62 [31].

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[13] An emergency response facility is a facility or location necessary for supporting an emergency response, for which specific functions are to be assigned at the preparedness stage, and which need to be usable under emergency conditions [13].

OTHER DESIGN CONSIDERATIONS FOR AN EFFECTIVE RADIATION PROTECTION PROGRAMME FOR ACCIDENT CONDITIONS

6.5. The possible radiological consequences of design basis accidents and design extension conditions, including severe accidents, is required to be determined to demonstrate compliance with design targets related to protection of site personnel, the public and the environment (see Requirements 5, 19 and 20 of SSR-2/1 (Rev. 1) [1] and para. 6.28 of this Safety Guide).

Protection of site personnel under accident conditions

6.6. To complete the design for the radiation protection of site personnel in accident conditions, an assessment should be made of the magnitudes, locations, possible dispersion mechanisms and exposure pathways of the radiation sources that will be present during and after accident conditions. All potential accident scenarios, including severe accidents, should be considered in this assessment.

6.7. The operating organization is required to ensure the safety of all persons involved in emergency response in the event of a radiological emergency (see Requirement 11 of GSR Part 7 [11]). 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 necessary 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: see IAEA Safety Standards Series No. SSG-34, Design of Electrical Power Systems for Nuclear Power Plants [65]) are located (or spaces adjacent to such rooms), areas where manual actuation may be needed, on-site sampling facilities (e.g. for the containment and for the stack) and laboratories. The design should ensure that adequate respiratory protective equipment is readily available and properly maintained for access to these areas, as necessary. For this purpose, plant operating instructions for accident management actions and emergency response actions should be developed. Design modifications should be based on the findings of the habitability assessments, in accordance with the relevant requirements established in GSR Part 7 [11] and with the recommendations provided in IAEA Safety Standards Series No. GS-G-2.1, Arrangements for Preparedness for a Nuclear or Radiological Emergency [66]. See also Ref. [67].

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

6.9. At multiple unit sites, to ensure habitability and accessibility in accident conditions, the design is required to take due account of the potential for specific hazards arising in one unit and affecting several or even all units of the site simultaneously (see para. 5.15B of SSR-2/1 (Rev. 1) [1]). The operating organizations of neighbouring units should share information, as recommended in para. 2.72 of SSG-54 [56]. The aim is to ensure the protection and safety of all persons on the site in a nuclear or radiological emergency, and provisions should include measures in case of simultaneous severe accidents at several power plant units (or any other nuclear facilities located at the site) accompanied by disturbances of infrastructure outside the site (e.g. blockage of access roads, failures of the power supply, communication failures).

6.10. The postulated hazardous conditions[14] in which emergency workers may need to perform response functions on or off the site should be identified. Arrangements should be made for taking all practicable measures to provide 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. These arrangements include:

(a) The assessment, control and recording of the doses received by emergency workers;
(b) Procedures to ensure contamination control;
(c) The provision of appropriate personal protective equipment (which depends on the severity of the hazard), procedures and training for emergency response in the postulated hazardous conditions;
(d) Arrangements for the provision and secure storage of sufficient amounts of consumables (e.g. respiratory protection and protective clothing) that will be necessary during the postulated event.

6.11. In addition to the radiation shielding provisions required during operation, provisions should be made to ensure that personnel can access and occupy the relevant working places in order to operate and maintain equipment that needs

[14] In addition to nuclear and radiological risks, these hazardous conditions may also result from internal and external hazards such as fires, releases of hazardous chemicals, anoxia risk, storms or earthquakes.

to remain operable to prevent the escalation of an accident or to limit further radioactive releases (e.g. pumps in water cooled reactors or gas circulators in gas cooled reactors, which are needed to maintain core cooling) 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 [11][15] and to operate equipment that is necessary for monitoring the state of the plant after an accident. This includes access to equipment where maintenance or repair may be necessary after an accident. In general, provision should be made to avoid the need for direct intervention by operators by installing automatic or remotely controlled equipment (e.g. remotely controlled valves).

6.12. In anticipation of movements of source material (e.g. handling of fuel assemblies), consideration should be given 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.

6.13. Provision should also be made to minimize airborne radioactive contamination in areas to which access is needed to ensure the safety of 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). This may be achieved by operating the ventilation system in a recirculation mode. In this case, heat removal would have to be achieved by cooling the air in the 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 is expected to be too high to permit occupancy of the area 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. Control room habitability should be ensured by maintaining overpressure, air cooling and oxygen supply (e.g. by using compressed air bottles) and should be maintained under conditions of releases of gaseous chemicals. Reliable isolation (dampers) and habitability should be ensured in the case of loss of power supply. Further recommendations regarding ventilation systems for the main control room, supplementary control room and other emergency response facilities under

[15] In the event of an emergency, radiation dose limits for normal operation may be exceeded. Use should then be made of dose levels described in paras 4.14–4.17 of GSG-7 [3] for emergency workers.

accident conditions are provided in paras 4.160–4.167 of SSG-62 [31]); see also GS-G-2.1 [66].

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

6.15. Consideration should also be given to the provision of safe locations for vehicles that are involved in monitoring in accident conditions and are equipped, for example, with systems or equipment for dose rate measurements, airborne contamination measurements, radionuclide analysis, a global positioning system (GPS) and adequate filtration.

6.16. Provision should be made for alerting and assembling site personnel and for — at least provisionally — sheltering site personnel not involved in emergency response until their evacuation. Diverse communication systems (e.g. satellite phones) should be provided between the control room, the supplementary control room, other emergency response facilities and locations, and assembly points for personnel.

6.17. The identification of rooms as 'ready to act' using clearly posted signs and the removal of any obstacles to the free movement of site personnel in passageways should be ensured as a means of decreasing the duration of exposures during safety related actions in accident conditions. These factors should be taken into consideration at the design stage.

6.18. Areas should be identified within the plant in which radiation exposures are expected to remain low during accidents. These areas may be used in evacuating site personnel and monitoring them for contamination (see Ref. [67]). Personal dosimeters (including spare devices) for individual monitoring should also be stored here.

6.19. Arrangements should be made in advance to ensure that relevant information related to the protection of workers is recorded and retained for use during accident conditions, in evaluations conducted afterwards, and for the long term health monitoring and follow-up of emergency workers who might be affected (see paras 5.52, 5.59 and 5.67 of GSR Part 7 [11]).

6.20. The design should also consider the long term maintenance of items that are needed to operate for an extended period following an accident. Appropriate assessments of the long term dose rates should be provided to demonstrate that maintenance would be feasible following all types of accident, including severe accidents.

DESIGN PROVISIONS FOR THE PROTECTION OF THE PUBLIC IN ACCIDENT CONDITIONS

6.21. Provisions for shielding should be incorporated into the design to protect the public in 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, on the basis of topographical and meteorological data (see paras 8.27–8.31).

6.22. Compliance with acceptance criteria for accident conditions is required to be assessed by means of safety analyses (see Requirement 14 of GSR Part 4 (Rev. 1) [8] and Requirement 42 of SSR-2/1 (Rev. 1) [1]). In cases where the preliminary safety analysis shows that the acceptance criteria have not been met, additional protective features are to be incorporated into the design, or operational measures are to be developed to meet the acceptance criteria in terms of doses to the public and releases to the environment.

6.23. 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 unlikely. However, the possibility of accidental release of radioactive material to the aquatic environment should be considered for each plant, including the contamination of groundwater by direct leakages. It should be postulated that releases to the atmosphere might end up in the surface water through rain and/or wind, or due to washing out of radioactive material inside the plant, increasing the content of fission products in the aquatic environment.

6.24. It should be taken into account that the dispersion of radionuclides into the atmosphere during and after an accident mainly depends on the composition of the source term, the kinetic attributes of the release, the release point and the meteorological conditions. It is usual to assume that an unfavourable meteorological situation will prevail during and after the accident. 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 [26]. A methodology for the calculation of doses to the public should be developed in accordance with the requirements of the regulatory body, and this methodology should be carefully validated, as recommended in GSG-10 [20]. Design targets are usually set so that restrictions on the consumption of food, milk and drinking water are necessary, at least for design basis accidents. Thus, in such situations, the consumption of food, milk and drinking water that has been produced within the potentially affected area is used as an input to the dose calculation for the representative person.

6.25. In design basis accidents, conservative assumptions should be made with regard to the source term (see SSG-2 (Rev. 1) [24] and Annex I of this Safety Guide), duration of exposure, meteorological conditions, and living habits, 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.

6.26. Within the off-site areas where protective actions are planned in the event of an emergency (e.g. the precautionary action zone[16] and the urgent protective action planning zone[17]), arrangements are required to be made for promptly assessing releases of radioactive material, radioactive contamination and public doses for the purpose of determining or modifying urgent protective actions (see Requirement 14 of GSR Part 7 [11]).

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

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[16] The precautionary action zone is an area around a facility for which emergency arrangements have been made to take urgent protective actions in the event of a nuclear or radiological emergency to avoid or to minimize severe deterministic effects off the site. Protective actions within this area are to be taken before or shortly after a release of radioactive material or an exposure, on the basis of prevailing conditions at the facility [13].

[17] The urgent protective action planning zone is an area around a facility for which arrangements have been made to take urgent protective actions in the event of a nuclear or radiological emergency to avert doses off the site in accordance with international safety standards. Protective actions within this area are to be taken on the basis of environmental monitoring — or, as appropriate, prevailing conditions at the facility [13].

6.28. Design measures to protect the public in the case of radioactive releases in accident conditions include the following:

(a) Ensuring that event sequences and accident scenarios potentially leading to an early radioactive release or a large radioactive release are ‘practically eliminated’ (see para. 5.27 of SSR-2/1 (Rev. 1) [1] and the recommendations provided in SSG-88 [21]);
(b) Providing design means to minimize the scope of fuel damage and protect the barriers against releases of fission products from the fuel;
(c) Providing design means to maximize the integrity of primary system pipework;
(d) Achieving leaktightness, isolation and bypass prevention of the containment;
(e) Ensuring adequate safety features are provided to condense steam and/or maintain containment pressures within design limits following all accidents including severe accidents;
(f) 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;[18]
(g) 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;
(h) 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);
(i) Providing means of sealing the containment building or reducing the flow volume of exhaust air to provide for decay time within the building;
(j) Reducing both the mass and activity of radioactive material released by decreasing the discharge velocity of fluids or the closure time of valves;
(k) 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[19] (e.g. sodium tetraborate).

[18] For boiling water reactors, if steam from the pressure vessel is released to the atmosphere via the main steam relief safety valve in the early stages of an accident, a significant amount of volatile radioactive materials might be released through the water contained in the suppression chamber.

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

6.29. In addition, several types of safety related design measures (which may be based on probabilistic safety analyses) are to be taken, including the following:

(a) Developing or upgrading safety systems and safety features for design extension conditions to minimize equipment malfunctions and operator errors, and to prevent and/or mitigate severe accidents;
(b) Ensuring that power is available for essential equipment, instrumentation (including health physics instruments) and protection systems.

6.30. 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 (see para. 4.31 and Requirement 12 of GSR Part 7 [11]).

6.31. For design extension conditions, an analysis should be performed to demonstrate the scope and duration of the emergency protective actions to be implemented; this demonstration should be based on established criteria for protective actions such as those presented in Appendix II of GSR Part 7 [11]. The use of protective actions (e.g. sheltering, iodine prophylaxis, relocation of people) may be considered in the safety demonstration of the design. Such considerations should be limited in area and time and in accordance with national regulations. Dose reduction factors can be applied, provided that clear instructions are given in emergency plans and that sufficient time and other conditions ensure, with a high level of confidence, that the emergency protective actions can be implemented.

6.32. Certain items important to safety should be qualified for the harsh environmental conditions to which they will be exposed either during operational states or in accident conditions, as recommended in IAEA Safety Standards Series No. SSG-69, Equipment Qualification for Nuclear Installations [68]. The environmental conditions to be considered in the qualification should include effects of radiation and potential synergistic effects (e.g. radiation and temperature).

7. SPECIFIC FEATURES OF RADIATION PROTECTION IN DESIGN FOR DECOMMISSIONING

7.1. The recommendations provided in this section are applicable for the initial design stage and for the revision of the design of a nuclear power plant before decommissioning. Decommissioning is required to be taken into consideration at the design stage of the plant in order to minimize the production of radioactive waste and to optimize the dose received by workers at the time of the decommissioning (see Requirement 12 of SSR-2/1 (Rev. 1) [1]). Recommendations on decommissioning during the design and construction stages are also provided in paras 7.5–7.9 of SSG-47 [34].

7.2. Spent nuclear fuel and radioactive waste generated during the operational stage of a facility are required to be removed from the facility prior to decommissioning. However, if such removal is not possible during the period of transition between permanent shutdown and decommissioning, the final decommissioning plan is required to be adjusted (see para. 8.10. of GSR Part 6 [10]).

7.3. Decommissioning actions included in the initial, updated and final decommissioning plan are designed to achieve a progressive and systematic reduction in radiological hazards during decommissioning. The design of these actions should include robust planning and assessment, in accordance with regulatory requirements, and is required to ensure the protection of workers, the public and the environment (see para. 1.4 of GSR Part 6 [10]) and to demonstrate that the decommissioned facility achieves the defined end state (see Requirement 10 of GSR Part 6 [10]).

7.4. A summary of decommissioning plans should be included in the safety analysis report of different plant stages and submitted to the regulatory body when applying for different authorizations (e.g. for construction, commissioning, operation and decommissioning).

7.5. Tasks are different in different phases of decommissioning and are conducted in a work environment that is continuously changing; this should be considered in the design, especially in terms of the provisions for radiation protection.

7.6. Prior to decommissioning, an assessment of radiological hazards and imposed risks is required to be performed (see Requirement 3 of GSR Part 6 [10]). Safety should be reassessed 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.

7.7. In the design of the plant layout, a careful assessment should be made of the access needed for decommissioning of equipment. The design of the layout should take into account the need to optimize the exposure of site personnel during decommissioning. This optimization can be achieved by providing enough space for cutting and segmenting operations.

7.8. Before the start of decommissioning, the plant layout should be reviewed to assess whether changes are needed for radiation protection purposes. For example, more space and more flexible egress routes for waste and items might need to be provided by the creation of new openings. Consideration should be given to the design of the facility structure in order to facilitate the creation of such openings during decommissioning.

7.9. For detailed decommissioning planning, information is needed about the radiological conditions, the arrangement of structures, systems and components, the course of dismantling and planned radiation protection measures. Estimates of doses should cover not only dismantling but also preparatory work, cleanup after dismantling and activities related to waste handling and monitoring activities (e.g. sampling, measurement of samples). Layouts, schemes and pictures of the areas to be decommissioned should be made available. Decontamination and dismantling techniques should be carefully planned to ensure that the radiation exposure of workers is optimized. Feasible dismantling techniques should be compared to select the optimal option on the basis of individual and collective doses.

Classification of areas and zones for decommissioning

7.10. Each controlled area should have a minimum practicable number of access and exit points for personnel as well as for the materials and equipment necessary for decommissioning. Provisions for controlling access to and from the area and for monitoring persons and equipment leaving the area should be regularly reviewed. Each decommissioning phase should be considered in the design. Before decommissioning starts, the design should be reviewed with a view to identifying, re-evaluating and, as necessary, modifying the access and exit points for personnel, materials and equipment, and the zones for decommissioning.

7.11. Recommendations for the classification of areas and zones for operational states are provided in paras 5.7–5.11 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.

7.12. During the design of the plant, consideration should be given to minimizing the number and size of contaminated areas to facilitate decontamination during decommissioning, and also to minimize radioactive waste and optimize the radiation exposure of workers.

7.13. It may be necessary during decommissioning to reclassify certain areas, either temporarily or permanently. In this regard, particular attention should be paid to the planning of access routes. Under such conditions, the designation of zones within controlled areas should be re-evaluated.

Changing areas and related facilities for decommissioning

7.14. Paragraphs 5.12 and 5.13 provide recommendations on radiation protection aspects for the design of changing areas: these recommendations are also applicable to decommissioning.

Control of access and occupancy during decommissioning

7.15. Paragraphs 5.14–5.20 provide recommendations on controlling access to and from controlled areas in operational states. These recommendations should be considered for the decommissioning stage. Before decommissioning starts, the design of access points should be reassessed, including their classification, to ensure that the controls provided to protect workers remain adequate. As the decommissioning progresses through different phases, the arrangements for access control should be adapted to reflect the changing risk profile on the site.

7.16. The recommendations provided in para. 5.15 in relation to access to areas of high dose rates or high levels of contamination are also applicable to decommissioning. Before the decommissioning starts, the design of the applicable systems should be reassessed, including their safety classification, and the systems that are provided to protect workers from doses above dose constraints should be modified, as necessary.

7.17. Access routes for personnel through radiation zones and contamination zones should be minimized during decommissioning, thereby minimizing the time spent in transit through these zones.

7.18. Structures, systems and components provided to protect workers from doses above dose limits during decommissioning may need safety classification in accordance with SSG-30 [57].

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

7.20. As far as reasonably practicable, the design of a nuclear power plant should aim to limit the possible spread of contamination during decommissioning and to facilitate the implementation of temporary containments needed for decontamination and dismantling activities.

7.21. The design (or the revised design) for decommissioning should be such that the occupancy time in radiation areas and contamination areas is consistent with the principle of optimization of protection. This can be achieved, for example, by the following:

(a) 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 in which such equipment is necessary during decommissioning.
(b) 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.
(c) Facilitation of access to structures, systems and components, including compartmentalization of processes (e.g. through incorporation of hatches and large doors (see para. 7.6(b) of SSG-47 [34])).
(d) Provision of adequate space in the working areas to carry out decommissioning actions.
(e) Provision of ladders, access platforms, crane rails or cranes in areas where their use is foreseen for the removal of plant components during

decommissioning. Consideration should be given to access by robotic demolition machines as well as human access. Features to facilitate the installation of temporary shielding should be included in the design.

(f) 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, and visualization modelling should be performed 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.
(g) Provision of means for the quick and easy removal of shielding and insulation during decommissioning.
(h) Provision of special tools and equipment to facilitate work during decommissioning and thus reduce exposure times.
(i) Provision of remote controlled equipment.
(j) Provision of a suitable system for communication during decommissioning with the site personnel working in radiation areas or contamination areas.
(k) Access controls for areas where dose rates and/or contamination can be temporarily high.
(l) Provision of decontamination facilities and storage facilities with sufficient space for radioactive waste.

OTHER DESIGN CONSIDERATIONS FOR AN EFFECTIVE RADIATION PROTECTION PROGRAMME FOR THE DECOMMISSIONING OF NUCLEAR POWER PLANTS

System design for decommissioning

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

7.23. The following measures for reducing radiation exposure during decommissioning should be adopted in the system design (see para. 7.6 of SSG-47 [34]):

(a) The use of modular construction to facilitate the dismantling of structures, systems and components;
(b) Facilitation of the removal and/or decontamination of material and 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;
(c) 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;
(d) Consideration of provisions for the installation of 'test specimens' to facilitate the radiological characterization of structures, systems and components;
(e) Measures (e.g. flushing) to avoid the sedimentation of radioactive sludge in piping and containers used during decommissioning;
(f) Development of a waste management concept, especially concerning treatment of radioactive material for clearance or disposal, and options for logistics;
(g) Consideration of provisions for water supply and drainage systems.

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

7.25. 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 (see para. 7.6(c) of SSG-47 [34]).

7.26. Provisions for the draining and flushing of tanks to reduce contamination in operational states should be assessed, and, if necessary, additional provisions should be made for decommissioning. These should be in combination with pipework design that ensures that hold-up points (e.g. U-bends) are minimized. Ideally, systems should self-drain to a low point to enable the system to be completely drained.

Component design for decommissioning

7.27. The recommendations provided in paras 5.33–5.39 of this Safety Guide also apply to component design for decommissioning. Components used in areas of high contamination levels should be designed to be easily decontaminated by chemical or mechanical means after operation and easily removable during decommissioning.

Radiological support facilities for decommissioning

7.28. The plant design should include the auxiliary facilities that are necessary for effective radiological control in decommissioning of the nuclear power plant, including 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 performing workplace monitoring and individual monitoring, for providing the workers with the necessary protective equipment, and for managing other health physics operations during decommissioning. These radiological support facilities should include the features listed in para. 5.119, as appropriate.

7.29. During the design stage, the identification and reservation of locations for new facilities (and/or the repurposing of existing buildings) to support decommissioning (e.g. for waste management, for dismantling equipment) should be considered as part of the decommissioning strategy required by Requirement 8 of GSR Part 6 [10].

7.30. The following equipment should be provided, and should be available before the decommissioning of the nuclear power plant begins:

(a) Protective clothing (e.g. coveralls, extra coveralls, helmets, eye protection, puncture resistant gloves, puncture resistant safety shoes, boots, shoe covers);
(b) Respiratory protective equipment;
(c) Air samplers and equipment for measuring airborne activity concentrations;
(d) Portable dose rate meters with an alarm at variable settings and devices for monitoring personnel contamination and surface contamination;
(e) Portable shielding, signs, ropes, stands and remote handling tools;
(f) Communication equipment;
(g) Meteorology instruments;
(h) Equipment for monitoring individuals for intakes of radionuclides;
(i) Temporary containers for solid radioactive waste and special containers for radioactive liquids;
(j) Emergency equipment (including additional specialized protective clothing, air samplers with built-in power sources and emergency vehicles);
(k) First aid equipment;
(l) Equipment for sampling and analysis around waste storage areas, such as borehole monitoring equipment for underground storage facilities for radioactive waste;

(m) Transport and handling equipment provided in the nuclear power plant design (for the operating stage), for dismantling structures, systems and components and for waste management during decommissioning;
(n) Personal dosimeters for monitoring individual external exposure.

Remote techniques during decommissioning

7.31. 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 to ensure that their use is not precluded. It is likely that there will be improvements in remote control techniques over the lifetime of the plant as well as between the initial stage and later stages of decommissioning. The best practicable techniques that are available when the work is conducted should be used. The recommendations in paras 5.41–5.44 of this Safety Guide should also be considered, as appropriate, for decommissioning. Access to high dose rate areas before and during decommissioning should be avoided as far as practicable by optimizing the use of the remote visual inspection equipment used in operational states, such as radiation resistant and radiation tolerant cameras.

Decontamination during decommissioning

7.32. The need for decontamination is required to be considered at the design stage (see Requirement 12 of SSR-2/1 (Rev. 1) [1]). The decontamination facilities and activities included in the design for operational states should be assessed in the decommissioning plan. If necessary, new decontamination facilities should be designed for decommissioning.

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

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

7.35. The system of active floor drains should be extended to all areas where there are systems that will contain radioactive fluids during decommissioning. Sumps should be provided with liquid level detectors that actuate a high-priority alarm. Paragraphs 4.269–4.285 of SSG-62 [31] provide recommendations related to drainage systems that should also be considered. The tank volume for liquid radioactive waste in the design for operational states should be assessed in the decommissioning plan to ensure that any releases of liquid radioactive effluents from decommissioning activities to the environment will remain within the limits established for decommissioning.

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

7.37. There should be an adequate tank volume for storage of contaminated water. The tank volume should also be sufficient for the storage of liquid radioactive effluents resulting from decontamination during decommissioning. It should be ensured that any releases to the environment will remain within the authorized limits for discharges established for decommissioning.

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

7.39. Decontamination facilities should be provided during decommissioning for removing radioactive material from the surfaces of components, tools and equipment.

7.40. Provision is required to be made for the decontamination of personnel (see para. 6.76 of SSR-2/1 (Rev. 1) [1]), and this should include the decontamination of reusable protective clothing for decommissioning.

7.41. Drains from decontamination facilities should be connected to the treatment systems for radioactive effluents.

Ventilation during decommissioning

7.42. A dedicated active ventilation system should be provided for maintaining appropriate clean conditions during decommissioning in workspaces within controlled area. Specific circumstances that arise during decommissioning (e.g. modification of the ventilation system) should be taken into account at the design stage (e.g. inclusion of local connection points). The availability of the relevant ventilation systems should be assessed in the final decommissioning plan, prepared in accordance with Requirement 12 of GSR Part 6 [10]. Plans for removing components of the ventilation system during decommissioning should also be considered and should include the capacity to rebalance the system after the removal of individual components.

7.43. For the purposes of radiation protection, the primary objective of providing a ventilation system should be to control the levels of airborne radionuclides and to reduce the need to wear respiratory protection. It should be taken into consideration that dismantling activities might result in elevated activity concentrations of airborne radionuclides (see para. 8.21 of SSG-47 [34]).

7.44. The spread of contamination and the magnitude of radioactive releases to the environment should be restricted by providing features such as air cleaning filters, and by maintaining appropriate pressure differentials and the efficiency of filter systems within design specifications. It should be taken into consideration 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. 8.20 of SSG-47 [34]). To ensure that pressure differentials and the efficiency of filter systems remain within design specifications, the design is required to allow for suitable periodic tests and/or measurements (see para. 6.63 of SSR-2/1 (Rev. 1) [1] and para. 5.69 of this Safety Guide).

7.45. The ventilation system should also provide suitably conditioned air to ensure the comfort of personnel during decommissioning.

7.46. In designing a ventilation system to control airborne contamination during decommissioning, account should be taken of the following (see para. 5.74 of this Safety Guide for comparison):

(a) Mechanisms of thermal and mechanical mixing;
(b) The limited effectiveness of dilution in reducing airborne contamination;
(c) Extraction of air from areas of potential contamination at points near the source of the contamination;
(d) The use of exhaust rates that are commensurate with the potential for contamination in the area;
(e) 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 taken into consideration in designing and reassessing relevant ventilation systems.

7.47. The airflow in the ventilation system used during decommissioning should be such that it is directed from regions of lower airborne contamination levels to regions of higher contamination levels. Particular sources that might arise during decommissioning (and their specific location and characteristics) and potential changes in pressure conditions due to activities during decommissioning should be also considered in the design.

7.48. Portable ventilation systems (e.g. fans, filters, tents) should also be used in areas where airborne contamination might arise during decommissioning, and provision should be made for sufficient space in which to operate such systems. The portable ventilation systems should also be used in areas where dismantling works produce metal dust and combustion gases. Additionally, where contaminated building structures are to be demolished, specific techniques such as water spraying, and the use of local containment systems should be applied in order to reduce the exposure of decommissioning workers and the impact on the environment (see para. 8.21 of SSG-47) [34].

Waste treatment systems used during decommissioning

7.49. 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 decommissioning starts, in preparation for the final decommissioning plan, an estimate should be made of the radionuclide content in treated waste, and of the maximum radiation levels that could arise in each area of the waste treatment system. Consideration should be given to the sources that give rise to the highest radiation levels (e.g. ion exchange resins, discarded radioactive components, filter waste).

7.50. During the finalization of the decommissioning plan, the extent to which the waste treatment systems will be used should be assessed. If there are planned changes in the conditions that apply in operational states for the decommissioning stage, the design of these systems should be reassessed. Other safety related considerations (e.g. fire hazards) with radiation protection consequences should be incorporated into the assessment, and the design should be modified accordingly.

7.51. The design should be such that, during decommissioning, it is possible to perform reverse flow flushing, washing, regeneration and change of resins for the final removal of contamination by remote control.

7.52. The use of hazardous substances in the design and construction of nuclear power plants that could result in mixed hazardous and radioactive waste during decommissioning should be avoided (see para. 7.6(k) of SSG-47 [34]).

Storage of radioactive waste at a nuclear power plant during decommissioning

7.53. As stated in para. 5.93, and in accordance with para. 6.59 of SSR-2/1 (Rev. 1) [1], facilities are required to be provided for the storage of radioactive waste that arises at the plant, with account taken of its form, radionuclide content and the extent to which it has been processed. The design 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, including during decommissioning. Further recommendations are provided in SSG-40 [4].

7.54. The design of storage facilities for radioactive waste generated during decommissioning should incorporate the following functions (see also para. 5.95 of this Safety Guide for comparison):

(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 for retrieval and packaging of waste for transport off the site;
(e) Further design features against incidents and accidents.

7.55. Storage facilities should provide protection for 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 decommissioning. This should be achieved by means of design features, the selection of appropriate materials, and maintenance and repair or replacement.

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

(a) Generation of hazardous gases caused by chemical effects and the buildup of overpressure;
(b) Generation of combustible or corrosive substances.

7.57. The possibility of accidents during decommissioning should be taken into account in the design of storage facilities. Features designed for this purpose may differ from, but should be complementary to, the features designed for the facilities provided for normal operation.

7.58. Non-radiological hazards (e.g. fire or explosion), which might contribute to radiologically significant consequences, should also be considered in the design of storage facilities for the decommissioning stage.

7.59. Where appropriate, equipment should be provided with suitable interlocks or other physical measures to prevent dangerous or incompatible operations during decommissioning. Such measures should be designed to prevent undesirable movements (e.g. the movement of waste into an area occupied by site personnel).

7.60. The need for remote handling during decommissioning should be considered in cases where waste containers give rise to high dose rates or where there is a risk that radioactive aerosols or gases could be released into the working environment. The design of remote handling devices should include means for their maintenance and repair, such as the provision of a shielded service room to keep occupational exposures as low as reasonably achievable during decommissioning.

PROTECTION OF THE PUBLIC DURING DECOMMISSIONING

Authorized limits on discharges during decommissioning

7.61. As stated in para. 5.104, the operating organization is required to ensure that doses to members of the public do not exceed the dose limits and that the optimization principle is applied (see Requirement 5 of SSR-2/1 (Rev. 1) [1]). This also applies during the decommissioning stage.

Waste stream source reduction during decommissioning

7.62. The design measures taken to control sources of radioactive material in the plant during decommissioning (i.e. to protect site personnel) might also affect the activity of 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.

Effluent treatment systems during decommissioning

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

7.64. The flows and the activity concentrations of liquid and gaseous effluents during decommissioning are required to be monitored and controlled to ensure that the authorized limits on discharges are not exceeded (see para. 6.81 of SSR-2/1 (Rev. 1) [1]). Liquid and gaseous waste treatment facilities that are based on best practicable means are required to be provided during decommissioning (see Requirement 78 of SSR-2/1 (Rev. 1) [1]). GSG-10 [20] and GSG-9 [23] provide recommendations on the calculation of public exposure resulting from radioactive discharges.

Liquid waste treatment systems during decommissioning

7.65. The major sources of contaminated water that need 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; laundries and changing room showers; and chemistry laboratories. These sources produce effluents that are essentially aqueous in nature; where non-aqueous liquid waste is generated in sufficient volumes, the provision of a separate waste treatment system should be considered. Further recommendations on the treatment of aqueous and non-aqueous liquid waste are provided in SSG-40 [4].

7.66. Proven methods of treating radioactive wastewater 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 sufficient flexibility to deal with liquids of different origins and unusual compositions and to enable retreatment if the authorized limits for discharges are 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 leakage from the primary circuit to the secondary circuit in the steam generator. In this case, it may be necessary to treat the water from the secondary circuit 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 treatment systems during decommissioning. The volumes of liquid that need treatment should be reduced to as low as reasonably achievable by the careful design of the circuits that contain radioactive water, to prevent leakage and to minimize the potential for the plant to need decontamining during decommissioning. By minimizing the production of solid waste, the treatment should be appropriate for the level and type of contamination in the water. This should be achieved by segregating the waste from different sources into waste streams. Each waste stream should contain waste with similar characteristics in terms of its chemical and particulate content so that each stream can be treated optimally. 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.

Gaseous waste treatment systems during decommissioning

7.68. Discharges of radionuclides to the atmosphere are required to be below the authorized limits on discharges and to be as low as reasonably achievable (see para. 6.61 of SSR-2/1 (Rev. 1) [1]). The system for the treatment and control of gaseous waste during decommissioning should be based on the available best practices.

7.69. The gaseous waste treatment and control system should be designed to collect all the radioactive gas that is produced in the plant during decommissioning, and to provide the necessary treatment before it is discharged to the environment.

7.70. Particulate material from the treatment system for gaseous waste and from the ventilation systems should be removed using filters. It is good practice to ensure that all possibly radioactive gas discharged from the plant 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.

Shielding for decommissioning

Design of shielding for decommissioning

7.72. During the initial design of shielding and when designing the shielding for a specific radiation source for decommissioning, the target dose rate should be set, taking into account the expected frequency and duration of occupancy of the area during decommissioning. When this target dose rate is set, account should be taken of the uncertainties associated with the source term and with the analysis made to determine the expected dose rate.

7.73. Specifications for shielding for decommissioning should take account 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 design of shielding for decommissioning should be carried out iteratively, starting with the design of shielding without penetrations.

7.75. The choice of shielding materials should be made on the basis of the nature of the radiation, the shielding properties of the materials, the mechanical and other properties of the materials, and space and weight limitations during decommissioning.

7.76. Losses in shielding efficiency might 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. For example, water filled (and shielded) systems are emptied for dismantling. Effects that should be taken into account are those occurring as a result of the interactions of gamma rays with the shielding, those resulting from 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 at the time of the decommissioning should be considered.

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

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

Penetrations through the shielding during decommissioning

7.79. Consideration should be given to the necessary penetrations through the shielding that are needed for decommissioning, such as those for temporary pipes, cables and access ways, and the provisions to be made to maintain the effectiveness of the shielding. The recommendations on controlling dose rates due to penetrations, point of access by personnel and equipment provided in paras 5.62–5.65 are also applicable to decommissioning.

Shielding and barriers during decommissioning

7.80. The shielding that is incorporated into the design to protect site personnel during dismantling 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. Any temporary waste storage units located at the periphery of the site should be designed to have a low radiological impact to workers and the public.

8. DESIGN OF RADIATION MONITORING FOR OPERATIONAL STATES, ACCIDENT CONDITIONS AND DECOMMISSIONING OF NUCLEAR POWER PLANTS

8.1. The operating organization of a nuclear power plant is required to establish and implement a radiation monitoring programme (see Requirement 14 of GSR Part 3 [2]). Associated requirements for the operational aspects of such a radiation monitoring programme are established in Requirements 20–32 of GSR Part 3 [2] and in Requirement 82 of SSR-2/1 (Rev. 1) [1].

8.2. The radiation monitoring programme should include the following elements, as appropriate:

(a) Area monitoring systems within the plant;
(b) Individual monitoring of external exposure and internal exposure of workers;
(c) Monitoring of discharges;
(d) Environmental monitoring, including measurements of background radioactivity in the environment;
(e) Monitoring of other parameters important for the assessment of public exposure (e.g. environmental and meteorological conditions at the site);
(f) Process system monitoring systems within the plant.

8.3. The monitoring programme should include the quantity to be monitored, the monitoring methods and frequency, the type of data to be collected, the methodology for data collection (including the location and frequency of data collection), the necessary resolution and precision of measurements, data backup, and data processing and analysis.

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

8.5. The operating organization is required to verify 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 (see Requirement 30 of SSR-2/1 (Rev. 1) [1]).

8.6. Monitoring for purposes of radiation protection is necessary during both plant operation and decommissioning, and the recommendations in this section apply to 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 decommissioning activities. The design of the monitoring system should therefore be reviewed before each phase of decommissioning begins. Responsibilities and measures for radiation monitoring, discharge control and radiological surveys during decommissioning and after its completion should be indicated in the decommissioning plan. If the decommissioning end state involves restrictions on future use, appropriate monitoring programmes for surveillance should be established.

8.7. Installed and portable equipment for radiation measurement should be used to ensure the protection of plant personnel, the public and the environment during both plant operation and decommissioning. This protection can be 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. Measurement quantities include radiation dose rates, radiation doses and levels of radioactive material in systems and rooms within the plant and in releases of radioactive material.

8.8. Stationary monitoring systems are required to be provided to detect radioactive material in the air (see para. 6.79 of SSR-2/1 (Rev. 1) [1]). This should include the monitoring of ventilation systems at the plant. Measurements should be taken in process streams to monitor the transport of radioactive material in liquid and gas systems inside the plant (see also paras 6.47 and 6.80 of SSR-2/1 (Rev. 1) [1]). Measurements of radioactive releases are required to be made to monitor both liquid and gaseous radioactive effluents from the plant (see para. 6.81 of SSR-2/1 (Rev. 1) [1]).

8.9. Equipment and systems for performing radiation monitoring are required to be provided in the design of a nuclear power plant (see Requirement 82 of SSR-2/1 (Rev. 1) [1]). The rationale and the design basis for the measurement channels, their measuring ranges and detector locations should be documented. These items are often subject to 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. Monitoring systems important to safety should incorporate redundancy to ensure that monitoring is always possible and should have reliable and diverse data communication systems with an independent power supply that can operate autonomously in the event of a loss of power for a period consistent with the safety analyses performed (see para. 2.115 of SSG-54 [56]). In some 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.

8.10. Monitoring data on operational safety performance, including radiation doses and the generation of radioactive waste and effluents are required to be collected and assessed (see Requirement 19 of GSR Part 4 (Rev. 1) [8]). Such data should be used, as appropriate, to update the safety assessment.

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

(a) Range of dose rates or activity concentrations to be measured;
(b) Physical quantities and units displayed and archived;
(c) Device sensitivity and accuracy;
(d) Uncertainty of measurement results;
(e) Types of radiation or radionuclides to be monitored;
(f) Provision of threshold alarms;
(g) Power supply and backup power supply;
(h) Environmental conditions;
(i) Provision for testing, calibration and maintenance;
(j) Provision for functioning in all plant states, including accident conditions;
(k) Response to overload conditions;
(l) Failure mode indication;
(m) Potential for interference with or corruption of monitored data due to other radiation sources present in the area, for example in the case of monitoring for neutron radiation and tritium;
(n) Connectivity of the equipment for data transfer;
(o) Provisions for saving and accessing results.

8.12. Recommendations on instrumentation and control systems are provided in SSG-39 [33]. Technical specifications for the design of instrumentation and devices are given in the standards of the International Electrotechnical Commission and the International Organization for Standardization.

8.13. Radiation monitoring systems should be designed to maintain their operability under specified environmental conditions. The range of conditions, including temperature, pressure, humidity, vibration and ambient radiation fields, should be specified.

8.14. Continuity of power for the monitoring of the key plant parameters and for the completion of short term actions necessary for safety is required to be maintained in the event of loss of alternate power sources (see para. 6.44D of SSR-2/1 (Rev. 1) [1]).

8.15. A system that displays relevant data on measured radiation levels within the plant should be provided in the main control room, the health physics room, at appropriate local control points and in the plant process computer. Alarm signals should be provided that reflect the design goals of the radiation monitoring systems. All data relevant for radiation protection, together with the time of the measurement and the location of the measurement, should be available in real time in the emergency response facility.

AREA MONITORING SYSTEMS WITHIN A NUCLEAR POWER PLANT

8.16. The type and frequency of workplace monitoring is required to be sufficient to evaluate the radiological conditions in and around the plant (see para. 3.97 of GSR Part 3 [2]). Supporting recommendations are provided in paras 3.112–3.115 of GSG-7 [3].

8.17. Area monitoring includes the measurement of dose rates and airborne activity concentration at set locations in and around the nuclear power plant and the measurement of surface contamination on persons and equipment exiting areas. Area monitoring normally provides results in real time. However, passive detectors (e.g. thermoluminescent dosimeters) might be installed for backup and retrospective evaluation of radiological conditions. The design of area monitoring systems should be based on the anticipated radiation and contamination levels.

8.18. In controlled areas, continuously operating fixed instruments with a local alarm and an unambiguous readout are required to be installed so as to give information on radiation dose rates and airborne contamination in selected areas (see paras 6.78 and 6.79 of SSR-2/1 (Rev. 1) [1]). To ensure the habitability of the main control room in the event of a radioactive release on the site, the radiation monitoring system should monitor the air inlet to the main control room ventilation system and, as necessary, actuate the iodine and particulate filters (see paras 4.86 and 4.160–4.167 of SSG-62 [31]). For monitoring special maintenance operations of short duration, and especially for monitoring in areas where high dose rates may vary, portable dose rate meters should also be provided, with alarms to notify if pre-set values are exceeded. When designing audio alarm systems, the likely noise level in the relevant areas should be taken into account: the use of visual alarm signals should also be considered.

8.19. Surface contamination monitors should be provided for special maintenance operations, especially operations with the potential for contamination. Tools and methods for sample collection, sample preparation and laboratory measurement of samples should also be developed and available.

8.20. Radiation monitoring systems should be installed in the following locations:

(a) The reactor containment;
(b) Rooms that are adjacent to the upper part (refuelling area) of the containment;
(c) Buildings that are housing systems connected to the reactor coolant system;
(d) The spent fuel storage facility;
(e) The fuel handling machine;
(f) The treatment and storage facilities for radioactive waste;
(g) The decontamination facilities;
(h) The transport routes for fuel and waste;
(i) Areas for the handling and storage of fresh mixed oxide fuel or fresh fuel containing reprocessed uranium;
(j) The main control room and supplementary control room;
(k) Emergency response facilities;
(l) Outdoor locations in and around the site.

8.21. With regard to stationary air monitoring systems (see para. 6.79 of SSR-2/1 (Rev. 1) [1]), the activity concentration in air should (at a minimum) be determined for accessible rooms of the controlled area where airborne radioactive material might 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:

(a) The containment;
(b) The fuel storage facilities;
(c) The auxiliary building;
(d) The radioactive waste building.

8.22. In selecting air monitors, the physical form (i.e. gaseous or particulate) 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.

8.23. Provision should also be made for the monitoring of air contamination and surface contamination at the entrances and exits of areas where radiation work is to be conducted.

INDIVIDUAL MONITORING OF PERSONNEL AT A NUCLEAR POWER PLANT

8.24. Facilities are required to be provided for monitoring the exposure and contamination of operating personnel at a nuclear power plant (see para. 6.83 of SSR-2/1 (Rev. 1 [1]). Equipment for monitoring individual doses to workers should include the means necessary to measure, evaluate and record the doses received from external radiation and from internal radiation, as appropriate. The contributions of alpha, beta, gamma and neutron radiation should be taken into consideration. Such monitoring equipment and systems are typically subject to national regulatory requirements. Detailed recommendations on the monitoring and assessment of individual doses are provided in GSG-7 [3].

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

(a) Equipment for monitoring contamination of skin and clothing;
(b) Equipment for monitoring contamination of any objects or material being removed from the area.

8.26. Background radiation levels in the rooms and areas where contamination monitoring is planned should be minimized to decrease the uncertainty of the measurements (see para. 9.30 of GSG-7 [3]).

MONITORING OF DISCHARGES

8.27. Equipment is required to be provided to monitor and record all discharges of radioactive liquid and gaseous effluents to the environment (see para. 6.81 of SSR-2/1 (Rev. 1) [1]). In addition, equipment should be provided to monitor systems that may contribute significantly to radioactive releases from the plant. Design specific aspects should be taken into consideration. Monitoring of the discharges from the following systems should be provided where applicable:

(a) Plant off-gas system;
(b) Vent header of radioactive waste tanks;
(c) Building ventilation with the potential for contamination;
(d) Liquid effluents of the radioactive waste system;
(e) Condenser cooling water discharge.

8.28. Monitoring programmes are required to be designed to demonstrate that discharges are in compliance with the authorized limits (see para. 3.137 of GSR Part 3 [2]). The results of monitoring should also be used to check the assumptions used to evaluate doses to the representative person, in accordance with para. 5.75 of GSG-9 [26].

8.29. As recommended in para. 5.77 of GSG-9 [26], monitoring programmes should be developed and conducted in accordance with a graded approach.

8.30. 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. Recommendations on monitoring of effluents are provided in GSG-9 [26].

8.31. The regulatory body can make provision for independent monitoring. The characteristics of such monitoring and the resources devoted to it 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

8.32. Paragraph 6.84 of SSR-2/1 (Rev. 1) [1] states:

> "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."

Monitoring programmes should involve 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 operational states, in design basis accident conditions and in design extension conditions.

8.33. Environmental monitoring, together with discharge monitoring, should provide sufficient information to determine whether the levels of public exposure comply with the dose limits and to demonstrate that protection and safety are optimized. Environmental monitoring should be designed to verify the results of discharge monitoring and to confirm predictions of radionuclide transfer in the environment.

8.34. The environmental monitoring programme should be established at the pre-operational stage (see NS-G-3.2 [19] and GSG-10 [20]). More detailed recommendations on environmental monitoring are provided in IAEA Safety Standards Series No. RS-G-1.8, Environmental and Source Monitoring for Purposes of Radiation Protection [69].

PROCESS MONITORING

8.35. Process sampling systems and post-accident sampling systems are required to 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 (see Requirement 71 of SSR-2/1 (Rev. 1) [1]). The purpose of these measurements is to detect fuel failures and the release of radioactive material from or to a process system, for example due to leakages. The recommendations on process radiation monitoring provided in paras 4.74–4.93 of SSG-62 [31] should be taken into account.

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

8.37. 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.

8.38. 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. Special attention should be paid to alpha emitting radionuclides during radiochemical processing because they are difficult to measure and pose a higher risk in the case of internal contamination.

8.39. Auxiliary systems that might also become contaminated include:

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

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

8.40. 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 fission products that are 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. This 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.

8.41. The design is also required to include means for 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 (see para. 6.68A of SSR-2/1 (Rev. 1) [1]).

RADIATION MONITORING UNDER ACCIDENT CONDITIONS

8.42. The radiation monitoring systems at a nuclear power plant should include provisions that are relevant to postulated accidents and, to the extent necessary and practicable, they should also be operable during severe accidents. Provision should be made for portable monitoring instrumentation (for monitoring dose rates and surface and airborne contamination) with ranges that are appropriate for severe accidents. The aim should be to provide 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 accident conditions.

8.43. Requirements on emergency response arrangements are established in GSR Part 7 [11], and supporting recommendations are provided in GS-G-2.1 [66]. Further information is provided in Ref. [67].

8.44. Arrangements should be made to promptly assess releases of radioactive material, the radiological conditions on and off the site and the associated radiation doses to workers and the public. 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 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 that form the basis for classifying the emergency. The response of instrumentation and other relevant systems at the facility should be adequate for the full range of emergencies, including severe accidents, as approved by the regulatory body.

8.45. The operating organizations should be aware of the performance of the radiation monitoring systems under the environmental conditions that might 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.

8.46. An assessment should be made of all areas of the plant in which radionuclides might concentrate — and of the radioactive releases that might occur — as a result of accidents, including the radionuclide composition of the releases and the expected environmental contamination, to ensure that the design of the monitoring instrumentation is adequate to achieve its purpose. This is especially important for severe accidents, namely those that might produce dose rates of up to 10^6 Gy/h and activity concentrations of up to 10^{15} Bq/m^3.

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

8.48. 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 performed using either mobile equipment or equipment in a laboratory that is operable under accident conditions. Provision should be made in advance for the transport of mobile monitoring equipment.

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

8.50. 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 communication 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 needed. Provision should be made for the direct transfer of relevant data to the emergency response centre.

8.51. Following an accident, there should be a means for taking representative samples of both gases and liquids 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 high activity samples.

8.52. An automatic external radiation measuring network should be installed close to the site. This type of measuring system will provide the operating organizations and the emergency response organization with real-time data on environmental radiation levels. Such data 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 IN NUCLEAR POWER PLANTS

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 are to be applied below any limits or constraints established on risk or dose for a nuclear power plant. Optimization arguments are not to be used to justify levels of risk or dose above any such limits or constraints.

A.3. There is no level for risk or dose below which optimization is not required. However, if it has been demonstrated that a further reduction in risk or dose could not be achieved at a reasonable societal or economic cost, attempts for further reductions may then be considered not warranted.

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 process should result in an overall balance of safety, with account taken of regulatory requirements, 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 optimal 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 (see SSG-47 [34]), and decision aiding techniques should be used, including identification of high dose activities that should be prioritized for optimization.

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

(a) 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.

(b) Progress has been made in understanding the phenomena that determine the production and transport of radioactive material within the plant.
(c) 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 the information described in para. A.5 is available, a differential cost–benefit analysis or other appropriate methods (see Refs [70, 71]) 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 Ref. [72].

A.7. If a differential cost–benefit analysis is performed, a monetary value for the averted dose needs to be established, which might need to be approved by the regulatory body. Different values are used in different States (see Refs [73, 74]).

A.8. The results of the analyses described in this Appendix 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 doses that are under consideration.

A.9. Evolutionary designs that have been designed with account taken 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 standardization, as a wider pool of experience will inevitably provide better feedback for future improvement in safety, and this should also be considered.

A.10. 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 (see Ref. [75]).

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Annex I

SOURCES OF RADIATION AND SOURCE TERMS IN NUCLEAR POWER PLANTS AND THEIR MINIMIZATION

I–1. The accidents presented in this Annex 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.

I–2. One of the early tasks for the design is to identify all sources of radiation and potential releases of radioactive material (source term) because these can affect radiation levels throughout the plant and in its surroundings. The radiological consequences 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 in the reliability of components.

I–3. Steps for assessment of radiological consequences 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. Radiation sources are located in many places (e.g. the reactor core, reactor coolant system, fuel stores, systems for treatment and storage of radioactive wastes), the spectrum of radionuclides is very large, and the mechanisms resulting in the potential source term to the environment vary depending on the specific conditions. All these steps are strongly determined by the plant design.

I–4. A major source in a given operational regime or plant state can become a minor one in a different operational regime or plant state (and vice versa). Some isotopes that are of minor importance in terms of dose rate during operation become of major importance during decommissioning. Also, even when dealing with reactors of the same type, changes in the design could have a strong influence on the relative importance of different sources.

I–5. This Annex provides examples of different items important for radiation protection of workers and the public for different plant states and for various designs of nuclear power plant. Anticipated operational occurrences are not discussed as a special category of plant states in this Annex, since phenomenologically they are similar to design basis accidents.

OVERVIEW OF POTENTIAL SOURCES OF RADIATION IN NUCLEAR POWER PLANTS

Fission products in the nuclear fuel in the core

I–6. The main sources of radiation in a nuclear power plant are fission products and actinides produced in the nuclear fuel in the core. The inventory of fission products and other radionuclides in the reactor fuel and core depends on several factors, in particular on the following:

(a) The quantity of fissile material;
(b) The fuel power and burn-up;
(c) The neutron flux distribution in the core;
(d) The operational power history (including transients) and fuel management;
(e) Decay time after reactor shutdown.

The number of radioisotopes produced by fission is very large; however, experience from the analysis of radiological consequences indicates that consideration of about 30–40 radionuclides is generally sufficient.

I–7. Particular attention is to be paid to the use of nuclide inventory software for accidents. In general, the libraries of radionuclide analysis software are focused on normal operation or measurements in the 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–8. 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, including the number and size of cladding defects, the local power in the vicinity of the defect and the burnup of the fuel. In modern reactors, the occurrence of fuel cladding defects is extremely rare. Furthermore, the main cause of cladding defects (~80%) is interaction with small migrating objects (debris), which is considerably reduced when a filtering grid is installed in the lower part of the fuel assembly.

I–9. Radionuclides 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) as well as from the uranium content of the cladding (a few ppm). A limit for uranium contamination (‘tramp uranium’) therefore needs to be specified.

I–10. Defects in the fuel cladding can 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–11. For pressurized water reactors, a spiking phenomenon is observed for fission products during shutdown and other reactor transients. The effect of spiking is further enhanced by an increase in fuel temperature (e.g. in reactivity accidents and steam line breaks). The fission products that are accumulated in the spaces in the fuel elements (in fractures in the fuel pellets, in the gap between the fuel pellets and the cladding and in the expansion chamber) can 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–12. 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) Selecting materials to minimize the concentration of nuclides (particularly of cobalt in steel) that are known to become major sources of radiation;
(c) Providing removal systems (such as particulate filters and ion exchange resins);
(d) Minimizing the concentration of nuclides in feedwater that can be activated in the core;
(e) Providing surface treatments to reduce the adherence of corrosion products to pipes and components.

I–13. The presence of materials with a high cobalt content (which produces the activation product ^{60}Co), such as Stellite, needs to be reduced. 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 a high cobalt content needs to be minimized in components of the feedwater system that are situated after the condensate purification system. 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, the presence of these material poses another important source of corrosion products (crud) in the feedwater circuit downstream of the condensate purification system. Special attention needs to be given to the choice of heater material for the feedwater, and due consideration given to the possible installation of filters in the feedwater or core coolant return circuit close to the core inlet.

I–14. Special attention also needs to be given to the selection of materials and to the coolant chemistry, which make an important contribution to the reliability of the steam supply system for the nuclear power plant. The compatibility of materials and coolant, which is of the utmost importance to minimizing the amount of maintenance, repair and 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–15. Hardfacing materials with a lower cobalt content, which have been used in plants such as Konvoi and EPR, can be considered as a dose reduction measure in some primary circuit components of pressurized water reactors, replacing Stellite. Stellite and potential replacement materials do not have identical physical properties. Consequently, hardfacing materials for primary circuit components need to be carefully selected, including consideration of the necessary material properties as well as the consequences to occupational exposure of the material selected.

I–16. The cobalt content of stainless steel and nickel-based alloys in contact with the reactor primary coolant and/or under neutron flux needs to be as low as reasonably achievable. The maximum cobalt content is expected to be specified and strictly controlled. Other materials such as silver (e.g. in control rods and seals) and antimony (used in seals and pump bearings) that produce activation products contributing significantly to occupational exposure need to also be reduced or eliminated.

I–17. Primary circuit chemistry needs to be designed to reduce corrosion and releases from components (especially steam generator tubing) that are in contact with the primary coolant and hence to reduce the levels of corrosion products.

I–18. 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. The capacity of such filters needs to be adequate to cope with the enhanced release of corrosion products (‘crud bursts’), including where initiated by hydrogen peroxide injection, and fission products (‘spiking’) that occurs during the startup and cooldown stages.

I–19. Systems to remove corrosion products, both radioactive and non-radioactive, are to be provided for the primary coolant for all types of reactor, whether the coolant is liquid or gas.

Activation products in the reactor coolant

I–20. 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. 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 of seven seconds. In such cases, other activation products in 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 letdown 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–21. In water cooled reactors, in particular 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–22. In the case of light water reactors, activation products are 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 source pins, in the pressure vessel itself, in the water and its impurities, and in the thermal shield. In the case of gas cooled reactors, the activation products are 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-free and lithium-free chemistry can significantly reduce the tritium source term.

I–23. For fast breeder reactors with sodium as a coolant, in which 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–24. Water in the fuel storage pool is maintained at a low activity level by means of a cleanup 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–25. In the cleanup systems of water cooled and moderated reactors (e.g. light water reactors, heavy water reactors), there is an accumulation of radioactive material in filters and ion exchange resins. This consists of fission products such as iodine and caesium that have escaped to the coolant through fuel cladding defects[1] and 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, generate very high activities that need shielding. Radioactive noble gases may be formed in these filters by the decay of iodine isotopes. In heavy water reactors, photoneutrons are

[1] 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.

produced in the heavy water by the photons from ^{16}N. This source is significant in determining the shielding of the coolant circuit external to the core. In gas cooled reactors, the gas treatment system accumulates activated corrosion products, such as ^{58}Co and ^{60}Co, and fission products, such as iodine and caesium, and it becomes an important source of radiation.

Secondary coolant system

I–26. 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 reach the steam and turbine system only if leaks occur between the primary and secondary circuits. Provided that the leak rates are monitored (e.g. by measuring 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 very low.

I–27. 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 might then be introduced into the secondary system.

I–28. In fast breeder reactors, the secondary sodium coolant might become activated with ^{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 short compared with the half-lives of ^{22}Na and ^{24}Na.

Liquid waste treatment system

I–29. 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–30. The composition of liquid waste (i.e. activity concentration and solid and chemical content) varies in accordance with its origin. It is general practice to segregate and treat liquid waste in accordance with its expected composition in order to improve the treatment efficiency. The liquids in the liquid waste treatment system therefore have a wide range of activity concentration. The segregation of liquid wastes could be done in accordance with the following categories:

(a) High purity (e.g. leakage wastes from the primary circuit of pressurized water reactors during power operation);
(b) High chemical content (e.g. decontamination agents);
(c) High solid content (e.g. liquid wastes from floor drains);
(d) Liquid wastes containing detergent (e.g. liquid wastes from laundry drains and personnel showers);
(e) Liquid wastes containing oil (e.g. in gas cooled reactors, liquid wastes from floor drains from the area of the lubricating oil tank for the circulator);
(f) Liquid wastes with a very high tritium content (from pressurized heavy water reactors).

I–31. 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–32. In light water reactors, before treatment, some of the liquid wastes may have a radionuclide content as high as that of the reactor coolant (except for radionuclides with short half-lives, 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 accumulates in parts of the system such as filters, ion exchangers and evaporators.

I–33. In most cases, the accumulated radionuclides consist of activation products such as ^{60}Co, ^{58}Co, ^{51}Cr, ^{54}Mn and ^{59}Fe (depending on 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–34. 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 stay in the coolant only 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 is essential to reduce the activity in systems that may have to be opened during shutdown.[2] 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, in this case, consists 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 (e.g. in a pressurized water reactor that is operated at a low degassing rate), isotopes with long half-lives constitute the most significant fraction.

I–35. Components such as hold-up tanks, hold-up 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–36. 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 could 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 also increases the delay time of a given system by a factor of about ten. 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–37. Apart from the fuel, the following constitute the major solid radioactive wastes in terms of activity and volume that arise during operation:

(a) 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] In such plants, gases are usually removed by the purification system.

(b) Irradiated components of the fuel assembly from gas cooled reactors (in which the assemblies are dismantled at the nuclear power plant);
(c) Ion exchange resins, filter material, filter coating material, catalysts, desiccants and similar;
(d) Concentrates from evaporators, precipitates;
(e) Contaminated tools;
(f) Contaminated clothing, towels, plastic sheet, paper and similar.

I–38. 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 in solid waste.

I–39. Solid waste needs to be carefully managed to allow its volume to be minimized. However, reducing releases to the environment to very low levels results in an increase in the volume of solid waste.

Irradiated fuel

I–40. Irradiated fuel has a very high radionuclide content owing to the fission products and transuranic nuclides 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–41. For wet (pool) fuel storage and handling systems, water cleanup systems with particulate filtration and ion exchange are 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–42. 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–43. When fuel is manufactured from fresh uranium, the activity of fresh (unirradiated) fuel is low.[3] Since most of the radiation emitted by the fuel is not penetrating, it is largely absorbed by the fuel cladding. Thus, the external exposure is of minor significance.

I–44. 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 is a significant source of both neutron and gamma radiation and it needs to be shielded and contained at all times until it is inserted into the reactor. The magnitude of the neutron source term depends on the time that has elapsed since the plutonium was created, since actinides that emit neutrons are produced as the plutonium decays.

I–45. 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 needs to be shielded and contained at all times until it is inserted into the reactor.

Decontamination facilities

I–46. The radioactive material in the waste solutions consists mainly of 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 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–47. There are also other sources of radiation at nuclear power plants, such as neutron startup sources, corrosion samples, in-core and ex-core

[3] 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.

detectors, calibration sources for instruments and sources that are used for radiographic inspections.

SOURCES OF RADIATION AND SOURCE TERMS DURING NORMAL OPERATION AND DECOMMISSIONING OF A NUCLEAR POWER PLANT

I–48. The radiological consequences of normal operation and decommissioning include impacts on persons and the environment that are attributable to both direct radiation and the releases of radioactive materials from the plant. The sources of radiation include sources at the site and also discharges from the site. In the assessment of the radiological impact, all pathways of external exposure are to be assessed, including direct ionizing radiation from buildings and from transportation, from gaseous and liquid discharges from the facility, and from any deposition of radionuclides on the ground or other surfaces. Internal exposure from inhalation or ingestion of radionuclides are also to be assessed.

I–49. The most significant radionuclides 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 the isotopes of strontium and plutonium may also be important. When the reactor is at power, the fuel elements emit neutron and gamma radiation 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 small contribution to dose rates around the reactor coolant circuits because the source term is small and the deposition rate is low. However, their contribution to dose rates can 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) might be released into the coolant. In this situation, the alpha activity of the water and of the deposits might 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 proportion of the activated corrosion products deposited on the fuel are released to the coolant, and the activity concentration of the water might 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 occurs 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 (e.g. ^{58}Co in steam generators, ^{110m}Ag on cold parts of the circuits, such as the non-regenerative heat exchanger). After purification, the main contributors to the dose rates are the activated corrosion product deposits.

I–54. The neutron and gamma radiation emitted by the core represent a very intense source of radiation. The residual neutron flux outside the primary shielding is a source of activation of structural materials. It can therefore induce a buildup 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 permitted 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 radiation 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 might not be negligible when the individual dose rate target is less than 10 μSv/h. 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 to the neutron flux for a sufficiently long time. 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 hours).

I–57. In cases where there is a separate oxygen containing fluid moderator system (e.g. in a pressure tube reactor), the isotope that is the major source of radiation during reactor operation is ^{16}N. After shutdown, the radiation levels around the primary coolant system are 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 also has to be taken into account in the design of light water reactors, 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, is the major source of radiation during power operation. The sky shine effect needs to be carefully checked 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 is volatile fission products, mainly the noble gases, and volatile fission products such as iodine and caesium. During power operation, this source is 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. 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 ^{35}S in the form of carbonyl sulphide, tritium 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 radiation hazard only if inhalation or ingestion 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. If these components penetrate the shield, considerable shielding is needed 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 sodium coolant can be covered by an inert gas such as argon. The activation of the cover gas gives rise to ^{39}Ar and ^{41}Ar, which might 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 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 reduces the content of short lived isotopes in the effluent but does 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 released are ^{85}Kr and ^{14}C.

I–64. The ventilation of buildings might be a source of gaseous release and, to a lesser 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 as follows:

(a) The calandria vault gas (in pressure tube reactors).
(b) 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).
(c) Coolant gas leaking into sections that contain air in gas cooled reactors.
(d) Air that has entered the pressure vessel of a light water reactor after it has been depressurized and the water level has been lowered prior to opening the vessel.

Vents for these gases need to be located such that the radioactive material they contain is kept away from operating personnel. In the case of advanced gas cooled reactors 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–66. During the handling and storage of irradiated fuel, some radionuclides are released to the surrounding coolant. Radioactive corrosion products might 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 might flake from the surface of fuel assemblies as a result of thermal or mechanical shock. In addition, defective fuel might release fission products, of which isotopes of noble gases, iodine, caesium and strontium are the most significant.

SOURCE TERMS IN NUCLEAR POWER PLANTS IN DESIGN BASIS ACCIDENTS

I–67. In a nuclear power plant, the main source of radiation 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 might 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 might fail due to overpressurization or overheating of the cladding material. Another example of an accident in which fission products might be released from the fuel rods is an accident in handling spent fuel, which might 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 on the assessment of accidents are presented in IAEA Safety Standards Series No. SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [I–1].

I–68. 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–69. 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–70. The number of fuel cladding failures that might 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. plate-out, deposition by dousing or spraying and iodine reactions within the building) need to be assessed, following the general rules for safety analysis (see SSG-2 (Rev. 1) [I–1]). Assuming 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–71. 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 the 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 (see Refs [I–4 to I–6]).

I–72. The behaviour of radioactive material released from the containment depends on 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 a small part of the leakage from the primary containment is released to the atmosphere (so-called secondary containment bypass), while most is confined by a secondary containment with a delayed release to the environment, preferably through filters. In other designs, the 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–73. The break of a main steam line in a boiling water reactor might 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; see 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 the case of a boiling water reactor. The severity of the accident depends on the diameter of the broken pipe and the characteristics of the plant safety systems.

I–74. If the location of the steam line break is within the containment, the event sequence is similar to that for a loss of coolant accident in a pressurized water reactor, possibly with a certain fraction of the fuel cladding damage. The fission product inventory for full power operating conditions is to be conservatively assumed. The design analysis for the potential radioactive release needs to consider the time necessary for containment isolation and the efficiency of the coolant purification system.

I–75. If the location of the steam line break is outside the containment and the main steam line isolation valves near the containment close immediately to isolate the reactor, only a fraction of the radioactive material present in the steam under operating conditions is released. Condensation of steam inside the building where the break occurs and the plate-out of substances other than noble gases results in a reduction in the amount of radionuclides released to the atmosphere. Usually, the release of coolant into a building other than the containment causes such an overpressure that radioactive material is released from the building to the environment or to other buildings via predetermined release paths (air or water) through potential release points breached by the overpressure. Mixing of the steam with the air in the building may be assumed if the possible pipe break and the release points of the building are not located in the close vicinity. After the overpressure is relieved, releases to the open atmosphere are not through uncontrolled release points but via the stack through the ventilation system and filters.

I–76. In some plants, leakage control systems have been added between the main steam isolation valves to limit the release of radioactive material by this path.

I–77. The possibility of direct releases from the building to the environment 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 cannot be restored by the ventilation system or by the natural draught of the stack.

Break of a steam line in a pressurized water reactor

I–78. Initially, the break of a steam line in a pressurized water reactor releases 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 might be released to the environment following a steam line break.

I–79. 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 might increase with time owing to the effects of fission product spiking.

I–80. Depending on 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 escapes through the broken steam line, has 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–81. 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 defined in the operating limits and conditions, the activity concentration of the escaping steam could be significant. This potential is even greater if failure of the fuel cladding occurs. The significance of the release for this event is due to the following:

(a) The high activity concentration for the leakage;
(b) The break being not fully isolable;

(c) The dry out of the affected steam generator that results in no partitioning of radioactive material within the steam generator.

I–82. After shutdown, the production of steam depends on the decay heat. The moisture content of the steam is 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, has 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 depend on the design.

Steam generator tube rupture in a pressurized water reactor

I–83. 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 the 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. Steam generator tube rupture has occurred in several operating nuclear power plants.

I–84. The design basis steam generator tube rupture is usually postulated to be a double ended break in one tube but could also involve 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 releases 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–85. 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 the radioactive release. 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 is terminated when the pressures of the primary and secondary circuits have equalized. The operating personnel cool down the plant using the intact steam generator(s). However, in the event 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–86. The nature of the transient depends on the automatic safety systems and the time at which operating personnel start 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 SSG-2 (Rev. 1) [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–87. 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 a conservative (i.e. high) estimate of the activity.

I–88. The minimum elapsed time between the shutdown of a plant and the beginning of fuel handling operations needs to be used to determine the maximum inventory in the fuel rods at the start of refuelling operations. The number of fuel rods that might become defective as a result of the impact needs to be determined either by means of a conservative estimate, by the evaluation of actual occurrences with similar fuel elements, or in experiments. The fraction of the noble gas inventory 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 might be leached out mainly by water penetrating the defective fuel rod, or the main release might be of ‘gaseous’ iodine, which is assumed to be present in the free space within the fuel rod.

I–89. 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 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–90. 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). 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–91. In addition to noble gases and iodine, up to a few per cent of the caesium inventory might be slowly leached out by water that penetrates the defective rods. This caesium is in ionic form in the water, and its transfer to the air above the pool water might be neglected.

I–92. The amounts of noble gases and iodine released to the environment are 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 can be taken into account by means of an appropriate decontamination factor determined on the basis of 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 achieved by operator action, a usual time delay (e.g. 10–30 min) can be assumed (see SSG-2 (Rev. 1) [I–1]).

Accidents in auxiliary systems containing radioactive material

I–93. Examples of accidents that might occur in auxiliary systems are pipe breaks in 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 depend on 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–94. One important type of accident is caused by a crack in the pipework of a residual heat removal system that 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 occurs as a result of the shutdown or that might have occurred before the break. In the analyses of such accidents, 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 are to be determined conservatively. A method for analysing accidents of this type is described in Ref. [I–8] and supporting references.

Accidents in heavy water reactors

I–95. Reactors using heavy water (deuterium oxide) as a moderator or coolant (or both) have the potential for the same type of accidental radioactive release 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. The rupture of a pressure tube in combination with a header or pipe break is not expected to be considered in the design basis accidents. However, accidents involving failure of steam generator tubes or heat exchanger tubes do need to be analysed.

I–96. The heavy water in the operating plant contains tritium, which is the activation product of deuterium. The tritium is in oxide form (i.e. water) and is not normally a significant 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–97. 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–98. Areas of the nuclear power plant in which other postulated initiating events resulting in releases of radioactive material to the environment might occur include the following:

(a) 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);

(b) The active effluent treatment plant;
(c) The treatment and cooling plant for fuel pool water;
(d) The coolant treatment plant;
(e) The storage area for solid radioactive waste;
(f) The fuel element debris vault;
(g) Ventilation filters.

SOURCE TERMS IN NUCLEAR POWER PLANTS UNDER DESIGN EXTENSION CONDITIONS

I–99. 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 might fail to perform their safety functions, potentially leading to significant core damage. The potential exists for a large release of radioactive material to the environment, in particular during a severe accident.

I–100. In accordance with general rules for deterministic safety analysis (see SSG-2 (Rev. 1) [I–1]), the source term in design extension conditions without significant fuel degradation can be determined using a best estimate approach. This means that plant operating parameters, the characteristics of plant systems and other assumptions used in the analysis can be selected realistically or at least less conservatively than in the case of design basis accidents.

I–101. The phenomena taking place in design extension conditions without significant fuel degradation do not differ from design basis accidents. The next example 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. A severe accident associated with containment bypass would result in a large radioactive release and thus is required to be practically eliminated (see para. 2.11 of SSR-2/1 (Rev. 1) [I–9]).

I–102. Because of the significant core damage during severe accidents, such accidents are analysed in detail to determine their possible radiological consequences, which might have a significant impact on the public and the environment. Such analyses can quantify the type and magnitude of the source term that could be released to the environment. For accidents with core melting, in addition to the gap inventory, a significant portion of the total core fission product inventory is 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–103. Recommendations on severe accident analyses, including quantification of the source term, are provided in SSG-2 (Rev. 1) [I–1] and further information is given in Refs [I–2, I–10 to I–13].

DETERMINATION OF SOURCE TERMS FOR OPERATION AND DECOMMISSIONING OF NUCLEAR POWER PLANTS

Phases of release of radionuclides from the fuel

I–104. Before an accident, most fission products are contained within the fuel, from where they might be released in several subsequent phases of the accident progression. The phases of a severe accident can be distinguished as follows:

(a) Coolant activity release. During this phase, (relatively) small amounts of radioactivity in the coolant itself are released from the reactor coolant system.
(b) Gap inventory release. This phase involves the release of radioactivity from the gap between the fuel pellets and cladding, including noble gases, iodine and caesium.
(c) In-vessel release. During this phase, which is associated with gradual melting and slumping of core materials, practically all of the noble gases and significant fractions of the volatile nuclides are released from the fuel.
(d) 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 the containment. In existing designs, the molten core interacts with the reactor cavity concrete, and the less volatile nuclides are likely to be released into the 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.

(e) Late in-vessel release. Simultaneously with ex-vessel release, some of the volatile nuclides that had deposited in the reactor coolant system during the in-vessel phase are released into the containment.

Fission product release from molten fuel

I–105. The releases from molten fuel depend on many parameters, such as temperature, oxidizing–reducing conditions, interactions with structural materials, burnup, fuel material and 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–106. Fractions of fission products released from the molten fuel are strongly determined by volatility. The most volatile fission products are naturally noble gases, followed by iodine, tellurium and caesium. In the very early stages of a core melt accident release, the activities of iodine and tellurium exceed caesium by several tens of times. Frequently used release fractions are described in Ref. [I–13], which states that, in all phases of severe accidents, all noble gases, about 75% of halogens and alkali metals, about 30% of the tellurium group, about 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 publications report different values.

I–107. In addition to fission products, other radionuclides in the primary coolant are also released, including activation products in the coolant or additives, and corrosion products. However, in comparison with the activity of fission products, the activity of these other radionuclides in a severe accident is not significant.

Behaviour of fission products in the reactor coolant system

I–108. It is essential that the chemical affinities of fission products, their redistribution within the fuel release kinetics in various atmospheres and their potential for retention in the reactor coolant system are understood to determine the release into the containment. The behaviour of iodine, caesium, ruthenium and tellurium need special attention; these are volatile species and can be almost completely released from the fuel.

I–109. 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 caesium iodide (CsI). The majority of the caesium (around 90%) is expected to be released from the reactor coolant system as caesium hydroxide (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 interacts with steel, diffusing into the inner chromium oxide layer and providing further potential attenuation.

I–110. Ruthenium is considered separately from other refractory materials, due to its distinct oxidation processes. Ruthenium has low volatility under reducing and low oxidizing conditions but is rapidly released in large amounts under high oxidizing conditions, with potential release fractions measured in tens of per cent. 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 delays transport to the containment compared with iodine and caesium.

I–111. The vapour species released from the fuel condense in the primary circuit due to cooler 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 behave primarily as aerosols. A number of physicochemical processes occur that affect the retention of aerosols in the reactor coolant system and thus the source term to the environment. Aerosol deposition occurs as a result of several processes such as diffusion, thermophoresis, diffusiophoresis, electrophoresis, sedimentation and inertial impaction.

I–112. Particles deposited on the surfaces of the primary circuit may resuspend to the gas stream, decreasing retention of radionuclides in the circuit. When aerosols are deposited on a dry, uncooled surface, the energy generated by fission product decay may be capable of reheating the deposited fission products to a sufficiently high temperature such that revaporization may occur. The significance of the revaporization 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 a sudden increase in the gas flow rate.

I–113. Several other phenomena can affect the release of fission products into the containment. If the reactor coolant system is at high pressure at the time of failure of the reactor pressure vessel, molten core materials could be ejected from the reactor pressure vessel into the containment at high velocities, adding a significant amount of radioactive material to the containment atmosphere.

I–114. 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). A steam explosion could lead to fine fragmentation of some portion of the molten core debris with an increase in the amount of airborne radionuclides.

I–115. 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, retention of fission product aerosols in the reactor coolant system can have a significant effect in terms of reducing the source term to the environment.

Behaviour of fission products in the containment

I–116. The 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 natural processes or active engineered safety features (containment sprays, ventilation) to remove fission products from the containment atmosphere.

I–117. 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 radioactive decay. Several processes leading to the production and removal of aerosols from the atmosphere take place in the containment. Aerosol removal is influenced by the natural depletion processes associated with sedimentation and diffusiophoresis, and to a lesser degree by thermophoresis, and is 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–118. The behaviour of iodine needs special attention due to its different physical and chemical forms. The majority (more than 90%) of the iodine entering the containment is in the form of CsI. Caesium iodide exists primarily in the form of aerosols for typical containment temperatures and ultimately settles out from the containment atmosphere via natural deposition processes. Active fission product removal processes, such as sprays, provide rapid removal of fission products from the containment atmosphere. Once the iodine enters the containment, however, additional reactions can occur that influence its ultimate importance to public dose. Once removed from the containment atmosphere in particulate form (as CsI), the iodine dissolves in containment water pools or plates out on wet surfaces in its ionic form [I–14]. Subsequently, iodine behaviour within the containment is dependent on time and on 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 is slowly converted to elemental iodine and is re-released into the containment atmosphere as elemental iodine.

I–119. Organic iodine is also produced slowly over time from reactions of 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 is converted to elemental iodine. Elemental iodine and organic iodine compounds are gaseous and are transported from the containment in a similar manner to noble gases. Elemental iodine is soluble and can be removed from the containment atmosphere by the operation of sprays. Thus, for several delayed containment overpressure sequences, a significant quantity of CsOH can revaporize and be available for release.

I–120. The deposition and resuspension of aerosols in the containment depends on the physical and chemical properties of the aerosols and the driving hydrodynamic and thermohydraulic conditions. In the absence of sprays, deposition by gravitation is the most important retention mechanism of aerosols in the containment, and the growth of particles is an important uncertainty affecting the aerosol removal rate. The most important parameters are atmospheric relative humidity and temperature gradients on 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–121. 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, which are 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–122. If available, containment sprays may also be used 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 a diameter typically of about 1 mm. When falling, the droplets partly agglomerate. There are also evaporation and/or condensation phenomena, depending on the relative humidity in the containment. As a result, the droplet size distribution changes during the fall.

I–123. 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 of the volatile fission products into the environment. Major hydrogen explosions, potentially leading to a loss of containment integrity, are conditions to be practically eliminated (see SSR-2/1 (Rev. 1) [I–9] and SSG-2 [I–1]). The processes associated with radionuclide behaviour have been described in a number of documents, for example in Refs [I–12 to I–18].

Releases of fission products from containment

I–124. 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 different containment points of release, where local concentrations of fission products may be significantly different.

I–125. More sophisticated calculations may take into account that the releases from the containment do not occur through an ideal nozzle with a given diameter but rather through microscopic cracks in the steel liner and in the concrete containment wall, with significant retention capacity for fission products in the form of aerosols. References [I–18, I–19] suggest a decontamination factor of 15 for iodine and 100 for caesium in dry conditions and almost complete retention of aerosols in wet conditions.

I–126. At least three different groups of radionuclides with significantly different behaviour are to be considered separately in the source term: noble gases; various forms of iodine; and all other fission products typically existing as aerosols. For noble gases there is neither a physical nor a chemical change of their forms.

I–127. Iodine can substantially contribute to the radiological consequences of a release, and consideration of its various forms is essential. Iodine exists in three different forms: iodine particulate (mostly as CsI), elemental iodine and HI, and organic iodine. These different forms behave differently during their transport in the environment and also are different in terms of the radiological consequences. Elemental iodine (I_2) and organic iodides (e.g. methyl iodide (CH_3I)) are volatile species that behave similarly to gases. The passage of a radioactive plume containing these forms of iodine causes an increase in the external dose rate, which then 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 are deposited on the ground surface. This is the main cause of environmental contamination in the area around the accident site. These forms of iodine also have a higher deposition rate than particulate forms. A higher proportion of elemental and organic forms of iodine produces higher doses in the vicinity of the nuclear power plant and correspondingly smaller doses at longer distance from the plant. A difficult problem is that iodine changes its form between its release from fuel and its release from the containment.

I–128. 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, such as containment basemat failure resulting in the opening of a direct pathway to the environment, pathways of this type are usually not considered in the determination of public dose.

I–129. The release of fission products to the environment in the event of basemat melt-through or basemat penetration (i.e. loss of containment integrity) needs special attention, including the potential release to the ground, transfer into the groundwater and subsequent transport to surface waters.

Containment filtered venting

I–130. Containment venting systems may be implemented to prevent containment failure in the case of a severe accident that induces a slow containment overpressurization. Timing for activation of the containment venting system can be predicted. The possibility of hydrogen generation, containment design pressure resistance and the evacuation of the public all need to be considered. Filter venting is used to minimize environmental contamination. In the case of boiling water reactors, filter venting can be divided into dry-well venting and wet-well venting. 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 by wet-well venting.

I–131. Different means of fission product filtration can be used to limit the source term released to the environment, including venturi scrubbers, gravel scrubbers, gravel bed or sand bed filters, multi-venturi scrubbers, metal fibre filters, and jet and packing filters. 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, needs charcoal filters. Reference [I–20] describes the results of measurements of the performance of eight potential containment-venting filtration devices.

METHODS OF CALCULATION OF SOURCE TERMS FOR NUCLEAR POWER PLANTS

I–132. 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–21], which contains extensive bibliographies. Where computer codes are needed to apply the method, these are available as described in Refs [I–22, I–23].

I–133. Recommendations on the performance of 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) [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–24, I–25].

Source terms for corrosion products

I–134. 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.

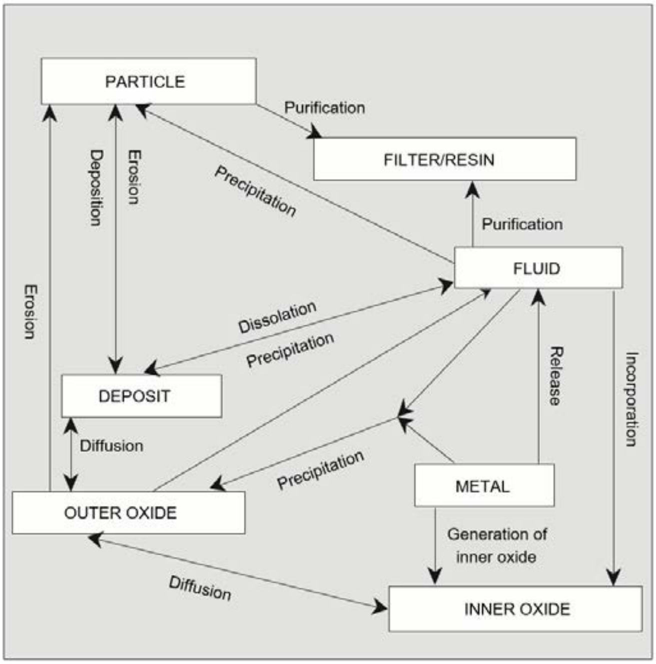

FIG. I–1. Flow chart of phenomena to be considered in modelling the behaviour of corrosion products.

I–135. 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–136. 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 in determining the behaviour of corrosion products in the primary coolant. The relevant parameters for coolant activity in pressurized water reactors are as follows:

(a) 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.

(b) The models that are used to describe the behaviour of corrosion products need to have the capability to model 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 $m \cdot s^{-1}$.
(c) The duration of one circuit (reactor → steam generator → reactor): ~10 s including ~1 s in flux.
(d) The variety of alloys: Zircaloy 4/Inconel 600, Inconel 690, Incoloy 800/Inconel 718/hardfacing materials (Stellite)/stainless steel.
(e) 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 as follows:
 — Release (average): 1 $mg \cdot dm^{-2} \cdot month^{-1}$;
 — Cycle duration: 12–18 months;
 — Area excluding Zircaloy (~ no release): 17 000 m^2;

 — ^{59}Co level (impurity): ~$5 \cdot 10^{-4}$ $g \cdot g^{-1}$;
 — ^{58}Ni in nickel-based alloys (Inconel 600, 690): ~$3 \cdot 10^{-1}$ $g \cdot g^{-1}$.

(f) The input to the reactor coolant during a ten month cycle is ~10 $g \cdot cycle^{-1}$ of ^{59}Co and ~5 $kg \cdot cycle^{-1}$ of ^{58}Ni.
(g) Wear of hardfacing materials (e.g. of parts in the internal structures of the core, pump bearings, valves, control rod drive mechanisms) is in addition to the figure for ^{59}Co.
(h) 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.

I–137. Some of the important phenomena that affect the source term for corrosion products are as follows:

(a) 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.
(b) 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–138. The models that are used need to be appropriate for the properties of the system that is being addressed.

I–139. Examples of other factors that need to be modelled include the following:

(a) When the concentration of oxides in the primary coolant is very low (in a pressurized water reactor it is typically a few 10^{-9} $g \cdot g^{-1}$);
(b) When the release of elements from alloys is not proportional to their composition;
(c) When chemical conditions vary throughout the fuel cycle within a specified range;
(d) When bulk coolant and surface temperatures need to be taken into account;
(e) When wear by friction is significant.

I–140. 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. More specifically, 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–141. 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 depends on how all of 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 of the other factors that have been mentioned. Collecting the most accurate operating experience involves taking regular measurements throughout the lifetime of the plant at exactly the same locations, including during transients such as reactor shutdowns.

I–142. 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 transitioning 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–143. For evaluating the source terms for the purposes of modifying or decommissioning a plant, the most useful information is provided by the results of the latest 100 measurements that have been made at the same plant at all the relevant locations.

Source terms for fission products

I–144. The usual approach to determining source terms for fission products is as follows:

(a) Calculating the inventory of fission products in the fuel — several well-known computer codes are available to perform this evaluation (e.g. the ORIGEN, FISPIN, APOLLO and THALES2 codes in the United States of America, United Kingdom, France and Japan, respectively).
(b) Determining the activity of radionuclides in the gas gaps in the fuel elements.
(c) Determining the total activity of the radionuclides released to the coolant through cladding defects.
(d) Determining 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.

I–145. A description of all of the 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 given in Ref. [I–26], and information on deterministic safety analysis of severe accidents is given in Ref. [I–12].

I–146. 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, CsI and 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–27].

I–147. A significant fraction of gaseous iodine has been found to migrate from the reactor cooling system to the containment when boron carbide (B_4C) control material is included in the core. In addition, monitoring after the Fukushima Daiichi nuclear power plant accident showed a significant amount of gaseous iodine along with particulate iodine [I–28].

I–148. 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–149. More accurate results for fission product releases are obtained with modern codes 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–150. 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 3–10 depending on the radionuclide. 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–151. An alternative method is to use reasonably bounding values derived from operating experience at relevant plants. The factors that determine the relevance of other operating plants include the design of the fuel elements and the rating and burnup of the fuel.

I–152. During power transients, fission products can be released to the coolant in a short time period through 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. Reference [I–7] presents a correlation of the release and the duration with the pre-transient coolant activity.

I–153. 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 I

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Annex II

EXAMPLES OF ZONING OF A NUCLEAR POWER PLANT FOR DESIGN PURPOSES

II–1. An example of radiation zoning in a nuclear power plant that may be used for design purposes is shown in Table II–1 [II–1].

II–2. An example of zoning that addresses dose rates and surface and airborne contamination levels in Swedish nuclear power plants is shown in Table II–2 [II–2]. Useful information can be found also in Ref. [II–3].

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	1 000	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.

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