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IAEA Safety Standards

for protecting people and the environment

Radiation Protection Aspects of Design for Nuclear Power Plants

Safety Guide

No. NS-G-1.13

IAEA

International Atomic Energy Agency

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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.

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Safety standards are coded according to their coverage: nuclear safety (NS), radiation safety (RS), transport safety (TS), waste safety (WS) and general safety (GS).

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 P.O. Box 100, A-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 e-mail to Official.Mail@iaea.org.

OTHER SAFETY 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.

Reports on safety and protection in nuclear activities are issued in other publications series, in particular the **Safety Reports Series**. Safety Reports provide practical examples and detailed methods that can be used in support of the safety standards. Other IAEA series of safety related publications are the **Provision for the Application of Safety Standards Series**, the **Radiological Assessment Reports Series** and the International Nuclear Safety Group's **INSAG Series**. The IAEA also issues reports on radiological accidents and other special publications.

Safety related publications are also issued in the **Technical Reports Series**, the **IAEA-TECDOC Series**, the **Training Course Series** and the **IAEA Services Series**, and as **Practical Radiation Safety Manuals** and **Practical Radiation Technical Manuals**. Security related publications are issued in the **IAEA Nuclear Security Series**.

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. NS-G-1.13

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FOREWORD

by Mohamed ElBaradei
Director General

The IAEA's Statute authorizes the Agency to establish safety standards to protect health and minimize danger to life and property — standards which the IAEA must use in its own operations, and which a State can apply by means of its regulatory provisions for nuclear and radiation safety. A comprehensive body of safety standards under regular review, together with the IAEA's assistance in their application, has become a key element in a global safety regime.

In the mid-1990s, a major overhaul of the IAEA's safety standards programme was initiated, with a revised oversight committee structure and a systematic approach to updating the entire corpus of standards. The new standards that have resulted are of a high calibre and reflect best practices in Member States. With the assistance of the Commission on Safety Standards, the IAEA is working to promote the global acceptance and use of its safety standards.

Safety standards are only effective, however, if they are properly applied in practice. The IAEA's safety services — which range in scope from engineering safety, operational safety, and radiation, transport and waste safety to regulatory matters and safety culture in organizations — assist Member States in applying the standards and appraise their effectiveness. These safety services enable valuable insights to be shared and I continue to urge all Member States to make use of them.

Regulating nuclear and radiation safety is a national responsibility, and many Member States have decided to adopt the IAEA's safety standards for use in their national regulations. For the Contracting Parties to the various international safety conventions, IAEA standards provide a consistent, reliable means of ensuring the effective fulfilment of obligations under the conventions. The standards are also applied by designers, manufacturers and operators around the world to enhance nuclear and radiation safety in power generation, medicine, industry, agriculture, research and education.

The IAEA takes seriously the enduring challenge for users and regulators everywhere: that of ensuring a high level of safety in the use of nuclear materials and radiation sources around the world. Their continuing utilization for the benefit of humankind must be managed in a safe manner, and the IAEA safety standards are designed to facilitate the achievement of that goal.

IAEA SAFETY STANDARDS

SAFETY THROUGH INTERNATIONAL STANDARDS

While safety is a national responsibility, international standards and approaches to safety promote consistency, help to provide assurance that nuclear and radiation related technologies are used safely, and facilitate international technical cooperation and trade.

The standards also provide support for States in meeting their international obligations. One general international obligation is that a State must not pursue activities that cause damage in another State. More specific obligations on Contracting States are set out in international safety related conventions. The internationally agreed IAEA safety standards provide the basis for States to demonstrate that they are meeting these obligations.

THE IAEA STANDARDS

The IAEA safety standards have a status derived from the IAEA's Statute, which authorizes the Agency to establish standards of safety for nuclear and radiation related facilities and activities and to provide for their application.

The safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment.

They are issued in the IAEA Safety Standards Series, which has three categories:

Safety Fundamentals

— Presenting the objectives, concepts and principles of protection and safety and providing the basis for the safety requirements.

Safety Requirements

— Establishing the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements, which are expressed as 'shall' statements, are governed by the objectives, concepts and principles of the Safety Fundamentals. If they are not met, measures must be taken to reach or restore the required level of safety. The Safety Requirements use regulatory language to enable them to be incorporated into national laws and regulations.

Safety Guides

— Providing recommendations and guidance on how to comply with the Safety Requirements. Recommendations in the Safety Guides are expressed as 'should' statements. It is recommended to take the measures stated 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. Each Safety Requirements publication is supplemented by a number of Safety Guides, which can be used in developing national regulatory guides.

The IAEA safety standards need to be complemented by industry standards and must be implemented within appropriate national regulatory infrastructures to be fully effective. The IAEA produces a wide range of technical publications to help States in developing these national standards and infrastructures.

MAIN USERS OF THE STANDARDS

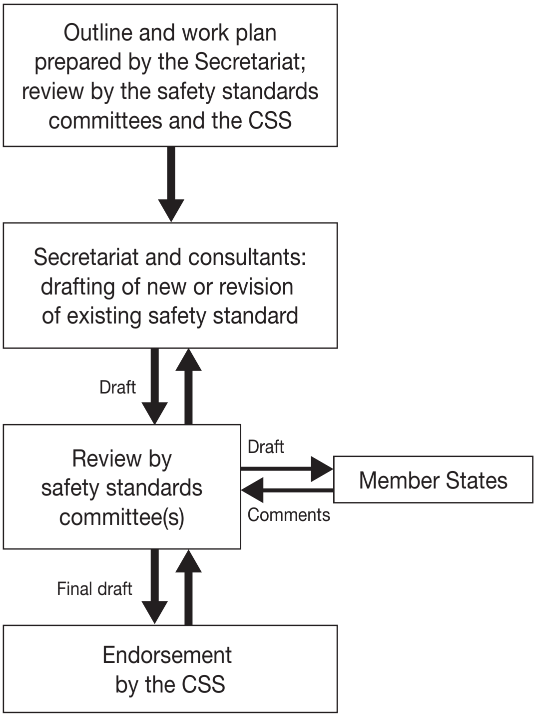
As well as by regulatory bodies and governmental departments, authorities and agencies, the standards are used by authorities and operating organizations in the nuclear industry; by organizations that design, manufacture and apply nuclear and radiation related technologies, including operating organizations of facilities of various types; by users and others involved with radiation and radioactive material in medicine, industry, agriculture, research and education; and by engineers, scientists, technicians and other specialists. The standards are used by the IAEA itself in its safety reviews and for developing education and training courses.

DEVELOPMENT PROCESS FOR THE STANDARDS

The preparation and review of safety standards involves the IAEA Secretariat and four safety standards committees for safety in the areas of 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 entire safety standards programme. All IAEA Member States may nominate experts for the safety standards committees and may provide comments on draft standards. The membership of the CSS is appointed by the Director General and includes senior government officials having responsibility for establishing national standards.

For Safety Fundamentals and Safety Requirements, the drafts endorsed by the Commission are submitted to the IAEA Board of Governors for approval for publication. Safety Guides are published on the approval of the Director General.

Through this process the standards come to represent a consensus view of the IAEA's Member States. 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 standards. Some 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 International



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

Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization.

The safety standards are kept up to date: five years after publication they are reviewed to determine whether revision is necessary.

APPLICATION AND SCOPE OF THE STANDARDS

The IAEA Statute makes the safety standards binding on the IAEA in relation to its own operations and on States in relation to operations assisted by the IAEA. Any State wishing to enter into an agreement with the IAEA concerning any form of Agency assistance is required to comply with the requirements of the safety standards that pertain to the activities covered by the agreement.

International conventions also contain similar requirements to those in the safety standards, and make them binding on contracting parties. The Safety Fundamentals were used as the basis for the development of the Convention on Nuclear Safety and the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. The Safety

Requirements on Preparedness and Response for a Nuclear or Radiological Emergency reflect the obligations on States under the Convention on Early Notification of a Nuclear Accident and the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency.

The safety standards, incorporated into national legislation and regulations and supplemented by international conventions and detailed national requirements, establish a basis for protecting people and the environment. However, there will also be special aspects of safety that need to be assessed case by case at the national level. For example, many of the safety standards, particularly those addressing planning or design aspects of safety, are intended to apply primarily to new facilities and activities. The requirements and recommendations specified in the IAEA safety standards might not be fully met at some facilities built to earlier standards. The way in which the safety standards are to be applied to such facilities is a decision for individual States.

INTERPRETATION OF THE TEXT

The safety standards use the form ‘shall’ in establishing international consensus requirements, responsibilities and obligations. Many requirements are not addressed to a specific party, the implication being that the appropriate party or parties should be responsible for fulfilling them. Recommendations are expressed as ‘should’ statements, indicating an international consensus that it is necessary to take the measures recommended (or equivalent alternative measures) for complying with the requirements.

Safety related terms are to be interpreted as stated in the IAEA Safety Glossary (http://www-ns.iaea.org/standards/safety-glossary.htm). 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 within the 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 main text (e.g. material that is subsidiary to or separate from the main text, is included in support of statements in the main text, or describes methods of calculation, experimental 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 standard. Material in an appendix has the same status as the main 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. An annex is not an integral part of the main text. Annex material published by the IAEA is not necessarily issued under its authorship; material published in standards that is under other authorship may be presented in annexes. Extraneous material presented in annexes is excerpted and adapted as necessary to be generally useful.

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide has been prepared as a part of the IAEA programme on safety standards for nuclear power plants.

1.2. This Safety Guide includes recommendations on how to satisfy the requirements established in paras 4.9–4.13, 5.61, 6.32, 6.87, 6.92–6.94 and 6.99–6.106 of the Safety Requirements publication on the Safety of Nuclear Power Plants: Design [1]. It addresses the provisions that should be made in the design of nuclear power plants in order to protect site personnel, the public and the environment against radiological hazards for operational states, decommissioning and accident conditions.

1.3. The recommendations on radiation protection provided in this Safety Guide are consistent with the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (BSS) [2], which were jointly sponsored by the Food and Agriculture Organization of the United Nations (FAO), the IAEA, the International Labour Organisation (ILO), the OECD Nuclear Energy Agency (OECD/NEA), the Pan American Health Organization (PAHO) and the World Health Organization (WHO).

1.4. This Safety Guide supersedes Safety Series No. 50-SG-D9, Design Aspects of Radiation Protection for Nuclear Power Plants, published in 1985.

1.5. Effective radiation protection is a combination of good design, high quality construction and proper operation. Procedures that address the radiation protection aspects of operation are covered in the Safety Guide on Radiation Protection and Radioactive Waste Management in the Operation of Nuclear Power Plants [3].

OBJECTIVE

1.6. The purpose of this Safety Guide is to provide recommendations for ensuring radiation protection in (1) the design of new nuclear power plants, (2) design modifications to operating plants, and (3) safety reviews of operating plants. They are provided to assist in meeting the requirements established in

Ref. [1] for which the first of three fundamental safety objectives is to protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defences against radiological hazards. This Safety Guide is for use by regulatory bodies[1] and by personnel of operating organizations and contractor organizations, including plant operators who are involved in planning, managing and carrying out the design and design modification of nuclear power plants. It can also be used for conducting safety reviews of operating plants.

SCOPE

1.7. This Safety Guide:

(1) Describes the applicable requirements of the system of dose limitation and optimization as a basis for the radiation protection measures that should be implemented in the design of nuclear power plants;
(2) Describes the measures to be taken in the design for the radiation protection of site personnel and the public;
(3) Outlines the methods that are used to calculate on-site and off-site radiation levels and to verify that the design provides an adequate level of radiation protection;
(4) Describes in annexes the important sources of radiation and contamination against which protection for site personnel, the public and the environment has to be provided in the design.

1.8. In addition to the measures that are required to protect site personnel and members of the public when the plant is in operational states and during decommissioning, this Safety Guide also deals with accident conditions, including severe accidents.[2]

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[1] Throughout this publication, the term 'regulatory body' is used to mean an authority or a system of authorities designated by the government of a State as having legal authority for conducting the regulatory process, including issuing authorizations, and thereby regulating nuclear, radiation, radioactive waste and transport safety. Earlier safety standards used the term 'Regulatory Authority'

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

1.9. Although the majority of the new designs for nuclear power plants are for water cooled reactors, this Safety Guide also deals with other types of operating reactors, and it is relevant to design issues associated with modifications to existing plants and their decommissioning.

1.10. This Safety Guide addresses radiation protection aspects of the handling, treatment and storage of radioactive waste. It does not specifically deal with the safety aspects of waste treatment relating to the form or quality of the waste product with regard to its longer term storage or disposal. These aspects are considered in a number of other safety standards [4–6].

STRUCTURE

1.11. Section 2 of the Safety Guide introduces the relevant requirements, such as those in respect of dose limits, the application of the principle of optimization of protection and the setting of design targets. Design approaches for operational states, decommissioning and accident conditions are described in Section 3, while Section 4 deals with the design features that protect site personnel in operational states and decommissioning. Section 5 covers discharge criteria, source reduction and systems for protecting the public in operational states and decommissioning. Sections 6 and 7 provide guidance on estimating radiation dose rates and on monitoring for the purposes of radiation protection under the same conditions. Guidance on radiation monitoring for processes and on auxiliary facilities is given in Sections 8 and 9. Section 10 deals with the principles of design for the protection of personnel at the site from radiation that might result from accident conditions and Section 11 covers radiation protection of the public under accident conditions. Guidance on the radiation monitoring system for accident conditions is given in Section 12.

1.12. Annexes I–III provide information on sources of radiation during normal operation and decommissioning as well as under accident conditions, while Annex IV deals with the determination of source terms for operational states and decommissioning. Annex V gives examples of zoning that may be used for design purposes.

2. SAFETY OBJECTIVES, DOSE LIMITATION AND OPTIMIZATION

SAFETY OBJECTIVES

2.1. In accordance with the principles of radiation protection, provisions are required to be made in the design to comply with the Radiation Protection Objective as given in para. 2.4 of Safety of Nuclear Power Plants: Design [1]:

> "To ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accidents".

Furthermore, provisions are required to be made in the design to comply with the following part of the Technical Safety Objective as outlined in para. 2.5 of Ref. [1]:

> "To take all reasonably practicable measures… to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits…"

AUTHORIZED DOSE LIMITS AND DOSE CONSTRAINTS[3] FOR OPERATIONAL STATES AND DECOMMISSIONING

2.2. The design of the nuclear power plant should be such as to ensure that authorized dose limits[4] and dose constraints for site personnel and the public will not be exceeded over specified periods (e.g. monthly, quarterly or annually) in operational states (normal operation and anticipated operational

[3] For internal exposures, such as those that result from the inhalation and ingestion of radioactive substances, the dose limits apply to the committed dose.

[4] An authorized dose limit or dose constraint is one that has been established or formally accepted by a regulatory body.

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

2.3. The authorized annual dose constraints for members of the public apply to the average dose to the critical groups of the population; that is, groups of persons that are reasonably homogeneous with respect to their exposure for a given radiation source and given exposure pathway, and are representative of individuals who receive the highest dose as a result of the nuclear operations that are being undertaken [8]. A critical group may be specific with respect to age or gender [9]. Pre-operational studies should be carried out to identify the critical groups and critical pathways for the exposure of such groups. Discharge limits for specific radionuclides in liquid and gaseous effluents (e.g. annual, quarterly, monthly, daily — the shorter periods permit increased release rates over short time periods and thus increased operational flexibility) should be derived from the application of the authorized dose constraints for members of the critical groups using the approved critical exposure pathways for all relevant nuclear operations. The discharge limits should ensure that the maximum individual dose for a critical group does not exceed the dose constraint.

APPLICATION OF THE OPTIMIZATION PRINCIPLE

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

— The radiation exposure should be reduced by means of radiation protection measures to values such that further expenditure for design, construction and operation would not be warranted (economic factors) by the associated reduction in radiation exposure.
— Issues such as reducing major disparities in the occupational doses received by workers of different types who work within the controlled area and avoiding arduous working conditions in radiation areas (social factors) should be taken into account in the design. The types of worker who could potentially receive the highest doses include maintenance and inspection personnel and health physics staff.

2.5. In general the optimization of radiation protection implies a choice from a set of protective measures such as shielding, remote operation and tooling to minimize radiation exposure time. To this end, feasible options should be identified, criteria for comparison and appropriate values for them should be determined and, finally, the options should be evaluated and compared. Details of different structured approaches to making decisions are given in Annex I.

2.6. 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. However, the techniques that are needed to take account of the probabilities of such events and their consequences are not yet well developed.

DESIGN TARGETS FOR OPERATIONAL STATES

2.7. To ensure that a design both reduces doses to levels that are as low as reasonably achievable and represents best practice, design targets should be set for the individual dose and collective dose to workers and for the individual dose to those members of the public who will receive the greatest doses. The setting of design targets for individual doses to site personnel and members of the public is consistent with the concept of dose constraints, which is discussed in paras 2.24 and 2.26 of the BSS [2]. The design targets should be set at an appropriate fraction of the dose limits.[5] The term ‘target’ or ‘target dose’ is used throughout this Safety Guide in respect of both individual and collective doses.

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

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

contributions to the dose and to identify aspects that contribute more to the dose than was envisaged initially.

2.9. The design target for the long term collective dose should preferably be expressed in terms of man Sv/unit of electricity generation, indicating the ratio of the radiation detriment to the benefit (the energy produced).

DESIGN TARGETS FOR ACCIDENTS

2.10. The adequacy of the design provisions for the protection of the site personnel and public under postulated accident conditions should be judged by means of the comparison of calculated doses with the specified dose criteria that constitute the design targets for accidents. In general, the higher the probability of the accident condition, the lower the specified design target should be. The regulatory body may recognize this principle by setting different design targets for accidents with different probabilities of occurrence. In addition, the regulatory body may define design targets by specifying frequency criteria for all accidents in specified dose bands. For design basis accidents, it is required that there is only a minor radiological impact outside the site boundary or the exclusion area, depending on the national regulatory requirements. The definition of minor radiological impact may be specified by the regulatory body. Typically it corresponds to very restrictive dose levels so as to preclude the need for evacuation.

2.11. It is beneficial to address separately:

— Design basis accidents (DBAs);
— Beyond DBAs (including severe accidents).

For severe accidents, the regulatory body may specify a risk criterion or a criterion associated with specified releases of radioactive substances.

3. RADIATION PROTECTION ASPECTS IN DESIGN

SOURCES OF RADIATION

3.1. The magnitudes and locations of the sources of radiation in operational states and during decommissioning should be determined in the design phase. Annex II briefly describes the main sources that cause radiation exposure in normal operation and during decommissioning. They are: the reactor core and vessel; the reactor coolant and fluid moderator system; the steam and turbine system; the waste treatment systems; irradiated fuel; the storage of new fuel; decontamination facilities; and miscellaneous sources such as sealed sources that are used for non-destructive testing. The largest sources are the reactor core, irradiated fuel and spent resins, and the design should therefore be such as to ensure that personnel are not exposed to direct radiation from these sources.

3.2. The magnitudes, locations, possible transport mechanisms and transport routes of the sources of potential radiation exposure under accident conditions should also be determined in the design phase of the plant. Guidance on the safety analysis to be carried out during the development of the design and for the final assessment is given in Ref. [10].

3.3. The main source of radiation under accident conditions for which precautionary design measures should be 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. In Annex III examples of methods for assessing radiation sources for selected accidents are described. The scenarios are selected for illustrative purposes and cover all the major categories of designs for nuclear power plants with LWRs, CO_2 cooled reactors with UO_2 metal clad fuel, HWRs and reactors with on-load refuelling.

DESIGN APPROACH FOR OPERATIONAL STATES AND DECOMMISSIONING

Human resources

3.4. The design team should be fully aware of the radiological protection measures that should be incorporated into the design.[6] A key issue is that design organizations should invite experts from relevant operating organizations to participate in activities in relation to the design of new plants and design modifications to an existing plant, to assist in ensuring that the requirements for radiation protection and waste management are met. Moreover, the applicable operating experience should be transferred to the design organization. In this way the interrelation between design aspects and operational procedures can be properly taken into account.

3.5. The optimization of protection and safety should be carried out at all stages of the lifetime of equipment and installations, from design and construction to operation and decommissioning. A structured approach should be taken to the radiation protection programme and the radioactive waste management programme to ensure the coherent application of the optimization principle in the operational stage of the plant [3].

3.6. In order to implement this structured approach, the design organization should have an optimization culture[7] in which the importance of radiation protection is recognized at each stage of the design. An optimization culture is established by ensuring that all participants in a project are aware of the general requirements for ensuring radiation protection and of the direct and indirect impact of their individual activities or functions on the provision of radiation protection for site personnel and members of the public.

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

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

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

— Knowledge of the practices that result in the occupational exposure of site personnel and members of the public;
— Achieving good feedback of operating experience to the design team;
— Familiarity with the main factors that influence individual and collective doses;
— Familiarity with the analytical methods that are available to assist in the optimization of the design;
— Recognition that specialists in radiation protection are to be consulted whenever necessary to ensure that aspects that will have implications for radiation protection are properly evaluated and taken into account in the design.

3.8. Specialists in radiation protection should be closely involved in the design process because of their:

— Expertise in all areas that affect the production of radioactive material and its transport on the plant and its transport in the environment;
— Ability to evaluate the different sources of radiation in the plant and the resulting doses using the best available analytical methods and data from relevant operating experience;
— Familiarity with the relevant regulations, guidance and best practices;
— Familiarity with maintenance, in-service inspection and other work in high radiation areas that make a major contribution to the radiation exposure of site personnel.

3.9. Because chemical parameters are very important in controlling the radioactive sources in the plant, specialists in radiochemistry should also be involved in the design process. Materials specialists should be involved in controlling the source term due to corrosion products.

Organizational aspects

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

— Rules or prescriptions for the layout and design of the plant;
— Written policies on such issues as the optimum use of respiratory protection;
— Checklists for use by engineers that can be reviewed by radiation protection specialists.

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

— Radiation protection specialists within the design organization should be consulted at the early stages of the design when options for the major aspects of the design are being evaluated. It may also be appropriate to consult with specialists from external organizations.
— The design should incorporate good engineering practices that operating experience has shown to be effective in reducing exposure; deviations from such practices should be accepted only when a net benefit has been demonstrated.
— Radiation protection specialists should review all decisions that may have a major influence on exposures.
— There should be an appropriate forum for proposing improvements and resolving disputes that may occur between design engineers and radiation protection specialists.

3.12. A systematic and structured quality assurance (QA) programme should be applied in the entire design process as required by Ref. [11].[8]

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

[8] The IAEA is revising the requirements and guidance in the area of quality assurance as established in Safety Series No. 50-C/SG-Q (1996) in new safety standards on management systems for the safety of nuclear facilities and activities involving the use of ionizing radiation. The term ‘management system’ has been adopted in the revised standards instead of the terms ‘quality assurance’ and ‘quality assurance programme’. The new standards will integrate all aspects of managing a nuclear facility, including the safety, health, environmental and quality requirements, into one coherent system.

Design strategy

General approach

3.14. As discussed in paras 2.7–2.8, design targets should be set at the start of the design process, and should include:

— Annual collective dose targets and individual dose targets for site personnel;
— Annual individual dose targets for members of the public.

The methods used for calculating doses should be subject to the approval of the regulatory body.

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

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

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

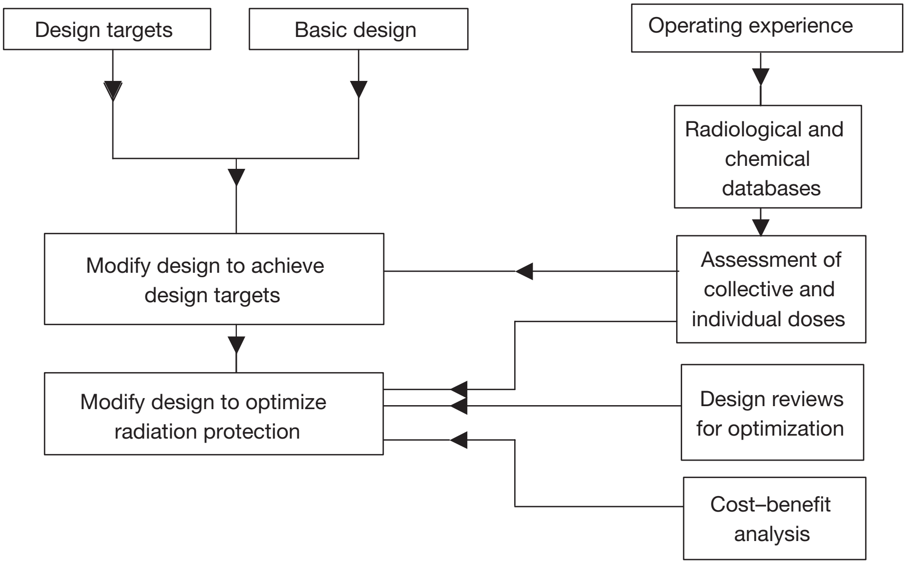

FIG. 1. Strategy for the optimization of radiation protection in the design of a nuclear facility.

Radiation protection design for site personnel

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

(1) A strategy for controlling exposures should be developed so that the most important aspects are considered early in the design and in a logical order. For example, in many types of reactor design, two areas in which there is a major potential for reducing exposures are scheduled and unscheduled maintenance. In some designs of PWRs, two of the plant items that are important contributors to exposures are the steam generators and valves. These should therefore be considered first and it should be ensured that the reliability of the design has been proven. This will reduce exposures to levels that are as low as reasonably achievable and will also help to improve the efficiency and therefore the economic performance of the plant.
The second area that should be considered is design features that minimize the production and buildup of radionuclides, since reducing these will reduce radiation and contamination levels throughout the

plant, whereas a local solution, such as increasing the shielding or improving ventilation, will have only a local benefit. Subsequently, local plant features should be considered, such as the plant layout, the shielding and the design of systems and components. An example of a simplified strategy for a PWR is shown in Fig. 2.

(2) The general requirements for the plant should be developed and documented. These should include the principles on which the layout of the plant will be based and restrictions on the use of specific materials in the design of the plant. These documents form part of the QA process for the design [11].

(3) A logical layout for the plant should be developed that is divided into zones based on predicted dose rates and contamination levels, access requirements and specific requirements such as the need to separate safety trains.[9] The dose rates may be calculated by using the source terms that are the basis for the radiation protection aspects of design (see Annexes II and IV), or they may be based on operating experience from similar plants provided that any changes in the relevant design and operating parameters are not significant. The zoning should be consistent with national legislation and regulatory requirements. It may be adequate to use the same definition of the zones as will be used when the plant is operational but it is found in many cases that a more specific definition of the zones is necessary for design purposes. Examples are given in Annex V.

(4) The maintenance programme and operational tasks should be defined, preferably on the basis of well established concepts. The number of staff for each task should be based only on the operational requirements and should not be artificially increased to comply with the regulatory requirements 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 is also identified. Types of worker include maintenance personnel, in-service inspection personnel, electrical staff, support staff (e.g. scaffolders), decontamination staff and health physics staff.

[9] 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.

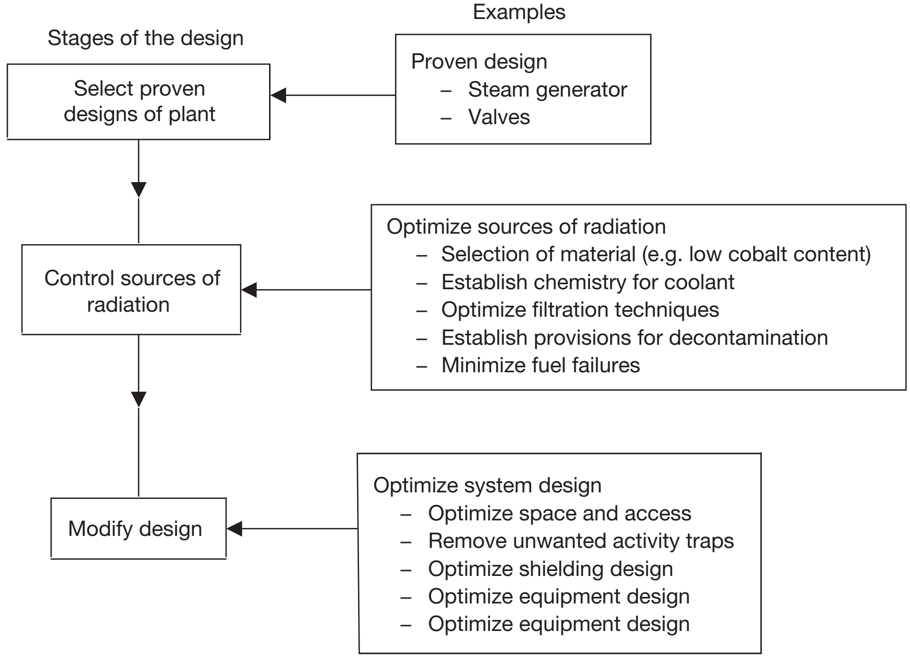

FIG. 2. A simplified strategy for the reduction of exposures in the (dashed of a PWR).

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

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

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

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

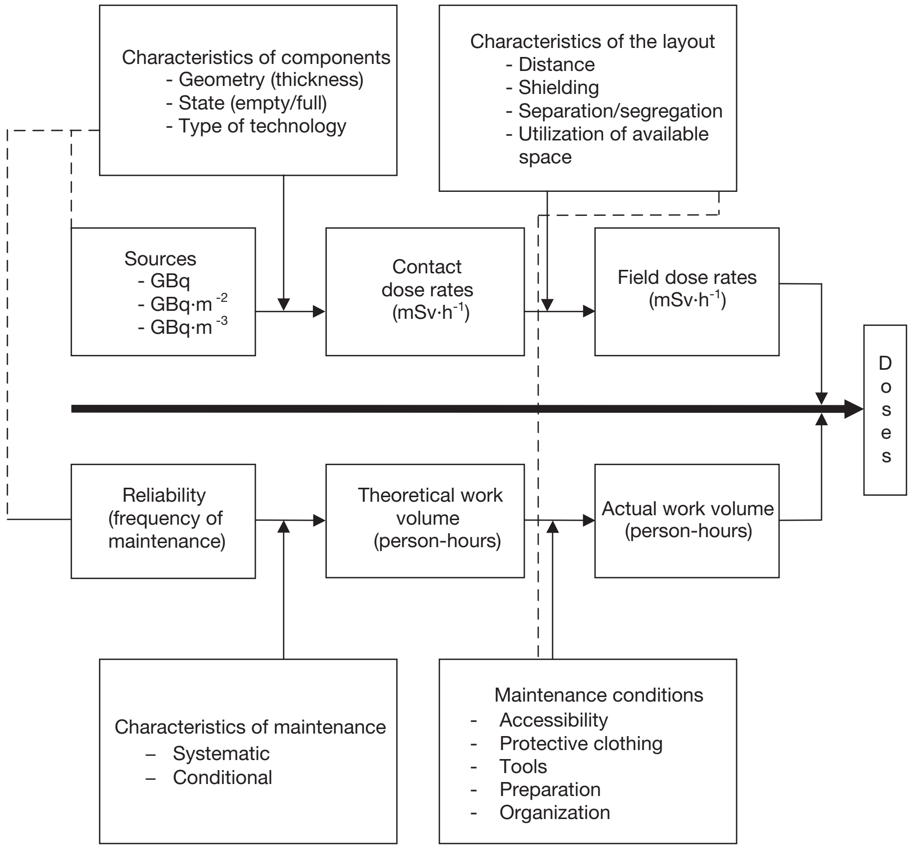

FIG. 3. Schematic flow chart of the origin of doses at a plant (dashed lines indicate the possible impact of certain blocks on others).

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

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

3.22. A preliminary decommissioning plan should be developed to ensure that the design includes the necessary features to reduce and control exposures

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

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

[a] Steps: The design of a complex project for which studies extend for several years is commonly divided into steps. The level of detail in the studies increases with the step number.

[b] Item: The line of the table indicates the main parameters that are to be considered.

[c] CDR: contact dose rate.

[d] EWV: exposed work volume.

[e] The information given in this line of the table is as follows: during step 1, an average dose constraint will be set (all craft/trade types included) as well as a collective dose target, including a margin; the optimization studies will result in a list of advantages and drawbacks of options; no zoning will be performed or contact dose rate calculations made; the exposed work volume will be estimated, with account taken of different options (the work is performed by workers or by robots).

[f] AST: accident source term; DST: design source term; RST: realistic source term.

[g] ‘Craft’ may be termed ‘trade’ in some States.

during decommissioning. In many cases, these features are the same as those necessary for operational states, but some additional special features may be necessary for decommissioning. If these are major, the necessary features for operational states and those for decommissioning should be optimized.

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

(1) Reduction of dose rates in working areas by:
- — Reduction of sources (e.g. by the appropriate selection of materials; decontamination measures; the control of corrosion, water chemistry, filtration and purification; and the exclusion of foreign material from the primary systems);
- — Improvement of shielding;
- — Increase in the distance between workers and sources (e.g. by the use of remote handling);
- — Improvement of ventilation in PHWRs.

(2) Reduction of occupancy times in radiation fields by:
- — Specifying high standards of equipment to ensure very low failure rates;
- — Ensuring ease of maintenance and removal of equipment;
- — Removing the necessity for some operational tasks by, for example, providing built-in auxiliary equipment and making provision in the design for permanent access;
- — Ensuring ease of access and good lighting.

Design for radiation protection for members of the public

3.24. As indicated in para. 2.7, design targets should be set at the start of the design process for annual individual doses to members of the public. Developments in the area surrounding the site and likely future population distributions should be taken into account as necessary.

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

- — Site specific features that affect the doses to members of the public should be identified at an early stage of the design process and taken into account in the design [12]. This should include the identification of the critical groups and the exposure pathways for these groups, which should be subject to the approval of the regulatory body.

— One possible approach would be to set targets for radioactive releases for which account is taken of operational experience and the use of best practicable means in the design of the treatment systems for radioactive effluents.
— The resulting doses to the critical groups should be evaluated to ensure achievement of the target.
— If the target is not met, other options should be evaluated.

3.26. The design should be such as to ensure that the contamination of material that leaves the plant can be adequately monitored.

Commissioning

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

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

APPROACH TO DESIGN FOR ACCIDENT CONDITIONS

3.29. 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 [1]. Accident prevention is not explicitly addressed in this Safety Guide, but reference should be made to the available relevant information [13, 14].

3.30. The design objectives for accident conditions are to limit to acceptable levels: (1) the risks to the public from possible releases of radioactive material from the nuclear power plant; and (2) the risks to site personnel from these releases and from direct radiation exposure. These design objectives should be achieved by means of high quality design and special features, such as safety systems and protection systems, that are incorporated into the design of the plant. Achievement of the design objectives should be confirmed by means of a safety analysis. Deterministic safety analyses and the associated dose assessments and probabilistic safety assessments for demonstrating compliance with the radiation dose limits should be based on conservative assumptions for the analysis of design basis accidents and realistic or best estimate assumptions for the analysis of severe accidents. These issues are discussed in Sections 10 and 11 of this Safety Guide and in Refs [1, 10].

3.31. To achieve the design objectives mentioned, the necessary design provisions and procedures (e.g. for access to the control room, maintenance of essential equipment or process sampling) should be such as to enable the plant operators to manage the situation adequately in an accident. Recommendations on how to protect site personnel under accident conditions are provided in Section 10.

3.32. Practices that are similar to those used for operational states should also be used to ensure that proper plant design is achieved to provide adequate radiation protection for site personnel and the public under accident conditions. A safety culture should be established to ensure that safety matters are given the highest priority and that regulatory requirements on releases of radioactive material under accident conditions are met with adequate margins.

3.33. The proper design of plant systems and components for radiation protection under accident conditions should be achieved by means of consultation with experts in radiation protection, plant operations, plant design and accident analysis, and regulatory matters. There should be continuous interactions among these groups throughout the design process to arrive at a design that provides radiation protection under accident conditions which is acceptable to the regulatory body. The design should also ensure that effective procedures for accident management can be implemented.

4. PROTECTION OF SITE PERSONNEL IN OPERATIONAL STATES AND DURING DECOMMISSIONING

OBJECTIVES

4.1. In this section, consideration is given to the design features for protecting site personnel from the radiation that results in operational states and during the decommissioning of a nuclear power plant, and the means of implementing the system of dose limitation as described in Appendix I of the BSS [2] and in Ref. [7]. Guidance given in Ref. [15] is also taken into account here.

CONTROL OF SOURCES OF RADIATION

General

4.2. As discussed in Section 3, one of the early tasks for the designer should be to optimize the protection against the sources of radiation in the plant since these affect the radiation levels throughout the plant, while most of the other aspects of the design affect the radiation levels in local areas only. For many reactor designs, the major sources of radiation are activated corrosion products, although fission products may also be significant if there are significant amounts of failed fuel cladding. These sources originate in the reactor core. The radioactive materials are then transported by the reactor coolant and by the moderator in liquid moderated reactors. Any practicable means should be employed by which the strength of sources or the transport of radioactive material can be reduced without excessive cost or reduction of the reliability of components. Leaktightness should be ensured as far as possible and leakage detection features should be provided, particularly for HWRs, for which the hazard due to tritium should be addressed. If seals are used, it should be ensured that these contain no antimony. Details are given in Annex II.

4.3. It should be recognized that, while consideration is given to decommissioning at the design stage, there will be significant and ongoing changes in conditions during decommissioning. Measures should be taken in the design to reduce the significance of these changes, but this factor should also be recognized in the operational arrangements. Equally, access will be necessary during decommissioning to areas that are not normally accessed. Consideration should be given to this factor in the design of facilities and equipment.

Corrosion products

4.4. 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 such as external pipe work and heat exchangers. This source should be minimized by the following means: (a) reducing the corrosion and erosion rate of circuit materials by the proper selection of materials and the control of the coolant chemistry; (b) selection of materials to minimize the concentration of nuclides (particularly of cobalt in steel) that are known from experience to become major sources of radiation; (c) providing removal systems (such as particulate filters and ion exchange resins); (d) minimizing the concentration in feedwater of nuclides that can be activated in the core.

4.5. The presence of materials with a high cobalt content, such as stellite, which are used for valve seats and bearings because of their hardness, in the primary coolant circuit and chemical control circuits, in the turbine systems of BWRs and in directly connected circuits should be reduced by applying the optimization principle. This is particularly important for components within the reactor core. In the case of direct cycle reactors, the use of materials with a high cobalt content should be minimized in components of the feedwater system that are situated after the condensate purification plant. For direct cycle, light water cooled, pressure tube reactors, for which the pressure tube and fuel cladding are made of zirconium or zirconium alloys of high purity and low activation cross-section, another important source of corrosion products (crud) is the feedwater circuit following the condensate polishing plant. Special attention should be paid to the choice of heater material for the feedwater, and consideration should be given to the possible installation of filters in the feedwater or core coolant return circuit close to the core inlet.

4.6. Special attention should also be paid to the selection of materials and to the coolant chemistry, which also make an important contribution to the reliability of the steam supply system for the nuclear plant. The compatibility of materials and coolant, which is of the utmost importance to minimizing the amount of maintenance, repair and statutory inspection necessary for primary circuit components, should be given careful consideration. Only those materials should 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.

4.7. In water cooled reactors, corrosion products are removed by treating the water with ion exchange resins to remove soluble species and by the installation of particulate filters. Their capacities should be adequate to cope with the enhanced release of corrosion products ('crud bursts') and fission products ('spiking') that occurs during the startup and cooldown phases.

4.8. Systems to remove corrosion products, both radioactive and non-radioactive, should be provided for the primary coolant for both water cooled and gas cooled reactors (GCRs). These systems are particularly important for gas cooled reactors with stainless steel clad fuel, such as advanced gas cooled reactors (AGRs), to minimize the amount of radioactive material available for deposition in the coolant circuit. In an AGR, active corrosion products arise in the coolant circuit primarily from the oxidation of fuel cladding. The oxides are released from the core surface in the form of particles when the fuel experiences a thermal shock as a result of reactor trip. Channel filters should be provided to remove this active oxide from the coolant so as to reduce deposition in the boilers and in other areas of the coolant circuit to which access for inspection and maintenance is required. Consideration should be given to treating the fuel cladding (e.g. by plating) to reduce spalling.

Fission products

4.9. Defects in the fuel cladding may result in the release of fission products to the coolant, which can add significantly to the activity of the coolant and contamination of the coolant circuit. Defective fuel elements should 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 should be provided for detecting failed cladding, appropriate limits should be set for the coolant activity and the plant should be shut down within a prescribed time interval if these are exceeded.

Activity in pond water

4.10. Water in the fuel storage pond should be 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 pond of a reactor in which there have been major fuel failures, the design should provide a means for containing any radioactive material that might otherwise leak into the pond water by bottling the fuel or some equivalent handling.

PLANT LAYOUT

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

Classification of areas and zones

4.12. The requirements for the classification of areas as controlled areas and supervised areas are established in the BSS, Appendix I, paras I.21–I.25 [2]. Each controlled area should have a minimum number of access and exit points for personnel and for materials and equipment.

4.13. Provision should be made for controlling the exit(s) from the controlled areas and for monitoring persons and equipment leaving the controlled areas.

4.14. Controlled areas should be divided into zones on the basis of the anticipated radiation levels and radioactive contamination levels (i.e. dose rates and activity concentrations for surface or airborne radionuclides; see Annex V). The greater the radiation or contamination related risks of a zone, the greater is the need to control access to that zone for the purpose of ensuring compliance with individual annual dose limits and taking account of dose constraints.

4.15. In the plant design stage, all rooms should be classified into planning zones on the basis of their likely dose rates, surface contamination levels and concentrations of airborne radionuclides. These zones constitute the controlled areas. The general practice is to divide the controlled areas of a nuclear power plant into three or more radiation and contamination zones, including zones that may not be accessible during operation.

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

Changing rooms, changing areas and related facilities

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

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

4.19. Within the changing rooms, a physical barrier should be provided to separate clearly the clean area from the potentially contaminated area. The changing rooms should be large enough to meet the needs during periods of maintenance work, and allowance should be made for temporary personnel employed as contractors.

Control of access and occupancy

4.20. The access by personnel to areas of high dose rates or high levels of contamination should be controlled by the provision of lockable doors and, where appropriate, the use of interlocks. Interlocks are provided to ensure that access is only possible when radiation levels are acceptably low and they should be designed to provide an alarm if they become inoperative.

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

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

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

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

(1) Provision of passageways of adequate dimensions for ease of access to plant systems and components. In areas where it is likely that site personnel will have to wear full protective clothing, including masks with portable air supplies or connections to a supply by air hoses, account should be taken of this in deciding on the dimensions of the passageways.
(2) Provision of clear passageways of adequate dimensions to facilitate the removal of plant items to a workshop for decontamination and repair or disposal. The routes for removing large items of plant during decommissioning should be planned at the design stage and the necessary provisions should be incorporated.
(3) Provision of adequate space in the working areas, to carry out repairs or inspections, for example.
(4) Provision of easy access to high radiation areas such as the water chambers of the steam generators in PWRs and the valves in the systems that contain primary coolant.
(5) Provision of 'waiting areas' in low radiation areas.
(6) Placement of components that are likely to be operated frequently, or to require maintenance or removal, at a convenient height for working.
(7) Provision of ladders, access platforms, crane rails or cranes in areas where it can be foreseen that they will be required to permit the maintenance or removal of plant components. Features to facilitate the installation of temporary shielding should be included in the design.
(8) Use of computer aided design models to optimize aspects of the design that affect working times. Video or photographic records should be made during the construction of the plant to facilitate the planning of work in areas of high radiation levels during operation and thus to shorten working times.
(9) Provision of means for the quick and easy removal of shielding and insulation where this is necessary to perform routine maintenance or inspection.
(10) Provision of special tools and equipment for facilitating work to reduce exposure times.
(11) Provision of remote controlled equipment;
(12) Provision of a suitable communication system for communication with the site personnel working in radiation areas or contamination areas.

SYSTEM DESIGN

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

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

(1) The work space in a zone of high radiation levels around components that require regular maintenance should be shielded from the radiation from other systems;
(2) Non-radioactive components that do not have to be mounted close to active components should be installed outside areas of high radiation levels;
(3) Methods for sampling radioactive liquids with minimal exposure should be provided;
(4) Methods for countermeasures (e.g. flushing) to avoid the sedimentation of radioactive sludge in piping and containers should be provided.

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

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

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

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

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

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

4.33. In direct cycle reactors, the design of the steam drying system should be such as to ensure that the levels of radiation and surface contamination in the turbine building are low.

COMPONENT DESIGN

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

4.35. The main approach in design to minimize radiation exposure is to provide for components of high reliability that require minimum surveillance, maintenance, testing and calibration, where applicable.

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

4.37. 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.

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

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

REMOTE TECHNIQUES

4.40. Remote techniques should be used wherever practicable to minimize the exposure of personnel. Techniques that should be considered include arrangements for remote inspection and for the removal and reinstallation of equipment. These techniques should be considered at the design stage. Techniques for the inspection and handling of plant items may be only semi-remote, in that personnel may still have to enter the controlled area to install equipment on rigs. An example of remote or semi-remote techniques is the provision of equipment for the ultrasonic inspection of welds. Access to the weld may be necessary in order to fit the scanner, but the operator can then move to a low radiation area to operate the equipment. For remote visual inspection, consideration should be given to the use of television cameras and windows shielded by lead glass or comparable materials.

4.41. Remote techniques may play a major part in the removal of the most radioactive items during decommissioning. The use of such techniques should be considered at the design stage and it should be ensured in the design that their use is not precluded. It is likely that there will be improvements in remote control techniques over the lifetime of the plant and between stages 1 and 3 of decommissioning. The best practicable techniques that are available when the work is carried out should be used.

DECONTAMINATION

4.42. The need for decontamination should be considered at the design stage. If it is considered that a worthwhile reduction in radiation exposure would result, the necessary provision for decontamination facilities should be made.

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

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

4.45. 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 high level alarm.

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

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

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

4.49. 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.

4.50. 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.

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

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

SHIELDING

Design of shielding

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

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

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

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

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

4.58. 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.

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

4.60. When neutrons are captured in the shielding, the gamma rays that are emitted as a consequence of the capture must 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.

4.61. Shields with the same mass per unit area provide approximately the same attenuation of a gamma ray flux, particularly at higher energies. The use of materials of a high density and high atomic number, such as lead, 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.

4.62. In relation to the formation of voids during construction, consideration should be given to the application of an appropriate quality assurance programme.

4.63. In the design of permanent shielding, account should be taken of seismic forces.

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

Penetrations through the shielding

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

(1) Minimizing the area and number of straight-through paths containing material of very low density (e.g. gases, including air);
(2) Providing shielding plugs;
(3) Providing zigzag or curved pathways;
(4) Filling the gaps with grouting or other compensatory shielding material.

4.66. In some cases, depending on the strength and location of the source with respect to the penetration, no additional shielding features may be necessary. In other cases, plugs or labyrinths of complex design should be incorporated and computer based shielding calculations should be made to justify the design.

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

VENTILATION

4.68. A dedicated active ventilation system should be provided for maintaining appropriate clean conditions in work spaces within the controlled area.

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

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

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

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

— Mechanisms of thermal and mechanical mixing;
— The limited effectiveness of dilution in reducing airborne contamination;
— Exhausting of the air from areas of potential contamination at points near the source of the contamination;
— The use of exhaust rates that are commensurate with the potential for contamination in the area;
— The need to ensure that the exhaust air discharge point is not close to an intake point of the ventilation system.

4.73. The airflow in the ventilation system should be such that the pressure in a region of lower airborne contamination is higher than the pressure in a region of potentially higher contamination. Thus the airflow in the ventilation system should be directed from regions of lower airborne contamination to regions of higher contamination and air should be extracted from the latter. The airflow should be such as to minimize the resuspension of contamination.

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

WASTE TREATMENT SYSTEMS[10]

4.75. 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. An estimate should be made of the expected radionuclide content in treated waste, and of the consequent maximum radiation level that can arise in each area of the waste treatment system. Consideration should be given to the sources that give rise to the highest radiation levels (such as ion exchange

[10] Requirements and recommendations on the management of radioactive waste before it is sent to a repository are established in IAEA Safety Requirements and Safety Guides [4–6].

resins, discarded radioactive components and filter waste). In the assessment of the sources and the estimation of radiation levels, account should be taken of the changes in the activity 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).

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

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

4.78. The design should be such that it is possible to carry out by remote control reverse flow flushing, washing, regeneration and change of resins.

STORAGE OF RADIOACTIVE WASTE AT THE PLANT

General design considerations

4.79. Facilities should be provided for the safe storage of the radioactive waste that arises at the plant, with account taken of its form (solid, liquid, gas or a mixture), its radionuclide content and its nature in terms of the extent to which it has been processed. The safe storage of waste will depend in part on the design, construction, operation and maintenance of the facility concerned. The design features of facilities should be such that the radioactive waste can be received, handled, stored and retrieved without causing undue occupational or public exposure or environmental effects. Further recommendations on this subject are provided in Ref. [6].

4.80. The design of storage facilities for radioactive waste should incorporate the following functions:

(a) Maintaining the confinement of stored materials;

(b) Maintaining subcriticality (in spent fuel storage facilities);
(c) Providing for radiation protection (by means of shielding and contamination control);
(d) Providing for the removal of heat (from spent fuel);
(e) Providing for ventilation, as necessary;
(f) Allowing the retrieval of the waste for transport off the site.

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

(a) Chemical stability against corrosion caused by processes acting within the waste and/or external conditions;
(b) Protection against radiation damage, especially stability under conditions of the degradation of organic materials and damage to electronic devices;
(c) Resistance to impacts caused by operational loads or due to incidents and accidents;
(d) Resistance to thermal effects, if applicable.

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

(a) Generation of hazardous gases caused by chemical and radiolytic effects (for example, the generation of hydrogen gas caused by radiolysis) and the buildup of overpressure;
(b) Generation of combustible or corrosive substances;
(c) Acceleration of the corrosion of metals (in particular, mild steel).

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

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

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

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

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

5. PROTECTION OF THE PUBLIC DURING PLANT OPERATION AND DECOMMISSIONING

DISCHARGE CRITERIA

5.1. To protect the public from radiological consequences due to the operation of the plant, plant operators are required to ensure that doses to members of the public arising from radioactive substances in the effluents and from direct radiation due to the plant do not exceed the prescribed limits, and that the optimization principle is applied [8]. In practice, radioactive discharges are generally regulated so that the best practicable means that do not involve excessive costs are employed for minimizing discharges. This is commonly done by specifying discharge limits for the most significant radionuclides, as described in para. 2.3.

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

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

SOURCE REDUCTION

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

EFFLUENT TREATMENT SYSTEMS

5.5. The flows and the activity concentrations of liquid and gaseous effluents need to be monitored and controlled to ensure that the authorized discharge limits are not exceeded [1]. Liquid and gaseous treatment facilities that are based on the best practicable means should be provided, as discussed in the following subsections. References [12, 16] provide recommendations and information on the calculation of the exposure of the public resulting from radioactive discharges.

Liquid treatment systems

5.6. The major sources of contaminated water that require treatment include: primary coolant that is discharged for operational reasons; floor drains that collect water that has leaked from the active liquid systems and fluids from the decontamination of the plant and fuel flasks; water that is used to backflush filters and ion exchangers; leaks of secondary coolant; laundries and changing room showers; and chemistry laboratories. The foregoing are essentially aqueous in nature and the guidance that follows is given on this basis. Where non-aqueous liquid waste is generated in sufficient volumes, the provision of a separate waste treatment system to deal with it should be considered. Further guidance on the treatment of aqueous and non-aqueous liquid waste is provided in Ref. [6].

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

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

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

Gas treatment systems

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

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

5.12. The isotopes of iodine, which usually have the greatest radiological impacts, are commonly removed by means of charcoal filters. Means should be provided for testing these filters using the most penetrating form of iodine to ensure their efficiency over the lifetime of the plant.

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

5.14. All radioactive gaseous effluents discharged to the atmosphere should be released from elevated points, with the topography of the site taken into account. There may be no need for these release points to be higher than the existing buildings (i.e. a stand-alone stack) if the radioactive content of the gaseous discharges can be reduced to the extent that some entrainment by the on-site buildings is acceptable. Such an approach should be justified in the optimization process, with consideration given to accident conditions.

SHIELDING

5.15. The provision for shielding that is incorporated into the design to protect site personnel during plant operation and to protect the public under accident conditions from direct or scattered radiation should also be designed to ensure adequate protection of the public during plant operation. 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.

6. GUIDELINES FOR ESTIMATING RADIATION DOSE RATES DURING PLANT OPERATION AND DECOMMISSIONING

OBJECTIVES

6.1. Recommendations on estimating radiation doses during operation and decommissioning are provided in this section in accordance with the scope of this Safety Guide.[11]

6.2. The first step in any calculation of dose rates should be to evaluate the source strength and its distribution. This may involve making calculations concerning the transport of radionuclides and their redistribution when

[11] This Safety Guide does not give guidance on the calculational methods or the values of the parameters to be used to evaluate the radiation dose rates that are expected to occur during operation and decommissioning.

activated corrosion products or fission products are carried in the reactor coolant (liquid or gas) and deposited away from the point of origin. The second step is to calculate the fluence rate (flux) at the dose point as a result of radionuclide transport from the source to the dose point and to calculate the radiation dose rate by multiplying the flux by the appropriate conversion factors.

SOURCE CATEGORIES

6.3. The sources of radiation in reactor systems during normal operation and decommissioning and the ways in which they arise are described in Annex II of this Safety Guide.

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

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

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

SOURCES AND PROPAGATION OF RADIATION: SPECIFIC SHIELDING DESIGN

The reactor core and its surroundings

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

6.6. An initial step in evaluating source strengths is to determine the fission rate, the neutron emission rate, and the spatial and energy distribution of the neutron flux within the core. This may be achieved by using computer codes in which account is taken of the spatial distribution of materials in the core and changes in fuel composition, the production of actinides and fission product poisons, and changes in control poisons (due to the positions of control rods, the heights of liquid moderators and poison concentrations) with fuel burnup. The neutron emission rate and neutron flux distributions that are calculated for the core are used as input data for computer calculations to determine the neutron flux energy and spatial distributions through the coolant and the structural and shielding materials surrounding the core. The neutron flux distributions are used in computer codes (which may be coupled with the neutron flux calculations) or in hand calculations to determine production rates for gamma ray sources in the core and surrounding materials. Production rates are determined for both prompt emission and delayed emission (activation) sources. In the case of activation sources, the decay of nuclides (half-life) and the irradiation time in the neutron flux are taken into account in determining the strength of gamma ray sources. In most cases, it is the gamma ray source that determines the dose rate to personnel.

6.7. The primary sources of radiation should be determined by using the procedures discussed in the references indicated in Annex II, Refs [II-2–II-4].

Reactor components

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

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

Activity of the coolant

6.10. 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:

— Corrosion products;
— Fission products;
— Activation products.

The first two are addressed separately in Annex IV. The details of the evaluation will depend upon the type of reactor under consideration. There are similarities between all reactor types, but in Annex III more specific details will be given for LWRs and PWRs, in particular. For most types of reactor, corrosion products are the major contributors to radiation levels at the plant during shutdown and thus to the occupational exposure of personnel. In PWRs, 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.[12] Therefore, accurate modelling of the source term for corrosion products is an important factor in optimizing the design. One important activation product is ^{16}N, which is a high energy gamma emitter with a half-life of 7 s and is a major source of radiation when the reactor is at power.

Radiation propagation through shielding

6.11. A detailed description of the methods and data that may be used to calculate the fluence from the radiation sources is outside the scope of this Safety Guide because of the volume of information involved. Some references are indicated in Annex II, Refs [II-2–II-4].

6.12. Calculations should be carried out for the propagation of radiation (mainly gamma rays) from the sources through simple, single material bulk shielding, or through shields of complicated geometry containing regions of low density (gases and voids) and low attenuation that present preferential transmission paths with scattering surfaces.

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

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

SOURCES FOR WHICH SHIELDING IS NOT PRACTICABLE

6.14. Some tasks have to be carried out in situations in which the provision of shielding is not practicable. Examples are work in the water chambers of PWR steam generators and the removal of insulation from, and the in-service inspection of, the primary circuit pipework of LWRs. 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 4.39 and 4.40.

SOURCES THAT DOMINATE DECOMMISSIONING DOSES AND WASTE VOLUMES

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

6.16. In a well designed and operated reactor, the major radiation source 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 one for tens of years after shutdown will be ^{60}Co arising from impurities in the steel and this will dominate until ^{63}Ni in the steel becomes important. In this case, the control of impurity levels that is exercised to control the magnitude of this source during operation will also be effective in controlling it during decommissioning.

6.17. In the case of concrete, the magnitude of the source term can affect both the doses to workers and the volume of radioactive waste that is generated. The source term in this case may be dominated by radionuclides that are not very important during operation, such as the rare earth isotopes, and control of such impurities may be an important aspect of the design process.

6.18. When a reactor has operated with defects in the cladding, the primary and auxiliary circuits may be contaminated by alpha emitters. The amount of irradiated fuel deposited on surfaces may reach a few tens of grams.[13] For such

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

situations, the risk of internal exposure by alpha emitters is a special hazard during maintenance, operation and decommissioning, and the relevant precautions, such as providing breathing protection, should be taken.

SPECIAL HAZARDS

6.19. A special hazard can be what are generally designated as 'hot spots'. Hot spots result from the activation of small objects present in the coolant. These objects may be:

— Particles of metal resulting from unusual wear of components and/or fuel assemblies;
— Debris left in the primary circuit or other circuits connected to it;
— Pieces of thick deposits on the fuel.

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

SOURCES THAT ARE IMPORTANT CONTRIBUTORS TO DOSES TO MEMBERS OF THE PUBLIC

6.21. As discussed in Annex II, it should be noted that the notion of 'important contributors to doses' is relative.

6.22. The important contributors to doses to members of the public are typically:

— ^{14}C, ^{3}H and ^{85}Kr, because the best practicable means available for their removal by waste treatment systems are not efficient and because their half-lives are long;
— ^{41}Ar is an important contributor, although its half-life is short, because it is released in large volumes of air (in venting of the containment during operation for AGRs and some PWRs);
— ^{133}Xe is a weak gamma emitter but it may be of importance when the reactor has been operating with a significant number of defects in the fuel cladding;
— Iodine, caesium and corrosion products.

6.23. Details of how to assess the radiation exposure of the public due to releases of radioactive substances to the environment are given in Refs [II-1, I-5] mentioned in Annex II.

7. MONITORING FOR RADIATION PROTECTION DURING PLANT OPERATION AND DECOMMISSIONING

GENERAL

7.1. For the effective implementation of the provision in the plant design for the radiation protection of site personnel and the public, a well planned radiation monitoring programme is required (Ref. [2], paras 2.38, 2.39). The requirements for the operational aspects of such a radiation monitoring programme are established in Appendices I and III of the BSS [2] and paras 6.105 and 6.106 of Ref. [1].

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

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

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

7.5. Equipment for performing these monitoring tasks should be provided in the design of a nuclear power plant. The rationale and the design basis for the measurement channels, their measuring ranges and detector locations should be documented. These systems are subject to national regulatory requirements. Safety significant equipment should be redundant to ensure that monitoring is always possible. Basic information on the electrotechnical and radiation measuring requirements for the design of instrumentation and devices is given in the standards of the International Electrotechnical Commission (IEC) and the International Organization for Standardization (ISO).

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

(1) Range of dose rate or activity concentration;
(2) Sensitivity;
(3) Radionuclides to be monitored;
(4) Provision of threshold alarms;
(5) Power supply and backup power supply;
(6) Environmental conditions;
(7) Provision for testing, calibration and easy maintenance;
(8) Provision for functioning during abnormal situations;
(9) Response to overload conditions;
(10) Failure mode indication;
(11) Potential for interference with or corruption of monitored data due to other radionuclides present in the area, particularly in the case of monitoring for neutrons, tritium and other beta radiation sources.

7.7. Measurement systems should be designed to maintain their operability under specified environmental conditions. The range of conditions of temperature, pressure, humidity, vibration and ambient radiation fields at least should be specified.

7.8. Measuring systems should be capable of detecting and indicating, within an acceptable margin, measurement results that are lower or higher than specified minimum and maximum values of an assessed measured variable. In special cases, it may be necessary to use two or more measuring channels to cover the specified range of measurement. In these cases, the measuring ranges should overlap sufficiently.

7.9. A system that gives relevant data on measured radiation values at the plant should be provided in the main control room, the health physics room, at some local control points and in the plant's computer information system. Alarm signals should be provided to the extent that is justified on the basis of the design goals of the radiation measuring systems.

7.10. Equipment for monitoring individual doses to workers should include the means necessary to measure, evaluate and record the doses received from external and internal sources. The general operational aspects of individual monitoring of doses (external radiation dosimeters, methods of assessment of internal dose, etc.) are considered in Refs [17, 18].

AREA MONITORING SYSTEMS WITHIN THE PLANT

7.11. Area monitoring includes the measurement of radiation dose rates and amounts of airborne radioactive material.

7.12. In the controlled areas, continuously operating fixed instruments with a local alarm and an unambiguous readout should be installed so as to give information on radiation dose rates and airborne contamination in selected areas. For monitoring special maintenance operations that last only a short time, and especially for monitoring in areas where high dose rates may vary, complementary portable dose rate meters should also be provided, with alarms to notify if preset values are exceeded. When designing audible alarm systems, the likely noise level in the relevant areas should be taken into account. Surface contamination meters should also be provided.

7.13. In LWRs, external radiation monitoring systems should be installed in:

- — The reactor containment;
- — The rooms that are adjacent to the upper part (refuelling area) of the containment;
- — The spent fuel storage facility;
- — The fuel handling machine;
- — The treatment and storage facilities for radioactive waste;

— The decontamination facilities;
— The transport routes for fuel and waste.

In other types of reactor, similar provisions should be made at the corresponding locations.

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

— The containment;
— The fuel storage facility;
— The auxiliary building;
— The radioactive waste building.

In other types of reactor, similar provisions should be made at the corresponding locations.

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

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

EFFLUENT MONITORING

7.17. Equipment is required to be provided to monitor and record all discharges of radioactive liquid and gaseous effluents to the environment [1]. In addition, equipment should be provided to monitor systems that may contribute large fractions of the overall releases of the plant. In water cooled reactors, monitoring of the following systems should be provided where applicable:

— Plant off-gas system;
— Vent header of radioactive waste tanks;
— Building ventilation with potential radioactive contamination.

7.18. In addition, in direct cycle reactors, provision should be made for monitoring of the condenser air removal system. In PWRs this is also useful for the detection of ruptures of steam generator tubes. In gas cooled reactors, provision should be made to sample and monitor all operational discharges of the reactor coolant.

7.19. The equipment for effluent monitoring should be capable of determining the total activity and the nuclide composition of the discharge. This may be done by on-line measurements and laboratory analysis. Guidance on monitoring of effluents is provided in Ref. [19].

8. PROCESS RADIATION MONITORING

8.1. Nuclear power plants should be fitted with installed radiation measuring systems for monitoring activity concentrations for process fluids and gases. The purpose of these measurements is to detect fuel failures and the leakage of radioactive material from or into a process system.

8.2. Installed radiation measuring equipment should be used for monitoring activity concentrations in the primary circuit water and secondary circuit of PWRs and of the primary coolant and main steam lines of BWRs. In indirect cycle reactors, the secondary systems operate at a lower pressure than the primary circuit systems and radioactive material may be transferred from the primary to the secondary side by leakage through heat exchangers. This may also occur in PWRs and fast breeder reactors. The activity in the secondary circuit should therefore be monitored. Large leaks, which may necessitate rapid action, may be detected by means of radiation monitoring of either the main secondary steam lines (response to ^{16}N) or the main condenser air exhaust lines (response to fission products).

8.3. Another method of detecting leaks to the secondary systems for PHWRs is to monitor the amount of make-up water supplied to the primary system, since the normal leak rate from the primary system is very small and any increase in this rate is apparent from the falling level in the make-up storage tank. For PHWRs, other effective methods of detecting leaks to the secondary system are (i) monitoring tritium activity and (ii) monitoring the concentration of heavy water.

8.4. 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.

8.5. 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.

8.6. Auxiliary systems that may also become contaminated are:

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

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

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

9. AUXILIARY FACILITIES

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

(1) A health physics operations office, including testing and calibration facilities for radiological instruments and protective equipment;
(2) A changing room for protective clothing;
(3) A personnel decontamination facility;
(4) An equipment decontamination facility;
(5) Laundry facilities for contaminated clothing;
(6) A first aid room;
(7) A radiochemistry laboratory (for the preparation of samples and the measurement of activity);
(8) A storage area for contaminated items and tools;
(9) A workshop for contaminated equipment;
(10) A store for radiation sources;
(11) Facilities for the management, conditioning and storage of waste;
(12) A dosimetry laboratory or dosimetry control if there is an external service provider;
(13) A data recording and storage system for creating relevant databases and updating them with the appropriate records as required;
(14) An alternative or remote health physics control centre;
(15) An assembly area at the plant for use during a plant emergency;
(16) An emergency response centre;
(17) An identified sheltering area for the plant personnel.

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

(1) Protective clothing, boots, etc.;
(2) Protective equipment for the respiratory tract;

(3) Air samplers and equipment for measuring airborne activity concentrations;
(4) Portable dose rate meters with an audible alarm at variable settings and devices for monitoring personnel contamination and surface contamination;
(5) Portable shielding, signs, ropes, stands and remote handling tools;
(6) Communication equipment;
(7) Meteorological instruments;
(8) Equipment for monitoring individuals for intakes of radionuclides;
(9) Temporary containers for solid radioactive waste and special containers for radioactive liquids;
(10) Emergency equipment (including additional protective clothing, self-powered air samplers and emergency vehicles);
(11) First aid equipment;
(12) Equipment for sampling and analysis around waste storage areas, such as borehole monitoring equipment for underground storage facilities for radioactive waste.

10. PROTECTION OF SITE PERSONNEL UNDER ACCIDENT CONDITIONS

10.1. This section deals with design for the protection of site personnel from radiation that arises from accident conditions. In the design process, a proper assessment should be made of the magnitudes and locations and the possible transport mechanisms and exposure pathways for the radiation sources that will be present in and after accident conditions. All potential accident scenarios including severe accidents should be considered in this assessment (see Annex III).

10.2. The design should be such that the operator can ensure the safety of all persons on the site in the event of an accident or radiological emergency, in compliance with international requirements for emergency preparedness [20].

10.3. An analysis should be made of the areas of the nuclear power plant in which it is necessary to maintain habitability for the purpose of taking both accident management measures and emergency preparedness measures. Areas to which access is expected to be required in emergencies include the control room, rooms where emergency systems are located (or spaces adjacent to such rooms), on-site sampling facilities (for the containment, the stack, etc.), the emergency control centre, the laboratories and the technical support rooms. For this purpose, plant operating instructions for actions for accident management, maintenance and emergency preparedness should be developed. Design modifications should be based on the findings of the habitability assessments, as discussed in Refs [21–23].

10.4. The anticipated hazardous conditions in which emergency workers may be required to perform response functions on or off the site should be identified. Arrangements should be made for taking all practicable measures to provide protection for emergency workers for the range of anticipated hazardous conditions in which they may have to perform response functions on or off the site. These arrangements should include: arrangements to assess continually and to record the doses received by emergency workers; procedures to ensure that doses received and contamination are controlled in accordance with established guidance and in compliance with international standards [20]; and arrangements for the provision of appropriate specialized protective equipment, procedures and training for emergency response in the anticipated hazardous conditions.

10.5. Provisions should be made for shielding the radiation sources, in addition to those provisions required during operation, to ensure that personnel can have access to and can occupy the plant control room or the supplementary control points (e.g. the location of the remote shutdown panel) so as to operate and maintain essential equipment[14] without exceeding established dose limits as specified in paras V.27–V.32 of the BSS [2] and paras 4.57–4.65 of Ref. [20].[15] This includes access to equipment in cases where maintenance or repair may be necessary after an accident. In general, provision should be made to render direct intervention by operators superfluous by installing automatic or remote controlled equipment (e.g. remote controlled valves).

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

[15] In the event of an emergency, radiation dose limits for normal operation may be exceeded. Use should then be made of dose levels given in para. 6.13 of Ref. [7] and other conditions as established in Section 6 of Ref. [7] for interventions in emergencies.

10.6. Consideration should be given in anticipation to movements of the source material (e.g. the transfer of the core to the base of the reactor building), 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.

10.7. Provision should also be made to minimize the airborne radioactive contamination in areas to which access will be required for ensuring the safety of the plant or the site personnel, such as the reactor building, the fuel storage area, the plant control room and supplementary control points. Such provision may be achieved by closing off the air intake and the exhaust. In this case heat removal would have to be provided by cooling the air in a recirculation system. An appropriate fraction of the circulation air should be filtered if the inward leakage of contaminated air may be expected to be too high to permit occupancy of the room without the use of respiratory protection. The spread of airborne contamination throughout the plant can be limited by means of secondary containment or by ducting to the atmosphere, through filters if necessary. Requirements for control room habitability in particular should be addressed, in terms of the oxygen supply and habitability under conditions of releases of gaseous chemicals.

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

10.9. Provision should be made for alerting and assembling site personnel and for — at least provisionally — sheltering site personnel not involved in accident control or firefighting. Communication should be possible between the control room, supplementary control points and assembly points for personnel.

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

10.11. In addition, areas should be identified within the plant in which radiation exposures are expected to remain low in accidents. These areas may be used in evacuating site personnel and monitoring them for contamination [23]. Recording devices for individual monitoring should also be stored here.

11. PROTECTION OF THE PUBLIC UNDER ACCIDENT CONDITIONS

11.1. The possible consequences of design basis accidents and severe accidents should be determined to demonstrate compliance with design targets.

11.2. Compliance with design targets for design basis accidents should be assessed by means of safety analyses [10]. In cases where the safety analysis shows that the acceptance criteria are not met, additional protective features should be incorporated into the design or operational measures should be developed to meet the acceptance criteria.

11.3. Generally, the releases that are evaluated for accident conditions are releases to the atmosphere, since an accidental release of radioactive substances to the aquatic environment is usually unlikely. However, this should be verified for each design or each plant, and consideration should be given, for instance, to the contamination of groundwater by leakage from the spent fuel pool, for example.

11.4. The dispersion of radioactive material that may be released into the atmosphere in an accident depends on the release point and the weather during the accident. It is the usual design practice to assume that an unfavourable weather situation prevails 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 weather and environmental conditions. A methodology for the calculation of doses to the public should be developed in accordance with the requirements of the national regulatory body and it should be carefully validated [12]. International guidance exists for the definition of a critical group [9]. Design targets are usually set so that no banning of food is assumed, at least for frequent events; and thus, for these situations, the consumption of food that has been produced within the potentially affected area is used as an input to the dose calculation for members of the public in the critical groups.

11.5. In demonstrating compliance with the design targets for doses to the public, conservative assumptions should be made with regard to the duration of the exposure, the weather conditions, and shielding of and occupancy by the public at the time of the accident.

11.6. Within the off-site areas where protective actions are planned in the event of a severe emergency (e.g. the precautionary action zone and the urgent protective action planning zone), arrangements should be made for promptly assessing any radioactive contamination, releases of radioactive material and doses for the purpose of determining or modifying urgent protective actions following a release of radioactive material (see the international safety requirements for emergency response [20] and the requirements of paras V.23–V.25 of the BSS [2]).

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

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

(1) Achieving leaktightness and isolation of the containment;
(2) Filtering the exhaust air in order to reduce the releases of airborne radioactive substances, with due account taken of the fact that some pathways for accidental releases may bypass the filtered exhaust system;
(3) 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;
(4) 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);
(5) Providing means of sealing the containment building or reducing the flow volume of exhaust air to provide for decay time within the building;
(6) Reducing the amount of radioactive substances released by decreasing the discharge velocity of fluids or the closure time of valves;
(7) 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[16];

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

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

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

(1) Developing or upgrading safety systems, reactor protection systems and instrumentation systems to minimize equipment malfunctions and operator errors that could potentially lead to beyond design basis events or severe accidents;
(2) Ensuring that power is available for essential equipment, instrumentation, including health physics instruments, and protection systems.

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

12. RADIATION AND CONTAMINATION MONITORING UNDER ACCIDENT CONDITIONS

12.1. The radiation monitoring systems for accident management at a nuclear power plant should include provisions that are relevant to postulated accident conditions and, to the extent that is necessary and practicable, they should also be operable during severe accidents. Provision should be made for having portable monitoring instrumentation (for monitoring dose rates and surface and airborne contamination) with ranges that are appropriate for severe accident conditions. The aim should be to enable the operator to have a quick and reliable way of assessing radiation levels throughout the plant and in its vicinity and, consequently, to take any action that may be necessary under such accident conditions. Further requirements and recommendations on accident management are given in other IAEA publications [20, 23]. Requirements in respect of the organization for planning and conducting the emergency response following an accident, and the monitoring that is necessary to ensure that access can be gained, where required, after an accident at a nuclear power plant are established in Ref. [20]. Special attention should be paid to the occupancy of the main control room and the necessary emergency response measures on the site.

12.2. In accordance with international recommendations [17, 18], arrangements should be made to assess promptly: abnormal conditions at the facility; exposures and releases of radioactive material; and radiological conditions on and off the site. This should include acquiring the information needed in support of mitigatory action by the operator, emergency classification, urgent protective actions on the site, the protection of workers and recommendations for urgent protective actions to be taken off the site. These arrangements should also include providing access to instruments displaying or measuring those parameters that can readily be measured or observed in the event of a nuclear or radiological emergency and which form the basis for classifying events. The response of instrumentation or systems at the facility should be adequate for the full range of postulated emergencies, including severe accidents, as agreed with the regulatory body.

12.3. Means should be provided so that the operator is aware of the performance of the radiation monitoring systems under the environmental conditions that occur as a result of an accident. The most onerous design requirements are associated with the radiation measurement systems that are within or close to the reactor containment. The design of the systems for sampling and the direct measurement of activity in gaseous effluents is also complex.

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

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

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

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

12.8. The radiation measurement data under accident conditions should be available in the main control room and in those locations, such as the emergency control room, where personnel would be implementing measures to manage the accident. Suitable communications systems should be provided to enable information and instructions to be transmitted between different locations and to provide external communication with such other organizations as may be required. Provision should be made for the direct transfer of relevant data to the emergency response centre.

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

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

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Annex I

APPLICATION OF THE OPTIMIZATION PRINCIPLE

I–1. 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. This is frequently a matter of judgement. In most cases, the optimization needs to achieve a balance, with account taken of the need for dose reduction, the need to ensure reliable energy production and the costs involved. Very often, a qualitative approach based on the utilization of the best available and proven technology may be sufficient for making decisions on the optimum level of protection that can be achieved. At the design stage of a plant, or for a major modification or decommissioning, where a large expenditure is involved, the use of a more structured approach is appropriate [I–1] and decision aiding techniques can be used.

I–2. For types of reactor for which significant operating experience is available, many of the criteria and input parameters that are required in such a decision making process can be quantified. This is because:

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

I–3. If such a database is available, a differential cost–benefit analysis or other appropriate methods [I–2 to I–5] needs to 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 are evaluated against several attributes. One such methodology is described in Ref. [I–6].

I–4. If a differential cost–benefit analysis is performed, a monetary value for the averted dose needs to be established, which may or may not be approved by the regulatory body. Different values are used in different States [I–7, I–8].

I–5. Where a monetary value of averted dose is used in the control of occupational exposure, a baseline value for the monetary value of dose and an increase in this value as the individual dose approaches the dose limit may be applied. This approach is consistent with the aim of avoiding major disparities in the doses that are received by personnel of different types who work in the controlled area. This relationship reflects the aversion to both such disparities and to the risk itself and ensures that the major effort is focused on those workers who may receive the highest doses.

I–6. The results of all such analyses are only a tool for use in the decision making process and do not provide the decision itself. There is still a major contribution from expert judgement. For example, an analysis may not be able to justify, on economic grounds, the provision of remote equipment to eliminate the need for personnel to enter areas with high radiation levels or contamination levels, but the decision may be taken to provide such equipment on social grounds. The level of sophistication with which these analyses are performed needs to reflect the magnitude of the dose that is under consideration.

I–7. In optimizing the design it needs to be recognized that radiation is only one of several types of hazard that will be experienced by site personnel. Measures to reduce radiation exposure need not to increase the total hazard [I–9].

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Annex II

SOURCES OF RADIATION DURING NORMAL OPERATION AND DECOMMISSIONING

GENERAL

II–1. In the context of radiation sources, it is important to understand that a major source in a given operational state may become a minor one in a different operational state. Similarly, the importance may vary with the issue that is being addressed. Some isotopes that are of minor importance for dose rate considerations during operation become of major importance during decommissioning. Also, even when dealing with reactors of the same type, changes in the design may have a strong influence on the relative importance of different sources.

REACTOR CORE AND VESSEL

II–2. During power operation, fission products and actinides are produced as a result of the fission process. The most significant isotopes in terms of doses to site personnel and members of the public are usually the isotopes of the noble gases, iodine and caesium, but others such as strontium and the isotopes of plutonium may also be important. In severe accidents, a much greater range of radionuclides need to be considered. When the reactor is at power, the fuel elements emit neutrons and gamma rays as a result of the fission process and the decay of fission products. Gamma rays are also emitted as a result of neutron capture in the core and the surrounding material. If the coolant contains oxygen, another major source of radiation during power operation will be ^{16}N, which is formed by the interaction of fast neutrons with ^{16}O that is present in the coolant in the vessel. In addition, in the case of heavy water moderated reactors, photoneutrons are emitted from the interaction of gamma rays with deuterium. 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.

II–3. The neutrons and gamma rays that are emitted by the core represent a very intense source. The residual neutron flux outside the primary shielding is a source of activation of structural materials. It can therefore induce a 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.

II–4. Whenever there exists a direct path through a radiation shield, neutrons and gamma rays will penetrate or stream through this path with little or no attenuation. This phenomenon gives rise to dose rates even at large distances from the core.

II–5. For fast breeder reactors with sodium as a coolant, where the coolant pumps and steam generators are inside the vessel, the secondary coolant and the structural materials of the components become activated. The most important radionuclides are ^{22}Na, ^{24}Na, ^{54}Mn, ^{58}Co, ^{60}Co and ^{59}Fe.

II–6. Even if the reactor building is not designed to allow access at full power for long periods, access for short periods under acceptable conditions has to be made possible since it may be necessary for operational reasons.

II–7. Other sources (including ^{41}Ar, airborne contamination by ^{3}H and volatile fission products and rare gases) need to be considered when access to the reactor building is authorized during the operation of the reactor. In a PWR, the activation of ^{40}Ar contained in the air is a source of ^{41}Ar which is a gamma ray emitter. The ventilation of the reactor cavity results (in some designs) in the ^{41}Ar contamination being transferred to the whole free volume of the reactor building above the operating deck. Although the corresponding dose rate (external exposure) is low, it may not be negligible when the individual dose rate target is less than 10 μSv/h or less. Hydrogen-3 is also an important possible source of airborne contamination in HWRs and in the fuel building of an LWR. Argon-41 is also produced in the CO_2 coolant of GCRs and in the systems of HWRs that contain helium gas, such as the liquid zone control system and the moderator cover gas system.

II–8. After shutdown of the plant, the main radiation source in the vicinity of the vessel is the gamma radiation from the fission products and activation products created in the vessel, in the metallic parts of the insulation and in any material that has been exposed for a sufficiently long time to the neutron flux. For some designs of HWR, neutrons produced by subcritical multiplication of the photoneutron source give rise to a significant power level accompanied by gamma radiation for a short period of time (about 24 h).

II–9. In the case of LWRs, activation products will be produced mainly in the materials of the structure of the fuel assemblies, in the cladding of the fuel pins, in the pressure vessel internal structures, in the control rods, in the primary and secondary neutron sources pins, in the pressure vessel itself, in the water and its impurities, and in the primary shield. In the case of GCRs, the activation products will be mainly in the fuel pin 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 HWRs of the pressure tube type, activation products are found mainly in fuel pin cladding, pressure tubes, calandria tubes, control tubes, the calandria tank and the shield tanks.

REACTOR COOLANT AND FLUID MODERATOR SYSTEM

II–10. If the coolant contains oxygen (such as in LWRs, HWRs and CO_2 cooled reactors), a major source of radiation during power operation will be ^{16}N, which is formed by the interaction of fast neutrons with ^{16}O as the coolant passes through the reactor core. Nitrogen-16 is a strong gamma emitter with gamma ray energies of 6 and 7 MeV. Since the half-life of ^{16}N is short (7.1 s), the significance of this isotope will be reduced where the transport time between the core and a component in the coolant system is long compared with the half-life. In this case, other activation products of the coolant such as ^{41}Ar (GCRs), ^{19}O and ^{18}F (water cooled reactors) may be the most important contributors to the radiation levels. In a PWR, 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.

II–11. In water cooled reactors, and particularly in HWRs, tritium is an important source of internal radiation exposure. In LWRs, tritium as HTO 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.

II–12. Fission products that are released from fuel pins with defective cladding are a source of radiation in the reactor coolant. The activity of this source depends on a large number of parameters: the number and size of cladding defects, the local power in the vicinity of the defect, the burnup of the fuel and others. However, in modern reactors, the occurrence of fuel cladding defects is extremely rare. Furthermore, the main cause of cladding defects (~80%), which is interaction with small migrating objects (debris), is considerably reduced when a filtering grid is installed in the lower part of the fuel assembly.

II–13. Fission products also enter the coolant from residual surface contamination of the cladding by uranium (the efficiency of the cleaning in the manufacturing process is not absolute) and also from the uranium content of the cladding (a few ppm). A limit for uranium contamination (‘tramp uranium’) therefore needs to be specified.

II–14. The main contributor to dose rates during maintenance and repair is activated corrosion products, such as ^{60}Co, ^{58}Co, ^{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. Fission products such as ^{131}1, ^{134}Cs and ^{137}Cs make a low contribution to dose rates around these circuits because both the source term and the deposition rate are low. However, this contribution to dose rates may increase significantly in situations where components such as heat exchangers and valves are opened or entered for maintenance and repair.

II–15. If a reactor operates with a significant number of fuel cladding defects, a non-negligible mass of fuel (a few grams to a few tens of grams) is released to the coolant. In this situation, the alpha activity of the water and of the deposits may not be negligible. 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.

II–16. In cases where there is a separate oxygen containing fluid moderator system (such as in a pressure tube reactor), the isotope that is the major source of radiation during reactor operation will be ^{16}N. After shutdown, the radiation levels around the primary coolant system will be due mainly to activated corrosion products. The tritium present in the water coolant or moderator contributes to the radiation hazard only if it is released from the system and becomes airborne. This hazard has to be taken into account in the design of LWRs also since operation with a limited leakage of primary coolant is tolerated.

II–17. In the case of PWRs 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 major changes in the physical (temperature, pressure) and chemical (from reducing to oxidizing conditions) conditions. The solubility of deposited oxides of corrosion products increases considerably. A large amount of the activated corrosion products deposited on the fuel are released to the coolant and the activity concentration of the water may be increased by two or three orders of magnitude. The release rate is not constant and it decreases when the temperature is decreased from hot conditions to 80°C. Metallic species are also released. For units that have large areas of alloy containing nickel, the total released mass is of the order of a few kilograms. The release increases sharply when peroxide water is injected and a spike is observed. The oxidizing conditions stop the release and 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 dose rate is unchanged. The corrosion products of high activity that are removed during this period accumulate mainly on the ion exchangers of the chemical and volumetric control system. The activity may be equal to the total activity accumulated 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, 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.

II–18. In addition, for PWRs, a spiking phenomenon is observed for fission products during the shutdown phase. The fission products that are accumulated in all the spaces in the fuel pin (in fractures in the fuel pellets, in the gap between the fuel pellets and the cladding, and in the expansion chamber) may be released to the coolant when the pressure is decreased. Water can enter the fuel pin and wash out the fission products when it is emitted. Thus, the release is not limited to gases and volatile species. The release depends mainly on the characteristics of the cladding defects.

II–19. In the cleanup systems of water cooled and moderated reactors (such as LWRs and HWRs), there will be an accumulation of radioactive material in filters and ion exchange resins. This will consist of fission products such as iodine and caesium that have escaped to the coolant through fuel cladding defects,[1] and of radioactive corrosion products that are transported by the coolant or moderator. Filters and ion exchange resins and, more generally, all

[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.

components in which an accumulation of radioactive products occurs, will generate very high activities that require shielding. Radioactive noble gases may be formed in these filters by the decay of iodine isotopes. In HWRs, photoneutrons are produced in the heavy water by the photons from ^{16}N. This source is significant in determining the shielding requirements of the coolant circuit external to the core. In GCRs, the gas treatment system will accumulate activated corrosion products such as ^{58}Co and ^{60}Co, and fission products such as iodine and caesium, and it will become an important source of radiation.

II–20. For fast breeder reactors with sodium coolant, the dominant sources are ^{22}Na and ^{24}Na. Sodium vapours may rise into primary components that may penetrate the cover plate shield of the reactor vessel. If these components penetrate the shield, considerable shielding is required to yield acceptable dose rates on the operating floor. Tritium that is generated in the fuel by ternary fission is released to the primary coolant through the stainless steel cladding of the fuel (the principal mechanism is diffusion). Fission products such as iodine and caesium are released to the coolant if cladding defects occur. The Na coolant may be covered by an inert gas such as argon. The activation of the cover gas gives rise to ^{39}Ar and ^{41}Ar, which may leak into the reactor building.

II–21. The coolant of some GCRs contains tritium, ^{35}S in the form of carbonyl sulphide and ^{14}C. The ^{35}S is produced mainly from the chlorine impurity in the graphite moderator, the tritium from the lithium impurity in the graphite and ^{14}C from the nitrogen impurity in the coolant and moderator. Because these are pure beta emitters, they present a health hazard only if inhalation or ingestion of the isotopes is possible.

II–22. Carbon-14 is produced in LWRs and HWRs 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 HWRs. This may be the main source term for this nuclide and a contributor to the global long term collective dose commitment. However, in some HWR 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.

STEAM AND TURBINE SYSTEM

II–23. In direct cycle water reactors, ^{16}N, which is carried over to the steam phase, will be the major source of radiation during power operation. The sky shine effect needs to be checked for carefully for buildings with potentially light structures, such as the roof of the turbine building. Downstream from the condenser, ^{19}O also needs to be considered a major source of radiation. In the event of fuel pin failures, an additional source of radiation will be volatile fission products, mainly the noble gases, and volatile fission products such as iodine and caesium. During power operation, this source will be of minor importance compared with ^{16}N, but after reactor shutdown these isotopes and their progeny (e.g. ^{140}Ba) will be the major radiation source in this system. Another source may be non-volatile corrosion products that are carried over with water droplets in steam.

II–24. In PWRs and PHWRs, the steam and turbine system is separated from the radioactive systems by a material barrier (the heat exchanger tubes). Thus, in these reactors radioactive material can only reach the steam and turbine system if leaks occur between the primary and secondary circuits. Provided that the leak rates are monitored (e.g. by measurement of the activity of the water or of ^{16}N in the secondary circuit) and kept to such a level that the activity in the secondary system is low, protective measures against direct and scattered radiation from this system are not necessary. Thus, the maximum leakage rate that can be tolerated between the primary and secondary circuits needs to be kept very low. However, provision needs to be made for cleaning the fluid circuits and for waste disposal from the secondary side in case primary to secondary leaks do occur. The leakage of primary coolant to the secondary circuit can also be detected by monitoring tritium in the feedwater. The presence of radioactivity in the feedwater can lead to the uncontrolled release of radioactive material to the environment through feedwater leaks as well as the venting of steam.

II–25. In direct cycle plants, an additional source of secondary system contamination that needs to be considered is leakage from equipment for concentrating radioactive waste that involves steam heating. One such source of contamination is through tube leaks that allow contaminated waste to enter the condensed heating steam. Contaminated condensed water from such steam may then be introduced into the secondary system.

II–26. In fast breeder reactors, the secondary sodium coolant may become activated to ^{22}Na and ^{24}Na. This can give rise to dose rates in parts of the buildings outside the containment if the delay for the sodium transport from the steam generator to these areas is not long compared with the half-lives of ^{22}Na and ^{24}Na.

WASTE TREATMENT SYSTEMS

Liquid waste treatment system

II–27. 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 to be granted, or disposed of safely in storage.

II–28. The composition of liquid wastes (i.e. activity concentration and solid and chemical content) varies according to their origin. It is general practice to segregate and treat liquid wastes according to their expected compositions. The liquids in the liquid waste treatment system therefore have a wide range of activity concentration. The segregation of liquid wastes could be made in accordance with the following categories:

— High purity (e.g. leakage wastes from the primary circuit of PWRs during power operation);
— High chemical content (e.g. decontamination liquors);
— High solid content (e.g. liquid wastes from floor drains);
— Detergent containing liquid wastes (e.g. liquid wastes from laundry drains and personnel showers);
— Oil containing liquid wastes (e.g. in GCRs, liquid wastes from floor drains from the area of the lubricating oil tank for the circulator);
— Very high tritium content liquid wastes (for PHWRs).

II–29. The mixing of a small volume of effluent with a high activity concentration with a large volume of effluent in the same category with a low activity concentration is to be avoided.

II–30. In LWRs, before treatment some of the liquid wastes may have a radionuclide content as high as that of the reactor coolant, with the exception of short lived nuclides, which will have decayed, and gases, which will have been evolved as a result of depressurization. Concentrations of up to a few 10^{10} Bq/m^3 may be found in such untreated liquids. Thus, since the liquid waste treatment system processes active liquids, radioactive substances will accumulate in parts of the system such as filters, ion exchangers and evaporators.

II–31. In most cases, the accumulated radionuclide content will consist of activated material such as ^{60}Co, ^{58}Co, ^{51}Cr, ^{54}Mn and ^{59}Fe (depending upon the composition and corrosion rates of the material used in the primary circuit). Fission products such as isotopes of iodine, caesium and strontium may be important if failure of fuel cladding occurs.

Gas treatment systems

Off gas system

II–32. 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 BWRs, these gases will only stay in the coolant for a short period of time before they are removed by the off-gas system. However, in indirect cycle systems such as PWRs, the removal of fission gases may be necessary only before shutdown of the plant, when it will be essential to reduce the activity in systems that may have to be opened during shutdown.[2] In the case of defective fuel being present in the core and a high degassing rate (e.g. in a BWR), 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 substances will, in this case, consist of short lived isotopes (e.g. with a half-life of less than 1 h). In cases where the average stay time of the gas in the primary circuits is long (as may be the case in a PWR that is operated at a low degassing rate), isotopes with long half-lives will constitute the most significant fraction.

II–33. Components such as holdup tanks, holdup pipes, charcoal delay beds or cryogenic devices are provided in the off-gas system to delay the release to the environment of the extracted gases for a time that is sufficient to allow for a large fraction of the radioactive material to decay.

II–34. Of major importance in the design of an off-gas system is the formation of radiolytic gas in a direct cycle BWR and the existence of high hydrogen

[2] In such plants, gases are usually removed by the purification system.

concentrations in the primary coolant of a PWR. For PHWRs, large amounts of hydrogen could build up in the cover gas of the moderator and to some extent in the primary circuit. This may lead to the formation of combustible gas mixtures in those parts of the plant where air may enter the system. A recombiner needs to be provided to avoid the formation of such combustible mixtures. The reduction of the gas volume by the recombiner will also improve the delay time of a given system by a factor of about 10. Other solutions are possible, such as the strict separation obtained by physical means and by the application of appropriate procedures for aerated and hydrogenated gaseous effluents.

II–35. Increasing the delay time will reduce the content of short lived isotopes in the effluent but will not significantly alter the content of isotopes with half-lives longer than the delay time. However, the increase of the delay time to 30 days considerably reduces the release of the rare gas effluents, particularly ^{133}Xe. In this case, the most important radionuclides that are released are ^{85}Kr and ^{14}C.

II–36. The ventilation of buildings may be a source of gaseous release and, to a less extent aerosols. The main isotopes are ^{3}H (from evaporation of the pools) and ^{41}Ar.

Process vents

II–37. In some cases it is not possible to prevent the dilution of radioactive gases with inactive gases such as air before they are processed. Examples of this are:

— The calandria vault gas (in pressure tube reactors);
— The cover gases of containers in which liquids with some content of volatile substances are stored (e.g. storage tanks for collected reactor coolant leakage water in LWRs and storage tanks or some other equipment in the liquid waste treatment system). In some cases, gases are formed by decay, e.g. the decay of iodine to xenon;
— Coolant gas leaking into sections that contain air in GCRs;
— Air which has entered the pressure vessel of an LWR after it has been depressurized and the water level has been lowered prior to opening the vessel.

II–38. Vents for these gases need to be so located that the radioactive substances they contain are kept away from the plant operators. In the case of

AGRs and the calandria vault gas of pressure tube reactors, the radioactive material is mostly ^{41}Ar. In the case of LWRs, fission product gases usually dominate. In pressure tube reactors, the same is true for process vents that are in direct contact with coolant (in storage tanks, etc.).

Solid waste

II–39. Apart from the fuel, the following constitute the major solid radioactive wastes in terms of activity and volume that arise during operation:

(1) Components and structures that become activated or contaminated and have to be removed (e.g. control rods, neutron source assemblies, defective pumps, flux measuring assemblies, structures or parts thereof);
(2) Irradiated components of the fuel assembly from GCRs (in these reactors the assemblies are dismantled at the nuclear power plant);
(3) Ion exchange resins, filter material, filter coating material, catalysts, desiccants and similar;
(4) Concentrates from evaporators, precipitates;
(5) Contaminated tools;
(6) Contaminated clothing, towels, plastic sheet, paper and similar.

II–40. 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 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.

II–41. The solid waste need to be carefully managed to allow its volume to be minimized. However, reducing releases to the environment to very low levels will result in an increase in the volume of solid waste.

IRRADIATED FUEL

II–42. The irradiated fuel has a very high radionuclide content owing to the fission products and transuranics that accumulate in it. For on-load refuelling systems, delayed neutrons that are emitted from the fuel while it is in the refuelling system also have to be taken into account. An additional source of radiation arises from activation of the materials that are used to construct the fuel assemblies or stringers.

II–43. During the handling and storage of irradiated fuel, some radionuclides are released to the surrounding coolant. Radioactive corrosion products may go into solution or be released as particles while the fuel is being transported or stored in water, or if part of the fuel route is dry, and particularly if the cladding is oxidized, activated material may flake from the surface of fuel assemblies as a result of thermal or mechanical shock. In addition, defective fuel pins may release fission products, of which isotopes of noble gases, iodine, caesium and strontium are the most significant.

II–44. For wet fuel storage and handling systems, water cleanup systems with particulate filtration and ion exchange need to be provided. They are usually combined with heat removal systems. The radioactive content of the water is removed by the 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.

II–45. In AGRs, a dry fuel handling system is used, with initial dry fuel storage of fuel assemblies prior to dismantling, followed by pond storage of the fuel elements. A similar fuel handling system may be used on future GCRs. 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. Similar circumstances are faced with the dry fuel storage for CANDU reactors.

STORAGE OF FRESH FUEL

II–46. Where 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 will be largely absorbed by the fuel cladding. Thus the external exposure is of minor significance.

[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.

II–47. However, in the case of mixed oxide fuel, the new fuel may be radioactive as a result of the recycled plutonium it contains and, in some fuels, recycled uranium may be used. In this case, the new fuel will be a significant source of both neutrons and gamma rays and it will need to be shielded and contained at all times until it is inserted into the reactor. The magnitude of the neutron source term will depend upon the time that has elapsed since the plutonium was created, since actinides that emit neutrons will be produced as the plutonium decays.

II–48. In the case of ^{232}Th–^{233}U fuel, the new fuel may be highly radioactive owing to the presence of ^{232}U progeny. It will need to be shielded and contained at all times until it is inserted into the reactor.

DECONTAMINATION FACILITIES

II–49. The radioactive material in the waste solutions consists mainly of the corrosion products containing radionuclides such as ^{60}Co, ^{58}Co, ^{51}Cr, ^{59}Fe, ^{54}Mn. This material arises from the decontamination of components, of contaminated areas, of reusable protective clothing and possibly also of personnel (see paras 4.41–4.50) in the facilities that are provided to remove radioactive contamination from the 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

II–50. There are also other sources of radiation at nuclear power plants, such as neutron startup sources, corrosion samples, in-core and ex-core detectors, calibration sources for instruments and sources that are used for radiographic inspections.

METHODS OF CALCULATION

II–51. Methods of performing the calculations to determine the primary sources of radiation and the data required can be found, for example, in Ref. [II–1]. Suitable computer codes for implementing the methods, where required, are generally available from the Radiation Safety Information Computational

Center, Oak Ridge National Laboratories, Tennessee, USA [II–2], and from the OECD/NEA Data Bank Computer Program Services [II–3].

II–52. 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. [II–1], which contains extensive bibliographies. Where computer codes are required to apply the method, suitable codes are generally available through the Radiation Safety Information Computational Center, Oak Ridge National Laboratories, Tennessee, USA [II–2], or the OECD/NEA Data Bank Computer Program Services [II–3].

II–53. Details of how to assess the radiation exposure of the public due to releases of radioactive substances to the environment are given in Refs [II–4, II–5].

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Annex III

SOURCES OF RADIATION UNDER ACCIDENT CONDITIONS

INTRODUCTION

III–1. The main source of radiation in a nuclear power plant under accident conditions for which precautionary design measures are adopted consists of radioactive fission products. These are released either from the fuel elements or from the various systems and equipment in which they are normally retained. Examples of accidents in which there may be a release of fission products from the fuel elements are loss of coolant accidents and reactivity accidents in which the fuel cladding may fail due to overpressurization or overheating of the cladding material. Another example of an accident in which fission products may be released from the fuel rods is a accident in handling spent fuel , which may result in a mechanical failure of the fuel cladding from the impact of a fuel element that is dropped. The most volatile radionuclides usually dominate the accident source term (the release to or from the reactor containment). Recommendations and guidance on the assessment of accidents are presented in Section 4 of Ref. [III–1].

III–2. 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.

III–3. In the following subsections, examples of methods for determining radiation sources are described for selected 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 particular, severe accidents scenarios are not addressed explicitly. These issues are plant specific but a generalized approach to evaluating the source term from severe accidents is given in Ref. [III–2].

LIGHT WATER REACTORS

Loss of coolant accidents

III–4. The highest number of fuel cladding failures that may be expected as a consequence of any of the potential range of loss of coolant accidents up to a double ended rupture of a main coolant pipe and the fraction of each fission product released from the failed fuel need to be calculated. The subsequent release of fission products from the coolant to the containment or an equivalent means of confinement and their behaviour within this building (e.g. plateout, deposition by dousing or spraying and iodine reactions within the building) need to be assessed. For the purposes of this assessment, it needs to be assumed that the reactor core has operated for a sufficiently extended period so that the maximum equilibrium 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). Although the containment isolation occurring as a result of the high pressure in the containment would minimize the releases to the environment, the potential for significant releases to occur before containment isolation 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 PWR is given in Ref. [III–3].

III–5. As an alternative to such an analysis of loss of coolant accidents, it is the practice in some States to specify the fractions of core inventory of fission products that are assumed to reach the containment atmosphere after the accident. This fraction is specified differently for different categories 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 [III–4 to III–6].

III–6. The behaviour of the radionuclides after their escape from the containment depends upon the design of the plant. In some designs, the radioactive substances may reach the atmosphere immediately; in others, they are confined by a secondary containment. In other designs they escape to a surrounding building, from which they are released via a stack at a low rate and only after passage through filters.

Break of a steam line in a BWR

III–7. The break of a main steam line in a BWR may have more severe consequences than the break of a coolant recirculation pipe, which was discussed in paras III–4 to III–6. This depends upon the diameter of this pipe and the characteristics of the plant safety systems. It is thus necessary to analyse both situations.

III–8. If the location of the steam line break is within the containment, the sequence of events is similar to that for loss of coolant accidents, but with a different fraction of the fuel cladding failing. The equilibrium concentration of fission products for full power operating conditions has to be assumed. The design analysis for the potential radioactive release has to consider the time needed for containment isolation to take place and the effectiveness of the coolant purification system.

III–9. If the location of the steam line break is outside the containment and the main steam line isolation valves near the containment immediately close to isolate the reactor, only a fraction of the radioactive substances present in the steam under operating conditions would be expected to be released. Condensation of steam in the building in which the break occurs and the plateout of substances other than noble gases, will result in a reduction of the radionuclides that are available for release to the atmosphere. The location of the release to the atmosphere depends upon the design of the plant. Usually, the release of coolant into a building, other than the containment will cause such an overpressure that radioactive substances will escape from the building either through predetermined release points (usually in the roof) or through doors or other weak structures which will be opened by the overpressure, or by leaks. Mixing of the steam with the air in the building may be assumed if the possible pipe break locations and the escape points from the building are not close together. After the relief of overpressure, release to the outside will not be through uncontrolled release points but via the stack through the ventilation system and filters.

III–10. In some plants, leakage control systems have been added between the main steam isolation valves to limit the escape of radioactive material by this path.

III–11. The possibility of direct releases from the building after the relief of overpressure needs to be considered if the overpressure relief openings will not close and the underpressure of the building relative to the atmosphere cannot be restored by the ventilation system or by the natural draught of the stack.

Break of a steam line in a PWR

III–12. Initially, the break of a steam line in a PWR will release only insignificant quantities of radionuclides that may be present in the secondary system during normal operation.

III–13. As a consequence of the steam line break, the integrity of the steam generator tubes, which depends on the pressure difference between the primary and the secondary sides, needs to be assessed. If the structural integrity of the steam generator tube cannot be assured, the amount of primary water that could enter the secondary side needs to be estimated. After the shutdown of the reactor, the radionuclide content of the leaking water may increase with time owing to the effects of fission product spiking, as discussed in Annex II.

III–14. Depending upon the design of the steam generator, the primary water that leaks into the secondary side may mix with the inventory of the secondary coolant in the steam generator. The steam produced shortly after the accident, which will escape through the broken steam line, will have a higher than normal moisture content because of the depressurization.

III–15. Even with essentially intact 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 and with the maximum primary to secondary leakage for the technical specification, 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: (1) the high activity concentration for the leakage for the technical specification, (2) the break being not fully isolable, and (3) the dryout of the affected steam generator which results in no partitioning of radioactive material within the steam generator.

III–16. After shutdown, the production of steam will depend on the decay heat. The moisture content of the steam will be low because of the low steam flow, and the efficiency of the steam separators and driers will be high. Thus the steam, which may be released by pressure relief valves, will have relatively low concentrations of water soluble substances such as iodine and caesium. The release of radioactive substances is expected to be minimized by the isolation of the defective steam generator and other safety actions that will depend on the design.

Steam generator tube rupture

III–17. The rupture of a steam generator tube in a PWR can potentially lead to releases of radioactive substances to the atmosphere. These releases could be significant because, if iodine spiking does not occur immediately before the initiation of this event, it will occur during the course of the transient. Actual incidents of steam generator tube rupture have occurred in at least 12 operating nuclear power plants.

III–18. The design basis steam generator tube rupture is postulated to be a double ended break in one or more steam generator tubes. The breach of this primary to secondary side barrier initiates the release of reactor coolant into the secondary side. Subsequent to a reactor trip, the actuation of the steam pressure relief valves on the secondary side would release contaminated steam to the atmosphere. A potential for radioactive releases exists even if the steam generator tubes are not uncovered because of direct carry-over of the primary coolant into the steam line. The sources of radiation during this event are the radioactive fission products that are present in the primary to secondary break flow. Maximizing the break flow therefore maximizes the amount of radioactive fission products that are available for release to the atmosphere through the secondary side pressure relief valves.

III–19. After a reactor trip, the magnitude of the decay heat and the operator actions to isolate the affected steam generator and depressurize the primary circuit determine the magnitude of radioactive releases. The release of radioactive substances to the atmosphere will be terminated when the pressures of the primary and secondary circuits have equalized. The operator will cool down the plant using the intact steam generator(s).

III–20. The nature of the transient depends on the automatic safety systems and the time at which the operator starts to take effective action. The time assumed for this varies. A value of between 10 and 30 min is recommended in Ref. [III–7]. A method for determining the radioactive release following the rupture of a steam generator tube is given in Ref. [III–8].

Fuel handling accidents

III–21. In a design analysis of the effects of a postulated fuel handling accident, such as the dropping of spent fuel during its transfer from the vessel to the storage pool, the first step is to determine the radioactive inventory of the fuel at the time of the accident. Assumptions about the details of the history of fuel irradiation need to be chosen so as to lead to conservative (i.e. high) estimates of the activity.

III–22. The minimum time that elapses between the shutdown of a plant and the beginning of fuel handling operations needs to be used to determine the maximum source term inventory in the fuel rods at the start of refuelling operations. The number of fuel rods that may become defective as a result of the impact needs to be determined either by means of theoretical considerations or by the evaluation of actual occurrences with similar fuel elements or in experiments. The fraction of the noble gas inventory that is released to the surrounding pool water depends on the volume of free space within the fuel rod. There is no general consensus as to which is the predominant mechanism of release of iodine to the pool water from rods with cracked cladding. Iodine may be mainly leached out by water penetrating the defective fuel rod, or the main release may be of ‘gaseous’ iodine, which is assumed to be present in the free space within the fuel rod.

III–23. The usual, conservative, approach is to neglect the solubility of noble gases in the pool water. However, a significant fraction of the iodine and caesium will be retained in the pool water. The release of iodine into the atmosphere above the pool may best be described in terms of a partition coefficient (the ratio of the volumetric activity concentrations (Bq/m^3) in air 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.

III–24. 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 water/air volume ratio, 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).

III–25. 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 specified[1] as a global figure for certain reactor designs.

III–26. In addition to noble gases and iodine, up to a few per cent of the caesium inventory may be slowly leached out by water that penetrates the defective rods. This caesium will be in ionic form in the water, and its transfer to the air above the pool water may be neglected.

III–27. The amounts of noble gases and iodine released to the environment will be controlled by the ventilation rate and by the type of pool air sweeping system used, if such a system is available. The reduction effected by filters in the concentration of iodine in the exhaust air will be taken into account by means of an appropriate decontamination factor based on the filter design. The release may be terminated by the isolation of the appropriate part of the plant, especially if the storage pool is situated within a containment. If this isolation is done by operator action, a time delay will usually be assumed (e.g. between 10 and 30 min) [III–7].

Accidents in auxiliary systems

III–28. Examples of accidents that may occur in auxiliary systems are pipe breaks in the auxiliary systems, the ignition of filters or absorbers, explosions in storage tanks, spilling of liquid radioactive wastes, and fires in radioactive waste systems. Their consequences may be as severe as those described in the preceding sections. The consequences will depend upon the design features of the systems concerned, for which there are significant differences in different designs of reactor. For this reason, the assumptions to be chosen for the purposes of accident analysis need to be made on a case by case basis.

III–29. One important type of accident is that caused by a crack in the pipework of the residual heat removal system when it 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 most important contribution to the source term is the fission product spike that will occur as a result of the shutdown or that may have occurred before the break.

[1] This fraction is the inverse of the 'decontamination factor' which is also sometimes used.

III–30. The analyses of such faults require that the leak rate from the affected pipe, the transport of radioactive gases through the auxiliary building and the active ventilation system, the behaviour of iodine and the efficiency of the filtration system under the accident conditions all be determined as a function of time.

III–31. A method for analysing accidents of this type is described in Ref. [III-9] and supporting references.

Severe accidents

III–32. Accidents that are compounded by multiple system and/or component failures and operator errors such that they have a very low probability of occurrence are classified as beyond design basis accidents. In some cases part of the core may melt and such accidents are referred to as severe accidents. The possible severity of the consequences of such accidents is characterized by the design of the plant and the nature of the failures and operator errors. In such cases, safety systems may fail to perform their required safety functions owing to the failures and errors, leading to significant core damage that challenges the integrity of the remaining barriers to the release of radioactive material from the plant. The potential therefore exists for large releases of radioactive material to the environment during a severe accident.

III–33. Because of the potential for significant core damage to occur during severe accidents, such accidents are analysed in detail to determine their possible radiological consequences, which may have a significant impact on public health and safety. Such analyses can quantify the type and magnitude of the radiological source terms for the inventory of radioactive substances that is available for release to the environment. Recommendations and guidance on performing severe accident analyses and on quantifying the source term inventory available for release are provided in Section 4 of Ref. [III-1] and in Refs [III-2, III-10].

CO_2 COOLED REACTORS WITH UO_2 METAL CLAD FUEL[2]

Single channel faults

III–34. For accidents involving fuel in the core, the significant sources of radiation are the fission products in the fuel and the activation products in the cladding. The design of the core and fuel with regard to fuel rating, cooling and stability of the core configuration is such that melting of the UO_2 in a design basis accident will not occur.

III–35. The type of accident that could lead to the largest release of radioactive substances is considered to be an accident that results in partial fuel clad melting, with an accompanying rise in fuel (UO_2) temperature above the normal operating temperature. Residual flow through the channel (even if a change of fuel configuration occurs), conduction of heat to the rest of the core structure and reduction in the power density of the fuel due to automatic trips all ensure that the UO_2 will not melt. Under these conditions, a substantial percentage (possibly even 100%) of the noble gases and iodine nuclides that are produced by fission is released to the coolant from the fuel pins with damaged cladding. In addition, the radioactive material in the molten cladding is assumed to be released to the coolant. The percentage release of fission products from the fuel with damaged cladding depends on its temperature history (i.e. variation of temperature with time) following clad failure and on the resulting oxidation from UO_2 to U_3O_8 by the CO_2 coolant. Appropriate values determined from experiments are used for the release percentages.

III–36. Some of the radionuclides present in the coolant as a result of the accident are released from the coolant circuit by coolant leakage. Provision needs to be made in the plant design to collect the leaking coolant by means of a ventilation system and to discharge it to the atmosphere via high efficiency particulate attenuation (HEPA) air filters.

III–37. After the radioactive substances are released to the coolant, the amount available for discharge will depend upon leakage, plateout, cleanup by the coolant treatment plant and radioactive decay. In the case of noble gases, plateout and removal by the coolant treatment plant are both nil. For the plateout of iodine, the possible existence of more than one species of iodine

[2] This section deals with the design philosophy in the United Kingdom, where such reactors are in operation.

and their different plateout behaviour need to be taken into account. Some of the iodine released to the coolant circuit will be in the form of elemental iodine attached to particles and the remainder will be in the form of methyliodine. These two species will deposit from the coolant at different rates. Total deposition will be limited by adsorption or resuspension of the deposited iodine, and this needs to be taken into account in determining the variation in the activity in the coolant with time. Appropriate values determined by experiment need to be used for the fraction of iodine in each form, for the deposition half-lives of the species and for the limiting plateout factor.

Depressurization accidents

III–38. In a depressurization (loss of coolant) accident, the cladding of some fuel pins may fail to remain leaktight, and these fuel pins may release a fraction of the inventory of fission product noble gases, iodine and caesium that is 'free' in the pin[3] to the coolant. The magnitude of this fraction depends on the power rating of the fuel (MW(thermal)), the fuel temperature and the burnup. In the case of noble gases and iodine, the fractions of stable fission product gases (Xe and Kr), ^{133}Xe and ^{131}I free in the fuel pins are calculated by using a computer code based on diffusion to the boundary of grains of UO_2 and 'bubble' formation at the grain boundary. The constants used in the code are adjusted to give calculated fractions in accordance with measurements. For caesium, the release from the fuel pin to the coolant may be determined on the basis of the observation that the fractional caesium release is about one-third of that of ^{131}I.

III–39. In the case of noble gases, the fraction of the releases from the fuel that is discharged to the atmosphere is determined by the half-life of the isotope and the rate of depressurization of the reactor. For iodine and caesium nuclides, which are released in molecular form, deposition on reactor surfaces reduces the concentration in the coolant and hence the discharge to the atmosphere. It is necessary to take account of both deposition and subsequent desorption. Important factors determining the deposition and desorption rates are the variations of coolant flow rate and surface temperatures with time and the extent of mixing of coolant in the reactor.

III–40. For GCRs, the design of the coolant circuit and the automatic shutdown system of the reactor and the design fuel rating need to be such that clad melting will not occur in a depressurization accident. It needs to be noted

[3] 'Free' here means released from the fuel (UO_2) matrix.

that the failure of a reactor's prestressed concrete pressure vessel is considered not credible, and that breaches in the coolant circuit can occur only as a result of the failure of a pressure vessel penetration (e.g. boiler steam pipe or water pipe penetrations) or the failure of an external coolant pipe (e.g. a pressure relief valve or pipes of the coolant treatment plant). The largest breach that could occur would result from the failure of a feed pipe or return pipe of the coolant treatment plant. To restrict the rate of depressurization, flow restrictors need to be provided within the pressure vessel penetrations for the pipes of the coolant treatment plant.

III–41. Tripping of the reactor by a low pressure trip, limitation by design of the maximum rate of depressurization, ensuring the minimum coolant flow that is required at atmospheric pressure and continued cooling by the heat exchangers are sufficient measures to ensure that the fuel clad temperatures do not rise above the maximum normal operating temperature. Maintaining the clad temperatures at low levels would minimize the possibility of fuel clad failures that might occur as a result of the depressurization event. Design limits for the clad temperature, the fuel temperature and the pressure of fission product gases within the fuel pins need to be such that only fuel pins with undetected manufacturing defects would fail to remain leaktight during a depressurization accident.

III–42. The discharge point to the atmosphere will depend on the location of the breach. In some areas where a large breach could occur, hot gas ducts are provided to conduct the gas to the atmosphere below the level of the roof of the reactor building. In other areas, the gas is discharged to the atmosphere above the level of the roof of the reactor building, via the ventilation air exhaust system for contaminated air. The discharge to the atmosphere is filtered by means of HEPA air filters. However, since a high collection efficiency for the gas discharged from the reactor cannot be guaranteed, it is a practice to assume that the discharge is not decontaminated by the filters. Hence, by virtue of not applying a filtration factor, the calculation is claimed to be very conservative.

HEAVY WATER REACTORS

III–43. Reactors using heavy water (deuterium oxide) as a moderator, a coolant or both have the potential for the same type of accidental release of radioactive material as the corresponding LWRs described above. For a pressure tube reactor, the analyses for loss of coolant accidents need to include ruptures of the pressure tubes as well as header or pipe breaks. Note that rupture of a pressure tube in combination with a header or pipe break is not required or considered in the design basis accidents. Accidents involving failure of steam generator tubes or heat exchanger tubes also need to be analysed.

III–44. The heavy water in the operating plant contains tritium, which is the activation product of deuterium. The tritium is in the oxide form (i.e. water) and is not normally an important factor in the potential radioactive hazard to the public following an accident. However, the presence of tritium needs to be taken into account for the protection of site personnel during and following certain accidents.

REACTORS WITH ON-LOAD REFUELLING

III–45. 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 pond, 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

III–46. Areas of the nuclear power plant in which other postulated initiating events resulting in releases of radioactive substances to the environment may occur include:

(1) Spent fuel handling areas (i.e. fuelling machines, the dry fuel store, the fuel dismantling cell, the fuel storage pond and the loading bay for fuel transport flasks);
(2) The active effluent treatment plant;
(3) The treatment and cooling plant for fuel pond water;
(4) The coolant treatment plant;
(5) The store for solid radioactive waste;
(6) The fuel element debris vault;
(7) Ventilation filters.

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Annex IV

DETERMINATION OF SOURCE TERMS FOR PLANT OPERATION AND DECOMMISSIONING

CORROSION PRODUCT SOURCE TERMS

IV–1. 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.

IV–2. The phenomena and the relationships that need to be modelled are illustrated in Fig. IV–1. In principle, the behaviour of corrosion products can be modeled by methods that range from hand calculations to the use of complex software that includes analytical and phenomenological models.

IV–3. In the case of LWRs, parameters relating to the solubility in water of oxides at the temperature and pH of the coolant are very important parameters that determine the behaviour of corrosion products in the primary coolant. More specific details of the relevant parameters for coolant activity in PWRs are given below:

— In the case of PWRs, parameters relating to the solubility in water of unsaturated nickel and cobalt ferrites at a coolant temperature range of 280 °C–340 °C and a pH range of 6.5–7.4 at 300 ºC are very important for determining the behaviour of corrosion products in the primary coolant.
— The models that are used to describe the behaviour of corrosion products need to have the capability of modelling a large interacting ‘water–metal’ system for which the following parameters are typical:

  — Area in contact with the primary coolant: ~22 000 m^2;
  — Mass of coolant: 200–300 t;
  — Velocity of coolant: 0.1–15 $m \cdot s^{-1}$;
  — Duration of one circuit (reactor ⇨ steam generator ⇨ reactor): ~10 s including ~1 s in flux;
  — Variety of alloys: Zircaloy® 4/Inconel® 600, Inconel® 690, Incoloy® 800/Inconel® 718/hard facing materials (Stellite®)/stainless steel.

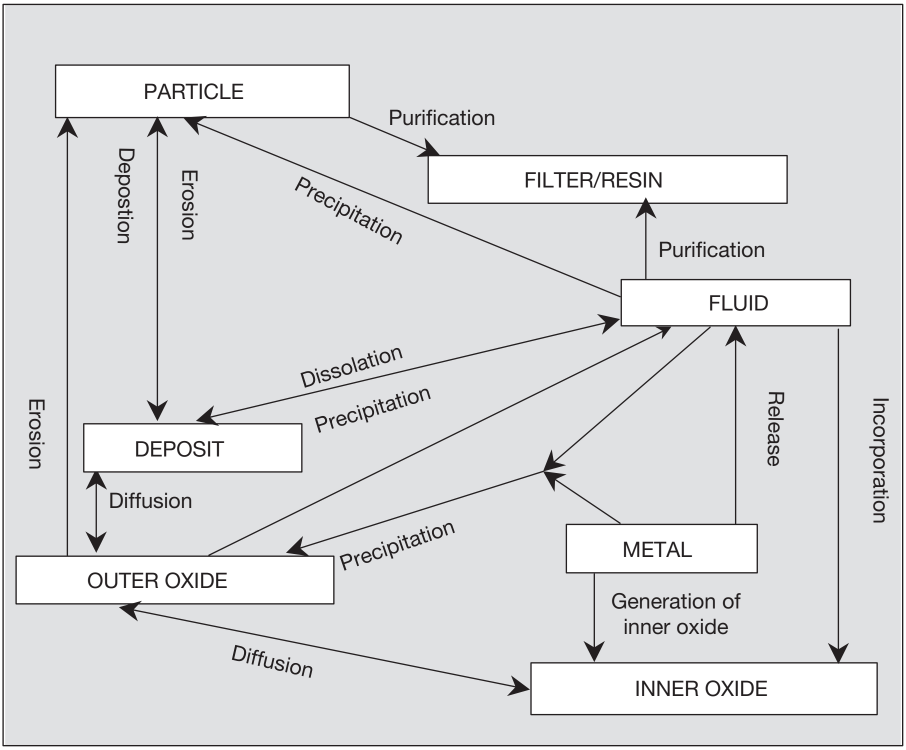

FIG. IV-1. Flow chart of phenomena required to be considered in modelling the behaviour of corrosion products.

— The order of magnitude of the mass of precursors of radioactive species (essentially ^{58}Ni (n, p) ^{58}Co and ^{59}Co (n, γ) ^{60}Co) is:

— Release (average):	1 mg/dm^2/month
— Cycle duration:	10 months
— Area excluding Zircaloy® (~ no release):	17000 m^2
— ^{59}Co level (impurity):	~$5 \cdot 10^{-4}$ g·g^{-1}
— ^{58}Ni in nickel based alloys (Inconel® 600, 690):	~ $3 \cdot 10^{-1}$ g·g^{-1}

— Therefore, the input to the reactor coolant during a ten month cycle is ~10 g·cycle^{-1} of ^{59}Co and ~5 kg·cycle^{-1} of ^{58}Ni;

— Wear of hard facing materials (of parts in the internal structures of the core, pump bearings, valves, control rod drive mechanisms, etc.) is in addition to the figure for ^{59}Co;

— As a result, approximately 10 g of ^{59}Co and 5 kg of ^{58}Ni are the origin of the ^{60}Co and ^{58}Co deposits, respectively, that are responsible for 90% of the dose rates and occupational exposures.[1]

IV–4. In the case of fast reactors, the secondary coolant circuit enters the neutron flux and it is necessary to evaluate the source terms that are due to corrosion of the secondary circuit. Some of the important phenomena that affect the source term for corrosion products are the following:

— Ionic species can precipitate and agglomerate to particles.
— These particles circulate in the fluid and are likely to form deposits either within the reactor core or on out-of-flux surfaces. By this process they become activated during circulation or after they have been deposited on in-core surfaces.
— Ions and particles can be removed from the primary coolant by the coolant purification system. The effectiveness of this process depends on the flow rate and on the decontamination factors of the filters and the ion exchange columns of the coolant purification system. If any of these factors are too low, the purification system will be ineffective.

Because the primary circuit is an almost closed and non-isothermal system, the above processes compete with reverse processes: for example, particles and deposits may dissolve.

IV–5. The models that are used need to be appropriate for the properties of the system that is being addressed. The main parameters of a PWR have already been given in para. IV–2.

IV–6. Examples of Other factors that should be modelled include the following:

— When the concentration of oxides in the primary coolant is very low (in a PWR it is typically a few 10^{-9} $g \cdot g^{-1}$);
— When the release of elements from alloys is not proportional to their composition;
— When chemical conditions vary throughout the fuel cycle within a specified range;

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

— When bulk coolant and surface temperatures need to be taken into account;
— When wear by friction is significant.

IV–7. The phenomena associated with the behaviour of corrosion products are so complex that the accuracy of both hand calculations and calculations made using computer codes that are based on analytical models is poor. However, the results of calculations made with codes in which the physical and chemical phenomena are taken into account are much more accurate. They do not give accurate results in absolute terms, but they correctly predict the relationships between the important design parameters and the source term. They are therefore a very important aid for optimizing the levels of the sources of ^{58}Co and ^{60}Co.

IV–8. Owing to the complex nature of the phenomena involved, another essential input to evaluating the source term due to corrosion products is operating experience at relevant plants. The relevance of an operating plant will depend on how all the relevant factors at that plant compare with those for the plant that is being designed. These factors include the materials of the coolant circuit and their impurities, the coolant chemistry, the shutdown procedures and all the other factors that have been mentioned. Collecting the most accurate operating experience involves making regular measurements at exactly the same locations throughout the lifetime of the plant, including during transients such as reactor shutdowns.

IV–9. To optimize the levels of sources of radiation for a plant that is being designed, it is also necessary to know the nature and composition of the radioactive material that is deposited on components at relevant operating plants. This is best achieved by using a collimated gamma spectrometer. The large changes that occur in the physical and chemical conditions of the coolant when going from operation at power to cold shutdown are the cause of a significant dissolution of corrosion products deposited on the fuel elements. The extent of the corresponding spiking of the coolant activity is a function of a large number of parameters. The spiking value is not predictable. However, for a given reactor type, a variation band can be indicated. Because deposits of corrosion products vary from one fuel cycle to another in the same plant, it is necessary to ensure that the operating data that are used are converted into values that are sufficiently bounding for design purposes.

IV–10. For evaluating the source terms for the purposes of modifying or decommissioning a plant, there is no substitute for the results of the latest measurements that have been made at the same plant at all the relevant dose points.

SOURCE TERMS FOR FISSION PRODUCTS

IV–11. The usual approach for determining source terms for fission products is:

— To calculate the inventory of fission products in the fuel — several well known computer codes are available to perform this evaluation[2];
— To determine the amounts of radionuclides, and the corresponding activity, that are available in all the voids in the fuel pins;
— To determine the total activity of the radionuclides that will be released to the coolant through cladding defects.

These determinations are complex, particularly the last item.

IV–12. 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 LWRs, 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.

IV–13. More accurate results are obtained with modern codes for fission product releases by including the dependence of the release coefficient (s^{-1}) on the half-life of each isotope and by taking into account the parameters that were omitted in the earlier approach. In this case, the agreement with operating experience is good, and the predictions made on the basis of such codes can be used as a significantly less conservative basis for the design of shielding.

[2] Examples are the ORIGEN, FISPIN and APOLLO codes in the USA, in the United Kingdom and in France, respectively.

IV–14. This improvement is important for the optimization of shielding because the difference between the two approaches can lead to source terms that differ by a factor of 3 to 10 depending on the isotope. For a point source emitting a 1 MeV gamma ray, a reduction in the source term by a factor of 5 would lead to a reduction in the thickness of a concrete shield of approximately 20 cm.

IV–15. An alternative method is to use reasonably bounding values that are derived from operating experience at relevant plants. The factors that determine the relevance of other plants that are operating include the design of the fuel elements and the rating and burnup of the fuel.

IV–16. During power transients, fission products are released to the coolant in a short time period through the cladding defects. This release is the cause of a spike in the activity of the coolant. The magnitude and period of the release are difficult to predict, but reasonably bounding values can be derived from operating experience. In Ref. [IV-1], a correlation of the release and the duration with the pre-transient coolant activity is reported.

IV–17. 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.

REFERENCE TO ANNEX IV

[IV–1] DUTTON, L.M.C., et al., Methods for Calculating the Release of Radioactivity following Steam Generator Tube Rupture Faults, Rep. EUR-15615-EU, EURATOM, Luxembourg (1994).

Annex V

EXAMPLES OF ZONING FOR DESIGN PURPOSES

V–1. A good example of radiation zoning that may be used for design purposes is shown below (Table V–1) [V–1].

V–2. A good example of zoning that addresses radiation, surface contamination and airborne contamination is given by the classification of zones within the controlled area in Swedish nuclear power plants (Table V–2) [V–2].

TABLE V–1. EXAMPLE OF RADIATION ZONING THAT MAY BE USED FOR DESIGN PURPOSES

Access requirement	Design dose equivalent rate (μSv/h)	
	Mean	Maximum
Uncontrolled areas on-site	—	1
Continuous (> 10 person-hours per week)	1	5
1–10 person-hours per week	10	50
< 1 person-hours per week	100	500
1–10 person-hours per year	1000	10 000
< 1 person-hours per year	10 000	[a]

[a] Dose rates in excess of 10 mSv/h are acceptable provided that the exposure time is correspondingly short.

TABLE V–2. CLASSIFICATION OF ZONES WITHIN THE CONTROLLED AREA IN SWEDISH NUCLEAR POWER PLANTS FOR RADIATION, SURFACE CONTAMINATION AND AIRBORNE CONTAMINATION

Zone identification	Blue zone	Yellow zone	Red zone
Radiation zones	< 25 μSv/h	25–1000 μSv/h	> 1000 μSv/h
Surface contamination zones	For total β < 40 kBq/m^2	40–1000 kBq/m^2	> 1000 kBq/m^2
	For total α < 4 kBq/m^2	4–100 kBq/m^2	> 100 kBq/m^2
Zones for airborne contamination	1 DAC [a]	1–10 DAC	> 10 DAC

[a] DAC: derived air concentration.

REFERENCES TO ANNEX V

[V–1] NUCLEAR ELECTRIC, Preconstruction Safety Report for Sizewell B Barnwood, Gloucester (1996).

[V–2] FORSMARK NUCLEAR POWER PLANT, Radiation Protection Instructions (2003), Instruction F-I-201, Forsmark Kraftgrupp AB, Östhammar (2003).

GLOSSARY

accident. Any unintended event, including operating errors, equipment failures or other mishaps, the consequences or potential consequences of which are not negligible from the point of view of protection or safety.

assessment. The process, and the result, of analysing systematically the hazards associated with sources and practices, and associated protection and safety measures, aimed at quantifying performance measures for comparison with criteria.

averted dose. The dose prevented by the application of a countermeasure or set of countermeasures, i.e. the difference between the projected dose if the countermeasure(s) had not been applied and the actual projected dose.

beyond design basis accident. Accident conditions more severe than a design basis accident.

commissioning. The process by means of which systems and components of facilities and activities, having been constructed, are made operational and verified to be in accordance with the design and to have met the required performance criteria.

confinement. Prevention or mitigation of releases of radioactive material to the environment in operational states or design basis accidents.

construction. The process of manufacturing and assembling the components of a facility, the carrying out of civil works, the installation of components and equipment and the performance of associated tests.

contamination. Radioactive substances on surfaces, or within solids, liquids or gases (including the human body), where their presence is unintended or undesirable, or the process giving rise to their presence in such places.

controlled area. A defined area in which specific protection measures and safety provisions are or could be required for controlling normal exposures or preventing the spread of contamination during normal working conditions, and preventing or limiting the extent of potential exposures.

critical group. A group of members of the public which is reasonably homogeneous with respect to its exposure for a given radiation source and is typical of individuals receiving the highest effective dose or equivalent dose (as applicable) from the given source.

decommissioning. Administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility (except for a repository or for certain nuclear facilities used for the disposal of residues from the mining and processing of radioactive material, which are 'closed' and not 'decommissioned'

derived air concentration (DAC). A derived limit on the activity concentration in air of a specified radionuclide, calculated such that Reference Man, breathing air with constant contamination at the DAC while performing light physical activity for a working year, would receive an intake corresponding to the annual limit on intake for the radionuclide in question.

design. The process and the result of developing a concept, detailed plans, supporting calculations and specifications for a facility and its parts.

design basis accident. Accident conditions against which a nuclear power plant is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

dispersion. The spreading of radionuclides in air (aerodynamic dispersion) or water (hydrodynamic dispersion) resulting mainly from physical processes affecting the velocity of different molecules in the medium.

dose constraint. A prospective restriction on the individual dose delivered by a source, which serves as an upper bound on the dose in optimization of protection and safety for the source.

emergency. A non-routine situation or event that necessitates prompt action, primarily to mitigate a hazard or adverse consequences for human health and safety, quality of life, property or the environment. This includes nuclear and radiological emergencies and conventional emergencies such as fires, release of hazardous chemicals, storms or earthquakes. It includes situations for which prompt action is warranted to mitigate the effects of a perceived hazard.

event. In the context of the reporting and analysis of events, an event is any unintended [by the operator] occurrence, including operating error, equipment failure or other mishap, and malicious act, the consequences or potential consequences of which are not negligible from the point of view of protection or safety.

exposure pathway. A route by which radiation or radionuclides can reach humans and cause exposure.

gap release. Release from a reactor core of fission products in the fuel pin gap, which occurs immediately after failure of the fuel cladding and is the first radiological indication of core damage.

intake. The act or process of taking radionuclides into the body by inhalation or ingestion or through the skin.

justification. The process of determining whether a practice is, overall, beneficial, as required by ICRP's System of Radiological Protection, i.e. whether the benefits to individuals and to society from introducing or continuing the practice outweigh the harm (including radiation detriment) resulting from the practice.

member of the public. In a general sense, any individual in the population except, for protection and safety purposes, when subject to occupational or medical exposure. For the purpose of verifying compliance with the annual dose limit for public exposure, the representative individual in the relevant critical group.

monitoring. The measurement of dose or contamination for reasons related to the assessment or control of exposure to radiation or radioactive substances, and the interpretation of the results.

individual monitoring. Monitoring using measurements by equipment worn by individual workers, or measurements of quantities of radioactive materials in or on their bodies.

workplace monitoring. Monitoring using measurements made in the working environment.

occupational exposure. All exposure of workers incurred in the course of their work, with the exception of excluded exposures and exposures from exempt practices or exempt sources.

operational states. States defined under normal operation and anticipated operational occurrences.

optimization of protection (and safety). The process of determining what level of protection and safety makes exposures, and the probability and magnitude of potential exposures, "as low as reasonably achievable, economic and social factors being taken into account" (ALARA), as required by the ICRP System of Radiological Protection.

potential exposure. Exposure that is not expected to be delivered with certainty but that may result from an accident at a source or owing to an event or sequence of events of a probabilistic nature, including equipment failures and operating errors.

practice. Any human activity that introduces additional sources of exposure or additional exposure pathways or extends exposure to additional people or modifies the network of exposure pathways from existing sources, so as to increase the exposure or the likelihood of exposure of people or the number of people exposed.

projected dose. The dose that would be expected to be incurred if a specified countermeasure or set of countermeasures — or, in particular, no countermeasures — were to be taken.

protection and safety. The protection of people against exposure to ionizing radiation or radioactive materials and the safety of radiation sources, including the means for achieving this, and the means for preventing accidents and for mitigating the consequences of accidents should they occur.

public exposure. Exposure incurred by members of the public from radiation sources, excluding any occupational or medical exposure and the normal local natural background radiation but including exposure from authorized sources and practices and from intervention situations.

radiation protection. The protection of people from the effects of exposure to ionizing radiation, and the means for achieving this.

radiation protection programme. Systematic arrangements which are aimed at providing adequate consideration of radiation protection measures.

radioactive discharges. Radioactive substances arising from a source within a practice which are discharged as gases, aerosols, liquids or solids to the environment, generally with the purpose of dilution and dispersion.

radioactivity. The phenomenon whereby atoms undergo spontaneous random disintegration, usually accompanied by the emission of radiation.

regulatory body. An authority or a system of authorities designated by the government of a State as having legal authority for conducting the regulatory process, including issuing authorizations, and thereby regulating nuclear, radiation, radioactive waste and transport safety.

risk. A multiattribute quantity expressing hazard, danger or chance of harmful or injurious consequences associated with actual or potential exposures. It relates to quantities such as the probability that specific deleterious consequences may arise and the magnitude and character of such consequences

safety culture. The assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receive the attention warranted by their significance.

source. Anything that may cause radiation exposure — such as by emitting ionizing radiation or by releasing radioactive substances or materials — and can be treated as a single entity for protection and safety purposes.

source term. The amount and isotopic composition of material released (or postulated to be released) from a facility.

supervised area. A defined area not designated a controlled area but for which occupational exposure conditions are kept under review, even though no specific protection measures or safety provisions are normally needed.

worker. Any person who works, whether full time, part time or temporarily, for an employer and who has recognized rights and duties in relation to occupational radiation protection. (A self-employed person is regarded as having the duties of both an employer and a worker.)

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