

IAEA Safety Standards

for protecting people and the environment

Deterministic Safety Analysis for Nuclear Power Plants

Specific Safety Guide

No. SSG-2 (Rev. 1)



IAEA

International Atomic Energy Agency



Documents

Document Sharing Hub

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

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

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

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

www.iaea.org/resources/safety-standards

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

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

RELATED PUBLICATIONS

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

Reports on safety in nuclear activities are issued as **Safety Reports**, which provide practical examples and detailed methods that can be used in support of the safety standards.

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

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

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

DETERMINISTIC
SAFETY ANALYSIS FOR
NUCLEAR POWER PLANTS



Documents
Document Sharing Hub

The following States are Members of the International Atomic Energy Agency:

AFGHANISTAN	GERMANY	PAKISTAN
ALBANIA	GHANA	PALAU
ALGERIA	GREECE	PANAMA
ANGOLA	GRENADA	PAPUA NEW GUINEA
ANTIGUA AND BARBUDA	GUATEMALA	PARAGUAY
ARGENTINA	GUYANA	PERU
ARMENIA	HAITI	PHILIPPINES
AUSTRALIA	HOLY SEE	POLAND
AUSTRIA	HONDURAS	PORTUGAL
AZERBAIJAN	HUNGARY	QATAR
BAHAMAS	ICELAND	REPUBLIC OF MOLDOVA
BAHRAIN	INDIA	ROMANIA
BANGLADESH	INDONESIA	RUSSIAN FEDERATION
BARBADOS	IRAN, ISLAMIC REPUBLIC OF	RWANDA
BELARUS	IRAQ	SAINT LUCIA
BELGIUM	IRELAND	SAINT VINCENT AND THE GRENADINES
BELIZE	ISRAEL	SAN MARINO
BENIN	ITALY	SAUDI ARABIA
BOLIVIA, PLURINATIONAL STATE OF	JAMAICA	SENEGAL
BOSNIA AND HERZEGOVINA	JAPAN	SERBIA
BOTSWANA	JORDAN	SEYCHELLES
BRAZIL	KAZAKHSTAN	SIERRA LEONE
BRUNEI DARUSSALAM	KENYA	SINGAPORE
BULGARIA	KOREA, REPUBLIC OF	SLOVAKIA
BURKINA FASO	KUWAIT	SLOVENIA
BURUNDI	KYRGYZSTAN	SOUTH AFRICA
CAMBODIA	LAO PEOPLE'S DEMOCRATIC REPUBLIC	SPAIN
CAMEROON	LATVIA	SRI LANKA
CANADA	LEBANON	SUDAN
CENTRAL AFRICAN REPUBLIC	LESOTHO	SWEDEN
CHAD	LIBERIA	SWITZERLAND
CHILE	LIBYA	SYRIAN ARAB REPUBLIC
CHINA	LIECHTENSTEIN	TAJIKISTAN
COLOMBIA	LITHUANIA	THAILAND
CONGO	LUXEMBOURG	TOGO
COSTA RICA	MADAGASCAR	TRINIDAD AND TOBAGO
CÔTE D'IVOIRE	MALAWI	TUNISIA
CROATIA	MALAYSIA	TURKEY
CUBA	MALI	TURKMENISTAN
CYPRUS	MALTA	UGANDA
CZECH REPUBLIC	MARSHALL ISLANDS	UKRAINE
DEMOCRATIC REPUBLIC OF THE CONGO	MAURITANIA	UNITED ARAB EMIRATES
DENMARK	MAURITIUS	UNITED KINGDOM OF GREAT BRITAIN AND NORTHERN IRELAND
DJIBOUTI	MEXICO	UNITED REPUBLIC OF TANZANIA
DOMINICA	MONACO	UNITED STATES OF AMERICA
DOMINICAN REPUBLIC	MONGOLIA	URUGUAY
ECUADOR	MONTENEGRO	UZBEKISTAN
EGYPT	MOROCCO	VANUATU
EL SALVADOR	MOZAMBIQUE	VENEZUELA, BOLIVARIAN REPUBLIC OF
ERITREA	MYANMAR	VIET NAM
ESTONIA	NAMIBIA	YEMEN
ESWATINI	NEPAL	ZAMBIA
ETHIOPIA	NETHERLANDS	ZIMBABWE
FIJI	NEW ZEALAND	
FINLAND	NICARAGUA	
FRANCE	NIGER	
GABON	NIGERIA	
GEORGIA	NORTH MACEDONIA	
	NORWAY	
	OMAN	

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

IAEA SAFETY STANDARDS SERIES No. SSG-2 (Rev. 1)

DETERMINISTIC SAFETY ANALYSIS FOR NUCLEAR POWER PLANTS

SPECIFIC SAFETY GUIDE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2019

COPYRIGHT NOTICE

All IAEA scientific and technical publications are protected by the terms of the Universal Copyright Convention as adopted in 1952 (Berne) and as revised in 1972 (Paris). The copyright has since been extended by the World Intellectual Property Organization (Geneva) to include electronic and virtual intellectual property. Permission to use whole or parts of texts contained in IAEA publications in printed or electronic form must be obtained and is usually subject to royalty agreements. Proposals for non-commercial reproductions and translations are welcomed and considered on a case-by-case basis. Enquiries should be addressed to the IAEA Publishing Section at:

Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 26007 22529
tel.: +43 1 2600 22417
email: sales.publications@iaea.org
www.iaea.org/publications

© IAEA, 2019

Printed by the IAEA in Austria

July 2019

STI/PUB/1851

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.

Title: Deterministic safety analysis for nuclear power plants / International Atomic Energy Agency.

Description: Vienna : International Atomic Energy Agency, 2019. | Series: IAEA safety standards series, ISSN 1020-525X ; no. SSG-2 (Rev. 1) | Includes bibliographical references.

Identifiers: IAEAL 19-01244 | ISBN 978-92-0-102119-9 (paperback : alk. paper)

Subjects: LCSH: Nuclear power plants — Safety measures. | Nuclear power plants — Risk assessment. | Industrial safety.

Classification: UDC 621.039.58 | STI/PUB/1851

FOREWORD

by Yukiya Amano
Director General

The IAEA's Statute authorizes the Agency to “establish or adopt... standards of safety for protection of health and minimization of danger to life and property” — standards that the IAEA must use in its own operations, and which States can apply by means of their regulatory provisions for nuclear and radiation safety. The IAEA does this in consultation with the competent organs of the United Nations and with the specialized agencies concerned. A comprehensive set of high quality standards under regular review is a key element of a stable and sustainable global safety regime, as is the IAEA's assistance in their application.

The IAEA commenced its safety standards programme in 1958. The emphasis placed on quality, fitness for purpose and continuous improvement has led to the widespread use of the IAEA standards throughout the world. The Safety Standards Series now includes unified Fundamental Safety Principles, which represent an international consensus on what must constitute a high level of protection and safety. With the strong support of the Commission on Safety Standards, the IAEA is working to promote the global acceptance and use of its standards.

Standards are only effective if they are properly applied in practice. The IAEA's safety services encompass design, siting and engineering safety, operational safety, radiation safety, safe transport of radioactive material and safe management of radioactive waste, as well as governmental organization, regulatory matters and safety culture in organizations. These safety services assist Member States in the application of the standards and enable valuable experience and insights to be shared.

Regulating safety is a national responsibility, and many States have decided to adopt the IAEA's standards for use in their national regulations. For 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 regulatory bodies and operators around the world to enhance safety in nuclear power generation and in nuclear applications in medicine, industry, agriculture and research.

Safety is not an end in itself but a prerequisite for the purpose of the protection of people in all States and of the environment — now and in the future. The risks associated with ionizing radiation must be assessed and controlled without unduly limiting the contribution of nuclear energy to equitable and sustainable development. Governments, regulatory bodies and operators everywhere must ensure that nuclear material and radiation sources are used beneficially, safely and ethically. The IAEA safety standards are designed to facilitate this, and I encourage all Member States to make use of them.

THE IAEA SAFETY STANDARDS

BACKGROUND

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

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

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

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

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

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

THE IAEA SAFETY STANDARDS

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

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

Safety measures and security measures¹ have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

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

Safety Fundamentals

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

Safety Requirements

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

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

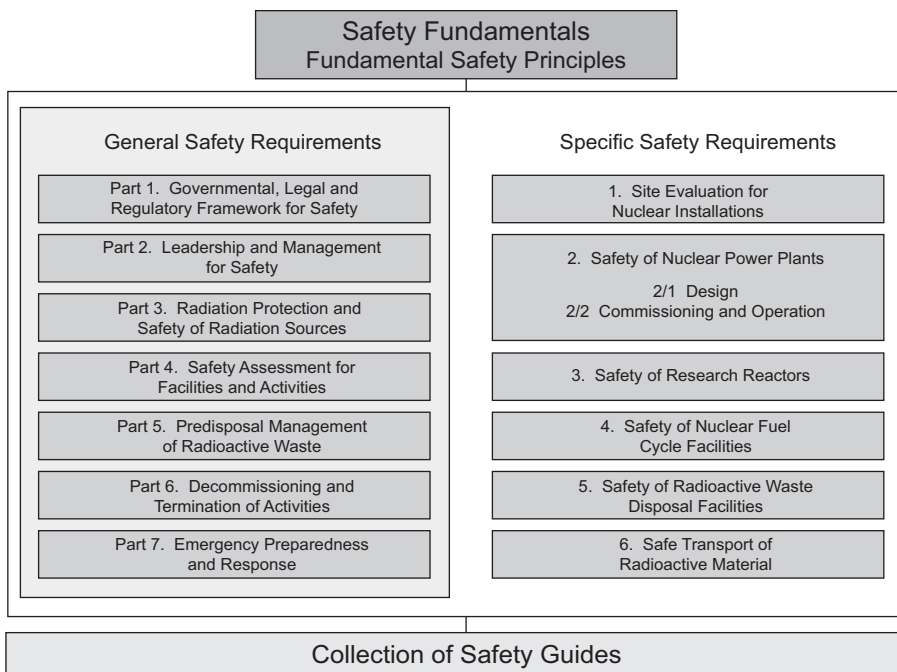


FIG. 1. The long term structure of the IAEA Safety Standards Series.

Safety Guides

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as ‘should’ statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

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

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

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

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

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

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

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

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

All IAEA Member States may nominate experts for the safety standards committees and may provide comments on draft standards. The membership of

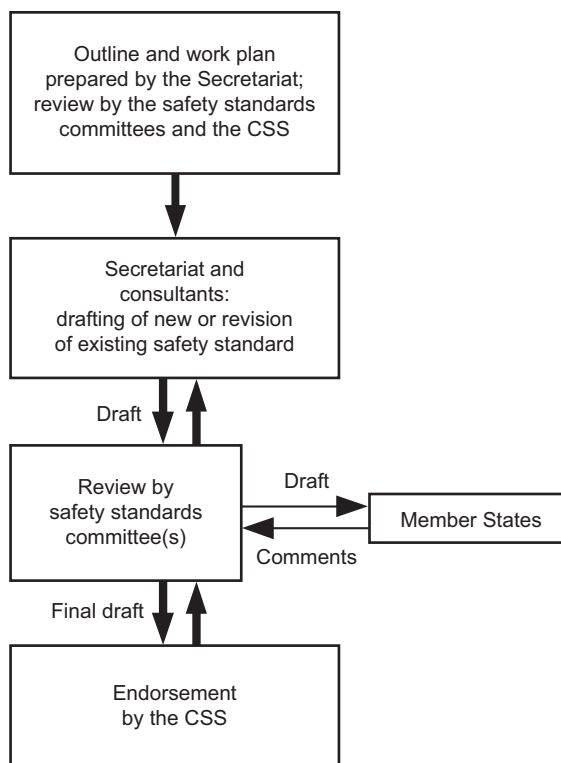


FIG. 2. The process for developing a new safety standard or revising an existing standard.

the Commission on Safety Standards is appointed by the Director General and includes senior governmental officials having responsibility for establishing national standards.

A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards. It articulates the mandate of the IAEA, the vision for the future application of the safety standards, policies and strategies, and corresponding functions and responsibilities.

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

The findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of international

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

INTERPRETATION OF THE TEXT

Safety related terms are to be understood as defined in the IAEA Safety Glossary (see <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 in the IAEA Safety Standards Series and its objective, scope and structure are explained in Section 1, Introduction, of each publication.

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

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

CONTENTS

1.	INTRODUCTION	1
	Background (1.1–1.3)	1
	Objective (1.4)	1
	Scope (1.5–1.14)	2
	Structure (1.15, 1.16)	4
2.	GENERAL CONSIDERATIONS	5
	Objectives of deterministic safety analysis (2.1–2.4)	5
	Acceptance criteria for deterministic safety analysis (2.5, 2.6)	6
	Uncertainty analysis in deterministic safety analysis (2.7)	6
	Approaches to deterministic safety analysis (2.8–2.15)	7
	Source term for a release of radioactive material to the environment (2.16–2.19)	9
3.	IDENTIFICATION, CATEGORIZATION AND GROUPING OF POSTULATED INITIATING EVENTS AND ACCIDENT SCENARIOS (3.1–3.7)	10
	Management system (3.8)	12
	Normal operation (3.9, 3.10)	12
	Postulated initiating events (3.11–3.22)	13
	Identification of postulated initiating events for anticipated operational occurrences and design basis accidents (3.23–3.36) ...	16
	General considerations for identification of design extension conditions (3.37, 3.38)	20
	Identification of design extension conditions without significant fuel degradation (3.39–3.44)	21
	Identification of design extension conditions with core melting (3.45–3.50)	23
	Identification of postulated initiating events due to internal and external hazards (3.51–3.54)	25
	Event sequences and accident scenarios to be ‘practically eliminated’ (3.55–3.57)	26
4.	ACCEPTANCE CRITERIA FOR DETERMINISTIC SAFETY ANALYSIS (4.1–4.18)	27

5.	USE OF COMPUTER CODES FOR DETERMINISTIC SAFETY ANALYSIS	31
	Basic rules for the selection and use of computer codes (5.1–5.6)	31
	Process management in connection with the use of computer codes (5.7–5.13)	33
	Verification of computer codes (5.14–5.20)	34
	Validation of computer codes (5.21–5.39)	35
	Qualification of input data (5.40)	39
	Documentation of computer codes (5.41–5.43)	39
6.	GENERAL APPROACHES FOR ENSURING SAFETY MARGINS IN DETERMINISTIC SAFETY ANALYSIS	40
	General considerations (6.1–6.11)	40
	Conservative approach and combined approach to deterministic safety analysis for anticipated operational occurrences and design basis accidents (6.12–6.20)	42
	Best estimate deterministic safety analysis with quantification of uncertainties for anticipated operational occurrences and design basis accidents (6.21–6.29)	45
7.	DETERMINISTIC SAFETY ANALYSIS FOR DIFFERENT PLANT STATES	47
	General considerations (7.1–7.5)	47
	Deterministic safety analysis for normal operation (7.6–7.16)	48
	Realistic deterministic safety analysis for anticipated operational occurrences (7.17–7.26)	50
	Conservative deterministic safety analysis for anticipated operational occurrences and design basis accidents (7.27–7.44)	52
	Deterministic safety analysis for design extension conditions without significant fuel degradation (7.45–7.55)	57
	Deterministic safety analysis for design extension conditions with core melting (7.56–7.67)	59
	Deterministic safety analysis in support of ‘practical elimination’ of the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release (7.68–7.72)	62

8.	DOCUMENTATION, REVIEW AND UPDATING OF DETERMINISTIC SAFETY ANALYSIS	64
	Documentation (8.1–8.14)	64
	Review and updating of deterministic safety analysis (8.15–8.18) ...	66
9.	INDEPENDENT VERIFICATION OF DETERMINISTIC SAFETY ANALYSIS BY THE LICENSEE (9.1–9.21)	67
	REFERENCES	73
	ANNEX I: APPLICATION OF DETERMINISTIC SAFETY ANALYSIS	75
	ANNEX II: FREQUENCY RANGES OF ANTICIPATED OPERATIONAL OCCURRENCES AND DESIGN BASIS ACCIDENT CATEGORIES	83
	CONTRIBUTORS TO DRAFTING AND REVIEW	85

1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide provides recommendations and guidance on the use of deterministic safety analysis and its application to nuclear power plants in compliance with the requirements established in IAEA Safety Standards Series Nos SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [1], and GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [2].

1.2. Current developments for ensuring the stable and safe operation of nuclear reactors are closely related to the advances being made in safety analysis. Deterministic safety analyses for normal operation, anticipated operational occurrences, design basis accidents and design extension conditions, including severe accidents, as defined in SSR-2/1 (Rev. 1) [1] and in the IAEA Safety Glossary [3], are essential instruments for confirming the adequacy of safety provisions.

1.3. This Safety Guide supersedes the 2009 version of SSG-2¹. The modifications incorporated into this Safety Guide reflect recent experience of deterministic safety analysis included in safety analysis reports for designs for new nuclear power plants and in the application of deterministic safety analysis to existing nuclear power plants. The Safety Guide has also been updated to maintain consistency with current IAEA safety standards, including those Safety Requirements publications updated to reflect lessons learned from the Fukushima Daiichi nuclear power plant accident.

OBJECTIVE

1.4. The objective of this Safety Guide is to provide recommendations and guidance for designers, operating organizations, regulatory bodies and technical support organizations on performing deterministic safety analysis and on its application to nuclear power plants. It also provides recommendations on the use of deterministic safety analysis in:

¹ INTERNATIONAL ATOMIC ENERGY AGENCY, Deterministic Safety Analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2, IAEA, Vienna (2009).

- (a) Demonstrating or assessing compliance with regulatory requirements;
- (b) Identifying possible enhancements of safety and reliability.

The recommendations are provided to meet the applicable safety requirements established in SSR-2/1 (Rev. 1) [1] and GSR Part 4 (Rev. 1) [2], and are supported by current practices and experience from deterministic safety analysis being performed for nuclear power plants around the world.

SCOPE

1.5. This Safety Guide applies to nuclear power plants. It addresses ways of performing deterministic safety analyses to achieve their purpose in meeting safety requirements. Such analyses are primarily required to demonstrate adequate fulfilment of safety functions by the design, to ensure that barriers to the release of radioactive material will prevent an uncontrolled release to the environment for all plant states, and to demonstrate the validity of the operational limits and conditions. Deterministic safety analyses are also required to determine the characteristics of potential releases (source terms) depending on the status of the barriers for different plant states.

1.6. This Safety Guide focuses primarily on deterministic safety analysis for the safety of designs for new nuclear power plants and, as far as reasonably practicable or achievable, is also applicable to the safety re-evaluation or reassessment of existing nuclear power plants when operating organizations review their safety assessment. The recommendations provided are intended to be consistent with the scope of applicability indicated in paras 1.3 and 1.6 of SSR-2/1 (Rev. 1) [1], and are particularly based on experience with deterministic safety analysis for water cooled reactors.

1.7. The recommendations provided in this Safety Guide focus on best practices in the analysis of all plant states considered in the design, from normal operation, through anticipated operational occurrences and design basis accidents, to design extension conditions including severe accidents.

1.8. This Safety Guide deals with human errors and failures of plant systems (e.g. systems in the reactor core, reactor coolant system, containment, fuel storage or other systems containing radioactive material) having the potential to affect the performance of safety functions and thus lead to loss of physical barriers against releases of radioactive material. An analysis of the hazards themselves, either internal or external (natural or human induced), is not covered by this Safety

Guide, although the effects and loads resulting from the hazards and potentially inducing failures in plant systems are taken into account in determining initiating events to be analysed.

1.9. This Safety Guide addresses the use of deterministic safety analysis for design or licensing purposes, aimed at demonstrating, with adequate margins, compliance with established acceptance criteria.

1.10. This Safety Guide addresses the different options available for performing deterministic safety analysis, namely the conservative approach, the best estimate approach with and without quantification of uncertainty, and a combined approach.

1.11. This Safety Guide focuses on neutronic, thermohydraulic, fuel (or fuel channel for pressurized heavy water reactors) and radiological analysis. Other types of analysis, in particular structural analysis of structures and components, are also important means of demonstrating the safety of a plant. However, detailed guidance on performing such analysis is not included in this Safety Guide, since such information can be found in specific engineering guides. Neutronic and thermohydraulic analysis provides the necessary boundary conditions for structural analysis.

1.12. This Safety Guide covers aspects of the analysis of releases of radioactive material up to and including the determination of the source term for releases to the environment for anticipated operational occurrences and accident conditions (paras 2.16–2.18). Radioactive gaseous and liquid effluents and discharges during normal operation are primarily controlled by operational measures and are not covered by this Safety Guide. Similarly, dispersion of radioactive material in the environment and prediction of the radiological effects on people and non-human biota is outside the scope of this Safety Guide (see IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [4]). While general rules for deterministic safety analysis also apply to the analysis of radiological consequences of anticipated operational occurrences and accident conditions, this Safety Guide does not provide specific guidance for such analysis. Such specific guidance can be found in other IAEA Safety Guides, for example IAEA Safety Standards Series No. GSG-10, Prospective Radiological Environmental Impact Assessment for Facilities and Activities [5].

1.13. This Safety Guide describes general rules and processes to be followed in performing deterministic safety analysis. The Safety Guide does not describe specific phenomena, nor does it systematically identify the key factors essential for

neutronic, thermohydraulic, fuel (or fuel channel) and radiological analysis. When such information is provided in this Safety Guide, it is intended as an illustration or example and should not be understood to be a comprehensive description.

1.14. Recommendations on nuclear security are outside the scope of this Safety Guide. In general, documentation and electronic records relating to deterministic safety analysis processes and outputs provide limited information regarding equipment location and vulnerability, and practically no information on cable routes and other aspects of plant layout. However, such information needs to be reviewed to identify any sensitive information that could be used to support malicious acts, and such information needs to be protected appropriately. Guidance on sensitive information and information security is provided in Ref. [6].

STRUCTURE

1.15. This Safety Guide comprises nine sections and two annexes. Section 2 introduces some basic concepts and terminology used in the area of deterministic safety analysis, as a basis for the specific recommendations provided in the other sections. The sequence of subsequent sections corresponds to the general process of performing deterministic safety analysis. Section 3 describes methods of systematic identification, categorization and grouping of postulated initiating events and accident scenarios to be addressed by deterministic safety analysis, and includes practical advice on the selection of events to be analysed for the different plant states. Section 4 provides a general overview of acceptance criteria to be used in deterministic safety analysis for design and authorization of nuclear power plants, and describes the rules for determination and use of acceptance criteria. Section 5 provides guidance on verification and validation, selection and use of computer codes and plant models, together with input data used in the computer codes. Section 6 describes general approaches for ensuring adequate safety margins in demonstrating compliance with acceptance criteria for all plant states, with a focus on anticipated operational occurrences and design basis accidents. Section 7 provides specific guidance on performing deterministic safety analysis for each individual plant state. Section 8 includes guidance on the documentation, review and updating of deterministic safety analysis. Section 9 provides guidance on independent verification of safety assessments, including verification of deterministic safety analysis.

1.16. Annex I indicates additional applications of the computer codes used for deterministic safety analysis, besides nuclear power plant design and authorization.

Annex II indicates the frequency ranges of anticipated operational occurrences and design basis accident categories used in some States for new reactors.

2. GENERAL CONSIDERATIONS

OBJECTIVES OF DETERMINISTIC SAFETY ANALYSIS

2.1. The objective of deterministic safety analysis for nuclear power plants is to confirm that safety functions can be performed with the necessary reliability and that the necessary structures, systems and components, in combination where relevant with operator actions, are capable and sufficiently effective, with adequate safety margins, to keep the releases of radioactive material from the plant below acceptable limits. Deterministic safety analysis is aimed at demonstrating that barriers to the release of radioactive material from the plant will maintain their integrity to the extent required. Deterministic safety analysis, supplemented by further specific information and analysis (such as information and analysis relating to fabrication, testing, inspection and evaluation of the operating experience) and by probabilistic safety analysis, is also intended to contribute to demonstrating that the source term and the potential radiological consequences of different plant states are acceptable, and that the possibility of certain conditions arising that could lead to an early radioactive release or a large radioactive release can be considered as ‘practically eliminated’ (see para. 3.55).

2.2. The aim of deterministic safety analyses performed for different plant states is to demonstrate the adequacy of the engineering design, in combination with the envisaged operator actions, by demonstrating compliance with established acceptance criteria.

2.3. Deterministic safety analyses predict the response of the plant to postulated initiating events, alone or in combination with additional postulated failures. A set of rules and acceptance criteria specific to each plant state is applied. Typically, these analyses focus on neutronic, thermohydraulic, thermomechanical, structural and radiological aspects, which are analysed with appropriate computational tools. Computational simulations are carried out specifically for predetermined operating modes and plant states.

2.4. The results of computations are space and time dependent values of selected physical variables (e.g. neutron flux; thermal power of the reactor;

pressures, temperatures, flow rates and velocities of the primary coolant; loads to physical barriers; concentrations of combustible gases; physical and chemical compositions of radionuclides; status of core degradation or containment pressure; and source term for a release to the environment).

ACCEPTANCE CRITERIA FOR DETERMINISTIC SAFETY ANALYSIS

2.5. Acceptance criteria are used in deterministic safety analysis to assist in judging the acceptability of the results of the analysis as a demonstration of the safety of the nuclear power plant. The acceptance criteria can be expressed in general, qualitative terms or as quantitative limits. Three categories of criteria are recognized:

- (a) Safety criteria: Criteria that relate either directly to the radiological consequences of operational states or accident conditions, or to the integrity of barriers against releases of radioactive material, with due consideration given to maintaining the safety functions.
- (b) Design criteria: Design limits for individual structures, systems and components, which are part of the design basis as important preconditions for meeting safety criteria (see Requirement 28 of SSR-2/1 (Rev. 1) [1]).
- (c) Operational criteria: Rules to be followed by the operator during normal operation and anticipated operational occurrences, which provide preconditions for meeting the design criteria and ultimately the safety criteria.

2.6. In this Safety Guide, only safety acceptance criteria are addressed. These acceptance criteria, as approved by the regulatory body, may include margins with respect to safety criteria.

UNCERTAINTY ANALYSIS IN DETERMINISTIC SAFETY ANALYSIS

2.7. The use of uncertainty analysis in deterministic safety analysis is addressed in paras 6.21–6.29. Several methods for performing uncertainty analysis have been published (e.g. Ref. [7]). They include:

- (a) Use of a combination of expert judgement, statistical techniques and sensitivity calculations;
- (b) Use of data from scaled experiments;
- (c) Use of bounding scenario calculations.

APPROACHES TO DETERMINISTIC SAFETY ANALYSIS

2.8. Table 1 lists different options currently available for performing deterministic safety analyses with different levels of conservatism associated with the computer code used (see Section 5), the assumptions made about the availability of systems, and the initial and boundary conditions applied for the analysis.

2.9. Option 1 is a conservative approach in which both the assumed plant conditions and the physical models are set conservatively. In a conservative approach, parameters need to be allocated values that will have an unfavourable effect in relation to specific acceptance criteria. The conservative approach was commonly adopted in the early days of safety analysis to simplify the analysis and to compensate for limitations in modelling and knowledge of physical phenomena with large conservatisms. It was assumed that such an approach would bound many similar transients in a way that the acceptance criteria would be met for all bounded transients.

2.10. Experimental research has resulted in a significant increase in knowledge of physical phenomena, and the development of computer codes has improved the ability to achieve calculated results that correspond more accurately to experimental results and recorded event sequences in nuclear power plants. Owing to the improved capabilities of computer codes and the possible drawbacks of the conservative approach (e.g. potential masking of important phenomena,

TABLE 1. OPTIONS FOR PERFORMING DETERMINISTIC SAFETY ANALYSIS

Option	Computer code type	Assumptions about systems availability	Type of initial and boundary conditions
1. Conservative	Conservative	Conservative	Conservative
2. Combined	Best estimate	Conservative	Conservative
3. Best estimate plus uncertainty	Best estimate	Conservative	Best estimate Partly most unfavourable conditions
4. Realistic*	Best estimate	Best estimate	Best estimate

* For simplicity, the terms ‘realistic approach’ or ‘realistic analysis’ are used in this Safety Guide to mean best estimate analysis without quantification of uncertainties.

and conservatisms in different parameters potentially cancelling each other out), Option 1 is rarely used now and is not suggested for current safety analysis, except in situations in which scientific knowledge and experimental support is limited. Option 1 remains relevant, however, as it may have been used in legacy analyses.

2.11. Option 2 is a combined approach based on the use of best estimate models and computer codes instead of conservative models and codes (para. 6.12). Best estimate codes are used in combination with conservative initial and boundary conditions and with conservative assumptions made about the availability of systems, assuming that all uncertainties associated with the code models are well established and that the plant parameters used are conservative, based on plant operating experience. The complete analysis requires use of sensitivity studies to justify the selection of conservative input data. Option 2 is commonly used for design basis accidents and for conservative analysis of anticipated operational occurrences.

2.12. Option 3 is a ‘best estimate plus uncertainty’ approach. This allows the use of best estimate computer codes together with more realistic assumptions. A mixture of best estimate and partially unfavourable (i.e. somewhat conservative) initial and boundary conditions may be used, taking into account the very low probability that all parameters would be at their most pessimistic value at the same time. Conservative assumptions are usually made about the availability of systems. In order to ensure the overall conservatism required in analysis of design basis accidents, the uncertainties need to be identified, quantified and statistically combined. Option 3 contains a certain level of conservatism and is currently accepted for some design basis accidents and for conservative analyses of anticipated operational occurrences.

2.13. In principle, Options 2 and 3 are distinctly different types of analysis. In practice, however, a mixture of Options 2 and 3 is often employed. This is because the tendency is to use best estimate input data whenever extensive data are available and to use conservative input data whenever data are scarce. The difference between these options is the statistical combination of uncertainties.

2.14. Deterministic safety analysis performed in accordance with Options 1–3 is considered to be conservative, with the level of conservatism decreasing from Option 1 to Option 3 (see paras 2.9–2.13).

2.15. Option 4 allows the use of best estimate models and computer codes, and best estimates of system availability and initial and boundary conditions. Option 4 is appropriate for realistic analysis of anticipated operational occurrences aimed

at the assessment of control system capability (see paras 7.17–7.44) and in general for best estimate analysis of design extension conditions (see paras 7.45–7.67), as well as for the purpose of justifying prescribed operator actions in realistic analysis. Deterministic analysis for operating events that may necessitate a short term relaxation of regulatory requirements may also rely on best estimate modelling. More detailed information with regard to modelling assumptions applicable for different options is provided in Section 7.

SOURCE TERM FOR A RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT

2.16. An essential component of deterministic safety analysis is the determination of source terms for releases of radioactive material as a key factor for prediction of dispersion of such material in the environment and ultimately of radiation doses to plant staff and to the public as well as the radiological impact on the environment. The source term is the “amount and isotopic composition of *radioactive material* released (or postulated to be released) from a *facility*” [3]; and it is used “in modelling *releases* of radionuclides to the *environment*, in particular in the context of *accidents* at *nuclear installations* or *releases* from *radioactive waste* in *repositories*” [3].

2.17. To evaluate the source term from a nuclear installation, it is necessary to identify the sources of radiation, to determine the inventories of radionuclides that are produced and to know the mechanisms by which radioactive material can travel from the source through the installation and be released to the environment. Under accident conditions, source term evaluation requires simulation codes capable of predicting fission product release from fuel elements, transport through the primary system and containment or spent fuel pool building, the related chemistry affecting this transport and the form in which the radioactive material would be released.

2.18. The source term is evaluated for operational states and accident conditions for the following reasons:

- (a) To confirm that the design is optimized so that the source term is reduced to a level that is as low as reasonably achievable in all plant states;
- (b) To support the demonstration that the possibility of certain conditions arising that could lead to an early radioactive release or a large radioactive release can be considered to have been ‘practically eliminated’;

- (c) To demonstrate that the design ensures that requirements for radiation protection, including restrictions on doses, are met;
- (d) To provide a basis for the emergency arrangements² that are required to protect human life, health, property and the environment in case of an emergency at the nuclear power plant;
- (e) To support specification of the conditions for the qualification of the equipment required to withstand accident conditions;
- (f) To provide data for training activities with regard to emergency arrangements;
- (g) To support the design of safety features for the mitigation of the consequences of severe accidents (e.g. filtered containment venting and recombiners of combustible gases; see IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants [11]).

2.19. General rules presented in this Safety Guide for deterministic safety analysis also apply to determination of the source term. Aspects associated with the determination of the source term are introduced in several paragraphs in this Safety Guide to remind readers of the applicability of the general rules to this specific application.

3. IDENTIFICATION, CATEGORIZATION AND GROUPING OF POSTULATED INITIATING EVENTS AND ACCIDENT SCENARIOS

3.1. In accordance with the definition of “plant states (*considered in design*)” from SSR-2/1 (Rev. 1) [1], the plant states considered in the deterministic safety analysis should cover:

- (a) Normal operation;
- (b) Anticipated operational occurrences;
- (c) Design basis accidents;

² The application and the establishment of such arrangements are beyond the scope of this Safety Guide. Requirements regarding these arrangements are established in IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [8]; and recommendations are provided in IAEA Safety Standards Series Nos GS-G-2.1, Arrangements for Preparedness for a Nuclear or Radiological Emergency [9], and GSG-2, Criteria for Use in Preparedness and Response for a Nuclear or Radiological Emergency [10].

- (d) Design extension conditions, including sequences without significant fuel degradation and sequences with core melting.

3.2. The deterministic safety analysis should address all postulated initiating events originating in any part of the plant and having the potential to lead to a radioactive release to the environment, both on their own and in combination with possible additional failures, for example in the control and limitation systems³ and the associated safety functions. This includes events that can lead to a release of radioactive material not only from the reactor core but also from other relevant sources, such as fuel elements stored at the plant and systems dealing with radioactive material.

3.3. Where applicable, the possibility should be considered that a single cause could simultaneously prompt initiating events in several or even all of the reactors in the case of a multiple unit nuclear power plant, or spent fuel storage units, or any other sources of potential radioactive releases on the given site (para. 5.15B of SSR-2/1 (Rev. 1) [1]).

3.4. The deterministic safety analysis should address postulated initiating events that can occur in all modes of normal operation. The initial conditions should assume a steady state with normal operation equipment operating prior to the initiating event.

3.5. Each configuration of shutdown modes, including refuelling and maintenance, should be considered. For these modes, possible failures or other factors that could occur during shutdown and lead to increased risk should be considered, such as:

- (a) Inability to start some safety systems automatically or manually;
- (b) Disabled automation systems;
- (c) Equipment undergoing maintenance or repair;
- (d) Reduced amounts of coolant in the primary circuit and, for some modes, in the secondary circuit;
- (e) Instrumentation switched off or non-functional so that measurements are not made;
- (f) Open primary circuit;
- (g) Open containment.

³ In this Safety Guide, the term ‘control and limitation systems’ refers not only to the instrumentation systems for control and limitation of the plant variables but also to the systems for normal operation and those for anticipated operational occurrences actuated by them.

3.6. For postulated initiating events relating to the spent fuel pool, specific operating modes relating to fuel handling and storage should be considered.

3.7. Postulated initiating events taking place during plant operating modes of negligibly short duration may be excluded from deterministic safety analysis if careful analysis and quantitative assessment confirm that their potential contribution to the overall risk, including the risk of conditions arising that could lead to an early radioactive release or a large radioactive release, is also negligible. Nevertheless, the need to prevent or mitigate these events with appropriate procedures or means should be addressed on a case by case basis.

MANAGEMENT SYSTEM

3.8. The performance of deterministic safety analysis and the use of the results should take into account the recommendations of IAEA Safety Standards Series Nos GS-G-3.1, Application of the Management System for Facilities and Activities [12], and GS-G-3.5, The Management System for Nuclear Installations [13], for meeting Requirements 1–3 of SSR-2/1 (Rev. 1) [1] and the requirements established in IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [14].

NORMAL OPERATION

3.9. Deterministic safety analysis should include an analysis of normal operation, defined as operation within specified operational limits and conditions. Normal operation should typically include operating conditions such as:

- (a) Normal reactor startup from shutdown, approach to criticality and approach to full power;
- (b) Power operation, including full power and low power operation;
- (c) Changes in reactor power, including load follow modes and return to full power after an extended period at low power, if applicable;
- (d) Reactor shutdown from power operation;
- (e) Hot shutdown;
- (f) Cooling down process;
- (g) Cold shutdown;
- (h) Refuelling during shutdown or during normal operation at power, where applicable;

- (i) Shutdown in a refuelling mode or maintenance conditions that open the reactor coolant or containment boundary;
- (j) Normal operation modes of the spent fuel pool;
- (k) Storage and handling of fresh fuel.

3.10. It should be taken into account that, in some cases during normal operation, the main plant parameters change owing to transfer to different plant modes or changes in the plant power output. A major aim of the analysis for transients occurring during normal operation should be to demonstrate that the plant parameters can be kept within the specified operational limits and conditions.

POSTULATED INITIATING EVENTS

3.11. The prediction of plant behaviour in plant states other than normal operation (anticipated operational occurrences, design basis accidents and design extension conditions) should be based on a plant specific list of postulated initiating events, possibly combined with additional equipment failures or human errors for specific event sequences.

3.12. A list of postulated initiating events should be prepared. The list should be comprehensive to ensure that the analysis of the behaviour of the plant is as complete as possible, so that **“all foreseeable events with the potential for serious consequences and all foreseeable events with a significant frequency of occurrence are anticipated and are considered in the design”** (Requirement 16 of SSR-2/1 (Rev. 1) [1]).

3.13. The list of postulated initiating events should take due account of operating experience feedback, including, depending on the availability of relevant data, operating experience from the actual nuclear power plant or from similar plants.

3.14. The set of postulated initiating events should be defined in such a way that it covers all credible failures, including:

- (a) Failures of structures, systems and components of the plant (partial failure if relevant), including possible spurious actuation;
- (b) Failures initiated by operator errors, which could range from faulty or incomplete maintenance operations to incorrect settings of control equipment limits or wrong operator actions;
- (c) Failures of structures, systems and components of the plant arising from internal and external hazards.

3.15. All consequential failures that a given postulated initiating event could prompt in the plant should be considered in the analysis of the plant response as a part of the postulated initiating event. These should include:

- (a) If the initiating event is a failure of part of an electrical distribution system, the analysis for anticipated operational occurrences, design basis accidents or design extension conditions should assume the unavailability of all the equipment powered from that part of the distribution system.
- (b) If the initiating event is an energetic event, such as the failure of a pressurized system that leads to the release of hot water or pipe whip, the analysis for anticipated operational occurrences, design basis accidents or design extension conditions should include consideration of potential failure of the equipment that could be affected by such an event.
- (c) For internal hazards, such as fire or flooding, or for failures caused by external hazards, such as earthquakes, the definition of the induced postulated initiating event should include failure of all the equipment that is neither designed to withstand the effects of the event nor protected from it.

3.16. In addition to the set of initiating failures and consequential failures, other failures are assumed in deterministic safety analysis for conservatism (e.g. single failure criterion in design basis accidents) or for the purpose of defence in depth (e.g. common cause failure). A distinction should be made between these failures and the failures that are part of, or directly caused by, the postulated initiating event. Finally, some failures may be added to bound a set of similar events so as to limit the number of analyses.

3.17. The postulated initiating events should include only those failures (either initial or consequential) that directly lead to the challenging of safety functions and ultimately to threatening the integrity of barriers to releases of radioactive material. Therefore hazards, either internal or external (natural or human induced), should not be considered as postulated initiating events by themselves. However, the loads associated with these hazards should be considered a potential cause of postulated initiating events, including multiple failures resulting from these hazards.

3.18. Paragraph 5.32 of SSR-2/1 (Rev. 1) [1] states:

“Where the results of engineering judgement, deterministic safety assessments and probabilistic safety assessments indicate that combinations of events could lead to anticipated operational occurrences or to accident conditions, such combinations of events shall be considered to be design

basis accidents or shall be included as part of design extension conditions, depending mainly on their likelihood of occurrence.”

3.19. The set of postulated initiating events should be identified in a systematic way. This should include a structured approach to the identification of the postulated initiating events, such as:

- (a) Use of analytical methods such as hazard and operability analysis, failure modes and effects analysis, engineering judgement and master logic diagrams;
- (b) Comparison with the list of postulated initiating events developed for safety analysis of similar plants (ensuring that previously identified deficiencies are not propagated);
- (c) Analysis of operating experience data for similar plants;
- (d) Use of insights and results from probabilistic safety analysis.

3.20. Certain limiting faults (e.g. large break loss of coolant accidents, main steam or feedwater pipe breaks, and control rod ejection in pressurized water reactors or rod drop in boiling water reactors) have traditionally been considered in deterministic safety analysis as design basis accidents. These accidents should be considered because they are representative of a type of accident against which the reactor has to be protected. They should not be excluded from the category of design basis accidents unless careful analysis and quantitative assessment of their potential contribution to the overall risk, including to conditions arising that could lead to an early radioactive release or a large radioactive release, indicate that they can be excluded.

3.21. Failures occurring in the supporting systems that impede the operation of systems necessary for normal operation should also be considered as postulated initiating events if such failures ultimately require the actuation of the reactor protection systems or safety systems.

3.22. The set of postulated initiating events should be reviewed as the design and safety assessment proceed, as part of an iterative process between these two activities. The postulated initiating events should also be periodically reviewed throughout the lifetime of the plant, for example as part of a periodic safety review, to ensure that they remain valid.