Safety Reports Series

No. 102

Safety Analysis and Licensing Documentation for Nuclear Fuel Cycle Facilities

IAEA

International Atomic Energy Agency

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

https://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.

SAFETY ANALYSIS AND LICENSING DOCUMENTATION FOR NUCLEAR FUEL CYCLE FACILITIES

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

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

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".

SAFETY REPORTS SERIES No. 102

SAFETY ANALYSIS AND LICENSING DOCUMENTATION FOR NUCLEAR FUEL CYCLE FACILITIES

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2020

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

Printed by the IAEA in Austria
April 2020
STI/PUB/1892

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Safety analysis and licensing documentation for nuclear fuel cycle facilities / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2020. | Series: IAEA safety reports series, ISSN 1020–6450 ; no. 102 | Includes bibliographical references.
Identifiers: IAEAL 20-01299 | ISBN 978–92–0–107119–4 (paperback : alk. paper) | 978–92–0–162419–2 (pdf)
Subjects: LCSH: Nuclear facilities. | Industrial safety. | License agreements.
Classification: UDC 621.039.58 | STI/PUB/1892

FOREWORD

This publication aims to provide practical information on methods and practices for performing safety analysis and for the preparation of licensing documentation for nuclear fuel cycle facilities. The approaches provided can be applied to all types of nuclear fuel cycle facility with the use of a graded approach that is commensurate with the potential hazards posed by the facility.

The publication covers all the steps of a safety analysis, including hazard assessment; identification and selection of initiating events and acceptance criteria; types of safety analysis; evaluation of event sequences and consequences; selection of computational tools; and presentation of the results of the analysis. Information is also provided on the use of safety analysis findings in various technical areas such as design verification and safety classification of structures, systems and components, operational limits and conditions, ageing management, accident management and emergency preparedness and response.

With respect to licensing documentation, this publication focuses on providing information on the content of the safety analysis report, considering it as a high level document that incorporates the information required at various steps in the licensing process for nuclear fuel cycle facilities.

This publication also discusses various aspects of the use of a graded approach in performing safety analysis and the preparation of a safety analysis report. Various factors that need to be considered to ensure the quality of the safety analysis and the safety analysis report are also discussed.

The information, methods and calculations that are described in this publication can be used for performing the safety analysis and preparing the safety analysis report for a newly designed nuclear fuel cycle facility or for modifications or upgrades to an existing one. They can also be used for updating the safety analysis for the facility and the corresponding licensing documentation.

This publication elaborates on the requirements for safety analysis and licensing documentation that are established in IAEA Safety Standards Series No. SSR-4, Safety of Nuclear Fuel Cycle Facilities. The information provided in the present publication is not intended to replace or supersede any of the requirements or guidance provided in the relevant IAEA safety standards; rather, it is to be used in close conjunction with them. Security topics and emergency preparedness and response are beyond the scope of this publication.

The information in this publication will be useful to the operating organizations, regulatory bodies and other organizations involved in the safety of nuclear fuel cycle facilities, including designers and technical support organizations.

The IAEA wishes to thank all those who contributed to the drafting and review of this publication. The IAEA officers responsible for this publication

were A.M. Shokr, R. Gater and M. Nepeypivo of the Division of Nuclear Installation Safety.

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

IAEA Safety Standards Series No. SSR-4, Safety of Nuclear Fuel Cycle Facilities [1], establishes requirements for the demonstration of facility safety based on the safety analysis process as established in IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [2], and the preparation of documents known as the licensing documentation (or safety case). This Safety Report elaborates on the safety analysis process established by Requirements 1, 14–16 and 19–22 of SSR-4 [1].

Safety analysis is an analytical study, undertaken to demonstrate that the facility design meets safety and regulatory requirements and that the design is based on the application of sound engineering practices, research and feedback from operating experience (para. 6.10 of SSR-4 [1]). Systematic and recognized methods of deterministic analysis are required to be used for nuclear fuel cycle facilities, complemented by probabilistic assessments where appropriate, in accordance with a graded approach [1].

The entire range of conditions for which a nuclear fuel cycle facility is designed, according to established design criteria, forms its design basis. Within the design basis, a range of conditions and events caused by credible technical failures, operator errors or human induced or natural events, whose potential consequences may be significant in terms of facility safety or environmental protection, are explicitly taken into account according to established criteria. The safety analysis process, therefore, includes the identification of accident conditions, determination of the facility's response to a range of postulated initiating events (PIEs) covering all facility states and the potential combination of PIEs, evaluation of event consequences and judgement of the acceptability of the results against the pre-established acceptance criteria.

The conditions caused by these events may be classified as anticipated operational occurrences (AOOs), design basis accidents (DBAs) or design extension conditions (DECs) on the basis of engineering judgement and the results of deterministic and probabilistic safety analyses. The DECs are accident conditions that are not considered for DBAs, but are considered in the design process of the facility, and for which releases of radioactive material are kept within acceptable limits. The DECs are used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their radiological or associated non-radiological consequences. Analysis of DBAs and DECs is also used to establish plans for emergency preparedness and response for the facility.

The safety analysis process for nuclear fuel cycle facilities is based on the same general principles as for other types of nuclear facility, as specified in IAEA Safety Standards Series No. SSG-2, Deterministic Safety Analysis for Nuclear Power Plants [3], IAEA Safety Standards Series No. SSG-20, Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report [4] and Safety Reports Series No. 55, Safety Analysis for Research Reactors [5]; however, its implementation at a specific facility could differ in several characteristics.

In accordance with Requirement 1 of SSR-4 [1], the safety of a nuclear fuel cycle facility has to be demonstrated through a set of documents known as the licensing documentation (or safety case), and this documentation has to include an adequate safety analysis report and the operational limits and conditions, as well as any other information required by the regulatory body. The licensing documentation provides the basis for the safety of the facility throughout its lifetime, and needs to be updated periodically to account for, among other things, modifications made to the facility and operating experience feedback. The IAEA Safety Standards covering the format and content of the safety analysis report for other types of nuclear facility are SSG-20 [4] and IAEA Safety Standards Series No. SSG-61, Format and Content of the Safety Analysis Report for Nuclear Power Plants [6]. However, different national regulatory requirements exist regarding the content of licensing documentation for nuclear fuel cycle facilities. Most of these national requirements define, in line with Requirement 1 of SSR-4 [1], the safety analysis report and the operational limits and conditions as key components of the documentation.

This publication addresses the features of safety analysis and licensing documentation specific to nuclear fuel cycle facilities, and has to be used in conjunction with the appropriate IAEA Safety Standards for nuclear fuel cycle facilities [1, 2, 7–12].

1.2. OBJECTIVE

The objective of this publication is to provide information on methods and practices, based on the IAEA safety standards and current international good practice, for performing safety analysis and preparing licensing documentation for nuclear fuel cycle facilities.

The approaches provided in this publication can be applied to different types of nuclear fuel cycle facility with use of a graded approach that is commensurate with the potential hazards and risks posed by the facility. The design and operational features of a specific nuclear fuel cycle facility also need to be considered as described in this publication.

The information in this publication is of practical use to the operating organizations, regulatory bodies and other organizations involved in the safety of nuclear fuel cycle facilities, including designers and technical support organizations.

1.3. SCOPE

This publication covers two topics: the safety analysis of nuclear fuel cycle facilities and their licensing documentation.

With respect to safety analysis, this publication covers all the steps in performing a safety analysis and in presenting and applying its results. A systematic methodology for performing safety analysis, as presented in this publication, comprises provisions for the establishment of acceptance criteria, hazard evaluation, identification and selection of PIEs, and transient and accident analysis, and also includes modelling of event sequences and consequences. This publication deals with external events and internal events originating in a nuclear fuel cycle facility or in its associated systems. External events are discussed in order to show how they fit within the overall safety analysis of a facility. The presentation of the results of the safety analysis as well as use of the results in the design and operational safety of the facility, including design verification, derivation of operational limits and conditions, safety classification of structures, systems and components (SSCs), modifications, ageing management and consideration of lifetime extension, are also covered.

Evaluation of source terms and modelling of the dispersion of radioactive material are beyond the scope of this publication.

The calculations and methods described in this publication may be used as deemed necessary for the preparation of safety analyses of newly designed nuclear fuel cycle facilities and for modifications and upgrades to existing ones, and may also be used for updating or reassessing previous safety analyses of operating facilities in the context of periodic safety reviews or consideration of long term operation.

With respect to licensing documentation, this publication provides an indication of the content of the safety analysis report, considering that the safety analysis report is a high level document that incorporates information (or cites documents) required at various steps in the licensing process (e.g. environmental impact assessment, radiation protection programme, emergency plans, management system). Information on methods used in the preparation of these documents is out of the scope of this publication. The authorization of these facilities from a nuclear security point of view is also out of its scope.

The establishment of a management system for safety analysis and licensing documentation for nuclear fuel cycle facilities is addressed in this publication.

This publication is intended to be applicable to all types of nuclear fuel cycle facility covered by the scope of SSR-4 [1]. Specific features of individual facility types are taken into consideration in the text of the main body or in the annexes, as appropriate.

The following processes are covered by this publication:

— Processing of uranium and thorium ores;
— Conversion and enrichment of uranium;
— Reconversion and fabrication of nuclear fuels of all types;
— Interim storage of fissile material and fertile material before and after irradiation;
— Reprocessing of spent nuclear fuel and breeder materials from thermal reactors and fast reactors;
— Associated waste conditioning, effluent treatment and facilities for interim storage of waste;
— Separation of radionuclides from irradiated thorium and uranium;
— Related research and development.

Some sites have multiple nuclear fuel cycle facilities, with the possibility for a specific facility to interact with other facilities nearby. This publication highlights those aspects of the safety analysis and licensing documentation that may be treated differently for such sites. However, the safety analysis and licensing documentation of multifacility sites are out of the scope of this publication.

Guidance provided here, describing good practices, represents expert opinion but does not constitute recommendations made on the basis of a consensus of Member States.

1.4. STRUCTURE

Sections 2, 3 and 4 deal with safety analysis. Section 2 describes general considerations in performing safety analysis, including types of safety analysis, methods for analysis of transients and accidents, acceptance criteria, specific features of nuclear fuel cycle facilities affecting safety analysis and use of a graded approach. Section 3 discusses the steps taken in performing safety analysis, including the identification and selection of PIEs and methods for grouping events, evaluation of event sequences and their consequences as well as comparison of the results against acceptance criteria. Section 3 also outlines the rules for analysis along with expectations for computational models and for presentation of the results of the safety analysis. Section 4 describes some applications of the results of safety analysis, including design verification, derivation of operational limits and conditions, safety classification of SSCs, modifications, ageing management and evaluation of long term operation. Section 5 provides information on the content of the licensing documentation, information needed at various steps of the licensing process and the content of the safety analysis report for nuclear fuel cycle facilities. Section 6 provides information for the establishment of a management system for safety analysis and for licensing documentation for nuclear fuel cycle facilities.

The Appendix provides indicative content of the safety analysis report for nuclear fuel cycle facilities. Annex I provides examples of factors to be considered and areas subjected to use of a graded approach in safety analysis of nuclear fuel cycle facilities. Annex II provides examples of factors to be considered and techniques used in hazard identification, while Annex III and Annex IV provide, respectively, a list of PIEs and examples of DECs in nuclear fuel cycle facilities. Annex V provides examples of the rules for the safety analysis of nuclear fuel cycle facilities. Annex VI provides considerations and examples of items to be covered by limiting conditions for the safe operation of nuclear fuel cycle facilities. Annex VII presents examples of the documentation to be submitted to the regulatory body in the licensing process for nuclear fuel cycle facilities.

2. GENERAL CONSIDERATIONS IN SAFETY ANALYSIS OF NUCLEAR FUEL CYCLE FACILITIES

2.1. PURPOSE OF SAFETY ANALYSIS

The analysis performed to demonstrate the safety of a facility can be used for different purposes and in a number of different areas, such as design, licensing, accident management and emergency planning, and as a part of operating experience feedback programmes. These applications of the safety analysis are summarized below. SSR-4 requires a site evaluation and environmental impact assessment to be provided in support of the safety analysis in accordance with a graded approach. These aspects are covered by other relevant IAEA publications, including IAEA Safety Standards Series No. NS-R-3 (Rev. 1), Site Evaluation for Nuclear Installations [13].

The processes for performing safety analysis for different purposes are similar. However, it is necessary to define the purpose of the safety analysis and its scope early in the process, because it can influence various aspects of the analysis, including the types of calculation required and the resources needed to obtain data and perform the analysis.

2.1.1. Safety analysis for design

Safety analysis is an integral part of the design process. The main objective of safety analysis for design is to confirm that the design will be able to meet relevant safety requirements. The safety analysis is best conducted in parallel with the design process, with iteration between the design and the analysis. The scope and level of detail of the analysis should increase as the design progresses so that the safety analysis reflects such progress. When conducted in parallel with design, the safety analysis is used to guide the design of a new facility or modifications to the design of an existing one. All the challenges that the facility may be expected to meet during its operational life are to be considered during the design process. These challenges are selected from all the foreseeable conditions and events related to factors such as facility stage or states, site characteristics, design requirements and modes of operation.

Facility designers recognize that challenges to all levels of defence in depth may occur, and design measures are provided to ensure that the safety functions are accomplished and the safety objectives can be met. Safety analysis will examine the necessity and adequacy of the application of design principles such as functional and physical independence and diversity. These challenges are selected from the PIEs, which are identified and selected for analysis as described in Section 3.

Safety analysis is performed in design to assist in setting characteristics such as equipment sizing and determining process parameters such as pressure, temperature, flow rates and electrical power. Safety analysis in design is also used to check at an early stage that the design will meet the national licensing requirements (including, for example, dose to workers and the public). Therefore, safety analysts need to work closely with facility designers to ensure that the design configuration of the facility is optimized in terms of safety and cost.

A safety analysis process is also needed for assessing the safety of modifications to an existing facility. Safety analysis for modification of an existing facility's design may be of limited scope compared to that for a new facility. As is the case for new designs, safety analysis for modifications is best conducted in parallel with their design.

2.1.2. Safety analysis for licensing and periodic safety reviews

Compliance with all applicable regulations and other relevant safety standards has to be demonstrated to the regulatory body by means of safety analysis for different stages of the facility's lifetime. The objective of safety analysis for licensing is to demonstrate to the regulatory body that the facility's design features and operational limits and conditions have been selected to ensure that the main safety functions of the facility are met and to ensure the protection of workers, the public and the environment from unacceptable radiological and non-radiological consequences. Safety analysis for licensing is used to provide evidence that the level of safety that can reasonably be achieved in the site evaluation, design, construction, commissioning, operation, modification, decommissioning and release from regulatory control of the nuclear fuel cycle facility is acceptable and in accordance with the national regulatory requirements.

Safety analysis supports licensing for the whole lifetime of the facility. This includes support of the application, siting and construction of the facility, operating licences and regulatory approval of modifications. At each licensing stage, the safety analysis has to reflect the actual status of the facility (e.g. as constructed, as built facility).

Furthermore, safety analysis is also performed as part of the periodic safety reviews of operating facilities. Periodic safety reviews are used in many countries as a basis for decisions on the extension of the lifetime of facilities or for renewal of their operating licences. Updating of the safety analysis within a periodic safety review is required by para. 4.27 of SSR-4 [1] to consider changes in the site characteristics, changes in the utilization programme (particularly for research and development facilities), the cumulative effects of ageing and modifications, changes to procedures, feedback from operating experience and technical developments. Safety analysis for relicensing may require new calculations, especially when new evidence arises from research and development or from operating experience feedback. The calculations could include the use of new computer codes or methods.

Safety analysis is also performed to support the authorization of facility modifications. The scope of such analysis typically corresponds to the safety significance of the modification.

2.1.3. Safety analysis supporting emergency planning and accident management

The safety analysis describes the facility's behaviour in accident conditions (including DECs) and provides the inputs necessary to specify the operator actions to be taken in such conditions. So, safety analysis is performed to support the development of arrangements for emergency preparedness and response. Safety analysis also plays an important role in the development and review of accident management guidelines, if applicable. A best estimate approach with realistic assumptions is normally used for safety analysis that supports accident management and emergency planning.

Owing to the limited possibility of using real transients for the validation of emergency operating procedures, analyses by computer codes are normally used to support the development and validation of these procedures and guidelines. The results of this type of safety analysis could also be used for identification of the hazard category of the facility and therefore its associated emergency class.

2.1.4. Analysis of operating experience

Safety analysis may be performed to support the investigation of incidents and accidents, and within programmes of operating experience feedback to identify the root causes of events occurring during the operation of the facility. This has the aim of establishing measures (organizational or technical) to prevent reoccurrences of such events and to identify solutions to potential safety issues. This also could be performed as a part of periodic safety reviews. The availability/use of operating experience from other similar facilities, including events reported to the Fuel Incident and Analysis System [14], can be particularly useful.

2.2. TYPES OF SAFETY ANALYSIS AND COMPUTATIONAL TOOLS

2.2.1. Types of safety analysis

There are two basic types of safety analysis: deterministic and probabilistic. The safety analysis of nuclear fuel cycle facilities is mainly performed using deterministic safety analysis. Probabilistic methods may be used as a complementary means for identifying weakness in the design and for improving the analysis. However, the probabilistic safety assessment (PSA) employed for nuclear fuel cycle facilities is quite different from that for nuclear reactors, as explained below.

Both deterministic analysis and PSA for nuclear fuel cycle facilities need to take account of the assessments of human performance necessary at a number of workplaces under varying process conditions. Analyses of human performance are used in the design of workplaces to optimise human performance and to prevent human errors. Human reliability analysis can be used to quantify the likelihood of human error for these tasks.

2.2.1.1. Deterministic safety analysis

Deterministic safety analyses are required to demonstrate adequate fulfilment of safety functions by the design, to ensure that uncontrolled exposure and release of radioactive and associated hazardous materials are prevented for all facility states. The analyses are also used to demonstrate the validity of the operational limits and conditions. Deterministic safety analyses are required to determine the characteristics of potential overexposures and releases (source terms) for different facility conditions [1]. In general, the principles of deterministic analysis described below are similar to those used for the analysis of reactors [3–5].

The aim of the analysis is to demonstrate the adequacy of the engineering design combined with the envisaged operator actions, by demonstrating compliance with established acceptance criteria. Deterministic safety analyses are performed to predict the response of the facility to PIEs, alone or in combination with additional postulated failures. A set of rules and acceptance criteria specific to each facility is applied (see Section 3.3 of this publication).

Different levels of conservatism could be applied to the computer codes used, the assumptions made about availability of systems, the initial and boundary conditions and the data applied for the analysis.

The degree of conservatism is commensurate with the safety analysis objectives and is dependent on the facility states to be analysed. Either a conservative or a best estimate approach with uncertainty analysis can be used for safety analysis in design and for licensing, while a best estimate approach with realistic assumptions is normally used for safety analysis that supports accident management and emergency planning.

A conservative approach has the following characteristics:

— It uses conservative or bounding data (i.e. parameters, initial conditions and assumptions about availability of equipment, accident sequences and conditions are chosen to give pessimistic results in relation to specific acceptance criteria).
— It uses models in a conservative manner (i.e. computational models that provide conservative estimates for physical processes, conservative input data or both). This also applies to use of correlations.
— It uses physically based models of behaviour, as distinct from statistical (or heuristic) models.
— It considers the most unfavourable facility configuration (e.g. maximum radioactive material inventory, spent fuel throughput, enrichment and burnup, acidity and temperature of spent fuel).

- It assumes that operator actions do not take place for a predefined period of time after initiation of an event.
- It considers uncertainties associated with the parameter of interest (e.g. geometric and material tolerances, chemical and physical changes in materials, errors in measurements of parameters such as temperature and pressure, and uncertainties in correlations).

It is important to note that the choice of overly conservative methods or assumptions can restrict the selection of valid options and unnecessarily limit the range of usefulness of the facility. The level of conservatism to be applied also needs to be proportional to the understanding of the physical phenomena and to the capability of the computational models. The analyst needs to have knowledge of and experience with the models used for accident analysis. Sensitivity studies are usually performed to supplement deterministic analysis. These studies assess the effects of changes to assumptions on the analysis results, such as identifying the worst single failures in various systems, or assessing the impact of using simplified models instead of more accurate and sophisticated approaches (requiring significant effort in the calculations). Sensitivity studies, with systematic variations in input variables or modelling parameters, are important to confirm that there are no 'cliff edge' effects (i.e. abrupt changes in phenomena as a result of small changes in input).

Operating experience may be used to verify the level of conservatism in the analysis (for example, the material corrosion allowances being considered may not conform to the actual status during operation of the facility). In a conservative analysis, the effect of any uncertainty has to be minimized by allowing a significant margin of safety, or eliminated based on the results of research and development.

In contrast, best estimate approaches provide a more realistic simulation of the physical process to a level that is proportional to the current state of knowledge. Some uncertainty is inevitable in the analysis of rare events. Best estimate approaches allow the review of the existing margins or limits on facility transient scenarios in relation to the safety analysis. Best estimate approaches are usually supported by the analysis of uncertainties (including uncertainties in computer code modelling and facility data).

2.2.1.2. Probabilistic safety assessment

PSA combines the likelihood of initiating events and potential scenarios in the development of sequences and their consequences into an estimation of source terms and risks from a nuclear fuel cycle facility. Given the distribution of materials throughout the facility, importance of human factors and other facility-wide considerations, and lack of reliability data, full scope PSA for nuclear fuel cycle facilities is not yet fully developed. However, PSA tools could be used to complement deterministic analysis for certain types of facility or activity. PSA tools combining analyses of the frequency of failure of components and systems (including human response) with estimates of consequential releases can allow comparisons with imposed risk targets to ensure balance in the safety argument.

Best estimate codes and data are normally used in PSA, since its focus is to provide realistic answers. The results of the analysis, however, may be bounded by the results of conservative deterministic analysis. The approaches and procedures for conducting PSA studies for non-reactor nuclear facilities are presented in Ref. [15].

2.2.1.3. Quantitative risk assessment

Quantitative risk assessment (QRA) is a generic risk based methodology that is used for the safety analysis of nuclear fuel cycle facilities. The methodology was initially developed for use in the petrochemical industry and is primarily utilized as a design tool. The use of QRA in the operational stage has mainly been limited to the assessment of the effect of major modifications.

QRA is different from PSA. Full scope PSA tends to be deep, combining a large number of individual risks that may interact with each other. PSA is tightly linked to a root cause (e.g. earthquake): common cause initiators are assessed and support systems (e.g. cooling systems) are explicitly included. Conversely, QRA usually includes consideration of risks to workers, the facility and the environment. Different types of risk may not be linked and the depth of analysis could be shallow, such as an exposure/consequence curve for a specific chemical. Generally, QRA covers a much wider range of events and consequences than PSA. The QRA may not be linked to an identifiable root cause and the consideration of uncertainty may be much simpler than in PSA.

If QRA is used, consequence analysis is carried out to identify the magnitude of the effect (e.g. the area covered by the chemical or radiological release, or the dose–distance relationship for external exposure) for an identified accident scenario. The last step of QRA is the combination of the likelihood of the occurrence of that event with the consequence, which will quantify the risk involved.

The results from PSA and QRA tools can be compared with the safety acceptance criteria established in advance. This may result in either accepting the design or operation of SSCs or incorporating additional safety features or mitigation measures to ensure that risks are balanced and within acceptable limits.

2.2.2. Calculation tools and data

Various calculation tools, including computer codes, are widely used for the safety analysis of nuclear fuel cycle facilities. The codes are employed for different applications, including the modelling of radiation protection and the dispersion of radioactive materials and chemicals, dose and release assessment, criticality safety analysis and analysis of fire safety, as well as for the development of emergency arrangements. The calculation methods and computer codes that are used to carry out the safety analysis are required to be verified, tested and benchmarked as appropriate to build confidence in their use and their suitability for the intended application (see para. 4.14 of GSR Part 4 (Rev. 1) [2]). The applicability of calculation tools to the analysed event and conditions needs to be demonstrated.

More detailed information on approaches to code qualification, verification, validation and documentation as applied to the safety analysis of nuclear reactors can be found in Refs [3–5].

Important steps in developing safety analysis are the collection of input data for the facility under consideration and the collection of the necessary documentation and other reliable sources of data to perform the safety analysis. Information relating to the site evaluation, design and operation of the facility, as well as its construction, commissioning and decommissioning, is required to support safety analysis at various stages of facility licensing.

Sources of data for safety analysis include the following:

— Documentation on facility design (including drawings);
— Technical specifications of equipment;
— Documentation gathered during the construction and commissioning of the facility;
— Operational documentation for the facility (e.g. operating instructions and records);
— ‘As built’ facility documentation.

The data necessary to produce the estimates and their associated uncertainties are collected and treated appropriately, and have to conform with the actual conditions around the site and inside the facility.

Parameters necessary for the modelling of potential accident sequences include phenomenological parameters relating to the amount, form and transport of radioactive material, and accident sequence specific data as required to predict damage to barriers and subsequent release of radioactive or other material to operating areas and to the environment. Data are also required relating to the modes of worker exposure to radioactive or toxic material (external and internal), as well as relating to the off-site migration of radioactive material and its uptake by members of the public or environmental receptors.

Applicable limits for the facility parameters that are used as initial and boundary conditions are identified. Recorded operational data could also be included, where applicable. This information covers SSCs, site specific characteristics, experience feedback data and off-site interfaces.

Uncertainties in the measurement or evaluation of the parameters need to be considered in the safety analysis.

All documents and other data sources used need to be clearly identified and cited. The need for and scope of the programmes for data collection, and the extent and detail of the specification and validation of the input data, as well as the procedure for documenting the sources of data and data archiving, are driven by the use of a graded approach.

2.3. ACCEPTANCE CRITERIA FOR SAFETY ANALYSIS

Acceptance criteria are required to be established for all facility states (normal operation, AOOs, DBAs and DECs) and used to judge the acceptability of the results of the safety analysis. For the design of items important to safety, acceptance criteria in the form of engineering design rules may be used (see para. 6.35 of SSR-4 [1]). Acceptance criteria need to be defined prior to starting the analysis, to allow judgements on safety. Compliance with acceptance criteria is usually demonstrated by deterministic safety analysis.

Acceptance criteria may vary from one State to another but have to establish limits for the protection of workers, the public and the environment from undue hazards in all stages and states of a facility. These criteria must be sufficient to meet the fundamental safety objective, apply the fundamental safety principles and meet the requirements of national regulatory bodies as well as the requirements of the nuclear fuel cycle facility's designers and operating organizations. Acceptance criteria are defined by the national regulations and national and international standards. They may also be proposed by the operating organization and approved by the regulatory body.

The acceptance criteria could include applicable industrial codes and standards, such as those for criticality safety, radiation protection, structure and mechanical design, and fire and explosion. Criteria other than those related to nuclear safety are also used, including those related to industrial safety and for environmental protection.

Acceptance criteria for safety analysis may include the following:

— Numerical limits on the values of predicted parameters;
— Conditions for facility states during and after an accident;
— Performance requirements for systems;
— Requirements on the need for, and the ability to credit, operator actions.

Acceptance criteria could be established at two levels: basic or specific.

Basic acceptance criteria are usually defined as limits and conditions set by a regulatory body. They are aimed at achieving a high level of safety and environmental protection. To demonstrate the safety of the facility, the following acceptance criteria are considered:

— Radiation and chemical exposure of workers and the public (radiological and conventional safety acceptance criteria) are within prescribed limits.
— Integrity of barriers against the release of radioactive or hazardous chemical material (e.g. means of confinement, shielding, cooling system) is maintained.
— Safety systems are able to perform their intended safety function, directly or indirectly (or operators are able to perform a safety action) in accident conditions.
— Large releases and early releases of radioactive material are practically eliminated (for new designs).

Radiological and conventional safety acceptance criteria normally include criteria such as the following:

— Radiological acceptance criteria:
  - Annual effective dose limits for workers and for members of the public;
  - Dose constraints and targets (individual and collective);
  - Dose limits for intervention in accident conditions;
  - Maximum allowable release to the environment.
— Conventional acceptance criteria:
  - Limits to control the chemical exposure;
  - Explosive and flammability limits.

For chemical effects on the public, the national criteria for the chemical industry or the internationally established system of Occupational Exposure Limits for normal operation and Acute Exposure Guideline Levels [16, 17] for accidents are used.

Specific acceptance criteria are often developed in addition to the basic criteria. Typically, they are used to confirm that there are adequate margins to the authorized limits established within the regulatory framework to allow for uncertainties. These may be derived exposure limits (e.g. on contamination levels) relating to exposure of workers, members of the public and the environment. Or they may be specific design or process parameters intended for safety. They may be developed by the designer or operating organization and approved by the regulatory body; or they may be established by the regulatory body. Specific criteria cover various conditions (AOOs, DBAs and DECs) and are normally developed based on physical evidence and well understood phenomena to limit the damage to different safety barriers. Compliance with these criteria ensures the prevention of unacceptable exposure or the release of radiological and other hazardous material in these conditions. Typical examples of such types of acceptance criteria include the following:

— Numerical limits on the values of the variables (e.g. temperature, pressure);
— Conditions for facility states during and after an accident (e.g. achievement of a long term safe state);
— Performance requirements for SSCs;
— Requirements for operator actions, with account taken of the specific accident environment (e.g. the reliability of the alarm system).

In establishing acceptance criteria, care needs to be taken to ensure that the initiating event does not escalate to a more serious condition without occurrence of a further independent failure.

2.4. FEATURES OF NUCLEAR FUEL CYCLE FACILITIES AFFECTING SAFETY ANALYSIS

The objectives of safety analysis for nuclear fuel cycle facilities are the same as for other nuclear installations, such as nuclear power plants and research reactors, as the fundamental safety objective, fundamental safety principles and criteria are the same for all these facilities. However, the design and operational features of nuclear fuel cycle facilities necessitate specific considerations when performing safety analysis, including regarding the selection of the approach to analysis and calculation methods. The following specific features of nuclear fuel cycle facilities need to be considered when performing a safety analysis:

— They employ a great diversity of technologies and processes, some of which can be unique. The nature and diversity of the processes associated with the facilities result in a broad range of hazardous conditions and possible accident conditions.

— They are characterized by a wide diversity of radioactive materials, including fissile material, and other hazardous materials that are toxic, corrosive, combustible or reactive, and that are mostly in dispersible forms (gases, powders and liquids). There is potential for a wide range of events that could lead to radiological and non-radiological impacts on people and the environment.
— Radioactive and other hazardous materials are often processed through a series of interconnected units and consequently can be found throughout the entire facility and transferred between vessels in different parts of the processes.
— They have a higher potential for nuclear criticality accidents compared to other nuclear installations.
— They are often characterized by frequent changes in configuration (including equipment and processes) and operation purpose that are necessary for production campaigns, new product development, and research and development.
— Operations at the facilities generally require more operator intervention than those at nuclear reactors, which may result in specific hazards for personnel. For instance, operator interaction with nuclear material in some routine processes may include the handling and transfer of nuclear material.
— Various forms and types of safety barriers between radioactive inventories and operators can make operators more vulnerable than at reactors (e.g. work in gloveboxes).
— The majority of nuclear fuel facilities worldwide were designed several decades ago and they may not now fully conform to the current standards and regulations. Many of these facilities also lack adequate documentation on their design and information on operating history.

These features of fuel cycle facilities require additional efforts and specific methods of analysis.

Owing to the nature of nuclear fuel cycle facilities it is important to ensure good coordination between the various regulatory bodies (e.g. nuclear safety, nuclear security and environmental). This is mainly to ensure that different regulations, including those related to protection against chemical hazards and environmental protection, are applied in a consistent manner, including with respect to safety analysis acceptance criteria and in the licensing process.

Event sequences tend to be much simpler for many nuclear fuel cycle facilities compared to reactors, and require less complex analysis. However, these events vary in nature and in potential consequences, particularly for workers, leading to fewer opportunities for grouping and ‘bounding’ these events as in the safety analysis of nuclear reactors. The specific hazards associated with nuclear fuel cycle facilities require the selection and analysis of specific PIEs. Further, there is little benefit in considering generic lists of PIEs owing to the huge differences between the varieties of nuclear fuel cycle facilities. The commonly used approach is to perform an analysis that is specific to the facility and cross-check against standard lists occasionally.

The models representing details of the accident progression, such as fault trees, are mostly much simpler for nuclear fuel cycle facilities than they are for nuclear reactors. However, these analyses usually contain a large number of disconnected event sequences.

Many nuclear fuel cycle facilities rely on a combination of static and dynamic containment for the confinement of radioactive material or hazardous chemicals. This inherently creates potential exposure pathways to the environment under abnormal operations and accident conditions. Releases of radioactive material or hazardous chemicals by different pathways can result in significant contamination and exposure, so potential intakes of radioactive material require specific consideration. Various factors such as the amount and rate of release of the radioactive material or hazardous chemicals, the distance between the individuals exposed and the source of the release, pathways for the transport of material to the individuals and exposure times need to be carefully considered in the safety analysis.

Chemical processing of materials may result in an inadvertent release of chemically hazardous material. In some Member States, release of hazardous chemicals and its consequences have to be covered by the safety analysis. This leads to the need for the use of specific analysis techniques (e.g. QRA) and acceptance criteria [16, 17]. For facilities with chemical hazards, the methods and approaches for analysis of industrial chemical facilities can be applied to nuclear fuel cycle facilities.

Fire and explosive risk could be significant for some nuclear fuel cycle facilities owing to the presence of organic solutions and combustible gases or operation under pressure or at high temperature. Therefore, fire and explosion hazards need to be covered by the safety analysis. Such an analysis normally has to cover identification of the causes of fires, assessment of potential consequences and estimation of the frequency or probability of occurrence of fires, as appropriate. It also needs to review the effectiveness of measures for eliminating and mitigating this type of risk.

Another important feature of nuclear fuel cycle facilities that affects safety analysis is that the number of event sequences leading to harmful consequences can be large and these consequences can be fatal for workers in the vicinity (e.g. in the case of criticality events), and associated non-radiological hazards can be significant (e.g. releases of UF_6 vapour).

As discussed earlier, human errors and human reliability play a more significant role in many event sequences and require greater consideration in the safety analysis of nuclear fuel cycle facilities compared to the safety analysis of other nuclear facilities. Specific rules for safety analysis are needed for operations with significant reliance on operator intervention, where accident scenarios could be initiated by the operators or where their intervention could make the accident worse. Designs that take account of human factors in operations and maintenance are usually more efficient and less prone to the types of anomaly that challenge safety systems and reduce defence in depth. In addition, human errors in maintenance are mostly not facility specific and operating experience feedback in this regard could be useful in the safety analysis of various types of nuclear fuel cycle facility.

The specific features of nuclear fuel cycle facilities discussed above need to be carefully considered in the identification of hazards, identification and selection of PIEs, and analysis of accident sequences. The transient modelling and methods used for the analysis need to take account of these possibilities.

Given the shortage of reliability data of sufficient breadth and quality, including data on human reliability, PSA tools can be used only to complement deterministic analysis for certain types of facility or activity. Consequently, QRA and human reliability analysis can be used for the analysis of risks in nuclear fuel cycle facilities.

It is also important to note that safety analysis in nuclear fuel cycle facilities requires a wide range of expertise (from facility designers, the operating organization and the regulatory body) owing to the nature and variety of hazards associated with these facilities. When safety analysis (or part thereof) is outsourced (through designers, contractors or technical support organizations), it is essential that an operating organization, in carrying out its prime responsibility for safety, own the results of the analysis and develop human resources to understand its basis so that the organization is able to implement the results of the analysis and to review and update it throughout the facility's lifetime.

2.5. USE OF A GRADED APPROACH IN SAFETY ANALYSIS

Nuclear fuel cycle facilities are of different types and sizes. These differences result in differences in the potential hazards posed by these facilities. Some types of facility can be shown to have a low hazard potential (e.g. research and development facilities with very low radioactive inventories and low criticality risk, and facilities handling natural uranium). Other nuclear fuel cycle facilities (e.g. reprocessing facilities or spent fuel storage facilities) may have a potential risk that is comparable to that for nuclear power plants, when the size of the inventory is considered. The description of the safety analysis and application of its results provided in this publication is applicable to all nuclear fuel cycle facilities. However, considering these variations in potential hazards, the safety analysis process needs to be commensurate with the potential hazard of the facility in accordance with the use of a graded approach.

It is important to mention that use of a graded approach in the application of a safety requirement does not mean relief from meeting this safety requirement; rather, the approach is used to determine the most appropriate way to implement the safety requirement. This mainly involves the optimization of the resources needed to perform such analysis and the highlighting of areas where additional resources and safety oversight may be needed.

A graded approach is applied, for example, by considering, using sound engineering judgement, the safety and operational importance of the topic and the maturity and complexity of the area involved. An adequate safety analysis is a requirement for all nuclear fuel cycle facilities regardless of their potential hazards, and such a need cannot be graded. However, some aspects of safety analysis may be subject to grading, including the scope, extent and details of the analysis, and the required human and financial resources, which may be significantly lower for low radioactive inventory nuclear fuel cycle facilities (typically ‘front end’ facilities) than for high inventory (‘back end’) nuclear fuel cycle facilities. It is also important to note that nuclear security considerations may change the grading of a facility.

It should also be noted that the application of a graded approach can be reviewed as the safety analysis progresses and a better understanding is obtained of the level of risk arising from the facility or activity. The scope, extent, level of detail and effort applied can be adjusted accordingly. For example, as the safety analysis progresses, it may emerge that the likelihood of a significant consequence is greater than originally considered and more effort may be needed to demonstrate meeting the safety requirements, or vice versa.

Examples of factors affecting the application of a graded approach and areas that could be subjected to grading in performing the safety analysis of nuclear fuel cycle facilities are provided in Annex I.

3. PERFORMING SAFETY ANALYSIS FOR NUCLEAR FUEL CYCLE FACILITIES

Having established the acceptance criteria for safety and defined the objectives of the analysis, the next stage of the process is to perform the safety analysis. This section describes the steps taken to perform the safety analysis of nuclear fuel cycle facilities.

The safety of the facility or activity for all facility states is required to be assessed in the safety analysis, including operational states (i.e. normal operation and AOOs) and accident conditions (i.e. DBAs and DECs) [1]. These states and the main elements of the design of SSCs are shown in Fig. 1. In the figure the areas of design basis, where a conservative approach is generally used, and design extension, where ‘best estimate plus uncertainty’ approach can be used, are shown.

The safety analysis has different objectives, acceptance criteria and analysis rules and assumptions for each of the facility states. These are discussed in Section 3.4 of this publication. Guidance on performing safety analysis for the normal operation of different types of nuclear fuel cycle facility can be found in the relevant IAEA Safety Standards publications in Refs [7–12] and is not further discussed in this publication. The discussions in this section focus on the safety analysis of AOOs and accident conditions.

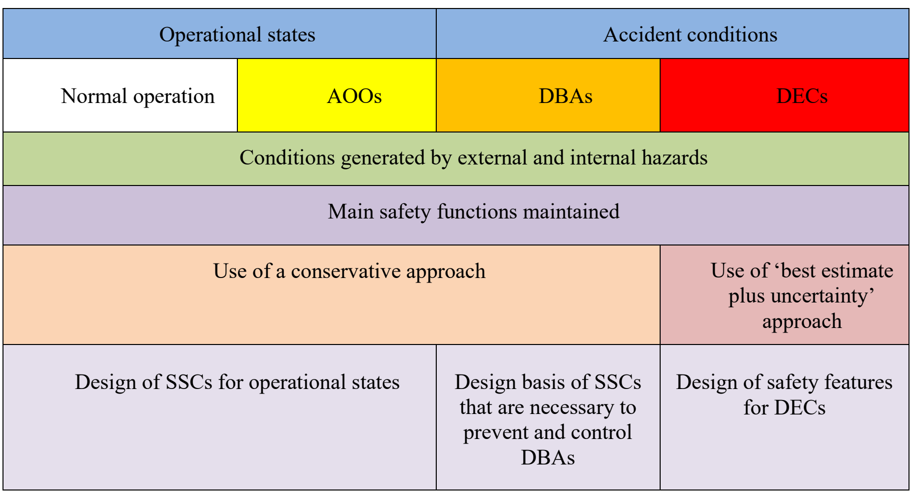

FIG. 1. Facility states for nuclear fuel cycle facilities. AOOs: anticipated operational occurrences; DBAs: design basis accidents; DECs: design extension conditions; SSCs: structures, systems and components.

The main safety functions to be performed by a nuclear fuel cycle facility [1] for all facility states are the following:

- Confinement and cooling of radioactive material and associated harmful materials;
- Protection against radiation exposure;
- Maintaining subcriticality of fissile material.

Incidents or accidents are postulated to occur whenever an external or internal event, or the failure, malfunction or incorrect operation of a system or component, challenges the fulfilment of one of these basic safety functions. Once a release of radioactive and chemical material is foreseen, either as a routine part of normal operation or as a consequence of an accident sequence, this release has to be controlled for the normal operation case and limited or delayed for the accident condition case. The safety analysis is used to demonstrate the safety of the facility (fulfilment of basic safety functions) and how the design of the facility and the related operational procedures will contribute to the prevention and mitigation of accidents. This analysis can be performed in accordance with the following steps, which are described in this section:

- Identification of hazards;
- Identification and selection of PIEs;
- Evaluation of event sequences (scenarios);
- Evaluation of event consequences;
- Comparison against safety analysis acceptance criteria (see Section 2.3 of this publication).

The process of safety analysis proceeds in an iterative manner with the design, until the levels of safety have been demonstrated to be acceptable. This iterative process results in the classification of SSCs with reference to their importance to safety (or the revising of the pre-established classification of these SSCs).

The results of the safety analysis could then be used in various applications, including the hazard categorization of the facility, derivation of operational limits and conditions, modification evaluation and justification, ageing management and lifetime extension, and for the purposes of accident management and emergency preparedness and response. These topics are discussed in Section 4 of this publication.

3.1. IDENTIFICATION OF HAZARDS

A systematic process has to be used for hazard analysis (or equivalent) to identify all DBAs and their associated initiating events that could challenge or cause the failure of the main safety functions and result in unacceptable consequences. The objectives of hazard analysis are as follows:

— Identify hazards: Internal and external hazards (in all modes of operation) are determined for foreseeable circumstances including fault conditions.
— Identify causes: Hazards identified during the hazard identification step are linked with the PIEs to produce event scenarios.
— Provide an overall assessment of the importance of the various hazards.

Some hazards might be obvious and can be directly identified from an established list (such as external hazards), but others require a process of postulating events and analysing event sequences to identify their significance. Annex II provides examples of factors to be considered and techniques used in hazard identification for safety analysis in nuclear fuel cycle facilities.

All types of hazard need to be considered, including those generated within the facility itself (internal hazards) and those generated outside the facility (external hazards). The hazards that are inherent to the process are a specific category of internal hazard that require particular attention in the safety analysis. These types of hazard need to be distinguished for the purpose of safety analysis.

3.1.1. Internal hazards

Internal hazards are those hazards that could affect the safety of the facility and originate within the site boundary. Internal hazards can be as significant as external hazards. The hazard identification process will normally employ information sources such as piping and instrumentation diagrams, process flow diagrams, plot plans, topographic maps and utility system drawings. The process needs to consider the combination of different types of hazard. For example, the release and ignition of an explosive material (chemical/fire hazard) could result in the release of radioactive materials (radiological hazard). Internal hazards that could be considered in nuclear fuel cycle facilities include the following:

— Explosions and uncontrolled chemical reactions of materials handled at the facility;
— Inadvertent nuclear criticality;
— Fires and the spread of smoke and hazardous gases generated as a consequence of a fire;

— Internal flooding/spray;
— Impacts from vehicular transport;
— Chemical hazards, corrosion, release of dangerous gases and liquids;
— Loss of shielding;
— Consequent effects of the failure of components, piping and tanks containing liquids or gases (missiles, jet forces, pipe whips, pressure waves);
— Dropped loads.

Some of the internal hazards are intrinsic (in some Member States intrinsic hazards are considered to be a separate category of hazard) to the processes employed in some nuclear fuel cycle facilities (e.g. conversion, fuel fabrication, spent nuclear fuel reprocessing). These may involve chemical reactions and thermal overrun, precipitation, sudden phase changes, failure of pressure vessels, red oil explosion or resin exhaustion. In evaluating these hazards, chemical properties such as reactivity, toxicity and incompatibility with other chemicals are considered. Additionally, hazard evaluation has to consider specific loads and load combinations, and the environmental service conditions of SSCs (e.g. temperature, pressure, humidity and radiation) as a result of internal events such as pipe breaks, impingement forces, internal flooding and spraying, internal missiles, load drop, internal explosions and internal fire.

3.1.2. External hazards

External hazards are those that can affect the safety of a nuclear fuel cycle facility but originate outside the site, and are unrelated to the processes conducted on the site. External hazards are characterized by information on the frequency of occurence and severity of the external events. SSR-4 [1] provides a list of external events that need to be considered in safety analysis for nuclear fuel cycle facilities. These events include earthquakes, floods, tornadoes and tornado missiles, sandstorms, hurricanes, storms and lightning, tropical cyclones, external explosions, aircraft crashes, external fires, toxic spills outside the facility, accidents on transport routes and hazards from adjacent facilities.

The safety evaluation of site characteristics that are specific to the facility site has to identify all of the possible external events in the region that have the potential to affect the safety of facilities and activities. These could include natural events (e.g. extreme weather conditions and external flooding) and human induced events (e.g. aircraft crashes and events due to hazards arising from transport and industrial activities, where appropriate). It is also important to perform a detailed local evaluation for external hazards that could have a major impact on safety (e.g. historical seismic activity, in order to eliminate the possibility of surface faulting). The regulatory body may also specify the level of a more challenging external hazard.

Credible consequential effects of events are considered as a part of the initiating event [1].

For sites with multiple facilities or multiple activities, account needs to be taken in the safety analysis of the effects of external events on all facilities and activities, including the possibility of concurrent events affecting more than one facility or activity, and of the potential hazards presented by each facility or activity to the others. The potential for multiple accident scenarios developing simultaneously from a single initiating event, as well as the potential interaction of multiple facilities or multiple accidents on the same site, need to be considered in the safety analysis [1].

More information can be found in the IAEA Safety Standards [7–12], which describe external and internal hazards for different types of nuclear fuel cycle facility.

3.2. IDENTIFICATION AND SELECTION OF POSTULATED INITIATING EVENTS

3.2.1. Postulated initiating events

PIE identification and selection follows hazard identification and builds upon its results. The identification of hazards and selection of PIEs are usually conducted simultaneously. The hazards identified during the hazard identification step are linked with the PIEs to produce event scenarios. Events considered in safety analysis as PIEs could be single PIEs, sequences of several consequential events or combinations of independent events. When developing accident scenarios, these PIEs are assumed to occur. The term ‘postulated initiating event’ (or simply ‘initiating event’) refers to an unintended event, such as an operating error, equipment failure or external influence, which directly or indirectly challenges basic safety functions. Typically, such an event necessitates protective actions (automatic or manual) to prevent or mitigate undesired consequences to facility equipment, facility personnel, the public or the environment.

Identification and selection of PIEs could be accomplished using one or more approaches, based on regulatory requirements, engineering evaluations, reference to previous sets of PIEs, deductive analyses and consideration of operational experience from the particular facility or from similar designs, complemented by adequate analysis (such as fault tree analysis) and engineering judgement. Examples include the following:

— List of PIEs as established by regulatory requirements.
— The list of PIEs established in the appendix to SSR-4 [1], which is reproduced in Annex III to this publication. This list could be used as a starting point to establish a facility specific list of PIEs. Additionally, lists of PIEs that are specific to various types of nuclear fuel cycle facility, which could be used as indicative lists of PIEs for safety analysis, also are provided in Refs [7–12].
— Existing lists of PIEs for the facility, or analysis of lists of events developed for safety analysis of another facility, as applicable.
— Engineering evaluation: A systematic review of the facility’s design, operations and site factors to identify occurrences that could lead to radiological and chemical hazards.
— Operational experience: Experience from similar facilities (including practices in industries such as the chemical industry) and experience from events reported to the Fuel Incident and Analysis System database [14].
— Logical analysis: An example is a top-down logical model (e.g. master logic diagram, fault tree analysis).
— Other credible data, including scientific literature and engineering handbooks.

It is worth mentioning that, due to the diverse nature of nuclear fuel cycle facilities, the consideration of PIEs from other facilities may not be as important as it would in a similar analysis for nuclear reactors. Nevertheless, comparison with the available information will contribute to the identification of a comprehensive list of applicable PIEs. Additionally, as nuclear fuel cycle facilities involve chemical and toxic hazards (in addition to radioactive hazards), it is also useful to consider additional approaches applied in identifying and characterizing PIEs.

The method used to identify PIEs and to select sets of particular events for further analysis has to ensure that the list of PIEs is as complete as possible, that PIEs are grouped in a logical manner to simplify the analysis and that limiting or bounding PIEs in each group can be selected for further analysis. The following methods could be applied for the identification of PIEs:

— Identification of all mechanisms of barrier failure (e.g. corrosion, overpressure or negative pressure, overflow, impact loads);
— Identification of all processes that could cause the failure mechanisms to initiate (e.g. fatigue, corrosion, thermal effects, mechanical loads, seismic events, inventory increase);

— Grouping of these processes by means of phenomenology (e.g. release of radioactive or chemical material, exposure);
— Identification of scenarios for each of the above groups;
— Identification of PIEs that lead to the above scenarios;
— Determination of original cause (e.g. internal events or external events, including their possible combinations).

When identifying and selecting PIEs, all permissible operating modes and configurations over the facility's lifetime have to be considered. To establish adequate safety margins or to ensure the robustness of the design, the regulatory body may request that certain PIEs be analysed as DBAs or as DECs. PIEs may change as the facility goes through different stages of its lifetime or because of ageing effects. Therefore, identified PIEs need to be regularly reviewed. The techniques mentioned above can be used or adapted for this purpose.

Some PIEs may be excluded from detailed consideration (e.g. because of their negligible contribution to exceeding safety criteria, or because they are bounded by another event that has been analysed). Such exclusion needs to be fully justified and such justification documented. The following PIEs are examples of those that may be exempted from further analysis:

— Events with minor consequences that are bounded by other more significant sequences;
— Non-credible events (PIEs that are not possible for the facility under analysis because the relevant process is not implemented or the event is excluded by design);
— Very rare events (PIEs whose frequency of occurrence may be so low that they could be candidates for rejection on a probabilistic basis using statistical data or conservative estimates, as agreed by the regulatory body).

3.2.2. Combination of events

Analysis is needed to consider credible combinations of events (which may occur either simultaneously or sequentially) in order to quantify their consequences. Types of combinations can include the following:

— Multiple independent failures in equipment important to safety;
— Multiple process system failures;
— Equipment failures with additional operator errors;
— Common cause events with additional operator errors.

External initiating events may also be accompanied or caused by other internal or external events. For example, an earthquake could be considered in combination with loss of off-site power, flood, tsunami or external fire. Certain events might be consequences of other events, such as facility equipment failures and challenges to subcriticality caused by flooding, or multiple leaks initiated by one external event. Credible consequential effects are required to be considered to be part of the initiating event [1].

Attention needs to be given to SSCs where the potential for a common cause failure could degrade the performance of multiple SSCs. Internal events that could result in common cause failure include fire, explosions and equipment failures that may generate missiles. Failures of interconnected and common systems and services such as ventilation or power supply could also be the cause of common cause failure and loss of safety functions. The effect of a failure in one part of such systems can have effects elsewhere that may be difficult to predict and could be overlooked in the analysis. External, naturally occurring events that are considered in safety analysis as initiators for facility equipment failures include earthquakes, flood, external fires and extreme weather conditions (e.g. temperature, precipitation, high winds and tornadoes).

Examples of nuclear fuel cycle facility features where this can be a potential issue include sumps, drains, vents and energy supplies. These are normally interconnected and are common services throughout the facility. An overload or failure in one part of such systems can have effects elsewhere that may be difficult to predict and could be overlooked in the analysis.

According to para. 6.76 of SSR-4 [1]:

> "Where the results of expert judgement and deterministic safety analyses complemented by probabilistic safety assessments (if available) indicate that combinations of events could lead to combinations of anticipated operating occurrences with other 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 and the magnitude of their potential consequences."

3.2.3. Grouping of events

Whenever possible, the analysis has to be based on an appropriate grouping and bounding of events and processes in order to simplify the safety analysis. For this the identified PIEs may be grouped into categories. It is expected that only representative or bounding events in each group of events will need to be analysed. The method used to group PIEs has to take into account the following:

- Grouping of PIEs and associated transients relating to facility states;
- Grouping of PIEs and their associated transients according to the frequency of the initiating events or the frequency of the accident scenarios;
- Grouping based on similarity of event or hazard type (e.g. loss of confinement, criticality, fire), initiating failures, the physical phenomena of the initiating events and the accident scenarios;
- PIEs that require similar safety actions and system and operator responses;
- PIEs that have similar influence on SSCs;
- Grouping or categorizing of PIEs that assists in the selection of limiting cases for analysis in each group;
- Grouping of external PIEs that have the potential for common cause impact on the whole facility.

See Annex III to this publication for a possible grouping of PIEs. Grouping of PIEs could also be specified for the type of the nuclear fuel cycle facility under analysis. For nuclear fuel cycle facilities, grouping initiating events and their associated transients according to the frequency of the initiating events or to the frequency of the accident scenarios, and to the similarity of the event or hazard type, are the most commonly used approaches.

PIEs in each group have to be evaluated to identify those events that would be limiting (sometimes referred to as 'bounding cases') and those that should be selected for further analysis. Since analysis of all possible accident scenarios is not practicable, a reasonable number of limiting cases, which are referred to as bounding or enveloping scenarios, need to be selected from each category of events. For many nuclear fuel cycle facilities, there may not be a single bounding event that bounds all PIEs due to the variety of processes, equipment and material to be used, such as in the case of nuclear fuel reprocessing facilities.

3.3. EVALUATION OF EVENT SEQUENCES

This step in performing the safety analysis includes all aspects of defining the accident scenario and modelling the accident sequence, such as the identification of relevant physical phenomena and performing the necessary calculations to evaluate the associated parameters. The model has to link the response of the facility's SSCs to these initiators, and identify the spectrum of end states of the facility. There may be several steps to modelling the scenario: prediction of the response of engineered SSCs; assessment of human performance; consideration of probabilities; and estimation of source terms.

Evaluation of the event sequence typically comprises the following steps:

- Identification of causes of the event;
- Transient and accident analysis;
- Modelling of sequence of events and systems operation;
- Specification of the end states.

Identification of causes includes a description of the occurrences that led to the PIE under consideration for each scenario evaluated. The scenario typically comprises a step by step chain of events and their consequences from initiation to the end state condition. Some sequences may result in radiological (or chemical) consequences and some others may not. Derivation of source terms and evaluation of radiological and non-radiological consequences follow as part of this analysis.

Transient and accident analysis comprise a detailed analysis of the nuclear fuel cycle facility system performance. Depending on the complexity of the process modelled, different methods can be applied. The most widely known methods for modelling event sequences in complicated systems and processes for nuclear fuel cycle facilities are based on such methods as the event tree analysis and fault tree analysis methods (see also Section 2). Many applications utilize a combination of event and fault trees to represent potential accident scenarios.

The rules of the safety analysis need to be established to determine the response of the facility's SSCs to a PIE, and have to be consistently applied throughout the safety analysis (for example, the application of single failure criterion, rules for crediting support systems, rules on the use of calculation models or computer tools). A set of rules for safety analysis is listed in Annex V. These rules help to do the following:

- Define which SSCs may be functioning during the course of an accident sequence and which ones may be considered to have failed;
- Determine those event sequences that are outside the design basis (and therefore excluded from further analysis, except for scenarios used for emergency preparedness and response);
- Standardize the codes and methods to be used for calculation.

These rules can be established by the regulatory body, or they can be proposed by operating organizations and reviewed by the regulatory body.

3.3.1. Modelling the scenario

The process of modelling a scenario begins with its definition. As well as identifying the initial status of the facility and the PIE to be considered, it determines relevant physical phenomena and human performance factors that should be analysed. The assessment will identify other key parameters and boundary conditions which should be documented.

For each scenario the following need to be analysed:

- The sequence of events and system operation in the course of the accident (from initial state to the final safe, stable state);
- Safety functions and relevant system performance, including actions of the facility systems and operator in response to the event and physical barrier performance;
- Facility state at the end of the event (may include damage to material or SSCs, etc.);
- Radiological and, as appropriate, non-radiological consequences.

The time span of a scenario extends up to the moment when the facility reaches a safe and stable end state. The analysis is also used for developing the performance requirements for SSCs important to safety, and the detailed analysis is used to identify bounding cases that provide the basis for the design and the operational limits for SSCs important to safety.

SSCs important to safety are considered in the event sequence. Examples of SSCs important to safety that are typically considered in safety analysis for nuclear fuel cycle facilities are listed in the relevant IAEA Safety Guides [7–12] and can include the following:

- Process equipment;
- Static containment systems and structures;
- Ventilation systems;
- Shielding systems;
- Cooling systems, if applicable;
- Instrumentation and control systems (e.g. process gas flow and pressure control systems, radiation monitoring systems, criticality control and alarm systems);
- Electrical power supply;
- Auxiliary systems (e.g. fire protection systems, chemical control, reagent feed systems, heating systems, supply of cooling water, steam, service air, compressed gas);
- Waste management systems and systems for controlling discharges;

— Handling systems (e.g. cranes and lifting equipment) and on-site transport systems;
— Other items listed in Refs [7–12].

As previously discussed, human actions, errors or human performance in general can initiate an event or make an already initiated event worse. Analysis of the potential consequences of PIEs caused by human error is best performed in parallel with the identification of PIEs and the description of the event sequence. The analysis rules often specify the type of operator response that can be expected and claimed. The level of human performance needs to be justified by an expert analysis using recognized techniques.

3.3.2. Identifying physical phenomena and evaluation of associated parameters

Physical phenomena and the values of associated parameters (or ranges of parameters) have to be determined for every scenario, using appropriate calculation tools. Sensitivity analyses may be used to supplement the calculations, in conjunction with engineering judgement, to evaluate the importance of a specific parameter or value to the overall assessment. Particular importance has to be placed on the identification of cliff edge effects.

3.3.3. Defining boundary and initial conditions

The boundary and initial conditions used in the analysis have to be identified. This includes the status and initial values of the physical parameters associated with the systems or processes that are involved in the analysis. Some examples are as follows:

— The facility operating mode;
— Environmental service conditions of SSCs (e.g. initial values of temperatures, pressures, material inventory);
— Performance characteristics of SSCs (initial values of variables);
— Performance of other facility equipment (status of SSCs such as pumps or valves that may be involved in the analysis, or initial values of associated variables);
— Weather conditions.

3.4. ANALYSIS OF FACILITY STATES

This section of the publication addresses the safety analysis performed for different facility states and discusses the selection of acceptance criteria and assumptions used for the analysis. The approaches described below are similar to the approaches used for reactors [3–5].

3.4.1. Anticipated operational occurrences

The main objective of the analysis of AOOs is to verify that the facility's operational systems can prevent occurrences from evolving into accident conditions and that the facility can return to normal operation following an AOO. It is necessary to demonstrate that safety functions are performed adequately and the limits specified in the design basis for AOOs are not exceeded.

While analysing AOOs it is assumed that the facility systems not affected by the PIE are available and operating according to the design. Planned operator actions performed in accordance with operating procedures for normal and abnormal operation are credited in the analysis. Typically, when correct operation of the control systems is assumed, there is no need for any operator action during the associated transient; otherwise realistic estimates for operator action times need to be used.

For AOOs, safety analysis is generally carried out using best estimate assumptions, data and methods. Where this is not reasonable, a rational degree of conservatism is used to compensate for the lack of adequate knowledge concerning the physical processes governing these events.

3.4.2. Design basis accidents

Analysis of DBAs (or equivalent) for a nuclear fuel cycle facility examines conditions that may occur in the lifetime of the facility in order to demonstrate that they are tolerable. The analysis is required to confirm that the risk of consequences from DBAs is acceptably low and that the likelihood of an accident has been minimized to the extent practicable [1].

Analysis of DBAs provides the design parameters for design measures, controls and mitigating measures. The aim of safety analysis for DBAs is to demonstrate that safety functions are performed adequately and the limits specified in the design basis for DBAs are not exceeded.

For uranium fuel fabrication facilities, the following DBAs are typically considered:

— Leakage/breakage of a UF_6 pipe;

— Explosion of reaction vessel;
— Explosion of a rotary kiln;
— Fire in a protected fire zone (fire compartment);
— Spillage of uranium powder;
— Collision of a truck with a barrel of powder outside a factory building;
— Collision involving a UF_6 cylinder;
— Drop of a storage container for radioactive waste from the maximum eligible height;
— Release of uranium after an earthquake.

An appropriately conservative approach is normally taken for the analysis of DBAs. The concept of conservatism is applied to the analysis to ensure that limiting assumptions are used. In addition, an adequate set of conservative or best estimate assumptions for the initial and boundary conditions could be used. Conservative analyses will demonstrate that no single failure could result in the loss of a safety function. For DBA analysis this criterion is applied to all equipment designated with a safety function. The relevant safety important systems are assumed to operate during postulated accidents with the performance established in the operational limits and conditions. If operator actions are credited, it is demonstrated that credible 'worst case' operator response time has been considered in the analysis.

The following assumptions are relevant for the analysis of DBAs for nuclear fuel cycle facilities:

— The time frame for release and the exposure time are considered in accordance with reasonable scenarios or performance of SSCs (e.g. response of detection systems and interlocks).
— The human interaction performance under the accident conditions is considered in a realistic way (e.g. the availability of workers).

Any uncertainties in the accident sequences need to be taken into account in a conservative manner. If the consequences are minor, simple methods (with less analytical detail) may provide conservative results. With more serious consequences, effort may be necessary to reduce the uncertainty, or a much larger margin of safety may need to be provided.

For nuclear criticality safety in design, the double contingency principle is the preferred approach [1]. By virtue of this principle, a criticality accident cannot occur unless at least two unlikely, independent and concurrent changes in process conditions have occurred. The analysis has to provide a documented technical basis that demonstrates that subcriticality will be maintained in operational states and conditions that are referred to as credible abnormal conditions, or conditions included in the design basis in accordance with the double contingency principle [18].

3.4.3. Design extension conditions

An analysis of DECs is required to be performed for existing facilities and new facilities for which there is a potential for a large release or an early release of radioactive material [1]. The main technical objective of considering DECs is to provide assurance that the design of the facility is such as to prevent accident conditions not considered DBAs, or to mitigate their consequences, as far as is reasonably achievable (see para. 6.73 of SSR-4 [1]).

The PIEs that lead to DECs are required to be analysed for their capability to compromise the ability to provide an effective emergency response (see para. 6.74 of SSR-4 [1]).

Acceptance criteria for DECs should meet the requirement established in para. 6.74 of SSR-4 [1], namely:

> "The design shall be such that, for design extension conditions, off-site protective actions that are limited in terms of times and areas of application shall be sufficient for the protection of the public, and sufficient time shall be available to take such actions."

In general, only systems shown to be operable for DECs are credited in the analysis. To ensure independence between the levels of defence in depth, the normal operation systems, including control and limitation systems, are not credited in the analysis of DECs. Having specified the conditions defining the DECs, the additional unavailability of safety features may not need to be considered. Non-permanent equipment is also not considered in demonstrating the adequacy of the facility design.

Best estimate assumptions may be used regarding operator actions for the analysis of DECs. However, some conservative assumptions, as described for DBAs, may be used if convenient.

Additional discussions and examples of DECs for these facilities are provided in Annex IV to this publication. Detailed discussion of DECs for different types of nuclear fuel cycle facility, including identification of these conditions and their assessment, can be found in Safety Reports Series No. 90, Safety Reassessment for Nuclear Fuel Cycle Facilities in Light of the Accident at the Fukushima Daiichi Nuclear Power Plant, published by the IAEA [19].

3.5. EVALUATION OF CONSEQUENCES

The radiological (or non-radiological, if appropriate) consequences of sequences (or bounding events, groups of PIEs) are evaluated in this step of the safety analysis. The consequences are then compared against pre-established acceptance criteria, and iteration between the safety analysis and design of the facility is performed until the safety of the workers and the public and the protection of the environment are demonstrated in accordance with the fundamental safety objective.

An evaluation of consequences covers the effect of an accident in terms of the quantity, type and chemical forms of radioactive material (or other hazardous material) released. These are quantified in terms of doses to workers and the representative public individuals for all possible exposure pathways (including external exposure, inhalation, ingestion, exposure due to dry and wet deposition, whole body doses, etc.).

Evaluation of the consequences normally involves the estimation of source terms, on-site and off-site radiation doses (including airborne releases), and, as applicable, associated chemical consequences. Conservative assumptions are usually adopted for the calculation of the source terms for the evaluation of consequences of AOOs and DBAs, including the use of the following:

- The highest specific radioactivity level of the material;
- The maximum inventory of radioactive material (normally the inventory allowed by the facility's licence);
- The maximum inventory of associated non-radioactive hazardous material;
- The maximum material throughput allowed to be processed by the facility;
- The minimum performances of barriers in the accident conditions.

Evaluating event consequences in the facility is a multistep process, which may require the following:

- Identification of major physical processes involving direct exposure, initial releases and subsequent transport of the material;
- Assessment of chemical forms and concentrations of radioactive material and toxic substances as they move from the release point to the receptor;
- Determination of exposure pathways (external, internal, dermal, ocular, etc.) and duration of exposure;
- Assessment of radiological consequences and associated chemical consequences (including toxicological effects of uranium and hydrogen fluoride).

3.6. COMPARISON AGAINST ACCEPTANCE CRITERIA

Safety analysis acceptance criteria are discussed in Section 2 of this publication. In this step, the results of the safety analysis are compared against the pre-established acceptance criteria. Where the consequences of accidents in the design basis exceed the acceptable limits, additional provisions (engineering or administrative) are required to be provided in accordance with the defence in depth concept (e.g. incorporating more robust design principles to improve the reliability of SSCs important to safety such as redundancy, diversity and physical protection) to reduce the frequency of the accident and/or to mitigate its consequences, so as to bring the consequences to acceptable values.

Events for which the consequences only just meet acceptance criteria may be analysed for the potential for optimization and additional protective measures may then be considered. Evaluation of the consequences of DBAs may also lead to modifications of the facility design, when the principle of optimization is considered, to decrease the consequences to an acceptable level as far as possible.

The process of making decisions on design and operational safety needs to be iterative. The safety analysis continues to the depth and extent necessary in an iterative process with design until it has been demonstrated that the acceptance criteria have been achieved and that all other performance and safety targets have been met. The amount of iteration depends on the stage of development of the facility and the nature of the decision to be made. Research and development may be needed to fill gaps in data and models and allow the analysis to proceed. The iteration needs to proceed until the assessment is judged to be adequate for its purpose.

3.7. PRESENTATION OF SAFETY ANALYSIS AND CONCLUSIONS

The safety analysis is documented for licensing, peer review and further use by the operating organization. The conclusions of the safety analysis provide the main basis for understanding the facility safety envelope, for the development of requirements for SSCs and for management procedures including operational limits and conditions. The assessment is also used in the evaluation of proposed modifications to the facility, as well as analysis of ageing management programmes. These aspects are described in Section 4 of this publication. The analysis also provides the basis for training programmes for operating personnel. The results of the analysis are reviewed by facility designers and used to ensure that quality requirements for SSCs important to safety are adequate. For an existing facility, the safety analysis provides evidence of the design bases for existing SSCs.

The safety analysis is presented to the regulatory body if required, along with the operational limits and conditions (sometimes in a separate document). These are used by the regulatory body in performing review and assessment for the licensing of the facility. Consequently, the conclusions of the safety analysis are important components of the licensing documentation. It is therefore essential that they be presented in a clearly structured, complete and comprehensive manner, consistently with other information provided by the licensing documentation (including the safety analysis report).

To ensure these objectives, the information on safety analysis could be presented in accordance with the following structure:

— Introduction: Objectives, approaches and methods used in the safety analysis.
— Facility characteristics: Facility parameters, boundary and initial conditions used.
— Identification of hazards and selection of PIEs: Information about hazard identification and the PIEs that were considered in the safety analysis.
— Evaluation of event sequences: Description of event sequences and the facility's system response to the event.
— Transient and accident analysis: Description of the analyses, including the radiological (and, as appropriate, non-radiological) consequences of the events.
— Summary and conclusions: Summary of the results of the safety analysis and comparison against the safety analysis acceptance criteria.

Any limiting assumptions for the analysis and any special interface requirements have to be discussed in the relevant sections. This structure is outlined further below.

3.7.1. Introduction

A statement of the objective of the safety analysis has to be presented, together with adequate information on the methods and approaches used in the safety analysis. This section includes a description of safety analysis acceptance criteria, rules used in the analysis, methods used for the identification and evaluation of hazards, and methods used for the identification and selection of PIEs.

3.7.2. Facility characteristics

A summary of the nuclear fuel cycle facility parameters and characteristics, as well as the range of operating conditions used in the analysis, is presented in this section. This also has to include the range of permitted facility parameters, including tolerances and associated uncertainties. A description of the facility's safety systems that are involved in the safety analysis (event sequences) and associated values of the system settings is also included here.

3.7.3. Identification of hazards and selection of PIEs

Adequate information needs to be presented in this part with respect to hazard identification and evaluation. A full list of the PIEs selected for the analysis has to be presented, along with a justification for the exclusion of any events or hazards from the assessment. The information could also provide a description of the grouping of PIEs and the basis of that grouping. Information on the identification and selection of bounding (or limiting) events is also provided in this section.

3.7.4. Evaluation of event sequences

This part provides detailed information on the consequences of selected PIEs, including the identification of causes, sequences of events and facility safety system operations. It provides detailed step by step descriptions of sequences, from the initiation of the event until the final stabilization. Operators' actions in the course of event sequences must also be included in the description.

3.7.5. Transient and accident analysis

A detailed analysis of facility systems and process performance is given here, including the methods used to characterize performance. Discussion must be included on the evaluation of any physical phenomena and parameters that are involved in the event sequence, and that may affect the safety performance of the facility, including physical barriers. This part presents estimates of the radiological source term and methods used in the evaluation of the radiological or toxicological consequences to the facility's workers, the public and the environment. Information on exposure pathways and models used in the analysis of the dispersion of released radioactive material into the environment and the exposure of workers and the public as a result of the event is also presented and discussed.

3.7.6. Summary and conclusions

This part discusses the results of the safety analysis, including descriptions of the bounding (or limiting) accident sequences and their consequences. Summaries of comparisons with acceptance criteria are presented in this part, with important conclusions drawn from the safety analysis. Evaluation of the safety acceptance of the facility design as well as the adequacy of the established operational limits and conditions are also provided in this part. This summary could also provide information on possible improvements to the facility's SSCs, and any administrative measures that could be established for improved safety.

The safety analysis is performed to demonstrate that an adequate level of safety has been achieved. This section concludes with a statement of the overall adequacy of the level of safety achieved for the assessed stage(s) of the lifetime of the facility.

4. APPLICATION OF SAFETY ANALYSIS

The safety analysis is used in a number of areas, including the determination of the facility hazard categorization, the classification of SSCs, the derivation of operational limits and conditions, the consideration of modifications, the establishment of programmes for ageing management and long term operation, and the development of emergency preparedness and response arrangements. This section discusses the application of the results and findings of the safety analysis to these areas.

4.1. HAZARD CATEGORIZATION OF THE FACILITY

Paragraph 6.28 of SSR-4 [1] requires a qualitative categorization of the nuclear fuel cycle facility on the basis of the potential hazards associated with the facility. The hazard category of the facility can be used to identify its performance targets. For the design of a new facility or its modification, the hazard categorization can be used for grading the application of safety requirements related to the siting, design, construction, commissioning and operation of the facility. It can also be used in the establishment of adequate regulatory supervision programmes for the facility.

The analysis performed to provide a basis for a qualitative hazard categorization needs to be facility specific. A preliminary assessment is performed to determine the hazard category of the facility, usually based on an estimate of unmitigated release from the inventory. The depth and extent of the safety analysis is influenced by the estimate of the unmitigated consequences of the identified accident sequences and the complexity of the process. The assessment of event consequences, undertaken as part of the safety analysis, can be used to define the hazard category of the facility or to confirm a preliminary category, as the safety analysis is developed in more detail.

4.2. SAFETY CLASSIFICATION OF STRUCTURES, SYSTEMS AND COMPONENTS

The safety classification of SSCs is performed during the facility design, system design and equipment design stages, and may be reviewed at a later stage in the facility's life.

The results of the classification are used to ensure that appropriate design codes, standards and procedures for manufacturing, testing, quality, installation, surveillance, maintenance, inspections and, where appropriate, qualifications are in place to ensure that SSCs adequately perform their intended safety functions. The classification of SSCs with respect to safety has to be justified based on the safety analysis. The method for classifying the safety significance of SSCs has to be based primarily on deterministic safety analysis, supported by engineering judgement and complemented, where appropriate, by probabilistic methods.

Guidance on the identification of SSCs important to safety and their classification for nuclear power reactors is provided by IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants [20]. A similar approach to the safety classification of SSCs for nuclear fuel cycle facilities was adopted by some Member States [21].

The classification of SSCs for nuclear fuel cycle facilities can also be based on the importance of the safety function(s) they provide, the consequences of failure to perform the safety function and related factors.

Examples of SSCs important to safety for different types of nuclear fuel cycle facility can be found in the relevant IAEA Safety Guides [7–11].

The iteration process between safety analysis and design also establishes the design basis for SSCs and their design verification. This includes applying design principles for enhancing reliability, such as redundancy, diversity and the fail-safe criterion, applying design provisions for facilitating maintenance, periodic testing and inspection, as well as establishing criteria for sharing common SSCs between facilities on the same site. The safety analysis can also be used to establish the specifications of design features of the SSCs, and the criteria for protection of these SSCs against internal and external events.

4.3. DERIVATION OF OPERATIONAL LIMITS AND CONDITIONS

To prevent situations arising in nuclear fuel cycle facilities that may lead to accident conditions, and to mitigate the consequences of accident conditions should they occur, operational limits and conditions have to be established.

The operational limits and conditions are a set of rules setting forth the parameter limits, the functional capability and the performance levels of equipment and personnel for the safe operation of a facility. Operational limits and conditions for nuclear fuel cycle facilities cover the following [1]:

— Safety limits;
— Safety system settings;
— Limiting conditions for safe operation;
— Periodic testing and surveillance;
— Administrative controls.

The operational limits and conditions are required to be derived from the safety analysis, with use of a graded approach, to ensure that the facility is operated in accordance with the design assumptions and intent, as well as in accordance with its licence conditions (para. 9.27 of SSR-4 [1]). The basis for the operational limits and conditions, with any assumptions used, is justified in the safety analysis report.

The results of the safety analysis are applicable to every operation mode and status (e.g. operation, maintenance, surveillance, shutdown) of the nuclear fuel cycle facility. The results include elements such as specifications of the safety limits, safety alarm settings (i.e. safety system settings) and requirements on the minimum amounts and combinations of safety equipment to be available during different operation modes. The availabilities provide the basis for the periodic testing and surveillance of the SSCs important to safety. Proposed modifications of operational limits and conditions need to be justified based on the safety analysis.

Discussions on the derivation and establishment of operational limits and conditions for different types of nuclear fuel cycle facility can be found in the relevant IAEA Safety Guides [7–12]. Considerations and examples of items to be covered by limiting conditions for safe operation for nuclear fuel cycle facilities are provided in Annex VI.

Attributes of operational limits and conditions and information on their presentation in the licensing documentation of the facility are provided in Section 5 of this publication.

4.4. MODIFICATIONS

In the context of this publication, a modification is a deliberate change in or addition to an existing nuclear fuel cycle facility, an SSC or an item of software important to safety. A modification may also involve a change in SSCs important to safety, process systems, operational limits and conditions or documentation (including operating procedures). Modifications with safety significance are evaluated with procedures for design, construction, commissioning and safety analysis that are equivalent to those for the facility itself, in order to ensure that they meet the same requirements as the existing SSCs.

Preparation of the safety analysis for modifications requires the review of the safety analysis originally developed for the facility. Modifications may entail hazards that are different in nature from, or more likely to occur than, those hazards previously considered. Some modifications (such as those related to process systems) may lead to new PIEs or to changes in the accident sequences of the originally developed safety analysis. It is necessary to assess the safety of the modification itself and the potential for a safety impact on the rest of the facility.

Additionally, when upgrading the instrumentation and control systems of nuclear fuel cycle facilities (including instrumentation or control systems related to criticality control, glovebox control, monitoring and control of discharges, etc.), improvements in the coverage of PIEs may lead to changes in the accident sequences and rules of safety analysis; therefore, such modifications require revision of the safety analysis. Furthermore, the technical and operational impact of proposed modifications (to SSCs or to a process) has to be evaluated for every related accident sequence that was considered in the original safety analysis of the facility.

4.5. AGEING MANAGEMENT AND LIFETIME EXTENSION

An effective ageing management programme needs to be established for nuclear fuel cycle facilities as part of providing assurance that SSCs important to safety are capable of performing their intended safety functions throughout the facility’s lifetime. Safety analysis plays a major role in defining the basis for identifying the mechanisms of ageing of SSCs and their effects on safety, and in the development of ageing management programmes.

The main elements of a systematic and proactive ageing management programme for nuclear fuel cycle facilities include the following (as specified in Ref. [22]):

— Identification and understanding of ageing degradation mechanisms;

— Screening of SSCs for ageing management;
— Minimization of ageing degradation (through design and operational practices);
— Detection, monitoring and trending of ageing degradation;
— Mitigation of ageing degradation.

Safety analysis provides a basis for practical actions and procedures covering the above elements. Screening of SSCs for the ageing management programme is performed based on their safety significance, which is normally defined by the safety analysis of the facility. The safety analysis also establishes the design basis of these SSCs and associated specifications that ensure the existence of appropriate safety margins to allow for the anticipated material properties at the end of their useful lifetime. Equipment qualification can also be considered to be part of the ageing management programme, and this may demand SSC qualification under AOOs, DBA conditions and, to the extent practicable, DECs. Activities related to the minimization and monitoring of ageing degradation include maintenance, periodic testing and inspection of SSCs important to safety, and are also defined based on the safety analysis.

The ageing management programme can make a substantial contribution to the lifetime of the facility.

The safety analysis can also be used (including in the context of periodic safety review) to identify practical and reasonable safety upgrades of the facility for long term safe operation or consideration of extending the facility lifetime.

4.6. EMERGENCY PREPAREDNESS AND RESPONSE

The results of the safety analysis are used for the development of the operators' actions in the case of accident conditions. Development of such procedures is mainly based on the analysis of the event sequences. Examples include operators' actions in the case of loss of electrical power supply, fire, earthquake or flood. Understanding of the event sequences will help to define the specific actions to be taken by the facility's operators in a step by step manner.

The results of the safety analysis (including analysis of DECs) establish the basis for development of emergency preparedness and response. Hazard assessment of the facility, and subsequently the facility's hazard category, are used to identify the emergency preparedness category of the facility, which provides the basis for developing appropriate arrangements for emergency preparedness and response (see IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [23], and guidance on the review of emergency arrangements for nuclear fuel cycle facilities [19]).

5. LICENSING DOCUMENTATION

5.1. GENERAL CONSIDERATIONS

Requirement 1 of IAEA Safety Standards Series No. SSR-4 [1] states:

“The operating organization shall demonstrate the safety of its [nuclear fuel cycle] facility through a set of documents known as the licensing documentation (or safety case). The licensing documentation shall provide a basis for the safety of the facility at all stages of its lifetime and shall be updated periodically, to take account of modifications made to the facility and other changes.”

The licensing documentation is submitted by the operating organization to the national regulatory body during the licensing process.

5.1.1. Scope and amount of information in licensing documentation

The regulatory processes for the licensing of nuclear installations, including nuclear fuel cycle facilities, vary between Member States. Typically, the licensing of nuclear installations involves discrete steps (e.g. site evaluation, design, construction, commissioning, operation and decommissioning). In some cases, some of these steps are combined (e.g. construction, commissioning and operation). Accordingly, the scope and the amount of information in the licensing documentation could be different depending on the licensing process and the licensing step, and varies between Member States.

A graded approach can be used in developing licensing documentation, where the scope and amount of information and depth of analysis is commensurate with the potential hazard of the nuclear fuel cycle facility, the complexity of its design, the maturity of the processes involved and the stage of its lifetime. Nevertheless, the scope and amount of information in the licensing documentation, at each step, has to be sufficient to demonstrate safety and to allow both the operating organization and the regulatory body to make a decision on the acceptability of the facility for that licensing step.

Section 5.2 of this publication discusses the scope and amount of information inlicensing documentation at various steps of the licensing process. In this publication, the ‘discrete steps’ licensing approach, as described in the IAEA safety standards (i.e. site evaluation, design, construction, commissioning, operation and preparation for decommissioning), is adopted.

5.1.2. Types of licensing documentation

The appendix to IAEA Safety Standards Series No. SSG-12, Licensing Process for Nuclear Installations [24], provides examples of the documents to be submitted during the licensing process for nuclear installations. Annex VII to the present publication reproduces these examples along with considerations for their application to nuclear fuel cycle facilities. In this regard, it is essential to note that the contents of the documents reproduced in Annex VII can be divided or combined into different documents, as appropriate. For example, 'general operating rules' and 'operational limits and conditions' are indicated as separate documents but could be combined into one document.

On the other hand, paras 3.5–3.7 of SSR-4 [1] define the safety analysis report and operational limits and conditions as parts of the licensing documentation for a nuclear fuel cycle facility (in addition to any other information that may be required by the regulatory body). The content of the safety analysis report described in these paragraphs includes elements that are indicated as separate documents in the appendix to SSG-12 [24] (see Annex VII to this publication). This includes, for example, application of safety principles, design criteria, design safety features, hazard analysis, accident analysis, safety limits and items important to safety, operating organization management systems and conduct of operation.

The practices in Member States with respect to the format, types and content of licensing documentation (safety case) and safety analysis reports vary, but they are generally in line with the approaches mentioned above. Examples of national practices in this regard can be found in Refs [25–29]. In this publication, the description of the content of the safety analysis report assumes that — in accordance with SSR-4 [1] and SSG-12 [24] — it is a high level document that covers the information needed for licensing purposes (see Section 5.3 of this publication). Some of this content may be presented in separate documents (e.g. environmental impact assessment, nuclear criticality safety assessment, operational limits and conditions, radiation protection programme, emergency plans and management systems). If so, this content needs to be discussed briefly in the safety analysis report and reference should be made to the appropriate separate documents. It needs to be ensured that there is consistency and continuity between the information provided in the licensing documents and that provided for subsequent stages of the life cycle in the safety analysis report. In cases in which a subsequent stage of the safety analysis report indicates results different to those in the report from the previous stage (e.g. because the information has improved or modifications have been made), the changes need to be explained and justified.

On that basis and in the context of this publication, the safety analysis report has the aim of demonstrating the safety of the design and operation of the nuclear fuel cycle facility. The safety analysis report also provides the basis for the safe operation of the facility and for interactions between the operating organization and the regulatory body in the licensing process. In addition, the preparation of a safety analysis report serves the following purposes:

— Ensuring the adequacy of the facility hazard category;
— Ensuring that the safety analysis and design of the facility are consistent;
— Supporting designers to confirm that the individual facility’s systems are integrated correctly, since design and the development of the safety analysis report are complementary and iterative processes;
— Supporting the derivation and establishment of operational limits and conditions;
— Identifying the mechanisms of ageing of the facility’s SSCs and their effects on safety, and providing for the development of ageing management programmes;
— Establishing the basis for development of emergency preparedness and response;
— Developing and implementing training programmes for facility staff.

5.1.3. Sites with multiple facilities

The suggested content of the safety analysis report for nuclear fuel cycle facilities that is presented in Section 5.3 applies to a single facility. However, there are sites that include multiple facilities. It is common practice in Member States to develop, in addition to the safety analysis reports on the facilities or activities within the site, a ‘site-wide safety analysis report’ for such sites.

The purpose of the site-wide safety analysis report is to demonstrate that the facilities and activities at the site are safe and to specify dependencies and claims made on site-wide provisions by the safety analysis reports of the individual facilities (e.g. facility interfaces, common services and emergency arrangements). This normally requires additional effort to ensure consistency, address gaps and integrate the information. In this case, a site-wide safety analysis report usually covers aspects related to management and the verification of safety, such as the organizational structure, organization capability, management system and safety culture, as well as the site emergency plan, common services and hazards applicable to the whole site. The site-wide report needs to include adequate references to the safety analysis reports for the individual facilities, and vice versa. The detailed content of the site-wide safety analysis report is out of the scope of this publication.

5.2. INFORMATION NEEDED AT VARIOUS STEPS OF THE LICENSING PROCESS

The preparation of the safety analysis report has to start as early as possible in the development of a new nuclear fuel cycle facility. This allows the facility's designers to derive the maximum benefit from the safety analysis process, as well as allowing the regulatory body to become familiar with the design and the safety features of the facility.

Effective management of the process of preparing and updating the safety analysis report requires adequate plans, which include timescales for the preparation or updating of specific parts of the report. This is particularly important because the regulatory review and assessment of the safety analysis report at a specific step requires time. The timescales for the preparation of the safety analysis report and the schedule for its submission have to be agreed between the operating organization and the regulatory body, including the period of time needed to complete the regulatory review and assessment before proceeding with the next stage.

5.2.1. Site evaluation

Sufficient information has to be provided to allow the regulatory body to assess the site characteristics of the proposed facility and their effect on the design. The practice in many Member States is that such information is provided in the site evaluation report. If required by the regulatory body, the site evaluation report could be supported by an environmental impact assessment and by a preliminary statement on potential radiological impacts on the site personnel, the public and the environment. This information could also be combined in the form of a preliminary safety analysis report.

5.2.2. Design and construction

The information included in the safety analysis report has to be sufficient to demonstrate that the design will result in a safe facility, and that a construction programme is in place to ensure that the design's intent is achieved. This information has to include a description of the design of the facility and its associated SSCs, including facility layout and process systems. The safety analysis (including its results and findings) has to be included. Any updated information on site characteristics and their effects on the design also needs to be included.

The information in the safety analysis report at this stage includes the main content of the facility construction programme. This may cover:

— The type of construction;
— Quality controls in manufacturing, construction and installation;
— The management system for the construction stage;
— Construction tests and verification;
— The management of interfaces;
— Control of contractors;
— Transfer of responsibility to the commissioning stage.

Outlines of the commissioning programme and decommissioning plan also have to be included in the safety analysis report at this stage. Aspects such as design provisions for the control of internal and external exposure, protection against non-radiological hazards, radioactive waste management, ageing management, emergency planning and decommissioning are also covered by the scope of the document.

Some of the information mentioned above may be provided in separate documents, but a summary of relevant information with adequate reference to the details needs to be included in the safety analysis report. This may also need updating as the facility's detailed design and construction proceed. Such updates could take the form of revised versions of documents, or in some cases technical supplements.

5.2.3. Commissioning

At this stage, the safety analysis report needs to be updated to include the 'as-built' drawings of the facility as well as any modifications that may have been introduced to the facility's SSCs or processes as a result of the construction work. These may require an update to the safety analysis. The main objective of the report is to provide sufficient information to support a key regulatory decision on operational readiness.

The information in the safety analysis report at this stage has to cover the commissioning programme (including the management system, commissioning stages, as applicable, commissioning tests and procedures, etc.), operational limits and conditions (which become active at the commissioning stage, if not before), operational radiation protection and waste management programmes, the maintenance programme, emergency preparedness and response, the training and qualifications of the facility operating personnel and any updates to the preliminary decommissioning plan.

5.2.4. Operation

At this stage, summaries and updated information on all the topics referred to in the preceding sections have to be included. These include the results and conclusions of commissioning and updated versions of the operational limits and conditions. The organizational structure for the facility operating stage, updates of the facility's operating procedures and the management system for the operating stage also have to be included.

Assessments of normal radiological and chemical exposures at this stage can be based on actual data.

The final safety analysis report needs to be prepared for the stage of application for an operating licence. This includes all the elements relevant to the site evaluation, design, construction, commissioning, operation and provisions for decommissioning of the facility. The safety analysis report has to be subject to regular review, in accordance with the requirements of the management system, to account for modifications and changes introduced to the facility and for operating experience feedback.

5.3. CONTENT OF THE SAFETY ANALYSIS REPORT

The content of the safety analysis report provided in this section was developed to be coherent with that for other types of nuclear installation, such as nuclear power plants and research reactors, as provided by the IAEA Safety Standards [4, 6] and considering practices adopted in Member States [25–29]. The Appendix to this publication elaborates on the content of the safety analysis report for nuclear fuel cycle facilities.

Alternative approaches exist in Member States with respect to the structure and content of the safety analysis report for nuclear fuel cycle facilities and, in accordance with use of a graded approach, to the amount of information and depth of the analysis provided by this publication.

It is important to take into account and properly manage the interfaces between the different chapters of the safety analysis report.

Following is an outline of the content of a safety analysis report:

(a) Chapter 1: Introduction and general description of the facility

This chapter introduces the safety analysis report and provides general information on the facility to give an adequate overall understanding of the facility. It has the following parts:

— Statement of objective of the safety analysis report;
— Stages of the facility's lifetime and its current status and existing authorization(s);
— General description of the facility: Includes consideration of applicable regulations, codes and standards, basic process routes, processes and technical characteristics, information on the layout and other aspects, operating modes and materials involved;
— Safety features: States the safety principles adopted for the design, construction and operation of the facility, as well as acceptance criteria for the safety analysis;
— Management of safety: Gives a brief description of the relevant management processes;
— Safety analysis overview.

(b) Chapter 2: Site characteristics

This chapter provides information on the geological, seismological, hydrological and meteorological characteristics of the site and the vicinity, in conjunction with present and projected population distributions and information on land use, site activities and planning controls. It includes a description of the site reference data, evaluation of site specific hazards, and information about site related issues in emergency planning and accident management and monitoring of site related parameters.

(c) Chapter 3: Safety objectives and general design requirements

This chapter provides descriptions of the following:

— Safety objectives of the facility;
— Design principles and criteria and engineering design requirements of SSCs important to safety;
— Classification of SSCs;
— Civil engineering works and structures;

— Equipment qualification and environmental factors;
— Human performance criteria;
— Protection against internal and external hazards;
— Compliance with national and international standards.

(d) Chapter 4: Processes and items important to safety

This chapter provides descriptions of the main processes at the facility and principal SSCs important to safety to demonstrate their conformance to the design requirements. It covers the following:

— Buildings and structures;
— Processes and process systems used at the facility;
— Containment and shielding systems;
— Cooling systems, as applicable;
— Instrumentation and control system;
— Electrical power systems;
— Auxiliary systems (e.g. fire protection systems, chemical control, heating systems, reagent feed systems, supply of cooling water, steam, service air, compressed gas);
— Human factors engineering;
— Waste management systems and systems for control of discharges;
— Handling and on-site transport systems and operations;
— Others.

(e) Chapter 5: Radiation protection

This chapter covers the description of the following:

— Radiation protection objectives, criteria and principles, including application of the principle of optimization of protection and safety;
— Radiation sources in the facility;
— Design features for radiation protection;
— Operational radiation protection programme, including monitoring programme and operating procedures;
— Dose assessment for normal operation of the facility.

(f) Chapter 6: Nuclear criticality safety

This chapter provides information on the policy, strategy, methods and provisions for control of criticality in design and operation of the facility.

(g) Chapter 7: Safety analysis

This chapter forms an important part of the safety analysis report. It provides the description and summary of safety analysis performed to confirm the ability of the facility to control the hazards and respond to events according to the acceptance criteria.

(h) Chapter 8: Operational limits and conditions

This chapter describes the operational limits and conditions of the facility:

— Bases for development;
— The specification(s) of the operational limits and conditions;
— Safety limits;
— Safety system settings;
— Limiting conditions for safe operation;
— Periodic testing and surveillance;
— Administrative control.

(i) Chapter 9: Conduct of operations

This chapter provides information on the following:

— Organizational structure for facility operation;
— Communications within the facility and the site;
— Operating instructions and procedures, including emergency operating procedures;
— Maintenance, periodic testing and inspection;
— Ageing management programme;
— Control of modifications;
— Training and qualification programme for operating personnel;
— Review and audits;
— Records;
— Feedback of operational experience;
— Nuclear safety and security interfaces.

(j) Chapter 10: Waste management and management of radioactive discharges

This chapter provides information on the management of radioactive waste, including control of radioactive waste, design provisions and practices to minimize the generation and accumulation of waste, waste processing and storage, preparing waste for disposal, as well as management of radioactive discharges from the facility.

(k) Chapter 11: Non-radiological hazards

This chapter provides information on protection against chemical hazards, fire and explosion and industrial hazards such as high pressure pipelines and vessels and lifting equipment. The information demonstrates that these hazards are prevented and controlled by design and operation measures according to the regulatory requirements.

(l) Chapter 12: Environmental impact assessment

This chapter provides information on the radiological and non-radiological impacts of the facility at all stages of its lifetime. This chapter has to demonstrate that licensing processes for safety and the environment are coordinated and potential environmental radiological and non-radiological impacts of the facility are acceptable and controlled in accordance with the national requirements. Non-radiological (toxic) effects associated with releases have to be considered in this chapter if required by national regulations.

(m) Chapter 13: Commissioning

This chapter describes the commissioning programme of the facility, including its results and conclusions. It has to demonstrate that the commissioning programme was carried out and that the safety functional requirements of the SSCs were adequately verified.

(n) Chapter 14: Management system

This chapter describes the management system at all stages of the facility lifetime, including the quality assurance programme required at different stages.

(o) Chapter 15: Emergency preparedness and response

This chapter states the emergency categorization of the facility and describes the arrangements for emergency preparedness and response. For facilities with a high hazard category, severe accident management guidance may be outlined in this chapter.

(p) Chapter 16: Preparation for decommissioning

This chapter provides information on the facility's design provisions for decommissioning, including operating practices and procedures to facilitate its ultimate decommissioning and elements of the decommissioning plan.

6. MANAGEMENT SYSTEM FOR SAFETY ANALYSIS AND LICENSING DOCUMENTATION

Quality needs to be ensured, through formal management system processes, in all activities related to the development and updating of safety analysis and licensing documentation. In accordance with para. 4.8 of SSR-4 [1]:

> "The operating organization [of a nuclear fuel cycle facility] shall establish and apply a single coherent management system in which all the constituents of the organization, including its structure, resources and processes, are integrated to enable the organization's objectives to be achieved".

This system has to integrate safety, health, environmental, security, quality and economic objectives for the nuclear fuel cycle facility. The management system documentation has to describe the system that controls the development and implementation of all aspects of the facility, including safety analysis and licensing documentation. The management system covers four functional categories: management responsibility; resource management; process implementation; and measurement, assessment and improvement. In general, these categories include the following:

(a) Management responsibility includes providing the means and support needed to achieve the organization's objectives.
(b) Resource management includes measures to ensure that resources essential to the implementation of strategy and the achievement of the organization's objectives are identified and made available.
(c) Process implementation includes those actions and tasks needed to achieve high quality.
(d) Measurement, assessment and improvement provide an indication of the effectiveness of management processes and work performance.

A management system for safety analysis and licensing documentation has to be implemented by the operating organization, and needs to include a means of establishing controls over activities related to safety analysis and licensing documentation to provide confidence that they are being performed in accordance with requirements. In setting up the system, a graded approach based on the relative importance to safety of each item or process (e.g. identification and selection of PIEs, evaluation of event sequences, preparation of safety analysis report) and on the potential hazard of the nuclear fuel cycle facility could be applied. The objective of the management system is to ensure that the facility meets the requirements for safety as derived from the requirements of the regulatory body, design requirements, assumptions made in the design and administrative requirements for facility management. The management system supports the development and maintenance of a strong culture for safety in all aspects of safety analysis and licensing documentation.

6.1. MANAGEMENT RESPONSIBILITY

The operating organization has the prime responsibility for the safety of the nuclear fuel cycle facility. The operating organization's management system for safety analysis and for licensing documentation has to provide a framework to manage and assess the preparation, performance, review and application of the results of the safety analysis, as well as activities for the preparation, update and review of licensing documentation. The organizational structure, along with the functional roles, responsibilities, authorities and communication lines of the groups and individuals performing these activities, have to be clearly identified and documented.

The following practices and factors have been shown to be effective for the successful implementation of activities related to safety analysis and licensing documentation:

(a) Planning and prioritization of work;
(b) Addressing all applicable regulatory requirements, codes and standards, including those related to chemical and industrial hazards;
(c) Establishing means (formal and informal) for interactions between the operating organization and regulatory body from the early stages of performing safety analysis and preparation of licensing documentation;
(d) Availability of qualified personnel with suitable skills;
(e) Availability of appropriate computational methods and tools;
(f) Availability of acceptance criteria for safety analysis;
(g) Availability of approved procedures;

(h) Establishment of effective means of exchange of information between designers, safety analysts and the facility's operating personnel;
(i) An environment with a strong safety culture.

Record keeping is an important aspect of the management of safety analysis and licensing documentation. Records essential to the performance and verification of these activities have to be controlled at all stages, including identification, approval, review, filing, retrieval and disposal.

6.2. RESOURCE MANAGEMENT

Adequate resources (both human and financial) have to be provided by the operating organization for performing the safety analysis and for preparing, reviewing and updating the licensing documentation for the facility. Safety analysts need to be trained and qualified for the job and their qualifications need to be adequately documented. Similarly, the personnel involved in the preparation, review and update of the licensing documentation need to be trained and qualified for these activities. The operating organization has to specify the required competence of staff performing these activities. Their responsibility extends to the supervision of personnel external to the organization who are involved in such activities, including ensuring that these personnel are adequately trained and qualified for the job.

Performing safety analysis, in particular transient and accident analyses, requires the use of tools (e.g. computer codes) that are validated for use for safety demonstration and for the conditions of the nuclear fuel cycle facility under consideration. These tools have to be identified, provided, maintained and controlled to ensure their proper use. In cases where equipment is needed to collect experimental or operational data for the purpose of safety analysis (e.g. for the verification and validation of computer codes), such equipment has to be calibrated as needed and qualified for the relevant environmental conditions.

6.3. PROCESS IMPLEMENTATION

Safety analysis and licensing documentation activities have to be carried out in accordance with written and approved procedures and instructions. These will include a documented method of safety analysis approved prior to performing the analysis. This documented method covers the models to be used, system assumptions, acceptance criteria and system nodalization. Review and approval (by management) of these items prior to performing the analysis reduces the risk of subsequently needing to re-perform the work due to errors or inappropriate use of tools or methods.

Additionally, the safety analysis and calculations supporting the licence application need to be documented to allow their independent review. Effective control of non-conformance with procedures, as well as of corrective actions, needs to be introduced. References to methods and tools used in the safety analysis need to be documented, as well as sources of data.

6.4. MEASUREMENT, ASSESSMENT AND IMPROVEMENT

The operating organization has to ensure that verification of the safety analysis is performed by individuals or groups independent from those carrying out the design and analysis. This could cover modelling and calculations related to transient and accident analysis. Individuals are considered to be independent if they have not participated in the parts of the design subject to safety analysis. This independent verification by the operating organization has to be performed in addition to the review and checks carried out by the design organization/entity.

Some operating organizations may request the support of technical organizations or independent peer review groups in reviewing the safety analysis or licensing documentation (or parts thereof). If this is the case, the operating organization, in carrying out its responsibility for safety, has to understand the results of such reviews and adopt their results, as appropriate.

Audits may also be performed with the objective of evaluating the adequacy and effectiveness of all aspects related to the safety analysis and licensing documentation, and compliance with the requirements of the management system. The operating organization has to evaluate the results of such audits and take necessary actions to make improvements.

Appendix

INDICATIVE CONTENT OF SAFETY ANALYSIS REPORT FOR A NUCLEAR FUEL CYCLE FACILITY

This Appendix elaborates on the information provided in Section 5.3 of this publication on the content of the safety analysis report for nuclear fuel cycle facilities.

The safety analysis report is a high level document that incorporates the information required at various steps in the licensing process of a nuclear fuel cycle facility. A graded approach can be applied to the content of the safety analysis report in accordance with the potential hazard of the facility, its size, the complexity of its design and the licensing stage. The scope, details and amount of information required, as well as the depth of the analysis, for a small or low-hazard fuel cycle facility (e.g. a research and development facility or radioactive waste management facility) could be substantially smaller than those needed for a complex facility with high potential hazards (e.g. nuclear fuel reprocessing). Discussion on the use of a graded approach is also provided in Annex I to this publication and in this Appendix.

The Appendix is divided into sections under headings that may be appropriate for the chapters of a safety analysis report. Alternative approaches may be used for the structure of the safety analysis report, provided that the content of the report is adequate to meet its purposes.

A.1. CHAPTER 1: INTRODUCTION AND GENERAL DESCRIPTION OF THE FACILITY

This chapter of the safety analysis report provides an introduction to the report and general information on the facility in order to give an adequate overall understanding of the facility.

A.1.1. Introduction

The following information is outlined:

(a) Description of the project that provides its specific objectives, background, various stages involved and current status;

(b) Statement of the main objective of the safety analysis report and a description of its structure, the objectives and scope of each of its chapters and the connections between them;
(c) Description of the lifetime stage of the facility and existing authorization status of the facility;
(d) General description of the facility and summary of the principal characteristics of the facility and its site;
(e) Operational history of the facility, including any major changes that have been introduced to the facility;
(f) Similarities to other nuclear fuel cycle facilities (any significant differences from other facilities could also be described here), including those in other countries, as appropriate;
(g) Identification of the owner, the operating organization and other responsible parties (including vendors, contractors, etc.).

This section of the report can also provide detailed lists of references, describe the structure of the report or give lists of tables or figures in the report.

A.1.2. General description of the facility

This section of the safety analysis report provides a general description of the facility, including specified basic technical characteristics, arrangements and layout, processes and technologies used, operating modes, safety justification of related activities and provisions for decommissioning. The description also has to cover applicable regulations, codes and standards, and source material referenced. Specifically, this section covers the following:

(a) Facility background and mission: This includes the purpose and mission of the facility and anticipated future changes in these.
(b) Overview of the facility: This includes a description of the location of the facility and of its physical and institutional boundaries, relationships and interfaces with nearby facilities, layout, and significant interfaces with external systems and operations such as utilities, fire response and medical assistance.
(c) Identification of (and basis for) the hazard category of the facility.
(d) Description of use of a graded approach in accordance with the identified hazard category of the facility.
(e) Consideration of applicable regulations, codes and standards.
(f) General description of inputs, outputs and materials in the facility:
    — Types and maximum quantities of radioactive and other hazardous materials involved and their chemical and physical characteristics;

— The isotopic composition or enrichment of nuclear materials;
— Quantities and types of waste and effluents generated.

(g) An overview of the facility operations, basic processes, process routes and technical specifications of the facility and its SSCs.
(h) Description of the facility operation including operational modes and characteristics.

A.1.3. Safety features

This section briefly states the safety principles adopted for the facility's design, construction and operation, including the safety criteria and acceptance criteria used for safety analysis. The key safety features of the facility and the SSCs that were examined in the safety analysis are identified and described here.

The report identifies how the hierarchy of design measures for protection against potential hazards has been applied (see para. 6.68 of SSR-4 [1]).

A.1.4. Management of safety

This section provides an overview of the facility's owner, operating organization, designer and vendor (as applicable), prime contractors and consultants. The management system of the operating organization is outlined and the ability of the organization to address the challenges associated with different stages of the facility lifetime is demonstrated.

This section briefly introduces the management of safety as an integral component of the management structure of the operating organization. The management structure of the operating organization is evaluated in this section of the report, and the procedures and processes that have been established to achieve adequate control of all aspects of safety throughout the lifetime of the facility are described. The aim is to demonstrate that the operating organization is able to fulfil its prime responsibility for safety throughout the lifetime of the facility. The description covers the specific aspects of the management processes and measures for the monitoring and review of safety performance.

A.1.5. Safety analysis overview

This section provides an overview of the results of the safety analysis, including basic safety functions, safety objectives, safety criteria and the design principles of SSCs. A description of the results of the safety analysis and its main conclusions has to be provided in this section to ensure that the design provides an adequate level of safety and meets the regulatory requirements.

A.2. CHAPTER 2: SITE CHARACTERISTICS

This chapter of the report provides information describing the characteristics of the site and its adequacy from the safety point of view. This requires the identification and assessment of site characteristics affecting, or potentially affecting, the facility and the effects that the facility has, or may have, on its surroundings (see para. 5.1 of SSR-4 [1]). It describes the location of the facility on the overall site, shows facility boundaries and identifies any nearby facilities that could affect the safety of operations. This chapter also provides information on external events, both natural and human induced, that supports the assumptions and conclusions of the safety analysis.

The information has to cover the characteristics of the site and, as necessary, the region (including geological, seismological, hydrological and meteorological characteristics), in conjunction with the present and projected population distribution, land use and site activities that are relevant to the safe design and operation of the nuclear fuel cycle facility.

If a separate report has been prepared for site evaluation of the facility, it needs to be cited and a summary may need to be included in this chapter. The site evaluation report may also include information on the environmental radiation monitoring programme for the site and other elements of the site evaluation, as required by the regulatory body.

This chapter has to include sufficient information to demonstrate compliance with applicable requirements for site evaluation, as established in SSR-4 [1] and NS-R-3 (Rev. 1) [13]. The description of the site characteristics needs to include the following, as appropriate:

(a) Geography and demography, including site location and facility area boundary, basic geographic information, uses of land and water resources, and population distribution.
(b) Site reference data, including detailed area maps (at suitable scales), as necessary.
(c) Specific features of the site relevant to the safety analysis:
    — Surface and subsurface hydrology;
    — Local and regional meteorology;
    — Geology, seismology and geotechnical engineering;
    — Evaluation of site specific external hazards.
(d) Proximity of industrial, transportation and other facilities, local and regional transportation routes (roads, railroad tracks and airports), electrical transmission lines, natural gas pipelines, oil or natural gas storage depots and local industrial facilities.

(e) Radiological conditions due to external sources, including baseline radiological levels.
(f) Site related assumptions made in prior environmental analyses or impact statements (or revised and updated environmental statements for the facility).
(g) Site related issues in emergency preparedness.
(h) Monitoring of site related parameters.

On multifacility sites, the safety analysis report has to consider the site as a whole to establish whether hazards from interactions between facilities have been taken into account.

A programme of monitoring throughout the lifetime of the facility to evaluate natural and human-made changes in the area, including changes in demographics, has to be described and reviewed periodically based on the results of monitoring and possible changes to site characteristics.

It has to be confirmed that appropriate arrangements are in place to periodically update the evaluations of site specific hazards in accordance with the results of updated methods of evaluation, monitoring data and surveillance activities.

This chapter of the safety analysis report has to demonstrate that hazards arising from external events have been considered in the design of the nuclear fuel cycle facility and can be compensated for by means of engineered features, site protection measures or administrative controls. The information has to confirm the suitability of the site for a nuclear fuel cycle facility, in sufficient detail to support the safety analysis relevant to the site.

The scope and details of the information presented can be proportionate to the potential hazard level of the facility and processes associated with it, in accordance with the use of a graded approach. This may mean that the amount of information that needs to be included in this chapter for low potential hazard nuclear fuel cycle facilities can be substantially reduced. For such facilities, it is not generally necessary to discuss in detail the geological, seismological or meteorological conditions, hydrology and off-site accident effects, because accident consequences are limited to the facility itself. However, if such a facility could release radioactive or other hazardous chemicals off-site, this chapter provides sufficient information on site meteorology and hydrology.

A.3. CHAPTER 3: SAFETY OBJECTIVES AND GENERAL DESIGN REQUIREMENTS

This chapter provides a description of the safety objectives and general design requirements that have been followed in the design of the facility in consideration of normal operation, AOOs and accident conditions. Compliance of the facility design with specific technical safety requirements is to be demonstrated in more detail in a separate report (in this case, that report needs to be summarized and cited in this chapter of the safety analysis report).

This chapter defines the safety objectives and design principles of SSCs, in particular the implementation of the defence in depth concept, use of inherent safety features or passive safety systems, use of high engineering standards in design (e.g. conservative safety margins) and extent of use of engineering design principles for high reliability (e.g. redundancy, diversity, physical separation, fail-safe), consideration of human factors and provisions for radiation protection. The design requirements for processes also have to be described in this chapter.

The following specific items have to be included in this chapter:

(a) Safety objectives, including safety philosophy and safety principles used in the project.
(b) Safety functions: Identification and justification of the fundamental safety functions to be fulfilled by the specific facility design.
(c) Design principles and criteria.
(d) Defence in depth: Description in general terms of the design approach adopted to incorporate the defence in depth concept into the design of the facility.
(e) Codes and standards employed for the design of the facility and justification of their applicability.
(f) Classification of SSCs: Basis for classification and list of the main SSCs important to safety, together with their related safety functions and associated criteria.
(g) Protection against external and internal hazards: Design criteria for the resistance of SSCs to external and internal events.
(h) General design aspects for civil engineering works of safety classified buildings and civil engineering structures.
(i) General design aspects for SSCs important to safety, including mechanical systems, instrumentation and control systems, and electrical systems.
(j) Specific design requirements of the facility (e.g. requirements for redundancy and maintenance).
(k) Equipment qualification: The scope of the qualification programme and the qualification procedures adopted.

(l) Human performance criteria.
(m) Compliance with national and international standards.

The last part of this chapter of the safety analysis report has to conclude that the facility is designed to meet the overall safety objective and that appropriate external and internal events, codes, standards and design methods have been considered in the design of the facility, including for the qualification of equipment.

A.4. CHAPTER 4: PROCESSES AND ITEMS IMPORTANT TO SAFETY

This chapter of the safety analysis report provides a description of the main processes used at the facility and the principal SSCs important to safety to demonstrate their conformance to the design requirements. It also needs to demonstrate that the SSCs and engineered safety features that are incorporated into the design are capable of ensuring the fulfilment of the main safety functions and meet the safety objectives of the facility and design requirements.

The description of SSCs important to safety has to include the following:

(a) Description of the SSCs and their functions and interfaces with other SSCs.
(b) Design bases and requirements for SSCs:
  — Safety functions and associated performance requirements as part of the defence in depth concept for various facility states;
  — Ambient conditions and the design criteria derived from these conditions;
  — Internal and external hazards that may affect SSCs;
  — Safety class of SSCs and the associated requirements of the engineering design principle (e.g. functional isolation, physical separation, redundancy, diversity);
  — Requirements on the construction materials for SSCs;
  — Description of the analyses, verification and tests (carried out or anticipated) for validation of the system and its structures and components;
  — Requirements for maintenance, inspections and testing.
(c) Operation and use of the system in different facility states.
(d) Interaction with other SSCs and auxiliary systems.
(e) A summary of the results of the failure analysis of the SSCs, including human errors and the prevention of fault propagation.

The information to be presented in this chapter of the safety analysis report depends on the particular type and design of the nuclear fuel cycle facility. The level of detail required in the facility description is based on the facility's hazard category, the safety class of the SSCs and the complexity of the safety analyses.

This chapter of the report also cites technical design documents for the facility that could aid understanding of the provided description. This description needs to be supported by engineering drawings, as necessary. The processes involved in the facility have to be described, with supportive illustrations such as drawings, flow diagrams and tables. Information that is necessary for understanding the assumptions and event sequences of the safety analysis has to be clearly given. Details of the selection of materials for SSCs, as well as consideration of aspects such as ageing, irradiation and contamination, also have to be provided in this chapter (or supporting technical documents have to be referenced).

For nuclear fuel cycle facilities, the items that can be described in this chapter of the safety analysis report include the following:

(i) Buildings and structures;
(ii) Processes and process systems used at the facility;
(iii) Containment and shielding systems;
(iv) Cooling systems if applicable;
(v) Instrumentation and control systems;
(vi) Electrical power systems;
(vii) Auxiliary systems;
(viii) Human factors engineering;
(ix) Waste management systems and systems for management of discharges;
(x) Handling and on-site transport systems and operations.

This list depends on the type of facility. The other SSCs important to safety may be described here, as applicable for the specific facility.

SSCs providing functions that are relevant to other chapters of the safety analysis report may be described under the corresponding chapter and referenced here.

A.4.1. Buildings and structures

This section of the safety analysis report provides a description of the facility's safety classified buildings and civil engineering structures (e.g. process building and internals, supporting structures, auxiliary buildings and other structures important to safety).

The information has to include the following:

(a) Applicable codes, standards and other specifications;
(b) Loads and load combinations;
(c) Design and analysis procedures;
(d) Structural acceptance criteria;
(e) Materials, quality control and special construction techniques;
(f) Zoning for fire protection, radiation protection and other purposes;
(g) Description of penetrations;
(h) Testing and in-service inspection requirements.

The description has to include the design basis of the building and structures, together with the design basis of the building penetrations (air locks, doors, etc.) in relation to their resistance to internal and external events. The characteristics of the buildings and structures of the facility that influence the equipment and processes needed for safe functioning have to be presented to show how they have influenced the design of the remainder of the facility and its operations and to demonstrate the adequacy of the safety of the facility.

Sufficient information, including construction details such as floor plans, equipment layout, construction materials and dimensions, and features relevant to hazard and accident analysis, has to be provided to allow for an overall understanding of the facility's structure as it pertains to safety analysis.

It is helpful to include technical drawings to support the description mentioned above. Features of the facility layout that interact with other elements of the safety analysis can be highlighted here (e.g. emergency exits and those security features that are not classified).

A.4.2. Processes and process systems

This section defines the main technological processes within the facility. For each process, the information provided has to include the following information:

(a) Process routes.
(b) Major material inventories; types and quantities of radioactive and other hazardous materials.
(c) Process flow diagrams and description of process steps.
(d) Process equipment.
(e) Control of the process: associated instrumentation and control systems and equipment.

(f) Process operations including major interfaces between SSCs and considerations relating to human performance and the human–machine interface.
(g) Process parameters, operational limits and conditions.
(h) Principal chemicals and energy sources used.

Hazards inherent to the process and discussion on the control of these hazards and associated protection measures may be summarized or referenced in this section.

Information has to be provided on the process design configuration, dimensions, materials of construction, pressure and temperature limits, corrosion allowances and any other operating limits in sufficient detail to provide support to the safety analysis. The typical process routes and processes for various types of nuclear fuel cycle facilities are provided in the relevant IAEA Safety Guides [7–12].

Descriptions of the SSCs that are part of the process also have to be included in the safety analysis report, including their safety functions, functional requirements, performance criteria and evaluation as described earlier in this section. Depending on the type of the nuclear fuel cycle facility, the process systems could include systems such as feed and delivery systems, dissolving systems, extraction systems, calciners, fluidized beds, furnaces, separators, and off-gas treatment systems. For each process system, the information has to be based on the description of SSCs specified above and include the following:

(i) The main parts and components of the system;
(ii) Interfaces with other systems of the facility, including auxiliary systems;
(iii) Process and instrumentation diagrams;
(iv) Monitoring and control of the process system functions;
(v) Operating parameters in different operational conditions (e.g. pressures, temperatures, flow rates, cooling capacities);
(vi) Levels of alarms and interlocks;
(vii) Protection functions and limits related to the operation of the system.

The information needs to be sufficient to demonstrate that the processes can be performed safely, and the process systems are capable of performing their intended functions and meet the design requirements.

A.4.3. Containment and shielding systems

The identification and description of SSCs that perform confinement and shielding functions is provided in this section of the safety analysis report.

For nuclear fuel cycle facilities, containment systems provide confinement of radioactive materials or hazardous chemicals and include static containment (such as building containment, process pipework, fittings and vessels, gloveboxes, casks) and dynamic containment (ventilation).

An adequate description of containment and shielding systems needs to be provided based on the description of SSCs specified earlier in this section and covering the following:

(a) For shielding systems:
   — Description of shielding means for each radiation source, including the dimensions, layout and materials.
(b) For confinement:
   — Zoning for contamination and location of barriers with pressure differentials employed.
(c) For ventilation systems:
   — Description of ventilation systems for each building within the facility, including the building volume, flow rates and specifications of filters;
   — Description of the ventilation system operation modes;
   — All other safety functions of the ventilation system, such as dilution of hazardous gases, heat removal and control of smoke in a fire.

These systems can be considered in chapter 5 of the safety analysis report with an appropriate reference here.

A.4.4. Cooling systems

Some nuclear fuel cycle facilities have cooling systems for the removal of heat from radioactive decay or chemical reactions (e.g. from spent fuel, plutonium and high activity processing and storage systems). In this case, this section of the safety analysis report provides a detailed description of the cooling system and its characteristics, modes, system parameters (flow rate, temperatures, pressure, etc.) and the support systems necessary for safety.

Any engineered safety features employed (e.g. those ensuring natural convection cooling in the case of electrical power supply cut) must be justified. Relevant characteristics of the system that have been used in safety analysis of the facility must be described.

Descriptions of the system need to be supported by process and instrumentation diagrams as well as flow diagrams. Specifications of the SSCs involved in the cooling systems (e.g. pumps, heat exchanger valves, fans) have to be provided (possibly in a supporting technical document). In some facilities, ventilation systems also provide cooling. In this case, reference could be made to a description of the ventilation cooling function.

It has to be demonstrated that the cooling system has been designed and built to prevent overheating resulting in the loss of other safety functions and the consequent release of radioactive and other hazardous materials. It has to be shown that its capacity, availability and reliability are consistent with the safety analysis, and meet design and regulatory requirements.

A.4.5. Instrumentation and control systems

A detailed description of the instrumentation and control systems required for monitoring and control of the process parameters necessary for safe operation in all operational states is provided in this section of the report. This includes the facility control systems, indicating and recording instrumentation, alarm systems and communications systems, including, as applicable, control rooms.

For nuclear fuel cycle facilities, the instrumentation and control systems could include process control systems, systems for monitoring and controlling static and dynamic containment, systems for monitoring of internal and external hazards and their related alarms, systems for monitoring of effluents and instrumentation for accident monitoring. Process parameters that need to be reliably controlled can include pressure, temperature, level, flow and speed. Internal hazard monitoring may include radiation detection, criticality monitoring and criticality alarm systems, fire detectors and hazardous gas monitoring. External hazard monitoring may include instrumentation for seismic, flood and other external hazards.

Adequate descriptions need to be provided, with the following information:

(a) The design basis for the instrumentation and control system;
(b) Description of the overall instrumentation and control system architecture, with schematic diagrams including system interfaces, interactions between systems and connections to the outside environment;
(c) Safety classification of instrumentation and control systems, including hardware and software classification;
(d) System calibration, testing and surveillance;
(e) The human–machine interface;
(f) Prioritization of the commands given by the instrumentation and control systems;
(g) Functional and physical separation between instrumentation and control for normal operations and accident conditions;
(h) Software quality, including its verification, validation and life cycle processes;

(i) Instrumentation and control in the main control room as applicable.

This section has to provide relevant information to demonstrate that the instrumentation and control systems meet design requirements for all facility states and modes of operation.

Some instrumentation and control systems (e.g. radiation monitoring systems, criticality control and alarm systems, chemical monitoring) could be considered in other sections of the safety analysis report with the relevant references here.

A.4.6. Electrical power systems

The information in this section of the safety analysis report provides a detailed description of the electrical power supplies, including on-site alternating current and direct current power systems, off-site power systems and emergency power supply where necessary, with an emphasis on their dependability and their relationship to safety. This description needs to be supported by adequate diagrams and includes the following:

(a) General design principles and approaches;
(b) A main diagram outlining the integrated structure of all electrical systems;
(c) The structure and operating parameters of each system (e.g. voltages);
(d) Monitoring and control of the systems;
(e) Switching between different operating conditions.

Ageing effects on items such as cables, panels and switches that could impact safety also have to be discussed.

The adequacy of each power supply needs to be demonstrated. The off-site electrical power supply and emergency power supply have to be described, with adequate information on their design and performance characteristics. It has to be demonstrated that electrical power supply systems have the required availability, sustainability and reliability, with provisions for an emergency power supply where necessary.

A.4.7. Auxiliary systems

This section of the safety analysis report covers all systems that support the operation of the nuclear fuel cycle facility. These include fire protection systems; chemical control systems (including sampling and analysis systems); reagent feed systems; systems supplying cooling water, steam, service air and compressed gas; and heating systems. The purpose of each system has to be described, along with an overview of the system and its principal components, functions and operations, and an analysis of performance. Technical design documents are included or cited here, if required. Descriptions are given at the level of detail necessary for understanding the utility distribution philosophy and facility operations and demonstrating that these systems perform their intended function with the required reliability and meet the design requirements.

For nuclear fuel cycle facilities that rely on ‘site-wide’ support services, organizations and procedures, their interfaces with the facility have to be specified.

A.4.8. Human factors engineering

This part of the safety analysis report describes the human factors engineering applied in the facility design to meet Requirement 27 of SSR-4 [1]. All operational states and accident conditions need to be considered at facility areas where interactions between the operators and the facility have safety implications. The human factors engineering considerations presented in the safety analysis report cover the following:

(a) The choice of human–machine interface design, with account taken of human factors engineering;
(b) The human factors analysis methods that were applied;
(c) Description and resolution of human factors engineering issues that were identified during the design stage and the assumptions made during analyses;
(d) A description of how the human–machine interface design has been implemented;
(e) A description of the strategy for monitoring human safety performance.

It has to be demonstrated in this section that human and organizational factors and human–machine interface issues have been properly considered throughout the design, assessment and management and cover all operational modes and states.

A.4.9. Waste management systems and systems for management of discharges

This section covers the facilities and equipment dealing with collection, monitoring and treatment of solid, liquid and gaseous radioactive wastes, as applicable. The section also includes systems and components which control the discharges to the environment.

A.4.10. Handling and on-site transport systems and operations

The section contains information on the handling of radioactive material and its transport on the facility site. The information is provided in accordance with the description of SSCs specified above and will also include the following:

(a) Description of the operations and equipment used for handling and on-site transport, including lists of the SSCs and vehicles used;
(b) Description of measures to prevent damage to or dropping of packages;
(c) Mitigating measures for accident conditions during handling and transport operations.

The strength, durability and reliability of handling and transport equipment in external and internal impacts needs to be demonstrated, as well as its compliance with operational controls and design requirements.

A.5. CHAPTER 5: RADIATION PROTECTION

The safety analysis report has to contain sufficient information to demonstrate compliance with the requirements on radiation protection as established by Refs [1, 30]. As a minimum, the following topics are addressed in the discussion:

(a) Radiation protection objectives, criteria and principles;
(b) Radiation sources within the facility;
(c) Design features, methods and provisions for radiation protection;
(d) Dose assessment;
(e) Operational radiation protection programme.

The radiation protection programme can be presented in a separate document. If this is the case, a summary has to be included in this chapter of the safety analysis report with adequate reference to the document describing the radiation protection programme.

A.5.1. Radiation protection objectives, criteria and principles

This section describes the operating organization’s policy for radiation protection and the application of the principle of optimization of protection in the design and operation of the nuclear fuel cycle facility. The information has to cover the radiation protection objectives of the design and a description of the dose limitation system for workers and the public, including requirements for the optimization of protection. In particular, this section has to summarize the authorized dose limits for both occupationally exposed personnel and the public, as well as the effluent limits based on these dose limits. The application of appropriate dose constraints and reference levels also has to be described.

A.5.2. Radiation sources within the facility

All potential radiation sources (including airborne radioactive material) due to nuclear fuel cycle facility operation have to be identified and described in this section. These sources are used as bases for radiation shielding calculations, design of ventilation systems, dose assessment, waste management and determination of effluent releases. Sources of radiation inside the facility and within the equipment and processes used, as well as their physical and chemical forms, quantities, activity and locations, have to be identified and described.

A.5.3. Design features for radiation protection

This section provides a detailed description of the design of the nuclear fuel cycle facility and appropriate equipment to ensure that radiation protection and contamination control are adequately provided for operational states of the facility. Demonstration that possible external and internal radiation exposure of workers and the public is controlled by the radiation protection policy has to be provided.

Information on safety features for radiation protection has to cover:

(a) Facility design and design provisions provided for radiation protection (including shielding, work area classification, ventilation);
(b) Radiation protection equipment;
(c) Features to minimize contamination.

The following information has to be provided in this section: access control and zoning of the facility from a radiation protection point of view, shielding for identified radiation sources, radiation protection aspects for ventilation, and radiation monitoring systems and the criteria for their selection and placement. This section also has to address design provisions for decontamination of equipment, if necessary.

A.5.4. Dose assessment

This section contains an evaluation of occupational exposure and doses to the public for normal operation to demonstrate the acceptability of the radiation protection arrangements and programmes and the design features at the nuclear fuel cycle facility.

Information on occupational exposure to be provided includes estimated annual occupancy data for the controlled areas of the facility. It has to be demonstrated that expected doses (including from inhalation of airborne radioactive material) are acceptable for the facility operation, experiments, maintenance and in-service inspections.

Information on doses to the public needs to be provided to demonstrate that the combined effects of direct radiation and of releases of radioactive material from the facility do not result in off-site doses that exceed authorized limits.

In addition, measures to reduce exposure based on the optimization principle have to be described.

A.5.5. Operational radiation protection programme

The operational radiation protection programme of the facility is described in this section. The description covers the administrative organization, equipment, instrumentation and facilities and procedures for radiation protection. The main elements of the information on the operational radiation protection programme are the following:

(a) Organization, staffing and the assignment of responsibilities for protection.
(b) Classification of work areas and access control.
(c) Radiation protection procedures, local rules and other documents relevant to personnel, and supervision of the work.
(d) Training on radiation protection.
(e) List of operating procedures for radiation protection, including for radiation monitoring; sampling; and decontamination of personnel, areas and equipment; and for operating and maintenance tasks that involve radiation exposure/contamination risk.
(f) Radiation monitoring programmes:
    — Personnel monitoring programme (internal and external);
    — Monitoring of the workplace within the main building and within any other buildings supporting its function, for control of radiation and contamination (surface and airborne) levels;
    — Effluent monitoring programme.
(g) Work planning and work permits.

(h) Protective clothing and protective equipment.
(i) The shielding of facilities and equipment.
(j) Radiation protection during on-site and off-site transport of radioactive materials.
(k) Establishing and maintaining records.
(l) Health surveillance.
(m) Audit and review.
(n) Investigation and reporting of any radiation accidents.
(o) Radiological protection aspects of emergency preparedness and response.

A.6. CHAPTER 6: NUCLEAR CRITICALITY SAFETY

Information needs to be given for those nuclear fuel cycle facilities where the risk of inadvertent criticality exists.

This chapter of the safety analysis report has to contain sufficient information to demonstrate compliance with the requirements for nuclear criticality safety. It has to demonstrate that the facility is designed and operated such that criticality is practically eliminated. The information to be provided includes discussions of the following:

(a) Criticality safety strategy and objectives, safety criteria and safety margins;
(b) Applicable codes and standards;
(c) Description of facilities, areas, processes, equipment, inventories, locations and quantities of fissile materials which are subject to criticality control;
(d) Criticality safety analysis process, methodology for conducting the criticality safety assessment, verification and validation of the calculation methods and nuclear data, results of the analysis;
(e) Design basis: Safety design limits, design features related to criticality prevention, control and mitigation;
(f) Nuclear criticality safety programme;
(g) Arrangements for emergency response to a criticality accident, criticality detection and alarm systems.

The information in this chapter has to cover all aspects of the prevention of inadvertent criticality that are employed by the design and operation of the facility, including the following:

(i) The criteria used to ensure subcriticality in operations and storage under operational states and credible abnormal conditions;

(ii) The parameters used for the prevention and mitigation of criticality for activities involving fissile material and the application of the double contingency principle for criticality safety;
(iii) The criticality safety design limits, their bases, and any design criteria used to ensure that criticality safety limits are not exceeded.

The expected information on the prevention of criticality accidents includes the following:

— Justification of the required subcriticality margin(s);
— Identification of conditions under which the subcriticality margin needs to be maintained;
— Criticality safety evaluations that provide evidence that subcriticality is maintained under each of the identified conditions, considering uncertainties in the input parameters and calculation method.

For nuclear fuel cycle facilities (or for activities within a given facility) where the amount of fissile material is low or the isotopic composition may be such that a full criticality safety assessment would not be justified, an explanation of the scope of the criticality safety analysis has to be provided.

The nuclear criticality safety programme also has to be described in this chapter. It needs to be demonstrated that safety measures for ensuring subcriticality are specified, implemented, monitored, audited, documented and periodically reviewed throughout the entire lifetime of the facility or activity. This programme could be described in a separate document. If this is the case, an outline of the programme needs to be provided in this section of the safety analysis report with adequate references to the separate document(s). Guidance on the elements of this programme can be found in IAEA Safety Standards Series No. SSG-27 [18].

This chapter will also include a description of the criticality detection and alarm system(s), or a justification for its omission, together with arrangements for responding to a criticality accident. The interface between these arrangements and the emergency response plan must be described for any facility where a criticality accident is credible. The emergency response plan and programme and the facility's capability to respond to credible criticality accidents also have to be discussed in this chapter of the safety analysis report, to demonstrate that all these arrangements are in place and meet the relevant requirements.

A.7. CHAPTER 7: SAFETY ANALYSIS

This chapter describes the results of the safety analyses performed to assess the safety of a facility in response to PIEs on the basis of agreed acceptance criteria. These include safety analyses performed for AOOs, DBAs and DECs. To ensure completeness of presentation and to facilitate review and assessment by the regulatory body, this chapter of the safety analysis report provides the following information as specified in Section 3.7 of this publication:

(a) Introduction: The objective of the safety analysis, acceptance criteria, general approach and methods used.
(b) Facility characteristics: The facility parameters as well as initial conditions used in the safety analysis.
(c) Identification of hazards and selection of PIEs: Hazard identification and evaluation and the PIE selection considered in the safety analysis, with a justification of this selection.
(d) Evaluation of event sequences: The sequences of selected PIEs considered in the safety analysis.
(e) Transient and accident analysis: Detailed analysis of facility systems and process performance and evaluation of consequences performed in the safety analysis.
(f) Summary and conclusions: A summary of the overall results and conclusions regarding their acceptability.

The detailed contents of this chapter of the safety analysis report (or the document on safety analysis) are outlined in Section 3.7 of this publication.

The description may be supported by reference material, where necessary. Where the safety analysis is provided in a separate document, this chapter of the safety analysis report can provide a summary of the separate document with a reference.

A.8. CHAPTER 8: OPERATIONAL LIMITS AND CONDITIONS

This chapter of the safety analysis report provides a clear and adequate description of the operational limits and conditions of the facility, together with their objectives, applicability and specifications, and the basis of these specifications. The components of the operational limits and conditions are discussed in Section 4 of this publication.

Although practices concerning the explicit inclusion of the operational limits and conditions in the safety analysis report differ among Member States, the limits and conditions are important because they are integral to the basis on which the operating organization is authorized (or licensed) by the regulatory body. In some Member States, the operational limits and conditions are presented in a separate document that is cited in the safety analysis report, where a summary of the separate document is presented.

Whatever approach is followed, it has to be demonstrated in the safety analysis report that the operational limits and conditions have been developed in a systematic way and that they are derived from a safety analysis specific to the facility. It has to be clear that the operating limits and conditions are based on the actual design of the facility with due account taken of the uncertainties in the process of safety analysis.

While Member States present information on the operational limits and conditions in different ways, a common style of presentation can be used. For clarity of presentation of the operational limits and conditions, it has been found useful to include the following attributes:

(a) Bases for development: To confirm that the operational limits and conditions are based on the safety analyses of the facility and its environment in accordance with the provisions made in the design.
(b) The objectives to be met by the establishment of operational limits and conditions (e.g. prevention of situations that may escalate to accident conditions).
(c) The applicability of the operational limits and conditions (e.g. facility status at which the limits or conditions are applicable, conditions of physical barriers against dispersion of radioactive material, equipment set up).
(d) The specification(s) of the operational limits and conditions (e.g. value of process parameter that may not be exceeded or specific conditions on SSCs).
(e) An explanation of the operational limits and conditions, particularly for the adopted specifications.

For the operational stage, this chapter has to confirm that appropriate measures are taken to ensure compliance with operational limits and conditions at all times, and that these are understood by staff, deviations are evaluated and reported as required, and the operational limits and conditions are regularly reviewed.

A.9. CHAPTER 9: CONDUCT OF OPERATIONS

This chapter of the safety analysis report provides information on the arrangements made by the organization operating the facility to ensure operational safety. Depending on the national practices in Member States, some of the topics discussed here may be included in separate documents (such as the organization structure or operating procedures). If this is the case, a brief description of these topics has to be included in this chapter with adequate references to the separate documents.

The following information on the conduct of operations for a nuclear fuel cycle facility is to be provided in this chapter:

(a) Organizational structure for operation of the facility, including communications within the facility and its site;
(b) Operating instructions and procedures;
(c) Maintenance, periodic testing and inspection programme;
(d) Ageing management programme;
(e) Control of modifications;
(f) Training and qualifications of the facility operating personnel;
(g) Review and audits;
(h) Records;
(i) Feedback of operational experience;
(j) Nuclear safety and security interfaces.

A.9.1. Organizational structure

The information has to include a clear description of the organization chart for operation of the facility with definitions of the roles, responsibilities and duties of groups and individuals within the organization and their lines of communication. Information on the operating organization's safety committee also has to be included, together with a description of the committee's composition, its terms of reference and relevant working procedures.

Sufficient information on any interfaces with other facilities and management on the site needs to be provided to justify any interdependencies. Typical interdependencies of relevance cover transfers of radioactive material and communication arrangements.

A.9.2. Operating instructions and procedures

The processes by which operating procedures are developed, verified and validated are described here. These include procedures for the conduct of normal, abnormal and emergency operations, and for maintenance, periodic testing and surveillance. The details provided need to cover the following:

(a) Administrative procedures: Description of administrative procedures used by the operating organization to ensure the safe management of the facility.
(b) Operating procedures: Description of the facility operating procedures for normal operation, ensuring that the facility is operated within operational limits and conditions.
(c) Procedures and guidelines for operating the facility during accidents: Description of the procedures that will be applied in AOOs and accident conditions; approach to accident management if applicable.

It is a common practice to provide a list of references to operating procedures in this section of the safety analysis report, while the procedures themselves are presented in separate documents.

A.9.3. Maintenance, periodic testing and inspection programme

Similar to operating procedures, the maintenance, periodic testing and inspection programme could be provided in one or more separate documents. In this case, this section of the safety analysis report has to provide a summary of the programme with adequate references to these separate documents. Design features of the facility that facilitate maintenance, testing and inspections may also be outlined in this section.

The arrangements that the operating organization has established (or intends to establish, based on the stage of the facility lifetime) to identify, control, plan, execute, audit and review maintenance, periodic testing and inspection of SSCs that influence safety need to be described in this section, including the following:

(a) The maintenance philosophy, objectives and organization;
(b) The assignment of responsibilities for specific maintenance functions within the maintenance organization, including training of maintenance personnel;
(c) The SSCs and equipment included in the formal maintenance programme;
(d) The management systems used to control maintenance activities;
(e) The interfaces between the maintenance group and other facility organizations (e.g. operations, engineering, quality assurance).

The information presented outlines the plans and provisions for initial and in-service testing, including assessment of the adequacy of these tests based on their type, scope, frequency and the time between successive tests. The organization of the maintenance group has to be provided together with a summary of maintenance workshops, tools and procedures. Control of external contractors also has to be described, including information on matters such as training needed and safety of personnel.

A.9.4. Ageing management programme

This section of the safety analysis report describes the ageing management programme. This description covers different stages of the facility lifetime (e.g. design, fabrication, construction, commissioning, operation) with respect to ageing management.

The elements of the ageing management programme are described, along with justifications for the selected practices (see Section 4.5 of this publication on the elements of a systematic ageing management programme).

The ageing management programme could also be provided in one or more separate documents. In this case, this section of the safety analysis report provides a summary of the programme with references to these separate documents.

A.9.5. Control of modifications

This section of the safety analysis report provides information on the processes provided for identifying, controlling, planning, executing, auditing, reviewing and documenting modifications to the facility throughout its lifetime. These processes have to take account of the safety significance of proposed modifications and the associated requirements for different types of modification. Descriptions of the required routes of approval, the safety analysis needed and the processes for design, installation and commissioning need to be included.

A.9.6. Training and qualification programme for the facility operating personnel

A description of the training and qualification programme for the facility operating personnel is provided in this section. The adequacy of the programme to achieve and maintain the required level of professional competence of staff throughout the lifetime of the facility needs to be demonstrated. The minimum qualifications required for the key positions within the facility organizational structure have to be specified. In this regard, the following has to be covered:

(a) Identification of job related knowledge, skills and abilities, as well as the minimum staffing requirements (including during shift operations), as appropriate;
(b) Requirements on certification of specific positions;

(c) Basis for allocating operational, emergency response and monitoring functions;
(d) Programmes or provisions for monitoring performance of the facility operating personnel, feedback mechanisms, line management and training provisions.

This section also has to provide summaries of training programmes for positions that have a bearing on safety. The scope of training (including basic knowledge, on the job training and refresher training), periodicity and training methods (i.e. self-training, classroom instruction, etc.) also have to be described, with references to supporting documentation. The information provided has to cover normal operations, maintenance and emergency situations.

The training and qualification programme could be provided in a separate document, with a brief description and adequate references to it in this section of the safety analysis report.

A.9.7. Review and audits

This section provides information on the review and audit of management systems and operational procedures at the facility aimed at verification of the facility's compliance with safety and environmental protection requirements. The description covers review and audit methods, the structure of the review and audit groups and their qualifications, items to be subjected to review and audit, frequency of reviews and audits, time intervals between reviews and audits, and procedures for reporting on review and audit findings. The interface between procedures for dealing with review and audit findings and the facility management system can be covered here.

A.9.8. Records

Requirements on record keeping for the facility (such as types of documents and records and duration of maintaining them) are described in this section. The details include provisions for creating, receiving, classifying, controlling, storing, retrieving, updating, revising and removing from service documents and records related to operation of the facility over its lifetime. Documents and records of safety significance that have to be kept include the following:

(a) Operating procedures and instructions;
(b) Operational data for the facility and its processes (including environmental discharges and records of compliance with or violation of the operational limits and conditions);

(c) Reports on events and incidents in the facility, including their analysis, root causes, lessons learned from events and corrective actions that may have been taken;
(d) Radiation protection data, including personal monitoring data;
(e) Data on amounts and movements of nuclear and other radioactive material;
(f) Communication with the national regulatory body (reference may be given to the overall licensing process);
(g) Records of maintenance, calibrations, periodic testing and inspections.

A.9.9. Feedback of operational experience

This section of the safety analysis report provides a description of the programme for operating experience feedback (or a summary of the programme with adequate references if the programme is provided in a separate document). This describes the measures taken to ensure that facility incidents and events are identified, recorded and investigated, as appropriate. The description includes how these measures are used to promote enhanced facility performance through the adoption of appropriate countermeasures to prevent recurrences of events. The programme also includes requirements on notifying the national regulatory body in case of incidents.

A.9.10. Nuclear safety and security interfaces

Security issues are usually described in a separate, confidential document. This section of the safety analysis report needs to confirm that safety requirements and security requirements have been addressed in the design and applied and managed during operation in a coordinated manner so that they do not compromise each other, as stated in Requirement 75 of SSR-4 [1]. It needs to be demonstrated that an effective system to address safety and security aspects in a coordinated manner, involving all interested parties, has been established and will be maintained throughout the lifetime of the nuclear fuel cycle facility.

A.10. CHAPTER 10: WASTE MANAGEMENT AND MANAGEMENT OF RADIOACTIVE DISCHARGES

This chapter of the safety analysis report has to present sufficient information to demonstrate the adequacy of measures for the safe management of radioactive waste of all types that is generated throughout the lifetime of the nuclear fuel cycle facility. This includes compliance with the applicable requirements for radioactive waste and effluent management as established by

SSR-4 [1] and by IAEA Safety Standards Series No. GSR Part 5, Predisposal Management of Radioactive Waste [31].

This section also has to provide descriptions of the facility design provisions and operating procedures and practices to minimize the generation of radioactive waste and effluents, as well as the arrangements for managing the radioactive waste generated, including segregation, monitoring, treatment, transport, storage and monitoring while in storage. The arrangements in place to ensure long term safety of radioactive waste, in terms of capacity and capability, have to be described. If national arrangements involve the transfer of responsibility for the waste to other organizations, then these arrangements and the associated responsibilities of the organizations involved also have to be clearly described.

The types of information in this section include the following:

(a) Main sources of solid, liquid and gaseous waste and estimates of their generation rate;
(b) Philosophy, objectives and the general process for management of the different forms of radioactive waste;
(c) Design provisions and practices to minimize accumulation of waste, and control of waste;
(d) Handling of radioactive waste and associated SSCs;
(e) Processing of radioactive waste and associated SSCs;
(f) Storage of radioactive waste and associated SSCs;
(g) Radioactive waste disposal routes.

Each radioactive waste management system, including for gaseous, liquid and solid waste, has to be described in accordance with the description of SSCs specified in chapter 4 of the safety analysis report to demonstrate its conformance to the design requirements.

For each waste form, the following information has to be presented:

(i) Radioactive waste sources and characteristics: How and where the radioactive waste is generated; handling of the radioactive waste; and quantities, chemical forms and characteristics, physical characteristics, and radiological or toxic/radiological composition of the radioactive waste, as appropriate.
(ii) Radioactive waste processing and storage systems: The methods employed to control or mitigate the potential impacts of the different radioactive waste forms; description of the operating principles, functions and performance objectives of radioactive waste processing and storage equipment; and engineering drawings indicating paths and locations of the relevant equipment and instrumentation.

(iii) Administrative and operational controls important to the effective management of the different radioactive waste forms.

Estimates of the quantity, volume and characteristics of secondary radioactive waste resulting from radioactive waste pretreatment or treatment in the facility (e.g. waste solutions generated from decontamination of the equipment, waste produced from laundry treatment) have to be included. The quantity, volume and form of incidentally generated radioactive waste need to be estimated and the equipment, provisions and plans for its management need to be described.

The effects of storage also need to be described (e.g. deterioration of packaging materials, gas generation and pressurization), including any requirements for chemical stabilization of the radioactive waste or maintenance/replacement of equipment over the lifetime of the storage. An explanation also has to be provided on whether radioactive waste storage is intended to be short or long term, and any requirements for processing of the waste to be carried out (see GSR Part 5 [31]) need to be included.

Possible disposal routes for the radioactive waste that is generated also need to be identified and presented in this section.

The discussion of aerial and liquid radioactive discharges has to cover the following:

— Authorized limits and operational targets for aerial and liquid discharges and associated hazardous chemicals (daily, monthly, annually);
— Measures to comply with established limits;
— Effluent monitoring programmes, including off-site monitoring for discharges to the environment;
— Record keeping.

As part of this discussion, the discharge sources and the volume of radioactive effluents (and, as necessary, associated hazardous chemicals) expected to be discharged from the facility in all operational modes and states have to be described, together with the inventory, the concentrations of each radionuclide and the total activity.

Process and effluent monitoring and sampling systems (on-site and off-site if applicable) have to be described in accordance with the description of SSCs specified in chapter 4 of the safety analysis report to demonstrate their conformance to the design requirements.

It has to be demonstrated that discharges of gaseous, liquid and particulate radioactive material and associated hazardous chemicals to the environment comply with authorized limits and are conducted in accordance with regulations for the protection of the public and the environment.

A.11. CHAPTER 11: NON-RADIOLOGICAL HAZARDS

This chapter of the safety analysis report contains sufficient information to demonstrate that relevant requirements related to industrial safety and protection against non-radiological hazards are met.

A.11.1 Use of pressure and lifting equipment

This section has to provide information that demonstrates safe design and use of systems involving high pressure equipment and lifting equipment in a nuclear fuel cycle facility, as these may impact the nuclear or radiation safety of the facility. Technical and administrative measures for the use of such equipment have to be described in enough detail to demonstrate safety. In this regard, the following have to be provided in this section:

(a) Description of the equipment, its functions, and its safety class;
(b) Codes and standards used for equipment design, construction, manufacture and operation;
(c) Operation and use of the equipment in the facility under different operational conditions;
(d) Qualification of equipment;
(e) Requirements for training and qualifications of personnel operating (or using) the equipment;
(f) Requirements for maintenance, periodic testing and inspection of the equipment;
(g) A summary of the results of the failure analysis of the equipment.

The information to be provided covers the conduct of work using the equipment and methods for diagnostics and the evaluation of its service life. Analysis of the dropping of heavy loads has to be provided in this section of the safety analysis report, together with the limits and conditions of use for lifting equipment.

A.11.2. Protection against chemical hazards

This section has to contain sufficient information to demonstrate the prevention and mitigation of chemical hazards associated with radioactive materials that can adversely impact the health and safety of the public or pose a risk to workers. The impact of releases of associated hazardous chemical materials to the environment is covered by the environmental report (see chapter 12 of the safety analysis report).

The discussion in this section has to cover the following:

(a) Regulations and standards used for the control of toxic chemicals;
(b) Description of processes involving hazardous chemicals, including description of facilities, equipment, inventories, locations and quantities of hazardous chemical materials;
(c) Programme for controlling the risks associated with chemical hazards to workers and the public (including operational rules, work planning, permits and supervision in accordance with the properties and quantities);
(d) Monitoring of individuals and the workplace and monitoring of chemical releases to the environment;
(e) Description of sequences of accidents involving chemical releases and evaluated consequences to workers, the public and the environment from chemical exposure from all possible exposure pathways (e.g. dermal, ingestion, inhalation);
(f) Summary of the results of the safety analysis related to chemical hazard prevention and mitigation;
(g) Control and recording of hazardous chemicals and reagents;
(h) Record keeping on chemical exposures;
(i) Protective clothing and protective equipment;
(j) Instrumentation and requirements on calibration and maintenance of instruments;
(k) Training and qualifications of the operating personnel of the facility on chemical safety aspects;
(l) Arrangements for response to emergencies.

The safety analysis report has to demonstrate that workers and the public are protected against toxic chemical exposures associated with radioactive material.

A.11.3. Protection against fire and explosion

This section has to provide sufficient information to demonstrate fire safety in the fuel cycle facility, particularly for those who use materials, including process chemicals, that are combustible or explosive. These may include metals, flammable liquids and gases, high temperature equipment, hot cells, gloveboxes and laboratories handling radioactive material.

The discussion in this section covers the following:

(a) Identification of fire risks.
(b) Fire hazard assessment (or a summary of it, if provided in a separate document).
(c) Design requirements for fire protection: Material selection, isolation, separation of redundant SSCs, fire zoning, fire barriers, segregation and barriers to prevent the spread of fire and smoke, building and process layouts, access and escape routes, fire resistance classes, ventilation system, electrical systems.
(d) Fire protection system, including automatic fire detection system, fire extinguishing system and operational firefighting.
(e) Safety organization, inspection, testing and maintenance, personnel training.
(f) Interfaces with other programmes such as criticality safety and operational radiation protection programmes (demonstrate that the arrangements for fire safety are consistent with the arrangements for nuclear and radiation safety).
(g) Emergency arrangements, including interfaces and communication with emergency response organizations.

The results of fire safety analysis have to be accompanied by the results of the analysis of fire consequences considering possible failures of fire suppression systems.

It has to be demonstrated in this section that the facility is designed and operated so as to prevent and control fires and to prevent explosions as required.

Fire safety (or part(s) of it, such as fire hazard assessment and fire protection plans) may be presented in a separate document, in which case this section contains a summary of the programme, with a reference to the document.

A.12. CHAPTER 12: ENVIRONMENTAL IMPACT ASSESSMENT

Practices among States may vary with respect to the inclusion of information on environmental impact assessment in the safety analysis report. If this is required, this chapter provides a brief description of the approach taken to assess the impact on the environment of the construction of the facility and its operation for operational states and, if required by the regulatory body, in accident conditions.

Usually, the overall environmental impact of the facility is covered by a dedicated environmental impact assessment report as a separate document. This chapter of the safety analysis report provides a link to the environmental impact assessment report. A summary of the report can be discussed in this chapter with adequate references to separate documentation.

The discussion of environmental impacts for nuclear fuel cycle facilities will cover both radiological and non-radiological impacts.

The discussion of environmental radiological impacts covers the following:

(a) Authorized limits and operational targets for liquid and gaseous discharges and, if applicable, for direct radiation.
(b) Measures to control discharges to the environment.
(c) Assessment of the radiological impact of such discharges on the environment and compliance with such limits (daily, monthly, annually).
(d) Environmental measurements and monitoring programmes, including off-site monitoring of contamination and radiation levels.
(e) Alarm systems that are required to respond to unplanned radioactive discharges and the measures to prevent such discharges, as applicable.
(f) Records of discharges: Methods for making, storing and retaining records of radioactive discharges.
(g) Availability of information to the authorities and the public: Means for making appropriate data available.

With regard to non-radiological effect on the environment, this chapter specifies the following:

(i) Chemical and physical nature of the discharges.
(ii) Authorized limits and operational targets for chemical discharges.
(iii) Measures to control discharges to the environment.
(iv) Assessment of the impact of such discharges on the environment.
(v) Records.
(vi) Environmental measurements and monitoring programmes: Off-site monitoring regime for pollution.
(vii) Alarm systems required to respond to unplanned discharges.
(viii) Availability of information to the authorities and the public.

Discharges to the environment and the resulting exposure to the public from all operational states need to be identified and quantified using suitably conservative methodology and compared with the relevant regulatory requirements. Sufficient information has to be provided to give assurance that risks from accidental exposures are controlled and within acceptable bounds.

This chapter also describes the provisions for monitoring of the environment and public exposure, as well as the equipment/processes needed for its functioning.

References and brief discussions of the site characteristics are provided insofar as they relate to the environmental impact of the licensing decision.

Information is also presented on the regime for monitoring the impact of gaseous and liquid discharges, including the scope of the surveys, the nuclides to be measured and the equipment and facilities needed to carry out the surveys. Where the facility is situated near another nuclear site, the arrangements could include provision for coordination of surveys and sharing of results. The description of methodologies for discharge monitoring (e.g. proportional sampling, batch sentencing) has to explain how the discharge of unmonitored or out-of-specification materials is prevented.

If it is decided that the facility or activity has little or no environmental effect, the decision has to be stated and justified.

A.13. CHAPTER 13: COMMISSIONING PROGRAMME

This chapter of the safety analysis report describes the technical and organizational aspects of the commissioning programme for the nuclear fuel cycle facility following construction or modification. The objective of this chapter is to demonstrate that the commissioning will verify that the SSCs meet the required design and performance criteria. It also has to indicate how the commissioning will be used to train the facility's operating personnel and transfer knowledge of design safety from the design organization to the facility personnel.

For operating nuclear fuel cycle facilities, this chapter provides the commissioning results and sufficient information to demonstrate that the commissioning programme has been carried out and that safety functional requirements of the SSCs have been adequately verified.

For nuclear fuel cycle facilities at the commissioning stage, the information to be provided in this section includes the following:

(a) A summary of the programme and objectives;
(b) Organization and responsibilities for commissioning;
(c) Details of the commissioning organization, including requirements on training of the commissioning team and the facility operating personnel;
(d) Management system for commissioning, including quality assurance programme for the commissioning stage (this covers processes for control of modification, control of non-conformances, acceptance criteria, commissioning tests and procedures, reporting on commissioning results, etc.);
(e) Schedule of major phases of the commissioning programme (cold commissioning and hot commissioning): This may also include description of

the criteria for transition between different commissioning phases, reporting on the commissioning results, a schedule for submitting documentation to the regulatory body, hold points for review of the commissioning results and a schedule for updating the facility documents with the results of commissioning;

(f) Commissioning procedures and reports, including provisions and procedures for reviews and verification;
(g) Operational limits and conditions;
(h) Emergency plans.

For new fuel cycle facilities that have completed the commissioning programme, this chapter of the safety analysis report has to summarize the commissioning results. The information expected includes technical and organizational changes made during commissioning, non-conformances important to safety that have been accepted together with the relevant corrective actions, and conclusions of the commissioning programme related to possible modification to SSCs, procedures or documentation. Provisions for the retention of commissioning data are also presented.

For existing nuclear fuel cycle facilities, this chapter can provide information similar to that for facilities after commissioning, in addition to a summary of the commissioning objectives and stages, and a list of tests that have been carried out during commissioning. The verification of safety functions for older facilities can also be described here.

The commissioning programme could be provided in separate document, with a summary in this chapter with adequate references to the document.

A.14. CHAPTER 14: MANAGEMENT SYSTEM

This chapter of the safety analysis report describes the management system applied to the facility in all stages of its lifetime. The requirements that are adhered to by the management system (e.g. regulatory requirements, IAEA safety standards and other relevant international requirements, stakeholders' requirements) are described. This section will describe or refer to the particular parts of the management system that have been established for the stages of design, construction, commissioning or operation, as appropriate.

The information in this chapter can be presented in terms of the four functional categories of the management system (i.e. management responsibility; resource management; process implementation; and measurement, assessment and improvement). Section 6 of this publication provides more information on the functional categories of the management system.

This chapter has to include information on the following:

(a) Safety policy of the operating organization;
(b) Responsibilities in management for safety (for different stages of the facility lifetime);
(c) Management system processes that apply to different stages of the facility lifetime;
(d) Management system procedures supporting the implementation of management system processes that cover the operation of the facility (e.g. radiation protection, criticality safety, chemical hazards, safety analysis, preparation of documents, control of design modifications, performing of calculations, maintenance, periodic testing and inspection, recruitment, training and qualifications of personnel, purchasing, emergency preparedness and response);
(e) Description of level of control and verification for achieving high quality and means available for it, including quality audits;
(f) Requirements for periodic review of the management system and improvements.

It has to be demonstrated that an integrated management system has been implemented and assessed, and is being continuously improved at all stages of the lifetime of the facility.

A.15. CHAPTER 15: EMERGENCY PREPAREDNESS AND RESPONSE

This chapter of the safety analysis report provides information on emergency preparedness and response arrangements. The information has to be sufficient to demonstrate that all actions necessary for the protection of workers (including emergency workers), the public and the environment could be taken in a nuclear or radiological emergency, and that these actions would be timely, coordinated and effective.

Design provisions for emergency preparedness and response also have to be described in this chapter. The emergency arrangements, in accordance with the identified emergency category, have to be described. Aspects related to chemical hazards with the potential to hinder the emergency response also have to be considered.

The emergency plan could also be presented in a separate document, but a summary has to be provided in this chapter with adequate references. The information in this chapter has to demonstrate that the emergency plan has been developed based on the safety analysis of the facility, including the analysis of the

DECs, and that the plan has identified actions to be taken in the facility, including its auxiliary buildings, on-site and off-site, as applicable. It also has to describe how the emergency plan for the facility is integrated and consistent with the site emergency plan and discuss the arrangements of external response organizations.

Requirements relating to the emergency arrangements for nuclear facilities are established in GSR Part 7 [23].

A.16. CHAPTER 16: PREPARATION FOR DECOMMISSIONING

Planning for decommissioning starts from the design stage and a decommissioning plan has to be available at the beginning of the operation stage of the nuclear fuel cycle facility. It has to be periodically updated to provide an increasing level of detail, to reflect new knowledge and technological developments in the field as well as new information and experience acquired from the operation of the facility. The decommissioning plan can be presented in a separate document, but a summary of this plan has to be included in the safety analysis report.

This chapter has to demonstrate that provisions and measures have been considered in the facility's design, construction, commissioning and operation to facilitate decommissioning (e.g. modular construction to facilitate dismantling, operational practices to reduce generation of radioactive waste, operation and maintenance record keeping, control of modifications). Information on conceptual plans for decommissioning has to be presented to demonstrate that adequate measures have been taken in the design and operation of the facility. This includes an evaluation of vulnerabilities to a spectrum of events to avoid unnecessary burdens and to minimize site or environmental contamination that would complicate decommissioning or limit the effectiveness of environmental restoration.

The discussion in this chapter can include the following:

(a) Design provisions to facilitate ultimate decommissioning;
(b) Operating practices and procedures to facilitate decommissioning;
(c) Organizational aspects of decommissioning, including a description of organizations responsible for decommissioning;
(d) Managing the transition period between operation and decommissioning;
(e) Decommissioning strategy and options;
(f) Provisions for safety during decommissioning;
(g) Definition of the end state of the facility;
(h) Planning for preliminary decommissioning work.

The information has to be sufficient to demonstrate that an appropriate decommissioning plan has been prepared and will be maintained throughout the lifetime of the facility, and that decommissioning can be accomplished safely and in such a way as to meet the defined end state.

The decommissioning plan, which is usually provided in a separate document, also includes elements such as information on the safety analysis for decommissioning, the final shutdown plan, and decommissioning stages and procedures. Safety requirements for decommissioning are established in IAEA Safety Standards Series No. GSR Part 6, Decommissioning of Facilities [32].

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. SSR-4, IAEA, Vienna (2017).

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

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

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report, IAEA Safety Standards Series No. SSG-20, IAEA, Vienna (2012).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Analysis for Research Reactors, Safety Reports Series No. 55, IAEA, Vienna (2008).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Format and Content of the Safety Analysis Report for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-61, IAEA, Vienna (in preparation).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Conversion Facilities and Uranium Enrichment Facilities, IAEA Safety Standards Series No. SSG-5, IAEA, Vienna (2010).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Uranium Fuel Fabrication Facilities, IAEA Safety Standards Series No. SSG-6, IAEA, Vienna (2010).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Uranium and Plutonium Mixed Oxide Fuel Fabrication Facilities, IAEA Safety Standards Series No. SSG-7, IAEA, Vienna (2010).

[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Fuel Reprocessing Facilities, IAEA Safety Standards Series No. SSG-42, IAEA, Vienna (2017).

[11] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Fuel Cycle Research and Development Facilities, IAEA Safety Standards Series No. SSG-43, IAEA, Vienna (2017).

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Storage of Spent Nuclear Fuel, IAEA Safety Standards Series No. SSG-15, IAEA, Vienna (2012).

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Site Evaluation for Nuclear Installations, IAEA Safety Standards Series No. NS-R-3 (Rev. 1), IAEA, Vienna (2016).

[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Fuel Incident Notification and Analysis System (2020),
http://finas.iaea.org

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Procedures for Conducting Probabilistic Safety Assessment for Non-reactor Nuclear Facilities, IAEA-TECDOC-1267, IAEA, Vienna (2002).

[16] AMERICAN INSTITUTE OF CHEMICAL ENGINEERS, Guidelines for Hazard Evaluation Procedures, 3rd edn, Centre for Chemical Process Safety of the American Institute of Chemical Engineers, New York (2008).
[17] UNITED STATES ENVIRONMENTAL PROTECTION AGENCY, Access Acute Exposure Guideline Levels (AEGLs) Values (2018), https://www.epa.gov/aegl/access-acute-exposure-guideline-levels-aegls-values
[18] INTERNATIONAL ATOMIC ENERGY AGENCY, Criticality Safety in the Handling of Fissile Material, IAEA Safety Standards Series No. SSG-27, IAEA, Vienna (2014).
[19] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Reassessment for Nuclear Fuel Cycle Facilities in Light of the Accident at the Fukushima Daiichi Nuclear Power Plant, Safety Reports Series No. 90, IAEA, Vienna (2016).
[20] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Classification of Structures, Systems and Components in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-30, IAEA, Vienna (2014).
[21] OFFICE FOR NUCLEAR REGULATION, Categorisation of Safety Functions and Classification of Structures, Systems and Components: Nuclear Safety Technical Assessment Guide, NS-TAST-GD-094, ONR, Bootle (2015).
[22] OECD NUCLEAR ENERGY AGENCY, CSNI Technical Opinion Papers No. 15: Ageing Management of Nuclear Fuel Cycle Facilities, OECD/NEA, Paris (2012).
[23] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL CIVIL AVIATION ORGANIZATION, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, INTERPOL, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, PREPARATORY COMMISSION FOR THE COMPREHENSIVE NUCLEAR-TEST-BAN TREATY ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, UNITED NATIONS OFFICE FOR THE COORDINATION OF HUMANITARIAN AFFAIRS, WORLD HEALTH ORGANIZATION, WORLD METEOROLOGICAL ORGANIZATION, Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GSR Part 7, IAEA, Vienna (2015).
[24] INTERNATIONAL ATOMIC ENERGY AGENCY, Licensing Process for Nuclear Installations, IAEA Safety Standards Series No. SSG-12, IAEA, Vienna (2010).
[25] FEDERAL ENVIRONMENTAL, INDUSTRIAL AND NUCLEAR SUPERVISION SERVICE OF RUSSIA, Federal Standards and Rules in the Field of Use of Atomic Energy, Requirements for Safety Analysis Report for Nuclear Installations of Nuclear Fuel Cycle, NP-051-04, Federal Environmental, Industrial and Nuclear Supervision Service of Russia, Moscow (2004).
[26] OFFICE FOR NUCLEAR REGULATION, The Purpose, Scope, and Content of Safety Cases, NS-TAST-GD-051 (Rev. 6), ONR, Bootle (2019).
[27] NUCLEAR REGULATORY COMMISSION, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste, 10 CFR 72, US Govt Printing Office, Washington, DC (2016).

[28] NUCLEAR REGULATORY COMMISSION, Domestic Licensing of Special Nuclear Material, 10 CFR 70, US Govt Printing Office, Washington, DC (2018).

[29] OFFICIAL JOURNAL OF THE FRENCH REPUBLIC, The General Rules Relative to Basic Nuclear Installations, No. 0033 of 8 February 2012, Official Journal of the French Republic, Paris (2012).

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

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

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

Annex I

FACTORS AFFECTING THE APPLICATION OF A GRADED APPROACH AND AREAS SUBJECTED TO GRADING IN SAFETY ANALYSIS OF NUCLEAR FUEL CYCLE FACILITIES

Factors affecting the application of a graded approach in the safety analysis of nuclear fuel cycle facilities include the following:

(a) The purpose of the safety analysis (i.e. in design, for licensing, for accident management, etc.);
(b) The scale of operations undertaken at the facility;
(c) The inventory of radioactive material, the amount and enrichment of fissile material or the inventory of transuranic elements and the perceived criticality risk;
(d) The radioactive inventory of a facility on the site, where appropriate;
(e) The amount, nature and physical and chemical forms of the radioactive materials that are used, processed or stored at the facility;
(f) Facility design, complexity of the site and the facility, inherent safety features in the design, maturity of the process;
(g) The ease or difficulty of changing the overall configuration of the facility;
(h) The presence of high pressure or high energy piping or the use of high temperature or high pressure processes;
(i) The quality (robustness) of the means of confinement (containment and ventilation systems, presence of high inventory glovebox suites, etc.);
(j) The facility utilization programme;
(k) The stage of the lifetime of the nuclear fuel cycle facility (including the physical status of the facility's SSCs);
(l) Any other inherent hazards (e.g. hydrogen, chemical and fire hazards);
(m) Siting (regional characteristics including location of reservoirs, dams and large water bodies and geological or meteorological conditions);
(n) Structural concept (above or below ground) and the proximity of other nuclear or non-nuclear industrial sites;
(o) Proximity of the nuclear fuel cycle facility to populated areas and the availability of off-site support to cope with accidents;
(p) The availability of safety analysis results from other similar facilities (or from the facility under consideration);
(q) The availability and reliability of information on the facility and its site as well as on the uncertainty associated with the process.

A graded approach could be applied to the selection of hazard identification and evaluation techniques (see also Annex II to this publication). Grading may be applied to the scope and the level of detail of the selection and identification of PIEs and the assessment of the accident conditions of a nuclear fuel cycle facility. Certain accident scenarios may not apply or may need only limited analysis in facilities with low radioactive inventory compared to those with high inventory. For example, the analysis of a loss of cooling event differs significantly depending on the radioactive material inventory within the facility. On the other hand, some facilities such as reprocessing plants are complex in design and require more safety measures to confine radioactive material (including dilution of radiolysis gases and removal of decay heat). These features require the analysis of initiating events such as loss of gas flow, loss of containment and loss of cooling of vessels or equipment, which may not be required in other types of nuclear fuel cycle facilities.

The scope of the assessment of human errors in the analysis of event sequences may also be subjected to grading, depending on the complexity of the facility and the level of operator intervention in processes. This is significantly lower in some facilities, which leads to a simpler analysis of events and transient modelling. A graded approach may also be applied to the selection of PIEs that are related to external hazards, as examination of some events may show that some of them pose minimal threat to the facility at that location.

A graded approach may also be used in the application of the safety requirements related to the levels of defence in depth. For example, if minimal confinement or containment is designed into a facility, this needs to be justified on the basis that there is no potential for release of radioactive or toxic material under accident conditions that might result in unacceptable off-site consequences. Consequently, the scope and level of detail of the calculations of the radiological or chemical consequences of some events may be much lower for these facilities.

Another area that may also be subjected to grading in the safety analysis process is the use of computational models and computer codes. Use of complex models and sophisticated computer codes in the analysis of accident transients may be needed, depending on the complexity of the facility and processes under consideration. For example, the computational models and computer codes that could be used in criticality safety analysis in fuel cycle research and development facilities may be simpler than those that may be needed for other types of facility such as spent nuclear fuel management or nuclear fuel reprocessing facilities.

Additionally, with respect to the use of computational models and tools in safety analysis, a graded approach may also be applicable to the extent, scope and level of detail of programmes for data collection. Requirements on operational record keeping in terms of scope, detail and duration may also be graded.

It is important to note that acceptance criteria for safety analysis may not be subjected to grading. For example, the maximum allowable radiation doses to workers or the public are the same in specific Member States regardless of the level of potential hazard of a nuclear fuel cycle facility. Similarly, the values of the minimum acceptable margins against accidental criticality would be the same for all nuclear fuel cycle facilities, regardless of their potential hazards. The acceptance criteria for safety analysis in general are not subjected to grading. However, grading may be applied to some other safety criteria, such as the use of a specific mechanical code or standards depending on the safety class of specific SSCs and the potential hazards of the facility.

A graded approach can be applied to the organizational aspects, including human and financial resources, of performing the safety analysis, and to the management of the implementation of the findings of the analysis. It can also be applied to the regulatory oversight of safety analyses. In this regard, areas that could be subjected to grading include the human resources needed for performing regulatory review and assessment and associated regulatory inspection activities. Nevertheless, certain organizational factors, such as safety culture, the qualifications of safety analysts and the implementation of conclusions, are required to be maintained by managers at the highest level and are not subjected to grading.

Annex II

EXAMPLES OF FACTORS TO BE CONSIDERED AND TECHNIQUES USED IN HAZARD IDENTIFICATION FOR NUCLEAR FUEL CYCLE FACILITIES

The following factors are examples of those that need to be considered in the identification of hazards for the safety analysis of nuclear fuel cycle facilities, along with suggested methods for hazard identification:

(a) Site characteristics, including geography, seismology, hydrology and meteorology.
(b) Surrounding facilities, the location of nearby industrial facilities and transportation routes, demography.
(c) Existing analysis of hazards: Examination of previous hazard analysis for similar facilities, including chemical facilities.
(d) Published hazard checklists: A review of standards and codes of practice that might refer to previously identified hazards that have caused accidents.
(e) Energy sources and energy flows: Examination of the basic energy sources, energy flows and high energy items in the system, together with the provisions for their control.
(f) Quantity, form and location of radioactive and other hazardous materials:
    — Inventory of radioactive, fissile and hazardous chemical material;
    — Radiological properties of concern (e.g. radioactive half-life, biological half-life and decay mode);
    — Chemical properties (e.g. toxicity, flammability and reactivity).
(g) Process conditions and interactions that can occur between hazardous materials in normal and abnormal conditions.
(h) Criticality hazard: Conditions under which a self-sustaining fission chain reaction may be initiated and associated analysis.
(i) Interface hazards: Hazards arising from incompatibilities in interfaces, including mechanical interfaces resulting from material incompatibility, or design errors that can cause failure in safety related interfacing components (e.g. software).
(j) Operator training material.
(k) Human performance under stress: Identification of all possible modes of operation in all environments.
(l) Human–machine interface: Examination of the modes of interaction between human and automated equipment and potential operator errors.

(m) Off-normal mode transitions: Identification of potential for accidents in transitions to non-routine operational modes, including startup, shutdown, testing, trials of new methods, breakdown, maintenance, repair, inspection, troubleshooting, modifications, changeovers, adjacent system change, non-standard input, stresses and adverse conditions.
(n) Scientific/technical investigation: Investigation of physical and chemical properties of the system, which may involve theoretical studies and small-scale tests.
(o) Process evaluation: Simulation of processes such as startup and shutdown step by step, attempting to anticipate what might go wrong and how incidents can be avoided or mitigated.
(p) Unexpected changes in the form or state of materials (e.g. liquid to solid, liquid to gas).
(q) Feedback from operating experience, including at the same facility as well as at similar facilities.
(r) History of incidents at the facility: Review of incident reports, hazard logs and lessons learned from the events.

For existing nuclear fuel cycle facilities, other possible sources of information supporting hazard identification include fire hazard analyses, health and safety plans, job safety analyses and occurrence reporting histories.

Bounding inventory values of radiological and hazardous materials are used that are consistent with the maximum quantities of material that are stored or used in facility processes. Inventory data may be obtained from flowsheets, vessel sizes, contamination analyses, maximum historical inventories and similar sources.

Hazard identification, particularly where complex activities are undertaken with highly hazardous material, requires a questioning attitude. Failures that are unlikely to occur need to be considered, and their potential to overcome safety features needs to be estimated. Only when a hazard is proven to have no significant effect can the need for further detailed analysis be ruled out. The list of identified hazards needs to be reviewed and updated on a regular basis.

Several techniques are used to support the identification of hazards in nuclear fuel cycle facilities. These include the following:

- ‘What-if’ and what-if/checklist analysis;
- Function failure analysis;
- Event tree analysis;
- Failure modes and effects analysis;
- Fault tree analysis;
- Cause–consequence diagrams;

— Hazard and operability studies;
— Human reliability analysis.

However, it is unlikely that application of a single technique will reveal all hazards. For instance, failure modes and effects analysis can produce a more reliable design, but hazards are only identified if all of the functional safety requirements have been captured in the specification of components, including human factors. It is better to combine top-down with bottom-up techniques in complex cases, such as hazard and operability studies, event tree analysis and fault tree analysis.

A graded approach could be applied to the selection of hazard identification and evaluation techniques (see also Annex I of this publication). The selection is based on several factors, including the complexity, type and size of operations being analysed, and the inherent nature of the hazards being evaluated. For example, what-if or what-if/checklist analysis are appropriate for analysing many low hazard category facilities, as well as simple operations for medium level hazard facilities such as waste packaging, storage or transport. More elaborate methods such as hazard and operability studies or failure modes and effects analysis need to be used for facilities with complex operations such as nuclear fuel reprocessing. For special situations that require detailed analysis of one or more hazardous conditions, fault tree analysis, event tree analysis and human reliability analysis are to be considered. The rationale for the selection of hazard evaluation techniques has to be discussed and justified in the licensing documentation of the facility.

Standard techniques used for the analysis of common cause and single mode failures (which are used for nuclear power plants) can also be used for the analysis of external hazards for nuclear fuel cycle facilities, but may not be sufficient to identify events associated with internal hazards at these facilities. It is therefore usual to include additional conditions in safety analyses of nuclear fuel cycle facilities and to provide additional protection in accordance with the optimization principle.

Additionally, as part of the hazard evaluation, an unmitigated hazard scenario is evaluated for each initiating event by assuming the absence of preventive and mitigation controls. Initial conditions may be necessary to define the unmitigated hazards. The consequences and likelihood of the unmitigated hazard scenario are estimated using qualitative or semi-quantitative techniques. Hazard scenario consequence estimates address potential effects on facility workers, co-located workers and the public, and compare them with pre-established acceptance criteria.

Annex III

SELECTED POSTULATED INITIATING EVENTS FOR NUCLEAR FUEL CYCLE FACILTIES

The following list of selected postulated initiating events for nuclear fuel cycle facilities is reproduced from the appendix to IAEA Safety Standards Series No. SSR-4, Safety of Nuclear Fuel Cycle Facilities [III–1].

(a) Loss of services:
- Loss of normal electrical power;
- Loss of compressed air;
- Loss of inert atmosphere;
- Loss of coolant;
- Loss of ultimate heat sink.

(b) Loss of criticality controls:
- Drop of fuel during handling;
- Loss of geometry;
- Flooding;
- Loss of neutron poison;
- Excess reflection or moderation;
- Unexpected change of phase;
- Failure or collapse of structural components;
- Maintenance error;
- Control system error;
- Over batching (double batching).

(c) Processing errors:
- Incorrect facility configuration;
- Insufficient reagent or coolant, or reagent or coolant added too slowly or too late;
- Excess reagent or coolant, or reagent or coolant added too fast or too early;
- Incorrect pressure or gas flow, rupture of pressure retaining vessels or pipes;
- Incorrect or extreme temperature;
- Unexpected change of phase leading to criticality or loss of confinement;
- Safety function not performed or performed too late.

(d) Facility and equipment failures:
- Loss of confinement or leakage;

- — Inadequate isolation of process fluids;
- — Blockage or bypass of a filter or column;
- — Spurious actuation of item important to safety;
- — Structural failures.

(e) Handling errors:
- — Hazardous load dropped;
- — Heavy load dropped on an item important to safety;
- — Failure on demand of a safety interlock;
- — Inadequate brakes or inadequate overspeed or overload protection;
- — Obstructed pathway leading to collision;
- — Failure of lifting component (e.g. hook, beam, cable);
- — Load remaining fixed to floor on lifting.

(f) Other internal events:
- — Internal fires or explosions;
- — Internal flooding;
- — Malfunction in experiment;
- — Criticality event;
- — Collisions with the facility building;
- — Fluid jets, pipe whip or internal missiles;
- — Exothermic chemical reaction;
- — Ignition of accumulated hydrogen;
- — Failure due to corrosion;
- — Loss of neutron absorption.

(g) External events:
- — Earthquakes (including seismically induced faulting and landslides);
- — Flooding (including failure of an upstream or downstream dam, blockage of a river and damage due to a tsunami or high waves);
- — Tornadoes and tornado missiles;
- — Sandstorms;
- — Hurricanes, storms and lightning;
- — Tropical cyclones;
- — External explosions;
- — Aircraft crashes;
- — External fires;
- — Toxic spills outside the facility;
- — Accidents on transport routes;
- — Effects from adjacent facilities (e.g. nuclear facilities, chemical facilities and waste management facilities);
- — Biological hazards such as microbial corrosion, structural damage or damage to equipment by rodents or insects;
- — Extreme meteorological phenomena;

— Power or voltage surges on the external supply line.

(h) Human errors:
   — Incorrect specification of incoming and transferred materials;
   — Operator error or omission;
   — Maintenance error or omission.

REFERENCE TO ANNEX III

[III–1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Fuel Cycle Facilities, IAEA Safety Standards Series No. SSR-4, IAEA, Vienna (2017).

Annex IV

EXAMPLES OF DESIGN EXTENSION CONDITIONS FOR NUCLEAR FUEL CYCLE FACILITIES

Analysis of design extension conditions (DECs) needs to consider the impact of accident conditions more severe than design basis accidents (DBAs) or those involving additional failures. Any consequential loss of basic safety functions due to extreme events (or combinations of them) that could result in damage to structures, systems and components (SSCs) or significantly reduce safety margins also has to be considered in the analysis. Examples of causes that could result in DECs include the following:

— Combination of anticipated operational occurrences or DBAs with a common cause failure of SSCs important to safety;
— Combination of SSCs and human failures that is considered significant in DBA analysis;
— Multiple failures of SSCs important to safety caused by rare extreme natural phenomena that are unlikely (but considered possible).

Generic examples of DECs for nuclear fuel cycle facilities could include the following:

— Loss of two independent criticality controls;
— Multiple failures leading to sustained and extended loss of the cooling of heat generating materials (spent fuel, high level waste and solutions);
— Extended blackout;
— Major disruption events (due to explosion, overheating or pressure increase) causing the catastrophic failure of a vessel or structure;
— External events more severe than design basis (e.g. earthquake, flooding, aircraft crash).

Examples of DECs that could be considered for specific nuclear fuel cycle facilities are described in the sections that follow. It is important to note that the examples are design dependent and the analysis of DECs is performed on a case-by-case basis for the specific facility.

IV–1. LOSS OF CRITICALITY CONTROLS

By virtue of the double contingency principle, if two unlikely, independent and concurrent changes in process conditions occur, the subcriticality margin of the system may not be ensured (a postulated margin in the effective multiplication factor might be reduced). Thus, the status of the system would be unknown, with potential for criticality.

IV–2. OVERHEATING OF IRRADIATED FUEL AND VITRIFIED HIGH LEVEL WASTE

Irradiated fuel and high level radioactive waste usually require cooling for several years after generation. Overheating of irradiated fuel or vitrified high level radioactive waste could lead to a failure of the cladding or fuel matrix. In this regard, it normally has to be demonstrated with a high degree of confidence (by providing additional independent, separate and diverse means to prevent overheating) that this possibility has either been prevented or is unlikely to occur.

IV–3. UNCOVERING OF IRRADIATED FUEL IN A WET STORAGE POOL

Inadvertent uncovering of irradiated fuel in a wet storage pool has to be prevented. Loss of water cover could cause high radiation exposure to workers and overheating of stored nuclear fuel. These conditions require the provision of additional features such as emergency water make-up, and could be considered DECs.

IV–4. EXTENDED BLACKOUT OF A LARGE REPROCESSING FACILITY

A large nuclear fuel reprocessing facility involves a range of monitoring and control SSCs. Loss of electrical power supplies (including emergency ones) necessitates evacuation of the facility. The design of the facility may be extended to provide additional power backup and control measures to prevent such a loss of monitoring and control functions from escalating into a more significant condition.

IV–5. LOSS OF COOLING OF HIGHLY ACTIVE LIQUOR

In nuclear fuel reprocessing facilities, the highly active liquor contains most of the radioactive material after recovery of some elements (e.g. U, Pu) in the first separation stage of reprocessing. Highly active liquor is stored as an aqueous solution in tanks that require cooling, pending further treatment and disposal. Overheating of highly active liquor following loss of cooling that could result in boiling has to be practically eliminated. Overheating of highly active liquor following a loss of normal and emergency cooling could be considered a DEC.

IV–6. HEATING OF AN OVERFILLED UF_6 STORAGE OR PROCESS VESSEL

Uranium hexafluoride (UF_6) is usually solid at normal temperatures, and expands as it melts. Storage vessels may be ruptured if they are overfilled and heated, as there is insufficient free volume to contain the expanded liquid. A consequential rupture of a cylinder containing UF_6 has to be practically eliminated. Such events have occurred in some facilities and caused serious injuries and release of toxic fluorides into the environment. Heating of an overfilled UF_6 vessel may be considered a DEC, owing to the severity of the local effects.

IV–7. MAJOR DISRUPTION EVENT WITHIN A FACILITY PRODUCING RECYCLED FUEL

A major disruptive event within a facility for production of recycled fuel (from ^{239}Pu, ^{233}U or mixed oxide fuel) has the potential to overload or bypass dynamic containment. Examples of such disruptive events include explosions, major fires and structural collapse. Since the consequences of these events are likely to be unacceptable, an internal disruption could be considered to be a DEC, in order to identify safety enhancements to strengthen the capability of the facility to withstand accidents more severe than DBAs.

Severe accidents that have occurred at other similar facilities (or at nuclear installations in general) have to be considered DECs for the purpose of emergency preparedness and response.

Annex V

EXAMPLES OF RULES FOR SAFETY ANALYSIS FOR NUCLEAR FUEL CYCLE FACILITIES

The following are examples of rules for the safety analysis of nuclear fuel cycle facilities:

- The application of single failure criterion.
- Rules for crediting systems with respect to system qualification (or lack of qualification) in the environment resulting from an accident.
- Rules for crediting safety systems, including reliability in quantitative terms, if appropriate.
- Rules for crediting support systems, such as normal and emergency electrical power systems and cooling water systems.
- Rules for considering concurrent failures.
- Rules for crediting redundancy trip parameters.
- Rules for crediting the actions of systems that are independent.
- Rules for accounting for diversity of monitoring variables (e.g. consideration of the second acting variable in each case (single failure in addition to the initial failure that led to the postulated initiating event).
- Rules for crediting operator actions (e.g. emergency planning and preparedness and response time).
- Rules for exclusion of postulated initiating events considered not applicable or sufficiently unlikely to occur. This may necessitate justification by design analysis or engineering judgement.
- Whether frequency or probability evaluations will be carried out to assess system response, the extent to which such methods will be used and the methodologies to be employed (including validation).
- Rules on the use of calculation models or computer tools: These normally require validation of methods and codes for their applicability and accuracy against relevant experimental data (including commissioning or operational data).
- Rules on the use of values of the input parameters used in the safety analysis: This may require selection of conservative predictions of consequences of each event, and use of uncertainties associated with each parameter.
- Rules on the use of empirical correlations used in the computer codes: These may include validation or demonstration of conservative use of the correlations, justification of any scaling from laboratory dimensions and controls to prevent their application beyond the range of experimental data.

Annex VI

CONSIDERATIONS AND EXAMPLES OF ITEMS TO BE COVERED BY LIMITING CONDITIONS FOR SAFE OPERATION OF NUCLEAR FUEL CYCLE FACILITIES

Limiting conditions for safe operation of a nuclear fuel cycle facility have to cover all equipment and operational parameters for safe operation. These could be operational constraints or administrative limitations that are imposed by the design, operational features and safety analysis of the facility. Examples of limiting conditions for safe operation of nuclear fuel cycle facilities are presented below. The list of these conditions is not intended to be comprehensive, but to provide guidance on items that may need to be considered in establishing limiting conditions for safe operation.

The items provided below on limiting conditions for safe operation are presented in groups by topic (such as by system or activity) for convenience. Although the presentation in the facility's documentation of these limiting conditions in groups provides logical arrangement and clarity, alternative approaches to grouping the limiting conditions exist among Member States. It is also important to note that some of the items provided below (e.g. type and frequency of periodic testing and inspection of equipment, instrumentation and automatic controls) could also be part of the periodic testing and surveillance requirements of the operational limits and conditions.

Examples of limiting conditions for safe operation of nuclear fuel cycle facilities include the following:

(a) Radioactive materials and their handling:
- Type, physical form, maximum capacity allowed in the facility or in system processes (e.g. gloveboxes), isotopic composition, nuclear material enrichment and burnup, etc.;
- Minimum margins against nuclear criticality during facility operation and during handling of fissile materials (including fissile contents, mass, geometry or shape, density and forms of materials, moderation, moisture content, reflection and neutron absorbers);
- Requirements on material movements (e.g. staffing, tools and measurements), including on-site and off-site transportation;
- Requirements on material handling tools and equipment, including cranes (maximum allowable loads and testing requirements);
- Requirements on storage containers (including geometry), levels of surface contamination and dose rate, mass, etc.

(b) Chemicals:
  - Types and maximum allowed quantity of chemicals inside the facility/processes, particularly flammable or explosive materials;
  - Conditions of storage, including maximum quantity, suitability of materials for use in the environmental conditions.

(c) Operation:
  - Minimum operability requirements of structures, systems and components, including structure, system and component configurations for different modes of facility/process operation;
  - Conditions of startup and shutdown of process, including relevant checklists, inspections and verifications, and conditions to restart after shutdown is triggered by automatic actions;
  - Conditions on physical parameters of processes such as temperatures, pressures and flow;
  - Requirements on cooling systems, as applicable, including coolant temperatures, flow rates and pressures in different cooling lines, coolant chemistry conditions (contents of solids, pH, conductivity, etc.), coolant availability, emergency cooling, removal of decay heat, leakage detection, etc.;
  - Requirements on performance of structures, systems and components, including periodic testing and inspections, and their types and frequency.

(d) Means of confinement:
  - Airflows (and, where appropriate, temperatures and humidity) within the facility and its processes;
  - Target pressure drops across filters;
  - Pressures within the facility buildings (rooms, cells or boxes, as appropriate) relative to the atmosphere (under normal and emergency conditions);
  - Isolation of means of confinement and starting of emergency ventilation;
  - List of operations that require confinement;
  - Configuration and minimum equipment for ventilation system;
  - Leak rate from the means of confinement;
  - Efficiency of filters.

(e) Instrumentation and control systems:
  - Type and minimum number of items of measuring equipment associated with safety controls.
  - Requirements for the calibration of instrumentation and its periodic control.

- — Types and frequency of periodic testing of instrumentation and automatic control actions (e.g. shutdown, interlock actions, as applicable), as well as time between two successive tests.
- — Types and minimum amount of startup instrumentation.
- — Display monitors.
- — Data acquisition systems.
- — Monitoring system and associated alarm settings: Values of the alarm settings for instrumentation in the facility and process equipment necessary for safety.

(f) Radiation protection and waste management:
- — Annual dose limits;
- — Dose constraints and targets (individual and collective);
- — Requirements on personal dosimetry, including frequency of measurements of personal detectors and bioassay whole body counting;
- — Limits on airborne concentration of radionuclides;
- — Limits for surface contamination;
- — Maximum allowable release to the environment in period;
- — Type and location of radiation monitoring equipment (including equipment for criticality monitoring) within the facility, including requirements on checks, verification and calibration;
- — Requirements on availability of portable radiation monitors (type and number), including calibration;
- — Requirements on working area classification from a radiation protection point of view, and the associated rules for accessing, working at and leaving controlled areas;
- — Requirements on radiation monitoring of the workplace (e.g. routine radiation and contamination monitoring within the facility, including its frequency);
- — Alarm setting for criticality and radiation monitoring instruments;
- — Requirements on monitoring of radioactive waste;
- — Storage capacity for liquid and solid waste.

(g) Electrical systems:
- — Requirements on emergency power supply and its testing frequency;
- — Availability of uninterruptable supply and diesel generators.

(h) Other systems and process auxiliaries:
- — Requirements on fire protection systems;
- — Requirements on process auxiliaries;
- — Communication systems;
- — Emergency lighting systems;
- — Emergency equipment.

(i) Administrative requirements:
- Requirements on staffing (e.g. minimum staffing);
- Prerequisites for activities with safety significance, including in particular the transport of radioactive or fissile material (on- and off-site).

Annex VII

EXAMPLES OF DOCUMENTATION TO BE SUBMITTED TO REGULATORY BODY IN VIEW OF LICENSING PROCESS FOR NUCLEAR FUEL CYCLE FACILITIES

The following list gives examples of documents to be submitted to the regulatory body during the licensing process (adapted for nuclear fuel cycle facilities) as specified in the appendix to IAEA Safety Standards Series No. SSG-12, Licensing Process for Nuclear Installations [VII–1]. The content of these documents may be divided or combined into different documents, as appropriate:

— A descriptive construction report (including a quality manual), which consists of a description of the basic information on the nuclear fuel cycle facility, the process and technologies used, justification of related activities and provisions for decommissioning;
— References to and benchmarks against other relevant nuclear installations, including those in other States, if any, and a summary of the most significant differences between the installations;
— A draft plan for the project, including phases and anticipated schedule (including technical research and development, if necessary), a prior economic study regarding the necessary financial investments and the expected costs;
— A site evaluation report, which may include a report on the environmental radiation monitoring programme and all or some of the elements dealing with the site evaluation;
— Reports on discharges into the environment, and a report on the environmental impact assessment;
— Public inquiry strategy plans and reports according to the State’s framework and practices;
— A report on the management and organization of the design and construction project, including responsibilities and a list of contractors;
— A report on the acquisition programme, including a list of the SSCs and their origin, and, as applicable, details of the manufacturing process for SSCs important to safety;
— The strategic plan for the licensing process, including the set of requirements, guides, codes and standards to comply with, which may be partly adopted from the vendor State (if any);

— A preliminary safety analysis report before authorization to begin construction, which may include information on site evaluation, the design basis, nuclear and radiation safety, deterministic analyses and complementary probabilistic safety assessment;
— Plans relating to the operating organization and its management system for all licensing steps;
— Technical design documents;
— Features of physical protection important to safety and measures on the interfaces between nuclear safety and security;
— Fire hazard assessment and fire protection plans;
— Plans for accounting for and control of nuclear material;
— Training and qualification programme plans for operations personnel;
— Proof of trustworthiness of all staff who will be engaged in responsible or sensitive positions;
— Commissioning programmes and reports dealing with the commissioning stage;
— Final safety analysis report, which may include all or parts of the elements relevant to the site evaluation, design, construction, commissioning and operation stages and provisions for decommissioning;
— Ageing management programme, including consideration of ageing management at different stages of the facility’s lifetime;
— General operating rules, including details of all elements dealing with the operation stage and operating procedures;
— Operating procedures for accident management;
— Technical specifications, including all operational limits and conditions (may be included in the general operating rules);
— A plan for collecting and applying feedback on operating experience;
— Plans for evaluating and improving safety performance;
— Reports and manuals on the radiation protection programme;
— Modification rules (may be included in the general operating rules);
— Details of the maintenance programme and the periodic testing programme;
— Reports of periodic safety reviews or other safety reviews;
— Reports on radioactive waste and spent fuel management, including a description of the system for the classification and characterization of waste, and rules and criteria to release waste;
— Emergency preparedness and response plans;
— Decommissioning plans and reports, including details of final shutdown, and decommissioning substages, actions and safety analyses.

REFERENCE TO ANNEX VII

[VII–1] INTERNATIONAL ATOMIC ENERGY AGENCY, Licensing Process for Nuclear Installations, IAEA Safety Standards Series No. SSG-12, IAEA, Vienna (2010).

ABBREVIATIONS

AOO	anticipated operational occurrence
DBA	design basis accident
DEC	design extension condition
PIE	postulated initiating event
PSA	probabilistic safety assessment
QRA	quantitative risk assessment
SSCs	structures, systems and components

CONTRIBUTORS TO DRAFTING AND REVIEW

Atack, S.	Nuclear Regulatory Commission, United States of America
Beeslar, F.	South African Nuclear Energy Corporation SOC Ltd, South Africa
Bhattacharya, R.	Atomic Energy Regulation Board, India
Bogdanova, T.	Federal Environmental, Industrial and Nuclear Supervision Service of Russia, Russian Federation
Brassier-Lecarme, C.	Institut de radioprotection et de sûreté nucléaire, France
Gater, R.	International Atomic Energy Agency
Glazbrook, D.	International Atomic Energy Agency
Groche, K.	AREVA, Germany
Khotylev, V.	Canadian Nuclear Safety Commission, Canada
Kilina, O.	State Scientific Technical Centre (Nuclear and Radiation Safety), Ukraine
Kotzalas, M.	Nuclear Regulatory Commission, United States of America
Nepeypivo, M.	International Atomic Energy Agency
Rovny, J.	International Atomic Energy Agency
Shokr, A.M.	International Atomic Energy Agency
Takanashi, M.	Nuclear Regulatory Authority, Japan
Vasilica, O.	Fabrica de Combustibil Nuclear, Romania

IAEA
International Atomic Energy Agency

No. 26

ORDERING LOCALLY

IAEA priced publications may be purchased from the sources listed below or from major local booksellers.

Orders for unpriced publications should be made directly to the IAEA. The contact details are given at the end of this list.

NORTH AMERICA

Bernan / Rowman & Littlefield
15250 NBN Way, Blue Ridge Summit, PA 17214, USA
Telephone: +1 800 462 6420 • Fax: +1 800 338 4550
Email: orders@rowman.com • Web site: www.rowman.com/bernan

Renouf Publishing Co. Ltd
22-1010 Polytek Street, Ottawa, ON K1J 9J1, CANADA
Telephone: +1 613 745 2665 • Fax: +1 613 745 7660
Email: orders@renoufbooks.com • Web site: www.renoufbooks.com

REST OF WORLD

Please contact your preferred local supplier, or our lead distributor:

Eurospan Group
Gray's Inn House
127 Clerkenwell Road
London EC1R 5DB
United Kingdom

Trade orders and enquiries:
Telephone: +44 (0)176 760 4972 • Fax: +44 (0)176 760 1640
Email: eurospan@turpin-distribution.com

Individual orders:
www.eurospanbookstore.com/iaea

For further information:
Telephone: +44 (0)207 240 0856 • Fax: +44 (0)207 379 0609
Email: info@eurospangroup.com • Web site: www.eurospangroup.com

Orders for both priced and unpriced publications may be addressed directly to:

Marketing and Sales Unit
International Atomic Energy Agency
Vienna International Centre, PO Box 100, 1400 Vienna, Austria
Telephone: +43 1 2600 22529 or 22530 • Fax: +43 1 26007 22529
Email: sales.publications@iaea.org • Web site: www.iaea.org/publications

This Safety Report provides practical information on methods and practices for performing safety analysis and for the preparation of licensing documentation for nuclear fuel cycle facilities. A systematic methodology is presented, covering the establishment of acceptance criteria, hazard evaluation, identification of postulated initiating events, and analysis of accident sequences and consequences. Information is provided on application of the results of the safety analysis in the design and operational phases, and on appropriate management system processes. The publication applies to all stages of the lifetimes of relevant facilities and for modifications and upgrades.

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA
ISBN 978–92–0–107119–4
ISSN 1020–6450