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INPRO Methodology for Sustainability Assessment of Nuclear Energy Systems: Safety of Nuclear Fuel Cycle Facilities

INPRO Manual

IAEA

International Atomic Energy Agency

INPRO METHODOLOGY FOR SUSTAINABILITY ASSESSMENT OF NUCLEAR ENERGY SYSTEMS: SAFETY OF NUCLEAR FUEL CYCLE FACILITIES

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INPRO MANUAL

INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was launched in 2000, based on resolutions of the IAEA General Conference (GC(44)/RES/21). One of the INPRO objectives is to help to ensure that nuclear energy is available in the twenty-first century in a sustainable manner. To meet this objective, INPRO has been proceeding in steps.

In Phase 1, INPRO developed a methodology for assessing the long term sustainability of a national or international nuclear energy system. This entailed establishing a set of basic principles pertaining to system sustainability, a set of user requirements in support of each basic principle, and a set of criteria for meeting each user requirement. The resulting INPRO methodology was documented in the form of a sustainability assessment guidance manual consisting of an overview volume and eight volumes covering economics, infrastructure, waste management, proliferation resistance, physical protection, environment, safety of reactors and safety of nuclear fuel cycle facilities. The first edition of that manual was published in 2008 as IAEA-TECDOC-1575/Rev.1.

In Phase 2, Member States participating in INPRO have been performing national and international nuclear energy system assessments (NESAs) using the INPRO methodology. The results of NESAs completed by 2009 were published at the end of 2009 as IAEA-TECDOC-1636. Included in that IAEA publication were several proposals on how to update the INPRO methodology based on the experience of the assessors. Further recommendations on how to update the methodology were developed in parallel by the INPRO steering committee, IAEA experts and the INPRO group.

All the proposals and recommendations were evaluated by internal and external experts at IAEA consultancy meetings in 2012 and 2015 and at an IAEA technical meeting in 2016. Based on the outcomes of those meetings, the INPRO sustainability assessment methodology was updated. This publication covers the updated INPRO methodology for the area of safety of nuclear fuel cycle facilities.

The IAEA officers responsible for this publication were A. Korinny and J. Phillips of the Division of Nuclear Power.

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SUMMARY

This report, which is part of the INPRO methodology manual, provides guidance on assessing sustainability of a nuclear energy system (NES) in the area of safety of nuclear fuel cycle facilities (NFCFs). The sustainability assessment approach described is not an application of the IAEA safety standards and in no way replaces the safety assessments to be performed as part of the pre-licensing and licensing processes for a country's NFCFs. The manual focuses instead on the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) methodology requirements in the area of fuel cycle facility safety that need to be fulfilled to demonstrate the long term sustainability of the NES assessed.

The INPRO methodology for assessing NES sustainability in the area of NFCF safety consists of one basic principle, seven user requirements, and twenty-three criteria. Using the concept of a graded approach, a different number of user requirements and corresponding criteria may be deemed applicable for different NFCFs.

The INPRO basic principle for sustainability assessment in the area of NFCF safety sets the goal for designers/developers to achieve an advanced facility design that is demonstrably safer than a comparable reference facility now in operation. If such a reference facility cannot be defined, the facility assessed needs at least to be demonstrated being state of the art with regard to safety. In making such safety comparisons, although safety risks and consequences are usually relatively independent of inventory or throughputs, it may be necessary to make appropriate allowances for the differences in the scale (capacity) of the facilities. To reach this goal, the INPRO methodology encourages the designer/developer to:

- Incorporate enhanced defence in depth into an advanced NFCF design as part of the fundamental safety approach and ensure that the levels of protection in defence in depth are more independent from each other than in the reference facility;
- When appropriate to minimize hazards, incorporate inherently safe characteristics and passive systems into advanced designs as part of a fundamental approach to excelling in safety and reliability;
- Take human factors into account in the design and operation of an NFCF and establish and maintain a safety culture in all organizations involved in a nuclear power program;
- Perform sufficient RD&D work to bring the knowledge of NFCF characteristics and the capability of analytical methods used for design and safety assessment of an assessed facility with innovative features to at least the same confidence level as for a reference facility.

The first four INPRO user requirements (UR1 to UR4) for sustainability assessment in this area are directly linked to the concept of defence in depth. Emergency preparedness and response, which is considered in Level 5 of the defence in depth concept, are dealt with in the INPRO methodology area of Infrastructure.

The first INPRO user requirement for sustainability assessment in the area of NFCF safety, UR1, is mostly related to the first level of defence in depth (DID), which is focused on preventing deviations from normal operation and preventing failures of items important to safety. UR1 asks for an increase of robustness in the design assessed relative to a reference design with regard to operation and systems, structures and components failures. The major means to achieve robustness are to ensure high quality in design, construction and operation, including human performance. Additionally, one of the criteria asks for an efficient implementation of the concept of optimization of worker radiation protection through the use of automation, remote maintenance and operational experience from existing designs.

The second user requirement for sustainability assessment in the area of NFCF safety, UR2, involves limited consideration of selected provisions in the first DID level and mostly relates to the second level of DID, which deals with detection and control of failures and deviations from normal operational states in order to prevent anticipated operational occurrences from escalating to accident conditions.

UR2 asks for an instrumentation and control (I&C) system that is capable of detecting anticipated operational occurrences and failures, providing an alarm for the operators, and intercepting deviations from normal operational states. For specific anticipated operational occurrences, the I&C system is expected to initiate mitigating measures automatically, e.g. in case of a fire, release fire suppressant.

The third user requirement for sustainability assessment in the area of NFCF safety, UR3, is mostly related to the third level of DID, which concentrates on controlling accidents, preventing releases of radioactive materials and associated hazardous materials, and preventing radiation levels that require off-site protective actions. UR3 asks for new NFCF designs to have less frequent design basis accidents (DBAs), longer grace periods for operator action after DBAs, and greater reliability of engineered safety features and on-site accident management measures than in a reference NFCF design. UR3 further stipulates that at least one physical barrier[1] against accidental releasees of nuclear material and/or toxic chemicals to the environment should remain intact in DBAs.

The fourth user requirement for sustainability assessment in the area of NFCF safety, UR4, focuses on accident conditions more severe than DBAs. UR4 is mainly related to the design extension conditions and to the fourth level of DID, which aims to mitigate the consequences of accidents that result from failure of the third level of DID and ensure that radioactive releases are kept as low as reasonably achievable. UR4 requires the assessed design to minimize the frequency of accidents with off-site contamination, i.e. to demonstrate that the accidental releases of nuclear materials and/or toxic chemicals to the environment have very low frequencies. The calculated source term of potential accidental release into the environment is expected to remain well within the envelope of the reference facility source term. The calculated consequences are not expected to require public evacuation.

The fifth INPRO user requirement for sustainability assessment in the area of NFCF safety, UR5, asks for increased independence of each level of defence in depth to be confirmed using appropriate methods (e.g. probabilistic and deterministic analysis) and for minimization of hazards by incorporating, if appropriate, inherently safe characteristics into the design assessed.

The sixth INPRO user requirement, UR6, asks that the safe operation of an NFCF be supported by an improved human machine interface and by the establishment and maintenance of a strong safety culture in all national organizations involved in a nuclear power program.

The seventh INPRO user requirement, UR7, asks the nuclear technology developer to perform sufficient RD&D for innovative design features to bring the knowledge of advanced NFCF characteristics and the capability of analytical methods to at least the same confidence level as for existing NFCFs.

INPRO methodology requirements related to nuclear law, institutional arrangements including the regulatory body, and emergency preparedness and response have been considered as part of

[1]Confinement shall be provided by two complementary containment systems — static (physical barrier) and dynamic (e.g. ventilation). The static containment of an NFCF (excluding mining and milling) shall have at least one physical barrier between radioactive material and operating areas (workers) and at least one additional physical barrier between operating areas and the environment.

the national infrastructure necessary to create and maintain a sustainable nuclear energy system and are therefore published in the INPRO manual covering the infrastructure area.

INPRO methodology user requirements are primarily recommendations to the developers of new nuclear technology on how to make a new NES more sustainable in relation to nuclear safety. The INPRO methodology justifies the requested increase in safety by noting that an assumed significant increase of installed nuclear power during the twenty-first century would theoretically increase the risks of nuclear power, unless nuclear technology is developed further with regard to enhanced safety. Therefore, the overall objective of the INPRO methodology in the area of NFCF safety is to demonstrate continuous improvement of NFCF safety characteristics in order to avoid or minimise a potential increase in the global risks of nuclear power.

In summary, the elements of the INPRO methodology described in this publication (i.e. sustainability assessment in the area of NFCF safety) evaluate safety enhancements in new designs but do not evaluate compliance with national or international safety standards. The INPRO assessment of sustainability in the area of NFCF safety is performed with respect to a reference design that is assumed to comply with applicable safety standards. Confirmation of compliance of either the reference or the new design with national or international safety standards is outside the scope of the INPRO methodology. If such confirmation is needed, a separate peer review[2] can be performed.

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[2] Peer review of the safety assessment report can be performed based on applicable national regulations and IAEA safety standards.

1. INTRODUCTION

1.1. BACKGROUND

This publication is an update of Volume 9, Safety of Nuclear Fuel Cycle Facilities, of the INPRO manual published as IAEA-TECDOC-1575 Rev.1, Guidance for the Application of an Assessment Methodology for Innovative Nuclear Energy Systems [1]. The update is based on recommendations presented by Member States participating in the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and supplemented by IAEA experts. The information presented in the INPRO methodology overview manual published in Volume 1[3] of Ref [1] should be considered as an integral part of this publication and the user is invited to become familiar with that information.

The concept of sustainable development was originally introduced in the 1980s. It defines sustainable development as development that meets the needs of the present without compromising the ability of future generations to meet their own needs. This concept embraces all environmentally sensitive areas of human activities, including different types of energy production. In the area of nuclear energy, the focus of sustainable development is on solving key institutional and technological issues including nuclear accident risks, health and environment risks, proliferation risks, economic competitiveness, radioactive waste disposal, sufficiency of institutions and public acceptability. Sustainable development implies demonstration of progress in the key issue areas. The INPRO methodology is the tool for assessing the sustainability and sustainable development of a nuclear energy system, that was originally created in 2003 under the aegis of the IAEA using broad philosophical outlines of the concept of sustainable development.

INPRO basic principles, user requirements and criteria have been defined for assessing NES sustainability in different areas, i.e. economics, infrastructure (legal and institutional measures), waste management, proliferation resistance, environmental impact of stressors, environmental impact of depletion of resources, and safety of nuclear reactors and NFCFs. The INPRO basic principles establish goals that should be met in order to achieve long term sustainability of a NES. An INPRO user requirement of sustainability defines what different stakeholders (users) in a NES should do to meet the goal defined in the basic principle. A criterion enables the assessor to determine whether a user requirement has been met. Using the INPRO methodology to assess the sustainability of a NES is a bottom up exercise. It consists of determining for each INPRO methodology criterion the value of each INPRO methodology indicator for that criterion and comparing that value with the corresponding INPRO methodology acceptance limit. The comparison then provides a basis for judging the capability of the assessed NES to meet the respective sustainability criterion. The ultimate goal of using the INPRO methodology is to check whether the NES assessed fulfils all the criteria defined for the assessment of sustainability and hence meets the user requirements and complies with the basic principle and therefore represents a long-term sustainable system for a Member State (or group of Member States).

One possible output from an assessment is the identification of areas where a given NES needs to be improved. Given the comprehensive nature of an assessment using the INPRO methodology, such an assessment would be expected to indicate clearly the specific attributes of a NES that need to be improved. The assessment could thus become an important input to

[3] An update of the INPRO methodology overview manual is in preparation at time of press.

the identification of necessary activities or desirable research, development and demonstration (RD&D) objectives.

The updated INPRO methodology manuals that have been already published are found in Refs [2–6].

1.2. OBJECTIVE

This volume of the updated INPRO manual provides guidance to the assessor of a planned NES (or a facility) on how to apply the INPRO methodology in the area of NFCF safety. The INPRO assessment is expected either to confirm the fulfilment of all INPRO methodology NFCF criteria, or to identify which criteria are not fulfilled and note the corrective actions (including RD&D) that would be necessary to fulfil them. It is recognized that a given Member State may adopt alternative criteria with indicators and acceptance limits that are more relevant to its circumstances. Accordingly, the information presented in Chapters 5 to 10 (INPRO methodology criteria, user requirements and basic principle for sustainability assessment in the area of safety of NFCFs) should be viewed as guidance. However, the use of such alternative criteria should be justified as providing an equivalent level of enhanced safety as the INPRO methodology.

This report discusses the INPRO sustainability assessment method for the area of safety of NFCFs. The INPRO sustainability assessment method for safety of nuclear reactors is discussed in a separate report of the INPRO manual[4].

This publication is intended for use by organizations involved in the development and deployment of a NES including planning, design, modification, technical support and operation for NFCF. The INPRO assessor (or a team of assessors) is assumed to be knowledgeable in the area of safety of NFCFs and/or may be using the support of qualified national or international organizations (e.g. the IAEA) with relevant experience. Two general types of assessors can be distinguished: a nuclear technology holder (i.e. a designer, developer or supplier of nuclear technology), and a (potential) user of such technology. The role of a technology user in an INPRO assessment is to check in a simplified way whether the supplier's facility design appropriately accounts for nuclear safety related aspects of long term sustainability as defined by the INPRO methodology[5]. A designer (developer) can use this guidance to check whether a new design under development meets the sustainability focused INPRO methodology criteria in the area of fuel cycle safety and can additionally initiate modifications during early design stages if necessary to improve the safety level of the design. The current version of the manual includes a number of explanations, discussions, examples and details so it is deemed to be used by technology holders and technology users.

1.3. SCOPE

This manual provides guidance for assessing the sustainability of a NES in the area of NFCF safety. This report deals with NFCFs that may be potentially involved in the NES, i.e. mining, milling[6], refining, conversion, enrichment, fuel fabrication, spent fuel storage, and spent fuel

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[4] The INPRO methodology user requirements and criteria are developed for the assessment of sustainability of nuclear energy systems and may incorporate new developments from different areas, not to be confused with the Vienna Declaration on Nuclear Safety.

[5] A technology user is assumed – in order to minimize its risk – to be primarily interested in installing NFCFs based on proven technology with designs that have been licensed (e.g. in the country of the supplier) and that have operated successfully for a sufficiently long time.

[6] Milling is also called processing.

reprocessing facilities. It is clear that operations of NFCFs are more varied in their processes and approaches than are nuclear reactor systems. Most significant of these variations is the fact that some countries pursue an open fuel cycle, i.e. spent fuel is treated as a waste, while some others have a policy of closing the fuel cycle, i.e. treating the spent fuel as a resource, and a number of states have yet to make a final decision on an open or closed fuel cycle. Further, diversity is large if one considers different types of fuels used in different types of reactors and the different routes used for processing the fuels before and after their irradiation depending upon the nature of the fuel (e.g. fissile material: low enriched uranium/ natural uranium/ uranium-plutonium/ plutonium/ thorium; fuel form: metal/ oxide/ carbide/ nitride) and varying burnup and cooling times. Taking into account this complexity and diversity, the approach adopted in this report has been to deal with the issues as far as possible in a generic manner, rather than describing the operations that are specific to certain fuel types. This approach has been chosen in order to arrive at a generalized procedure that enables the user of this report (the assessor) to apply it with suitable variations as applicable to the specific fuel cycle technology being assessed. In addition, it is recognized that the defence in depth (DID) approach and ultimate goal of inherent safety form the fundamental tenets of safety philosophy. The DID approach is applied to the specific safety issues of NFCFs.

As the safety issues relevant to the sustainability assessment of refining and conversion facilities are similar to those of enrichment facilities, the INPRO methodology criteria for those two types of facilities are combined in this manual and not discussed separately. Based on similar considerations, the assessments of uranium and uranium-plutonium mixed oxide (MOX) fuel fabrication facilities have likewise been combined[7]. However, particular care must be taken to ensure that using a graded assessment approach and enhanced safety measures for higher risk facilities (e.g. using plutonium or uranium with higher enrichments/criticality risks) will yield appropriately enhanced levels of safety.

It should be noted that for NFCFs the INPRO methodology includes the consideration of chemical and industrial safety issues, principally where these could affect facility integrity or radiological safely. Although otherwise beyond the scope of this guidance, it bears noting that care is required due to the different public perceptions of the risks posed by conventional and radiological events and releases and, conversely, the negative reactions that may be generated about an NFCF's radiological safety if conventional safety events occur.

In the current version of the INPRO methodology, the sustainability issues relevant to safety of reactors and safety of NFCFs are considered in different areas. Innovative integrated systems combining reactors, fuel fabrication and reprocessing facilities on the same site such as molten salt reactors with nuclear fuel in liquid form and integrated fast reactors with metallic fuel has not been specifically addressed. Reactor and NFCF installations of such integrated systems are expected to be assessed simultaneously and independently against corresponding criteria in the INPRO areas of reactor safety and safety of NFCFs. When more detailed information on the safety issues in integrated systems has been acquired, this approach can be changed in the next revisions of the INPRO methodology.

NFCFs processing nuclear materials in a given stage of the fuel cycle may be based on different technologies with different safety issues. Different kinds of fuel may be fabricated or reprocessed in different facilities serving different reactors. In this report, the discussion is restricted to the fabrication of fuels most commonly used in power reactors; however, the requirements and criteria have been formulated in a sufficiently generic manner and are therefore expected to be applicable to innovative technologies. Nevertheless, the fabrication or

[7] This approach is applicable to the sustainability assessment of NESs and is valid only within the INPRO methodology at its relatively generic criteria. The approach cannot be used for safety assessments per se.

reprocessing technologies for innovative types of fuels (e.g. TRISO fuel with carbon matrix, metal fuel, nitride fuel) may involve safety issues requiring the modification of specific INPRO methodology criteria or the introduction of new or complementary criteria. It is expected that the future accrual of more detailed information on safety issues in innovative NFCFs will give rise to proposed modifications of the INPRO criteria and that these will be considered in future revisions of the methodology.

In this version of the INPRO methodology, the transportation of fresh nuclear fuel, spent nuclear fuel, and other radioactive materials or wastes throughout the nuclear fuel cycle has not been generally considered as independent stages of the nuclear fuel cycle. The INPRO methodology does not define specific requirements and criteria for such transportation but assumes that the safety issues of transportation are to be considered as part of the INPRO assessments of those NFCFs from which such packaging and transportation activities originate, e.g. fuel fabrication facilities for fresh fuel transportation and spent fuel storage facilities for spent fuel transportation. The IAEA has developed a set of safety standards to establish requirements and recommendations that need to be satisfied to ensure safety and to protect persons, property and the environment from the effects of radiation in the transport of radioactive material [7–12].

This manual does not establish any specific safety requirements, recommendations or criteria. The INPRO methodology is an internationally developed metric for measuring nuclear energy system sustainability and is intended for use in support of nuclear energy system planning studies. IAEA safety requirements and guidance are only issued in the IAEA Safety Standards Series. Therefore, the basic principles, user requirements and associated criteria contained in the INPRO methodology should only be used for sustainability assessments. The INPRO methodology is typically used by Member States in conducting a self-assessment of the sustainability and sustainable development of nuclear energy systems. This manual should not be used for formal or authoritative safety assessments or safety analyses to address compliance with the IAEA Safety Standards or for any national regulatory purpose associated with the licensing or certification of nuclear facilities, technologies or activities.

The manual does not provide guidance on implementing fuel cycle safety activities in a country. Rather, the intention is to check whether such activities and processes are (or will be) implemented in a manner that satisfies the INPRO methodology criteria, and hence the user requirements and the basic principle for sustainability assessment in the area of safety of NFCFs.

1.4. STRUCTURE

This publication follows the relationship between the concept of sustainable development and different INPRO methodology areas. Section 2 describes the linkage between the United Nations Brundtland Commission's concept of sustainable development and the IAEA's INPRO methodology for assessing the sustainability of planned and evolving NESs. It further describes general features of NFCF safety and presents relevant background information for the INPRO assessor. Section 3 identifies the information that needs to be assembled to perform an INPRO assessment of NES sustainability in the area of NFCF safety. Section 4 identifies the different types of facilities that can form part of a nuclear fuel cycle. This section also provides an overview of the general safety aspects of those facilities. Section 5 presents the rationale and background of the basic principle and user requirements for sustainability assessment in the INPRO methodology area of NFCF safety. Criteria are then presented in Sections 6 to 10 along with a procedure at the criterion level for assessing the potential of each NFCF to fulfil the respective INPRO methodology requirements. The Annex presents a brief overview of the selected IAEA Safety Standards for NFCFs that are the basis of the INPRO methodology in this area. The Annex also explains the relationship and differences between the IAEA Safety Standards and the INPRO methodology.

Table 1 provides an overview of the basic principle and user requirements for sustainability assessment in the area of NFCF safety.

TABLE 1. OVERVIEW OF THE INPRO BASIC PRINCIPLE AND USER REQUIREMENTS FOR SUSTAINABILITY ASSESSMENT IN THE AREA OF NFCF SAFETY

INPRO basic principle for sustainability assessment in the area of NFCF safety: The planned NFCF is safer[8] than the reference NFCF. In the event of an accident, off-site releases of radionuclides and/or toxic chemicals are prevented or mitigated so that there will be no need for public evacuation[9].	
UR1: Robustness of design during normal operation	The assessed NFCF is more robust than the reference design with regard to operation and systems, structures and components failures.
UR2: Detection and interception of AOOs	The assessed NFCF has improved capabilities to detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions.
UR3: Design basis accidents (DBAs)	The frequency of occurrence of DBAs in the assessed NFCF is reduced. If an accident occurs, engineered safety features and/or operator actions are able to restore the assessed NFCF to a controlled state, and subsequently to a safe state, and the consequences are mitigated to ensure the confinement of radioactive and/or toxic chemical material. Reliance on human intervention is minimal, and only required after sufficient grace period.
UR4: Severe plant conditions	The frequency of an accidental release of radioactivity into the environment is reduced. The source term of accidental release into the environment remains well within the envelope of the reference facility source term and is so low that calculated consequences would not require public evacuation.
UR5: Independence of DID levels and inherent safety characteristics	An assessment is performed to demonstrate that the DID levels are more independent from each other than in the reference design. To excel in safety and reliability, the assessed NFCF strives for better elimination or minimization of hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics.
UR6: Human factors (HF) related to safety	Safe operation of the assessed NFCF is supported by accounting for HF requirements in the design and operation of the facility, and by establishing and maintaining a strong safety culture in all organizations involved in the life cycle of the facility.
UR7: RD&D for advanced designs	The development of innovative design features of the assessed NFCF includes associated research, development and demonstration (RD&D) to bring the knowledge of facility characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for operating facilities.

[8] The term 'safer' should be taken to mean 'better' or having lower risk throughout this document. That is, safer means a reduction in either event frequency or event consequences, or a combination of both, i.e. lower event frequency and reduced event consequences.

[9] Other protective measures still may be needed. Effective emergency planning, preparedness and response capabilities will remain a prudent requirement as discussed in the infrastructure area of the INPRO methodology.

2. NFCF SAFETY ISSUES RELATED TO NUCLEAR ENERGY SYSTEM SUSTAINABILITY

This section presents the relationship of the INPRO methodology with the concept of sustainable development, a comparison of NFCFs with chemical plants and nuclear reactors, and a summary of INPRO recommendations on the application of the DID concept to NFCFs.

2.1. THE CONCEPT OF SUSTAINABLE DEVELOPMENT AND ITS RELATIONSHIP WITH THE INPRO METHODOLOGY IN THE AREA OF NFCF SAFETY

The United Nations World Commission on Environment and Development Report [13] (often called the Brundtland Commission Report), defines sustainable development as "development that meets the needs of the present without compromising the ability of future generations to meet their own needs" (para.1). Moreover, this definition:

> "contains within it two key concepts:
>
> - the concept of 'needs', in particular the essential needs of the world's poor, to which overriding priority should be given; and
> - the idea of limitations imposed by the state of technology and social organization on the environment's ability to meet present and future needs."

Based on this definition of sustainable development a three-part test of any approach to sustainability and sustainable development was proposed within the INPRO project: 1) current development should be fit to the purpose of meeting current needs with minimized environmental impacts and acceptable economics, 2) current research, development and demonstration programmes should establish and maintain trends that lead to technological and institutional developments that serve as a platform for future generations to meet their needs, and 3) the approach to meeting current needs should not compromise the ability of future generations to meet their needs.

The definition of sustainable development may appear obvious, yet passing the three-part test is not always straightforward when considering the complexities of implemented nuclear energy systems and their many supporting institutions. Indeed, many approaches may only pass one or perhaps two parts of the test in a given area and fail the others. Where deficiencies are found, it is important that appropriate programmes be put in place to meet all test requirements to the extent practicable. Nevertheless, in carrying out an NFCF INPRO assessment, it may be necessary to make judgements based upon incomplete knowledge and to recognize, based upon a graded approach, the variable extent of the applicability of these tests for a given area.

The Brundtland Commission Report's overview (para.61 in Ref [13]) on nuclear energy summarized the topic as follows:

> "After almost four decades of immense technological effort, nuclear energy has become widely used. During this period, however, the nature of its costs, risks, and benefits have become more evident and the subject of sharp controversy. Different countries world-wide take up different positions on the use of nuclear energy. The discussion in the Commission also reflected these different views and positions. Yet all agreed that the generation of nuclear power is only justifiable if there are solid solutions to the unsolved problems to which it gives rise. The highest priority should be accorded to research and development on environmentally sound and ecologically viable alternatives, as well as on means of increasing the safety of nuclear energy."

The Brundtland Commission Report presented its comments on nuclear energy in Chapter 7, Section III. In the area of nuclear energy, the focus of sustainability and sustainable development is on solving certain well known problems (referred to here as 'key issues') of institutional and technological significance. Sustainable development implies progress and solutions in the key issue areas. Seven key issues are discussed:

1) Proliferation risks;
2) Economics;
3) Health and environment risks;
4) Nuclear accident risks;
5) Radioactive waste disposal;
6) Sufficiency of national and international institutions (with particular emphasis on intergenerational and transnational responsibilities);
7) Public acceptability.

The INPRO methodology for self-assessing the sustainability and sustainable development of a NES is based on the broad philosophical outlines of the Brundtland Commission's concept of sustainable development described above. Although three decades have passed since the publication of the Brundtland Commission Report and eighteen years have passed since the initial consultancies on development of the INPRO methodology in 2001 the definitions and concepts remain valid. The key issues for sustainable development of NESs have remained essentially unchanged over the intervening decades, although significant historical events have starkly highlighted some of them.

During this period, several notable events have had a direct bearing on nuclear energy sustainability. Among these were events pertaining to non-proliferation, nuclear security, waste management, cost escalation of new construction and, most notably, to nuclear safety.

Each INPRO methodology manual examines a key issue of NES sustainable development. The structure of the methodology is a hierarchy of INPRO basic principles, INPRO user requirements for each basic principle, and specific INPRO criteria[10] for measuring whether each user requirement has been met. Under each INPRO basic principle for the sustainability assessment of NESs, the criteria include measures that take into consideration the three-part test based on the Brundtland Commission's definition of sustainable development as described above.

The Commission Report noted that national governments were responding to nuclear accidents by following one of three general policy directions:

> "National reactions indicate that as they continue to review and update all the available evidence, governments tend to take up three possible positions:
>
> - remain non-nuclear and develop other sources of energy;
> - regard their present nuclear power capacity as necessary during a finite period of transition to safer alternative energy sources; or
> - adopt and develop nuclear energy with the conviction that the associated problems and risks can and must be solved with a level of safety that is both nationally and internationally acceptable."

These three typical national policy directions remain consistent with practice to the current day. Within the context of a discussion on sustainable development of nuclear energy systems, it would seem that the first two policy positions cannot result in development of a sustainable nuclear energy system in the long term since nuclear energy systems are either avoided altogether or phased out over time. However, it is arguable that both policy approaches can meet the three-part Brundtland sustainable development test if technology avoidance or phase-

[10] INPRO basic principles, user requirements and criteria for NES sustainability assessment.

out policies are designed to avoid foreclosing or damaging the economic and technological opportunity for future generations to change direction and start or re-establish a nuclear energy system. This has certain specific implications regarding long term nuclear education, knowledge retention and management and with regard to how spent nuclear fuels and other materials, strategic to nuclear energy systems, are stored or disposed of.

The third policy direction proposes to develop nuclear energy systems that "solve" the problems and risks through a national and international consensus approach to enhance safety. This is a sustainable development approach where the current generation has decided that nuclear energy is necessary to meet its needs, while taking a positive approach to develop enhanced safety to preserve the option in the future. In addition to the general outlines of how and why nuclear reactor safety is a principal key issue affecting the sustainability and sustainable development of nuclear energy systems, the Commission Report also advised that several key institutional arrangements should be developed. Since that time, efforts to establish such institutional arrangements have achieved a large measure of success. The Brundtland Commission Report was entirely clear that enhanced nuclear safety is a key element to sustainable development of nuclear energy systems. It is not possible to measure nuclear energy system sustainability apart from direct consideration of certain safety issues.

Understanding the psychology of risk perception in the area of nuclear safety is critical to understanding NES sustainability and sustainable development. In a real measured sense, taking into account the mortality and morbidity statistics of other non-nuclear energy generation technology chains (used for similar purpose), nuclear energy has an outstanding safety record, despite the severe reactor accidents that have occurred. However, it should not be presumed that this means that reactor safety is not a key issue affecting nuclear energy system sustainability. How do dramatically low risk estimations (ubiquitous in nuclear energy system probabilistic risk assessment) sometimes psychologically disguise high consequence events in the minds of designers and operators, while the lay public perception of risk (in a statistical sense) may be tilted quite strongly either toward supposed consequences of highly unlikely, but catastrophic disasters, or toward a complacent lack of interest in the entire subject? This issue has been studied for many years. What should be the proper metrics for the INPRO sustainability assessment methodology given that the technical specialist community has developed an approach that may seem obscure and inaccessible to the lay public?

With regard to nuclear safety, the public are principally focussed on the individual and collective risks and magnitude of potential consequences in case of accidents (radiological, economic and other psychosocial consequences taken together). In the current INPRO manual, the URs and CRs focus on assessment of the NES characteristics associated with the majority of these issues. Unlike several other key sustainability issues assessed in other areas of the INPRO methodology, Brundtland sustainability in the area of nuclear safety is intimately tied to public perception of consequence and risk. Continuously allaying public concern about nuclear reactor safety is central to sustainability and sustainable development of nuclear energy systems.

This report describes how to assess NES sustainability with respect to the safety of NFCFs.

2.2. HOW NFCFs COMPARE WITH NUCLEAR REACTORS AND CHEMICAL PLANTS

As stated in Section 3 of Ref [14], NFCFs imply a great diversity of technologies and processes. They differ from nuclear power plants (NPPs) in several important aspects, as discussed in the following paragraphs.

First, fissile materials and wastes are handled, processed, treated, and stored throughout NFCF mostly in dispersible (open) forms. Consequently, materials of interest to nuclear safety are more distributed throughout NFCF[11] in contrast to NPP, where the bulk of nuclear material is located in the reactor core or fuel storage areas. For example, nuclear materials in current reprocessing plants are present for most or part of the process in solutions that are transferred between vessels used for different parts of the processes, whereas in most[12] NPPs nuclear material is present in concentrated form as solid fuel.

Second, NFCFs are often characterized by more frequent changes in operations, equipment and processes, which are necessitated by treatment or production campaigns, new product development, research and development, and continuous improvement.

Third, the treatment processes in most NFCFs use large quantities of hazardous chemicals, which can be toxic, corrosive and/or combustible.

Fourth, the major steps in NFCFs consist of chemical processing of fissile materials, which may result in the inadvertent release of hazardous chemicals and/or radioactive substances, if not properly managed.

Fifth, the range of hazards in some NFCFs can include inadvertent criticality events, and these events can occur in different locations and in association with different operations.

Finally, in NFCFs a significantly greater reliance is placed on the operator, not only to run a facility during its normal operation, but also to respond to anticipated operational occurrences and accident conditions [15].

Whereas the reactor core of an NPP presents a very large inventory of radioactive material and coolant at high temperature and pressure[13]and within a relatively small volume, the current generation of NFCFs operate at near ambient pressure and temperature and with comparatively low inventories at each stage of the overall process. Accidents in NFCFs may have relatively low consequences when compared against nuclear power plants. Exceptions to this are facilities used for the large scale interim storage of liquid fission products separated from spent fuel and, where applicable, facilities for separating and storing plutonium.

In some cases in an NFCF, there are rather longer timescales involved in the development of accidents and less stringent process shutdown requirements are necessary to maintain the facility in a safe state, as compared to an NPP. Nevertheless, the INPRO area of NFCF safety applies the principles of the DID concept and encourages the NFCF designers to enhance the independence of DID levels[14] in new facilities. NFCFs also often differ from NPPs with respect to the enhanced importance of ventilation systems in maintaining their safety even under normal operation. This is because nuclear materials in these facilities are in direct contact with ventilation or off-gas systems. Various forms and types of barriers between radioactive inventories and operators may have different vulnerabilities. Fire protection and mitigation assume greater importance in an NFCF due to the presence of larger volumes of organic solutions and combustible gases. With fuel reprocessing or fuel fabrication facilities, the wide variety of processes and material states such as liquids, solutions, mixtures and powders needs to be considered in safety analysis.

From this point of view, the safety features of NFCFs are often more similar to chemical process plants than those of NPPs. In addition, radioactivity and toxic chemical releases and criticality

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[11] Except uranium or thorium mining and milling facilities.
[12] Exceptions include the molten salt reactors to be developed.
[13] Pressure is high in water and gas cooled reactors, but not in liquid metal or molten salt reactors.
[14] Taking into account the graded approach.

issues warrant more attention in NFCFs than in NPPs[15]. Further comparisons of the relevant features of an NPP, a chemical process plant and an NFCF are presented in Table 2.

TABLE 2. TYPICAL DIFFERENCES BETWEEN NPPs, CHEMICAL PROCESS PLANTS AND NFCFs (MODIFIED FROM REF [16])

Feature	NPP	Chemical Process Plant	NFCF
Type of hazardous materials	Mainly nuclear and radioactive materials	A variety of materials dependent on the plant (acids, toxins, explosives, combustibles, etc.)	- Nuclear and radioactive materials; - Acids, toxins, combustibles (nitric acid, hydrogen fluoride, solvents, process and radiolytic hydrogen, etc.)
Areas of hazardous sources and inventories	- Localized in core, fuel storage and spent fuel pool; - Standardized containment system, cooling of residual heat, criticality management	Distributed in the process and present throughout the process equipment	- Present throughout the process equipment in the facility; - Consisting both of nuclear materials and chemically hazardous materials; - Containment relies on both physical barriers and ventilation[16]
Physical forms of hazardous materials (at normal operation)	- Fuel in general is in solid form[17]; - Other radioactive materials in solid, liquid, gaseous form	Wide variety of physical forms dependent on the process, e.g. solid, liquid, gas, slurry, powder	- Wide variety of physical forms of nuclear and radioactive materials; - Wide variety of physical forms of chemically hazardous materials

As outlined above, from a safety point of view, NFCFs are characterized by a variety of physical and chemical treatments applied to a wide range of radioactive materials in the form of liquids, gases and solids. Accordingly, it is necessary to incorporate a correspondingly wide range of specific safety measures in these activities. Radiation protection requirements for the personnel are more demanding, especially in view of the many human interventions required for the operation and maintenance of an NFCF. The safety issues encountered in various NFCFs have been discussed in Refs [14, 15]. A comprehensive description of the safety issues of fuel cycle facilities is provided in Ref [17].

For most existing NFCFs, the emphasis is on the control of operations using administrative and operator controls to ensure safety as well as engineered safety features, as opposed to the emphasis on engineered safety features used in reactors. There is also more emphasis on criticality prevention in view of the greater mobility (distribution and transfer) of fissile materials. Because of the intimate human contact with nuclear materials in the process, which may include (open) handling and transfer of nuclear materials in routine processing, special attention is warranted to ensure worker safety. Potential intakes of radioactive materials require control to prevent and minimize contamination and thus ensure adherence to specified

[15] The objective of improving radiological safety remains unchanged. It should nevertheless be noted that the consequences of conventional toxic chemical releases, chemical fires and explosions usually include direct or indirect radiological impacts. The public perception of the risk of radiological hazards is much higher than that for chemical or industrial hazards and the design needs to ensure that all risks are appropriately addressed.

[16] For most types and design of NFCFs containment/confinement is provided by a combination of a 'static barrier' and a complementary 'dynamic' (e.g. provided by a ventilation system) barrier that, together, provide effective containment/ confinement.

[17] Except for molten salt reactors.

operational dose limits. In addition, releases of radioactive materials into the facilities and through monitored and unmonitored pathways can result in significant exposures.

The number of physical barriers in an NFCF that are necessary to protect the workers, the environment and the public depends on the potential internal and external hazards, and the consequences of failures; therefore the barriers are different in number and strength for different kinds of NFCFs (the graded approach). For example, in mining, the focus is on preventing contamination of ground or surface water with releases from uranium mining tails. Toxic chemicals and uranium by-products are the potential hazards of the conversion stage and for forms of in-situ mining. In enrichment and fuel fabrication facilities (with no recycling of separated or recovered nuclear material from spent fuel), safety is focused on preventing criticality in addition to avoiding contamination via low-level radioactive material.

It might be possible to enhance safety features in a nuclear energy system by co-location of front end (e.g. mining/ milling, conversion and enrichment, and fuel production facilities) and back end (reprocessing and waste management) facilities. This would have benefits through minimal transport, optimisation and alignment of processes, avoiding multiple handling of radioactive materials in different plants of the fuel cycle and comprehensive and integrated waste treatment and storage facilities.

Compared to safety of operating NPPs, only limited open literature is available on the experience related to safety in the operation of NFCFs. Examples of United States Nuclear Regulatory Commission regulation are provided in Refs [18-22]. Safety of and regulations for NFCFs have been discussed in IAEA meetings and conferences [14, 15]. Aspects of uranium mining have been reported extensively [23–30]. The Nuclear Energy Agency of the Organization for Economic Cooperation and Development published a comprehensive report on safety of nuclear installations in 2005 [31]. Safety guides on conversion/enrichment facilities, fuel fabrication, reprocessing and spent fuel storage facilities have also been published by the IAEA [32–36].

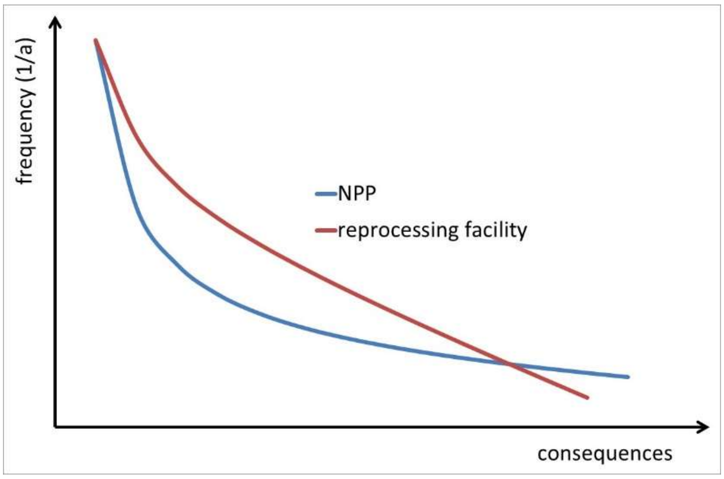

FIG.1. Conceptual comparison of safety characteristics between an NPP and a reprocessing facility.

It is obvious that in well-designed NFCFa, the safety related events that have a high hazard potential will have low frequency of occurrence and vice versa. For example, Fig. 1 (modified from Ref [37]) conceptually compares the relationship between potential consequences and frequency for safety related events in a nuclear power plant and a reprocessing facility.

The figure demonstrates that, compared to accidents in an NPP, an NFCF may have relatively higher consequences of accidents having higher probability of occurrence, e.g. accidental criticality. However, accidents with very high consequences have essentially lower probability than in NPPs and can only occur in a few high inventory NFCFs, typically large reprocessing plants and associated liquid high level waste interim storage facilities [38].

2.3. APPLICATION OF THE DEFENCE-IN-DEPTH CONCEPT TO NFCFs

The original concept of defence in depth was developed by the International Safety Advisory Group (INSAG) and published in 1996 [39]. Historically it is based on the idea of multiple levels of protection, including consecutive barriers preventing the release of radioisotopes to the environment, as already formulated in Ref [40]:

> "All safety activities, whether organizational, behavioural or equipment related, are subject to layers of overlapping provisions, so that if a failure were to occur it would be compensated for or corrected without causing harm to individuals or the public at large"

The application of DID to NFCFs takes into account their following features:

- The energy potentially released in a criticality accident in a fuel cycle facility tends to be relatively small. However, generalization is difficult as there are several fuel fabrication or reprocessing options for the same or different type of fuels;
- The power density in a fuel cycle facility in normal operation is typically several orders of magnitude less than in a reactor core;
- In a reprocessing facility, irradiated fuel pins are usually mechanically cut (chopped) into small lengths suitable for dissolution and the resultant solution is further subjected to chemical processes. This may create a possibility for larger releases of radioactivity to the environment on a routine basis as compared to reactors;
- The likelihood of a release of chemical energy is higher in fuel cycle facilities of reprocessing, re-fabrication, etc. Chemical reactions are part of the processes used for fresh fuel fabrication as well as for reprocessing of spent nuclear fuel.

The numbers of barriers to radioactive releases to the environment depend in different types of NFCFs on the forms, conditions, inventories and radiotoxicity levels of the processed nuclear materials. Table 3 gives a summary of the typical numbers of barriers to radioactive releases to the environment in existing NFCFs at different steps of nuclear fuel cycle.

TABLE 3. TYPICAL NUMBERS OF BARRIERS IN EXISTING NFCFS

Facility type		Number of barriers
Mining		0–1
Milling / Processing / Conversion		1–2
Enrichment		2
Fuel manufacture	Low radiotoxicity	1-2
	High radiotoxicity	2–3
Fresh fuel storage		2
Fresh fuel transportation		2
Spent fuel transportation		3
Spent fuel storage	Wet	2
	Dry	3
Reprocessing		3
Reprocessing product storage including waste	Low radiotoxicity	2
	High radiotoxicity	3

Table 4 summarises how INPRO uses the DID concept within this sustainability assessment methodology for the area of NFCF safety. The INPRO methodology applies this DID concept to all NCFCs as part of a graded approach that considers the level of risks in each individual facility.

TABLE 4. INPRO PROPOSALS FOR APPLYING THE DEFENCE-IN-DEPTH CONCEPT TO SUSTAINABILITY ASSESSMENT IN THE AREA OF NFCF SAFETY

Level	DID level purpose [17]	INPRO methodology proposals for NFCFs
1	Prevent deviations from normal operation and the failure of items important to safety.[18]	Enhance prevention by increasing the robustness of the design, and by further reducing human error probabilities in the routine operation of the plant. Enhance the independence among DID levels.
2	Detect and control deviations from operational states in order to prevent anticipated operational occurrences at the facility from escalating to accident conditions.	Give priority to advanced monitoring, alarm and control systems with enhanced reliability and intelligence. Together with qualified procedures for operators, the systems need to be able to anticipate and detect abnormal operational states, prevent their progression and restore normalcy. Enhance the independence among DID levels.
3	Prevent releases of radioactive material and associated hazardous material or radiation levels that require off-site protective actions.	Decrease the expected frequency of accidents. Achieve fundamental safety functions by an optimized combination of inherent safety characteristics, passive safety features, automatic systems and operator actions; limit and mitigate accident consequences; minimize reliance on human intervention, e.g. by increasing grace periods. Enhance the independence among DID levels.
4	Mitigate the consequences of accidents that result from failure of the third level of DID and ensure that the confinement function is maintained, thus ensuring that radioactive releases are kept as low as reasonably achievable.	Decrease the expected frequency of severe plant conditions; increase the reliability and capability of systems to control and monitor severe accident sequences; reduce the characteristics of the source term of the potential emergency off-site releases of radioactivity Avoid 'cliff-edge' failures of items important to safety. Enhance the independence among DID levels.
(5)	Mitigate the radiological consequences and associated chemical consequences of releases or radiation levels that could potentially result from accidents.	Emergency preparedness is covered in another area of the INPRO methodology called Infrastructure [2].

[18] Abnormal operation is identical to anticipated operational occurrences.

3. NECESSARY INPUT FOR A SUSTAINABILITY ASSESSMENT IN THE AREA OF SAFETY OF NUCLEAR FUEL CYCLE FACILITIES

3.1. DEFINITION OF A NUCLEAR ENERGY SYSTEM TO BE ASSESSED

In principle, a clear definition of the nuclear energy system (NES) is needed for an assessment using the INPRO methodology in all its areas. As described in the overview manual of the INPRO methodology[19], the NES is supposed to be selected, in general, based on an energy planning study. This study is expected to define the role of nuclear power (the amount of nuclear capacity to be installed over time) in an energy supply scenario for a country (or region or globally). Using the results of such a study, the next step is to choose the facilities of the selected NES that fit the determined role of nuclear power in the country (or region). The NES definition should include a schedule for deployment, operation and decommissioning of the individual facilities.

For a NES sustainability assessment in this area of the INPRO methodology, the NFCF to be assessed and a reference design have to be defined. Where possible, the reference design has to be determined as an NFCF of most recent design operating in 2013, preferably from the same designer as the assessed facility, and complying with the current safety standards. In such a case, the INPRO assessment in this area is expected to demonstrate an increased safety level to achieve long term sustainability in the assessed NFCF in comparison to the reference design. If a reference design cannot be identified within the same technology lineage, a similar existing comparable technology or, when other options are not available, an existing facility of different technology used for the same purpose can be used as a reference. If a reference design cannot be defined, it needs to be demonstrated through the assessment of RD&D results that the NFCF design employs the best international practice to achieve a safety level comparable to most recent technology and that the assessed facility is therefore state of the art.

3.2. INPRO ASSESSMENT BY A TECHNOLOGY USER

An INPRO assessor, being a technology user, needs sufficiently detailed design information on the NFCF to be assessed. This includes information relating to the design basis of the plant, engineered safety features, confinement systems, human system interfaces, control and protection systems, etc. The design information needs to highlight the structures, systems and components (important to safety) that are of evolutionary[20] or innovative[21] design [41] and this could be the focus of the INPRO assessment[22].

In addition to the information on the NFCF to be assessed, the INPRO assessor needs the same type of information on a reference plant design in order to perform a comparison of both designs. Details of the information needed are outlined in the discussion of the INPRO methodology criteria in the following sections.

If not available in the public domain, the necessary design information could be provided by the designer (potential supplier). Therefore, a close co-operation between the INPRO assessor

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[19] An update of the INPRO methodology overview manual is currently in preparation.

[20] An evolutionary design is an advanced design that achieves improvements over existing designs through small to moderate modifications, with a strong emphasis on maintaining proven design elements to minimize technological risks.

[21] An innovative design is an advanced design which incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice.

[22] It would seem reasonable to expect the designer to highlight not only the modifications (changed systems, structures and components) but also the unchanged ones with an explanation of why the decision to keep them or to change was made, i.e. safety or non-safety (economics) reasons.

as a technology user and the designer (potential supplier) is necessary as detailed in the INPRO methodology overview manual.

In addition, all relevant operational and maintenance data and history of the reference facility will be useful as well as any records of modifications, any failures and incidents in the reference NFCF or similar facilities.

3.3. RESULTS OF SAFETY ASSESSMENTS

To assess sustainability, the INPRO assessor will need access to the results of a safety assessment of a reference plant and to the basic design information of the NFCF to be assessed that includes a safety analysis that evaluates and assesses challenges to safety under various operational states, AOO and accident conditions using deterministic and probabilistic methods; this safety assessment is supposed to be performed and documented by the designer (potential supplier) of the NFCF to be assessed.

For an NFCF to be assessed using the INPRO methodology, the safety assessment would need to include details of the RD&D carried out for advanced aspects of the design. Such information is usually found in a (preliminary) safety report (or comparable document) that may be available in public domain or could be provided by the designer (potential supplier) of the NFCF. Thus, as stated before, a close co-operation between the INPRO assessor as a technology user and the designer (potential supplier) is necessary.

3.4. INPRO ASSESSMENT BY A TECHNOLOGY DEVELOPER

In principle, an INPRO assessment can be carried out by a technology developer at any stage of the development of an advanced NFCF design. This assessment can be performed as an internal evaluation and does not require results of the formal safety assessment. However, it needs to be recognized that the extent and level of detail of design and safety assessment information available will increase as the design of an advanced NFCF progresses from the conceptual stage to the development of the detailed design. This will need to be taken into account in drawing conclusions on whether an INPRO methodology sustainability requirement for safety has been met by the advanced design.

One potential mode for the technology developer’s use of the INPRO methodology is in performing a limited scope assessment. Limited scope INPRO assessments can be focused on specific areas and specific nuclear energy system installations having different levels of maturity. A limited scope study may assess the facility design under development and may help highlight gaps to be closed in on-going RD&D studies and define the scope of data potentially needed to make future judgements on system sustainability.

3.5. OTHER SOURCES OF INPUT

The assessor can use the IAEA Fuel Incident Notification and Analysis System (FINAS) and other international and national event reporting systems for specific and general information relevant to the technology type and detailed design of an advanced NFCF.

4. GENERAL FEATURES OF NUCLEAR FUEL CYCLE FACILITIES

This section presents some background information on NFCFs, discusses their main hazards, and compares NFCFs to nuclear reactors and chemical plants.

4.1. NUCLEAR FUEL CYCLE

A nuclear fuel cycle comprises a number of activities other than reactor operation[23], the possible combinations of which provide the various fuel cycle options. These activities[24] are:

- Uranium/ thorium mining and milling[25];
- Uranium/ thorium refining and conversion;
- Uranium enrichment;
- Fuel fabrication;
- Fuel transportation (including spent fuel transportation);
- Spent nuclear fuel storage;
- Spent nuclear fuel reprocessing including recovered/recycled nuclear material storage;
- Re-fabrication of nuclear fuel using fissile (and fertile) material from reprocessing;
- Radioactive waste management[26] including predisposal waste management and disposal.

Depending upon the requirements and preferences of the individual country, either an open (once through) or closed fuel cycle option can be chosen. In an open fuel cycle, the spent fuel is treated as a waste, i.e. it is (after storage) disposed of directly, without reprocessing. In a closed fuel cycle, spent fuel is reprocessed and the fissile and fissionable elements are used to produce new fuel, and the rest of the fuel elements, e.g. fission products, are disposed of. A comprehensive review of the activities related to nuclear fuel cycles is given in Refs [31, 42]. Developing trends on reactor fuels and their technologies are described in Ref [43].

The characteristics of nuclear fuel cycles depend mainly upon the type of nuclear reactors. A few examples illustrate this statement: Pressurized heavy water reactors (PHWRs) use mainly[27] natural uranium as fuel, whereas PWRs and BWRs use low enriched uranium (LEU) as fuel. Up till now, fast reactors used several types of fuel, such as high enriched uranium, mixed U/Pu oxide, uranium carbide or uranium /plutonium carbide, and uranium nitride, and metallic fuels. Metallic fuels are also used in thermal (and could be used in fast) gas cooled reactors. The currently predominant (partly) closed nuclear fuel cycle is based on U/Pu, where Pu is obtained by reprocessing of spent U fuel from thermal reactors, to be used either again in thermal reactors or in fast reactors. The $^{232}Th/^{233}U$ fuel cycle would require for large-scale use of thorium firstly the conversion of ^{232}Th into ^{233}U in (thermal) reactors with U fuel. A combination of fast reactors with U/Pu fuel cycle and thermal reactors with $^{232}Th/^{233}U$ fuel cycle provides one closed fuel cycle option [42].

Typical fuel cycle options currently deployed on an industrial scale are the open fuel cycle in heavy water reactors (HWRs) and light water reactors (LWRs), and mono recycling in LWRs. Within the next decades [44] examples of potential industrial developments of fuel cycles

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[23] Reactor safety issues related to the NES sustainability assessment are considered in a separate INPRO methodology manual ‘Safety of Reactor’.

[24] Every activity may involve one or several NFCFs which pass through seven main stages within the lifetime: siting, design, construction, commissioning, operation, decommissioning and release from regulatory control.

[25] Milling is also called processing.

[26] Radioactive waste management issues related to the NES sustainability assessment are covered in a separate INPRO methodology manual ‘Waste Management’.

[27] Some PHWRs use LEU for improvement of economic characteristics.

include: DUPIC (Direct Use of PWR spent fuel in CANDU reactors), multi recycling of spent fuel in thermal reactors, and application of a partitioning and transmutation concept [45]. Possible long-term industrial developments include: a reactor fleet with a mixture of LWR and fast reactors, a 100 % closed fuel cycle for fast reactors, a thorium fuel cycle [46], and a fuel cycle for molten salt reactors.

4.2. OVERVIEW OF HAZARDS IN NFCFs

4.2.1. Internal and external hazards

In this publication, hazards are considered to be those phenomena or the effects of phenomena that may cause damage to the systems, structures and components relevant to safety of an NFCF, prevent them from performing their functions or alter the processes characteristics beyond the limits defined for normal operation conditions. Generally, hazards are divided into two categories: internal and external. The external hazards can be further split into two sub-categories – natural hazards and human induced hazards.

Internal hazards may involve criticality, fire and explosion, leaks and flooding, radiation and chemical releases, collapse of structures and falling objects, corrosion and erosion, etc., originating from inadequacies in the original design, fabrication or modifications of systems, structures and components or from inadequate processes or procedures, process malfunctions and operational mistakes.

Natural external hazards are associated with meteorological, hydrological, geological and seismic events [47, 48] such as earthquakes, inundation in the flooding, tsunamis, natural fires, and extreme weather conditions, e.g. excessive snowfall, avalanches, tornadoes/ storms/ cyclones, lightning and extreme high or low temperatures. Precaution against these is required at the selection of the site as well as in the design and construction of civil works. The plant and machinery are expected also to be protected per recommended practices and as required by statutory and risk-managing organizations.

Human induced external hazards are associated with the damages/risks caused by man-made factors like potential loss of power supply and consequently loss of control, fire/explosion in the constituent units of the facility or units adjacent to the facility, flying missiles/debris from the neighbourhood/sky/space, accidental or wilful (terrorist) aircraft crash and civil strife including violent strikes, blockages and sabotage[28].

The risks associated with internal and external hazards are expected to be below acceptable levels. Thus, in the design analyses all potential sources of external hazards need to be identified and the associated event sequences affecting the facility determined. The associated radiological or chemical consequences of any damage caused by an event associated with external hazards need to be evaluated and compared to the acceptance criteria. In the following a few examples of external hazards are presented.

An NFCF is expected to be designed for *earthquakes* per appropriate standards to ensure that they would not induce a loss of confinement capability (especially of radioactive and/or toxic material such as UF_6 and HF) or a criticality accident by an induced loss of criticality safety functions, such as geometry and moderation, with possible significant consequences for site personnel or members of the public.

Hazards from external *fires and explosions* need to be covered in the design of an NFCF. They can arise from various sources in the vicinity of the NFCF, such as petrochemical installations,

[28] Civil strife including violent strikes, blockages and sabotage

forests, pipelines and road, rail or sea routes used for the transport of flammable material such as gas or oil.

Flooding of an NFCF is expected to be taken into account in its design, if flooding is a credible hazard. For example, the building of the NFCF needs to be robust enough to withstand the impact of a water wave, keeping its integrity as a confinement. In case the material with enrichment higher than 1 % is handled in the facility, the criticality accidents caused by flooding need to be excluded [32].

An NFCF needs to be protected against extreme weather conditions by including in the design:

- Sufficient strength of structures important to safety to withstand the loads (e.g. mechanical, thermal) caused by these conditions;
- Measures to prevent flooding of the facility, and
- If needed, measures to enable safe shutdown of the facility.

It is necessary to emphasize that the safety requirements adopted for a particular NFCF normally take into account the hazard potential and thus result in a graded approach ensuring that the design and operating philosophies are commensurate with the hazards [17]. The most significant general hazards in NFCFs are briefly discussed in the following sections (see also regulatory guides [18-22]).

4.2.2. Criticality hazard

Criticality safety is one of the dominant safety issues for NFCFs that handle uranium enriched above 1 % of ^{235}U, or other fissile material such as ^{233}U or plutonium (Pu). As stated before, these facilities employ a great diversity of technologies and processes. Thus, the materials of interest to nuclear safety are distributed throughout the facilities. The nuclear material may be used not only in bulk form (e.g. fuel pellets, fuel elements, fuel rods, fuel assemblies, etc.), but in a distributed and mobile form as well (e.g. in different kinds of solutions, slurries, gases, powders, etc.). As a result, fissile elements may accumulate in some parts of the equipment and may also spill as a result of equipment leakage. The distribution and transfer of potentially critical nuclear materials requires monitoring, alarm systems and operator attention to account for this material throughout the installation and to thus ensure that sub-criticality is maintained and thereby preventing the potentially lethal effects of gamma and neutron radiation doses to workers and the subsequent release of fission products from an inadvertent nuclear criticality.

Nuclear criticality safety is achieved by controlling one or more of the following parameters of the system within sub-critical limits during anticipated operational occurrences (AOO) (e.g. vessel overfilling) and design basis accident conditions:

- Mass and enrichment of fissile material present in a process (e.g. powder in rooms and vessels);
- Geometry of processing equipment (e.g. by safe diameter of storage vessels, separation distances in storage);
- Concentration of fissile material in solutions (e.g. in wet processes for recycling uranium);
- Presence of appropriate neutron absorbers (e.g. in the construction of storage facilities, drums for powder, fuel shipment containers);
- Moderation limitation (e.g. by control of moisture and amount of additives in powder);
- Control of neutron reflectors.

The general procedures to be followed in the criticality analysis are:

- The use of a conservative approach (considering uncertainties on physical parameters, physically possible conditions of optimum moderation, etc.);

- The use of appropriate and qualified (verified and validated) computer codes and cross section libraries within their qualified range.

NFCFs may be split into two groups with regard to criticality:

(1) Facilities where a criticality hazard is not credible — mining, milling and conversion of natural uranium facilities, and natural uranium fuel fabrication/storage/transportation.
(2) Those where criticality hazards may be credible — enrichment, reprocessing, uranium fuel fabrication, mixed oxide fuel fabrication, fresh fuel storage (and transportation), spent fuel storage (and transportation), waste treatment and waste disposal facilities.

Those facilities in group (2) need to be designed and operated in a manner that provides a high level of assurance that criticality parameters are controlled. Firstly, designs of such facilities need to ensure sub-criticality in all areas utilizing where possible 'criticality safe' designed equipment. Secondly, during operation of these facilities, measurement of criticality parameters has to be continuously maintained via monitoring, detection and alarm systems.

A review of some criticality accidents that occurred during operation of NFCFs is provided in Ref [49]. The criticality accident at Tokai Mura, Japan, was the highest level event in the International Nuclear Event Scale reported since 1991. Of the nearly 60 reported criticality accidents that have occurred since 1945, about a third occurred at NFCFs. Two of these occurred in 1997 and 1999. Twenty of these accidents involved processing liquid solutions of fissile materials, while none involved failure of safety equipment or faulty (design) calculations. The main cause of criticality accidents appears to be the failure to identify the range of possible accident scenarios during the design, particularly those involving potential human (operator) errors.

4.2.3. Chemical hazards

NFCFs may also pose hazards to workers (and the public) from releases of chemically toxic and corrosive materials during any of the chemical processing steps in a nuclear fuel cycle.

Chemical hazards differ considerably from facility to facility. The production of uranium hexafluoride (UF_6) (in a conversion facility) involves the use of significant amounts of hydrogen fluoride (HF), which is both a powerful reducing agent and chemo-toxic and thus poses a significant hazard to workers.

Other examples include the use of strong chemical acids to dissolve nuclear fuel and other materials. These acids are used to chemically dissolve spent nuclear fuel during reprocessing (also to recycle scrap pellets in fuel fabrication facilities), thereby removing the fuel cladding material and enabling separation of the plutonium and uranium from the residual fission products. In addition, residual fission products, which comprise approximately 99 % of total radioactivity and toxicity in spent nuclear fuel, pose a significant radiological hazard in what is typically a complex chemical slurry. During solvent extraction processes, strong acids and organic solvents are used to remove plutonium and uranium from these slurries. These processes can generate toxic chemical by-products that need to be sampled, monitored and controlled.

Other chemicals encountered at NFCFs in significant amounts include chemicals such as ammonia, nitric acid, sulphuric acid, phosphoric acid and hydrazine. It is important to recognize that unplanned releases of these toxic chemicals may adversely affect safety controls. For example, a release of hydrogen fluoride could disable an operator who may be relied upon to ensure safe processing.

Chemical hazards have caused operational problems and accidents at many NFCFs worldwide. The chemical toxicity hazards associated with UF_6 processing were evident in two incidents in 1986 in the USA and Germany [14] and in 2010 in Canada [50].

4.2.4. Fire and explosion hazards

Many NFCFs use flammable, combustible and explosive materials in their process operations, such as a tri-butyl phosphate-dodecane mixture for solvent extraction, bitumen for conditioning radioactive wastes, hydrogen in calcining furnaces and chemical reactors for oxide reduction. Some flammable and explosive substances may also be generated as bypass products in the production process or as a result of faulty operation when unexpected chemical reactions take place.

Many fire and explosion hazards have been recorded at NFCFs. In 1990, for example, there was an ammonium-nitrate reaction in an off-gas scrubber at a LEU scrap recovery plant in Germany, which injured two workers and destroyed the scrubber [14]. Fire is an especially significant player in accident scenarios because it can be both an initiating event for the accident sequence and can also disable or damage passive and active safety features. It can also provide an energy source to transport radiological and chemical contaminants into uncontrolled areas where they may pose risks to both workers and members of the public. An example of this situation is the fire and explosion at the Tokai Mura reprocessing plant in Japan in March 1997, which contaminated 37 workers with radioactive material.

Therefore, the design of NFCFs is expected to provide for minimum inventories of combustible materials and needs to ensure adequate control of thermal processes and ignition sources to prevent fire and explosions, or at least reduce their potential. For example, extreme care needs to be taken to prevent accumulation of radiolytic hydrogen, which is generated in high activity waste tanks in spent nuclear fuel reprocessing plants.

Fire can also lead to significant releases of radioactive and toxic material. Consequently, fire detection (alarm), suppression, and mitigation controls are usually required. The NFCF design and operation need to consider radiological and other consequences from fires and explosions. Suitable safety controls are supposed to be instituted to protect against potential consequences of fire and explosive hazards. These safety controls need to be designed to provide requisite protection during normal operations, anticipated operational occurrences and credible accidents at a facility.

Summarizing the statements above, it is noted that similar to chemical hazards, fires and explosions that could adversely affect nuclear safety measures need to be given adequate consideration in the design and operation of NFCFs.

4.2.5. Radiation hazards

Radiological safety is an important consideration at NFCFs. Special attention is warranted, when developing and using standards and establishing operational practices to ensure worker safety in operational processes, which may include the open handling and transfer of nuclear materials in routine processing. Although external exposures to radiation fields may be limited, potential intakes of radioactive material require careful control to prevent and minimize internal and external contamination and to adhere to operational dose limits. In addition, releases of radioactive material inside facilities through unmonitored pathways could result in significant exposures to workers, particularly from long lived radiotoxic isotopes. Some facilities, such as MOX fuel fabrication and reprocessing facilities require special shielding design, containment, ventilation and maintenance measures to reduce potential exposures to workers.

Fundamental principles whose effective application will ensure appropriate protection and safety in any situation that involves or might involve exposure to radiation are defined in Ref [51]. Based on these principles and objectives, requirements with respect to radiological safety for all types of nuclear installations are established in the International Basic Safety Standards for Radiation Protection and for the Safety of Radiation Sources [52].

4.2.6. Initiating events

Ref [53] defines an initiating event[29] as "an identified event that leads to anticipated operational occurrences or accident conditions".

The list of internal and external hazards (see Section 4.2.1) is normally used to select initiating events for detailed safety analysis. Postulated initiating events need to be identified on the basis of expert judgement, feedback from operating experience and deterministic assessment, complemented by probabilistic methods where appropriate. The resulting set of identified postulated initiating events has to be confirmed as comprehensive. [17]

Typical initiating events for various NFCFs are discussed in Refs [16, 32–34].

4.2.7. Decommissioning of NFCFs

The safety aspects of decommissioning of NFCFs deserve as much attention as the safety aspects of operation of these facilities. The decommissioning of an NFCF has to be factored into the design of the facility and a clear plan for decommissioning needs to be available even at the time of commissioning of the facility. The safety aspects of decommissioning have been dealt with by the IAEA in Refs [54, 55]. The safety aspects related to releasing NFCFs from regulatory control upon termination of practices are discussed in Ref [56].

In all phases of decommissioning, workers, the public and the environment have to be properly protected from both radiological and non-radiological hazards resulting from the decommissioning activities. Safety issues that are expected to be considered in the decommissioning of NFCFs include [54]:

- "The presence and nature of all types of contamination;
- Hazards associated with the possible in-growth of radionuclides (such as ^{241}Am);
- The potential for criticality hazards associated with the possible accumulation of fissile material in the process equipment during operation or during decommissioning actions (such as decontamination);
- The complexity of strategies for waste management owing to the diversity of waste streams;
- For multifacility sites, hazards associated with facilities that are not under decommissioning;
- Inaccessible areas and buried pipes;
- Separation and concentration of material stored in tanks;
- Hazardous chemicals located in SSCs and in buildings, soil, sediment, surface water and groundwater;
- Changes in chemical and physical forms of materials;
- Non-radiological hazards, such as fire or explosion, associated with decommissioning actions."

The specific characteristics of each type of NFCF will strongly influence the selection of the decommissioning option. A safety assessment is expected to form an integral part of the decommissioning plan. Non-radiological as well as radiological hazards associated with the decommissioning activities need to be identified and evaluated in the safety assessment and

[29] For consideration of hypothetical events considered at the design stage, the term postulated initiating event is used.

factored into the design of the facility. The extent and detail of the safety assessment shall be commensurate with the complexity and the hazards associated with the facility and its operation.

5. RATIONALE OF THE INPRO BASIC PRINCIPLE AND USER REQUIREMENTS FOR SUSTAINABILITY ASSESSMENT IN THE AREA OF SAFETY OF NUCLEAR FUEL CYCLE FACILITIES

This section presents some background on the INPRO basic principle (BP) and user requirements (UR) for sustainability assessment in the area of NCFC safety. It is noted that the INPRO methodology in this area was originally developed with a nuclear power plant in mind and had to be adapted, especially at the criterion level, to the individual NFCF[30].

5.1. INTRODUCTION TO THE DEFINITION OF INPRO BASIC PRINCIPLES

In the INPRO methodology area of NFCF safety, one basic principle (BP), also called INPRO basic principle in the following, a set of user requirements (UR), and a set of criteria (CR) for NES sustainability assessment have been defined, the focus of which is to reflect the expected changes in nuclear technology in the future. As will be shown in the discussion of the INPRO basic principle and user requirements, the methodology encourages the introduction of innovations that enhance the safety of NFCFs.

One of the basic assumptions of the INPRO methodology is the expectation that – to fulfil the needs of sustainable energy supply in the twenty-first century – the global number of nuclear reactors and NFCFs in operation will have to increase considerably compared to the situation today. Keeping the safety level of newly deployed reactors and NFCFs (deployed after 2013) at the same level as the global operating systems today would lead to an overall increase in the numerical risk of nuclear accidents. It is expected, however, that this increase in calculated risk would be compensated by the increased safety level of the newly deployed reactors and NFCFs, based in part on lessons learned from systems in operation. Therefore, the INPRO methodology evaluates enhancements in the safety of new designs but does not evaluate compliance with national or international (e.g. IAEA) safety standards. The reference design is assumed to comply with current regulatory standards[31]. Similarly, a new facility is assumed to be designed so that it complies with applicable safety standards. Confirmation of compliance of the reference or new design with national or international safety standards is outside the scope of the INPRO methodology area of safety of nuclear fuel cycle. When such confirmation is needed, a separate peer review can be performed.

The INPRO methodology's basic principle and its set of user requirements and criteria are expected to be applicable to any type of advanced design and to foster an appropriate level of safety that can be communicated to and be accepted by all stakeholders in nuclear energy.

The legal and organizational framework related to safety of nuclear reactors and NFCFs is dealt with in another report of the updated INPRO methodology focused on infrastructure [2].

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[30]This adaptation of the INPRO methodology at the criterion level is laid out in Sections 6 to 10 for each individual type of NFCF.

[31]Current standards are those prevailing at the time of the INPRO assessment and are normally met by recently licensed facilities.

5.2. INPRO BASIC PRINCIPLE FOR SUSTAINABILITY ASSESSMENT IN THE AREA OF SAFETY OF NUCLEAR FUEL CYCLE FACILITIES

INPRO basic principle for sustainability assessment in the area of NFCF safety: The planned NFCF is safer than the reference NFCF. In the event of an accident, off-site releases of radionuclides and/or toxic chemicals are prevented or mitigated so that there will be no need for public evacuation[32].

The main goal of the INPRO basic principle is to encourage the designer/developer to increase the safety level of a new facility to be installed after 2013. To achieve this goal, the INPRO methodology proposes that NFCF designers/ developers undertake the following key measures:

- Incorporate enhanced defence in depth into an advanced NFCF design as a part of the fundamental safety approach.
- Incorporate, when appropriate, inherently safe characteristics and passive systems into advanced NFCF designs as a part of a fundamental safety approach to excel in safety and reliability.
- Reduce the risk from radiation exposures to workers, the public and the environment[33] during construction/ commissioning, operation, and decommissioning of an advanced NFCF.
- Perform sufficient RD&D work to bring the knowledge of NFCF characteristics and the capability of analytical methods used for design and safety assessment of a plant with innovative features to at least the same confidence level as for a reference plant.
- Take human factors into account in the design and operation of an NFCF and establish and maintain a safety culture in all organizations involved in a nuclear power program.

The INPRO methodology has developed seven user requirements to specify in more detail the main measures presented above. These user requirements are to be fulfilled primarily by the designer (developer, supplier) of the NES but also in some cases by the operator. As stated before, the role of the INPRO assessor is to check, based on evidence provided by the designer and operator, whether they have implemented the necessary measures as required by the INPRO methodology.

The following sections provide rationale and background information for each user requirement (UR).

5.3. USER REQUIREMENT UR1: ROBUSTNESS OF DESIGN DURING NORMAL OPERATION

INPRO user requirement UR1 for sustainability assessment in the area of NFCF safety: The assessed NFCF is more robust than the reference design with regard to operation and systems, structures and components failures.

The first INPRO user requirement, UR1, for sustainability assessment in the area of NFCF safety is mostly related to the first level of DID, which is focused on preventing AOOs, i.e. deviations from normal operation and failures of items important to safety. AOOs are defined as those conditions of operation that are caused by events associated with internal or external hazards (see Section 4.2.1 and 4.2.6) expected to occur one or more times during the lifetime

[32] Other protective measures still may be needed. Effective emergency planning, preparedness and response capabilities will remain a prudent requirement as discussed in the INPRO methodology area of Infrastructure.

[33] Radiation and chemical exposure to the public and environment is covered in the INPRO methodology manual on environmental impact of stressors.

of an NFCF but that do not cause any significant damage to items important to safety nor lead to accident conditions requiring safety features (Level 3 of DID) to control.

In principle, the design (e.g. mechanical, thermal, electrical, etc.) of normal operating systems in any NFCF can be made more robust, i.e. reducing the likelihood of failures, by increasing design margins, improving the quality of manufacture and construction, and by using materials of higher quality. Sufficient margin in the design needs to be provided so that any small deviation (e.g. resulting from failure) of system parameters from normal operation will not lead to an accident.

It is acknowledged that increasing the robustness of an NFCF design is a challenging task that requires optimisation wherever enhancing one aspect can have a negative influence on other aspects in other areas (e.g. in economics, making the system uncompetitive, or in proliferation resistance). Thus, an optimum combination of design measures is necessary for increasing the overall robustness of a design.

It is important to note that for the assessment of all criteria of user requirement UR1 the INPRO assessor (a technology user) needs information on the facility to be assessed and on a reference facility. The assessed NFCF is expected to demonstrate a safety level superior to that of the reference facility. If a reference facility design is not available to the assessor, it needs to be demonstrated that the assessed facility incorporates the most recent technology and that international best practice has been used, i.e. that the facility is state of the art.

For an operating NFCF, the requirements for design, manufacturing, and operation (and decommissioning) are usually specified in (extensive) national standards or in adopted standards from other countries; the most widely known and used standards are the Nuclear Codes and Standards published by the American Society of Mechanical Engineers.

The major means to achieve an increase in robustness in an NFCF are to ensure a high quality of design, construction and operation, including human performance. For new (innovative or evolutionary) NFCF designs, the expected frequencies of AOOs are expected to be reduced relative to a reference design. This reduction could be achieved by such means as using improved materials, simplified designs to minimize failures and errors, improved design margins (mechanical, thermal, electrical, etc.), increased operating margins, increased redundancies of systems, lessened impacts from incorrect human intervention (the system needs to be tolerant of mistakes), more effective and efficient inspections, continuous monitoring of the plant health, etc. Examples of concepts with increased robustness against certain potential hazards are designs that use passive systems deemed potentially more reliable than active systems (e.g. natural convection cooling), higher reliability self-checking control systems (avoidance of deviations from normal operation), use of non-flammable materials[34] (avoidance of fires), etc. The use of inherent safety characteristics is a useful means of achieving robustness and has been highlighted as a separate user requirement, UR5 (Section 5.6).

For an NFCF under assessment, measures and features are to be developed that ensure that the robustness of the innovative design against internal and external hazards [39] will be comparable or superior to that of the reference design.

For (innovative) designs of NFCFs still under development and for which no standards may yet exist, at least for the first plant to be installed, a conservative design approach according to existing standards can be proposed as discussed for user requirement UR7 in Section 5.8.

User requirement UR1 considers occupational doses corresponding to Levels 1 and 2 of DID, i.e. at normal operation and for anticipated operational occurrences. It is important to note that

[34] The use of non-flammable materials is an example of an inherent safety characteristic.

UR1 does not consider radiation exposure of workers during accidents. Radiation exposure of workers, public and the environment during/after accidents is dealt with in user requirements UR3 and UR4. A similar approach is supposed to be established for limiting chemical doses to workers.

The need to avoid undue burdens from radiation and/or toxic chemical exposure of the public and the environment during normal operation and AOOs (in an NFCF or nuclear reactor) is covered in a separate area of the INPRO methodology focused on the environmental impacts of stressors [5].

In this context, it bears noting that the International Basic Safety Standards for Radiation Protection and for Safety of Radiation Sources in Ref [52] define acceptable levels of radiation exposure for workers and the public for planned and emergency (accident) exposure situations Additional detailed guidance on occupational radiation protection in NFCFs is provided in Ref [25]. Comparable (mostly national) standards exist for toxic chemicals [57].

5.4. USER REQUIREMENT UR2: DETECTION AND INTERCEPTION OF AOOs

INPRO user requirement UR2 for sustainability assessment in the area of NFCF safety: The assessed NFCF has improved capabilities to detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions.

The second user requirement, UR2, for sustainability assessment in the area of NFCF safety involves the limited consideration of selected provisions in the first DID level and mostly relates to the second level of DID, which deals with detection and control of failures and deviations from normal operational states in order to prevent AOOs from escalating to accident conditions. The objective is met if the plant returns to normal operation either automatically or through operator action after an AOO or component failure and a progression to higher levels of DID is avoided.

In the design of a new NFCF (to be installed after 2013), priority is expected to be given to advanced instrumentation and control (I&C) systems and improved reliability of these systems. The facility needs to be designed to give the operator a sufficient grace period after an AOO or failure. In the longer term, priority can be given to design-specific inherent safety features and to robust and simple (possibly passive) control as well as advanced monitoring and alarm systems.

The main function of the I&C system in this level of DID is to detect deviations from normal operation and failures, produce an alarm, and together with operator actions prescribed in detailed operating procedures, enable rapid return of the facility to normal operating conditions with, ideally, no consequences, e.g. no need for follow up inspections or regulatory event reports.

I&C systems process measurement data from several different kinds of instrumentation. Examples of I&C systems include: conventional process instrumentation, vessel fluid level measurement instrumentation, radiation monitoring and alarm instrumentation, accident instrumentation, and hydrogen detection and measurement instrumentation. These instrumentation sets contain channels of different importance to safety.

5.5. USER REQUIREMENT UR3: DESIGN BASIS ACCIDENTS

INPRO user requirement UR3 for sustainability assessment in the area of NFCF safety: The frequency of occurrence of DBAs in the assessed NFCF is reduced. If an accident occurs, engineered safety features and/or operator actions are able to restore the assessed NFCF to a controlled state, and subsequently to a safe state, and the consequences are mitigated to ensure the confinement of radioactive and/or toxic chemical material. Reliance on human intervention is minimal, and only required after sufficient grace period.

The third user requirement, UR3, for sustainability assessment in the area of NFCF safety is mostly related to the third level of DID, which concentrates on the control of accidents to prevent releases of radioactive materials and associated hazardous materials or radiation levels that would require off-site protective actions. The objective is met if the accident consequences are limited to within the design basis. The 'design bases' of a facility are the conditions and events taken into account in the NFCF design such that the facility can withstand them by the intended operation of engineered safety features, inherent safety features and prescribed operator interventions without exceeding authorized limits. Thus, a DBA is an accident causing conditions [53] for which a facility is designed, in accordance with established design criteria and conservative methodology, and for which releases of radioactive and/or chemically toxic materials are kept within authorized limits. Authorized limits of radiation exposure after accidents in nuclear facilities are expected to comply with the IAEA Safety Standards [25, 52]. Examples of limits for chemical exposure can be found in Ref [57].

A grace period needs to be available before human (operator) intervention is necessary to prevent the escalation of a DBA into an accident with large releases of radioactivity and/or toxic chemicals to the environment. This grace period depends upon the nature of the NFCF, the type of incident, and the system parameters at the time of the incident, etc. However, based on available international experience, a grace period of 10 to 30 minutes is given as the typical decision interval for the operator in the event of a DBA in an NPP [40]. A similar approach could be adapted for NFCFs other than mining and milling activities.

The term 'controlled state' is characterized by a situation in which either the facility's engineered safety features or its prescribed operator interventions are able to compensate for the loss of functionality resulting from a DBA. The term 'frequency of occurrence' as used in user requirement UR3 refers to the number of events per NFCF year that lead to a DBA as determined via probabilistic methods (PSA). In the context of DBAs (caused by postulated initiating events associated with internal or/and external hazards), the term 'grace period' refers to the time period during which no operator inventions are needed and solely the actions of automatic active (and/or passive) safety features will suffice to keep the analysed DBA from escalating to a severe accident with potentially large releases to the environment.

Passive safety features can provide additional safety gains. Safety features consisting solely of passive components are very often deemed more reliable than active safety features due to missing (or a reduced number of) active components. In addition, no (or very limited) human actions are needed and, thus, the likelihood of human errors is very low. Nevertheless, failures in passive safety features due to human error in design or maintenance, the presence of unexpected phenomena, and potential adverse system interactions, are expected to be analyzed and may need to be compensated by other design measures. It is acknowledged that some kinds of passive safety features can be difficult to design in NFCFs.

Ensuring the confinement of radioactive and/or chemically toxic materials means that the design of engineered safety features and/or operator actions (procedures) for mitigating the consequences of a DBA need to provide deterministically for the continued integrity of at least one barrier to the unacceptable release of radioactive and/or chemically toxic materials following any DBA.

5.6. USER REQUIREMENT UR4: SEVERE PLANT CONDITIONS

INPRO user requirement UR4 for sustainability assessment in the area of NFCF safety: The frequency of an accidental release of radioactivity into the environment is reduced. The source term of accidental release into the environment remains well within the envelope of the reference facility source term and is so low that calculated consequences would not require public evacuation.[35]

The fourth user requirement UR4 for sustainability assessment in the area of NFCF safety is focused on accident conditions more severe than those in DBAs. It is mainly related to the design extension conditions and to the fourth level of DID, which has the objective to mitigate the consequences of accidents that result from failure of the third level and ensure that radioactive releases are kept as low as reasonably achievable. A severe (nuclear fuel cycle) accident is any event affecting the facility that results in off-site radiological consequences equal to or greater than the high contamination level or radiation level criteria for design extension conditions,[36] i.e. an event more severe than a DBA.

An accidental release of radioactivity could occur if the magnitude of an initiating event (associated with external hazards) exceeds the design basis or additional failures of safety systems and/or operator interventions occur after an initiating event (associated with internal or / and external hazards) that lead to the design extension conditions with severe damage to equipment containing radioactive and / or chemically toxic materials. Consequence mitigation calls for keeping those radioactivity and / or toxic chemicals that are released from internal barriers damaged during an accident inside the NFCF containment/ confinement structure to the extent possible by avoiding any cliff-edge effects that could damage the remaining barrier(s) to external release.

Ref [58] identifies generic criteria for protective actions and other response actions in a nuclear or radiological emergency to reduce the risk of stochastic effects. Projected dose limits indicated as criteria for public evacuation can be used in the INPRO assessment when corresponding national criteria have not been established yet.

For new NFCFs, the capability and reliability of natural and/or engineered processes for controlling complex accident sequences with severe damage is expected to be increased, as well as the capability and reliability of associated instrumentation, control and diagnostic systems. Appropriate severe accident management procedures also need to be developed. Through these measures, the frequency of accidental releases of radioactive and chemically toxic materials can be reduced and the inventory and conditions of release are expected to be constrained to avoid any need to evacuate the population.

When the frequency of accidental releases cannot be calculated with a high level of confidence, the new NFCF design needs to demonstrate deterministically that the probability of an accidental release of radioactivity and/or toxic chemicals into the environment is lower than that for the reference facility, e.g. through improved engineered safety features, prescribed operator actions, and the use of additional inherent safety characteristics or further measures to minimize hazards, and that the consequences (doses, concentrations of toxic chemicals) from an accident would not require public evacuation except as a short term precautionary measure.

It is nevertheless acknowledged that also for new (and advanced) NFCFs, it will still be necessary to establish an emergency preparedness regime [52, 58, 59] regardless of the safety level of the new NFCF (as discussed in another area of the INPRO methodology focused on

[35] However, emergency preparedness and response still remain necessary as discussed in the infrastructure area of the INPRO methodology.

[36] IAEA Safety Standard SSR-2/1 (Rev.1) defines design extension conditions as postulated accident conditions that are not considered for DBAs, but that are considered in the design process for the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits.

infrastructure [2]) in order to meet the objective of the fifth level of the DID concept and the corresponding legal and regulatory requirements.

5.7. USER REQUIREMENT UR5: INDEPENDENCE OF DID LEVELS AND INHERENT SAFETY CHARACTERISTICS

INPRO user requirement UR5 for sustainability assessment in the area of NFCF safety: An assessment is performed to demonstrate that the DID levels are more independent from each other than in the reference design. To excel in safety and reliability, the assessed NFCF strives for better elimination or minimization of hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics.

As discussed in Section 2.3, the different levels of DID focus on facility conditions ranging from operational to accident states. The DID levels are arranged with increasing severity from operational states (Level 1) to the control of severe plant conditions, including the prevention of accident progression and the mitigation of severe accident consequences (Level 4). As stated in Ref [39], the general goal of DID is to ensure that even a combination of equipment or human (operator) failures at one level of defence "would not propagate to jeopardize defence in depth at subsequent levels". Thus, independence of the safety features designed to cope with processes in the different levels of defence is key to meeting this goal.

To confirm sufficient independence of the DID levels in the assessed NFCF design, a safety assessment is supposed to be performed by the designer (potential supplier) using a suitable combination of deterministic and probabilistic approaches, or hazards analysis.

INPRO user requirement UR5 covers also the role of inherent safety characteristics in new NFCF designs (to be installed after 2013). An inherent safety characteristic is defined in Ref [60] as a fundamental property of a design concept that results from the basic choices in the materials used or in other aspects of the design that assure that a particular potential hazard cannot become a safety concern in any way. The term inherent safety is normally used with respect to a particular characteristic, not to the plant as a whole; e.g. an area is inherently safe against internal fire if it contains no combustible material. An increased use of inherent safety characteristics in the design will strengthen accident prevention in advanced NFCFs by reducing hazards.

The design of a new NFCF is expected to be such that hazards are eliminated (if possible) or minimized, e.g. avoiding explosions by eliminating or minimizing the use of explosive gases. If hazards cannot be eliminated, appropriate equipment needs to be installed to prevent potential damage and to protect the installation, its personnel, the public and the environment. In addition, administrative measures need to be implemented to avoid operator errors to the extent possible.

The analysis of an inherent safety characteristic is difficult but can be possible with the application of adequate mathematical models and, in some cases, by experimental investigations. The analysis of hazards and their consequences needs to be performed using deterministic and probabilistic approaches. For the deterministic approach, engineering judgment, operating experience, validation of design tools and continuous exchanges of information with other industries are mandatory. For probabilistic approaches, the methods likewise need to be validated and the data used needs to be reliable. Analyses are expected to cover all operating states, including normal operation, shutdowns, and maintenance and repair intervals.

There are also external hazards (see Section 4.2.1) associated with the site of an NFCF. Examples of external hazards related to siting include earthquakes, flooding, storms, airplane crashes, and fires and explosions outside the plant. By selecting an appropriate site for an NFCF, these hazards can be minimized.

The necessary RD&D effort to achieve sufficient confidence in advanced designs with increased inherent safety characteristics is discussed in user requirement UR7 (Section 5.8).

5.8. USER REQUIREMENT UR6: HUMAN FACTORS RELATED TO SAFETY

INPRO user requirement UR6 for sustainability assessment in the area of NFCF safety: Safe operation of the assessed NFCF is supported by accounting for HF requirements in the design and operation of the facility, and by establishing and maintaining a strong safety culture in all organizations involved in the life cycle of the facility.

There are two aspects of safety covered in this user requirement. The first aspect focuses on the design of safety related equipment to minimize effects from human errors. The second aspect covers the attitude to safety of workers in the nuclear facilities and related organizations.

The importance of human factors to the safe and reliable operation of nuclear facilities is globally recognized and is an issue that needs to be dealt with systematically in an NFCF design. The designer of a new NFCF is expected to place increased emphasis on human factors to minimize the possibilities for human (e.g. operator or maintenance worker) error. Any relevant experience available from operating NFCFs and the best practices from other industries such as aircraft and chemical plants need to be considered for this process.

There are two perspectives on human factors. On the one side, the operating staff members are seen as valuable resources who play important roles in facility operation, inspection, testing and maintenance, and who sometimes compensate for deficiencies in automatic systems. On the other side, human intervention can also be seen as having limited reliability and a potential to cause disturbances whose consequences need to be taken into account in the design of all facility systems and functions in order to ensure sufficient levels of safety and availability of the facility.

The INPRO task group on safety has summarized the possible negative contributions to accident hazards from human actions into three groups:

- Human errors during plant operation, testing or maintenance that contribute to the failure or unavailability of systems;
- Human errors during plant operation, testing or maintenance that give rise to an initiating event; and
- Human interventions during incident or accident situations that negatively influence the sequence of events.

As a common design principle, it needs to be ensured that:

- Functions assigned to the operating staff constitute consistent tasks that align with the abilities and strengths of the operating staff (e.g. appropriate degrees of automation, appropriate numbers of tasks, appropriate sharing among centralized and local operating actions);
- The man-machine interface (e.g. control room, screen-based and conventional control means, processing of information to be presented to the operators) optimally supports the tasks of operators and minimizes the potential for human errors.

It is expected that the ability to predict human response to both normal and abnormal situations will improve significantly over the next decades and will have a major impact on facility design and operation. Simulator technologies are constantly improving and can thus allow more realistic representations (and progression predictions) of transient and accident plant states in expert systems.

A human factors engineering (HFE) program plan needs to be an essential part of the NFCF design process that helps to integrate the operating staff and facility systems and to minimise the frequency of potential human errors. Ref [61] has defined HFE as follows:

> "The application of knowledge about human capabilities and limitations to designing the plant, its systems, and equipment. HFE affords reasonable assurance that the design of the plant, systems, equipment, human tasks, and the work environment are compatible with the sensory, perceptual, cognitive, and physical attributes of the personnel who operate, maintain, and support the plant or other facility".

Listed below are examples of some design and operational features and assessments that are largely already implemented in existing NFCFs but can be subjected to further improvements in new NFCFs:

- Feedback from experience including a formal methodology;
- A probabilistic safety assessment (PSA) taking human error into account;
- Use of adequate (and quantitative) models that consider the causes of human error and, as such, may help the designer find appropriate measures to avoid the causes of human errors and thus minimize their occurrence;
- The existence of a separate main control room;
- Visualization of the status of facility equipment (components, systems, etc.), the dynamics of processes, the performance of automated processes and their relation with the state of the facility in a manner that helps guide operator actions;
- Monitoring by knowledge-based (expert) systems;
- Appropriate ambient conditions in safety relevant rooms (e.g. main control room);
- Appropriate plant operating procedures (e.g. alarm sheets, procedures for normal operations, incidents and accident situations);
- Formal verification of adequate design implementation;
- Management of human reliability (e.g. personnel selection, periodic training, etc.).

The term 'safety culture' was introduced in 1986 by the International Safety Advisory Group in a summary report of the post-accident review meeting on the Chernobyl accident [62] and was further elaborated in Refs [40, 63]. Ref [63] defined safety culture in the following way[37]:

> "Safety culture is the assembly of characteristics and attitudes in organizations and individuals, which establish that, as an overriding priority, protection and safety issues receive the attention warranted by their significance".

This definition emphasizes that safety culture relates to the structure and style of organizations (governmental institutions, owner/operator, and industrial entities) as well as to the habits and attitudes of individuals (managers and employees). Safety culture demands a commitment to safety on three levels: policy, management and individual [64–71]. The policy level requires a clear statement of safety policy, adequate management structures and related resources, and the establishment of self-regulation (by regular review). To fulfil their commitments, managers need to define clearly the responsibilities, accountabilities and safety practices for the control of work, ensure that staff are qualified and trained, establish a system of rewards and sanctions, and perform audits, reviews and benchmarking comparisons. In carrying out their tasks, individuals need to maintain an attentive and questioning attitude, adopt a rigorous and prudent approach, and participate in effective communications (see Fig. 2 taken from Ref [64]).

The importance of the management system for safety culture in nuclear facilities has been described in Ref [64], which defines this system as "those arrangements made by the

[37] A similar definition is given by the United Kingdom Advisory Committee on the Safety of Nuclear Installations.

organization for the management of safety in order to promote a strong safety culture and achieve good safety performance”.

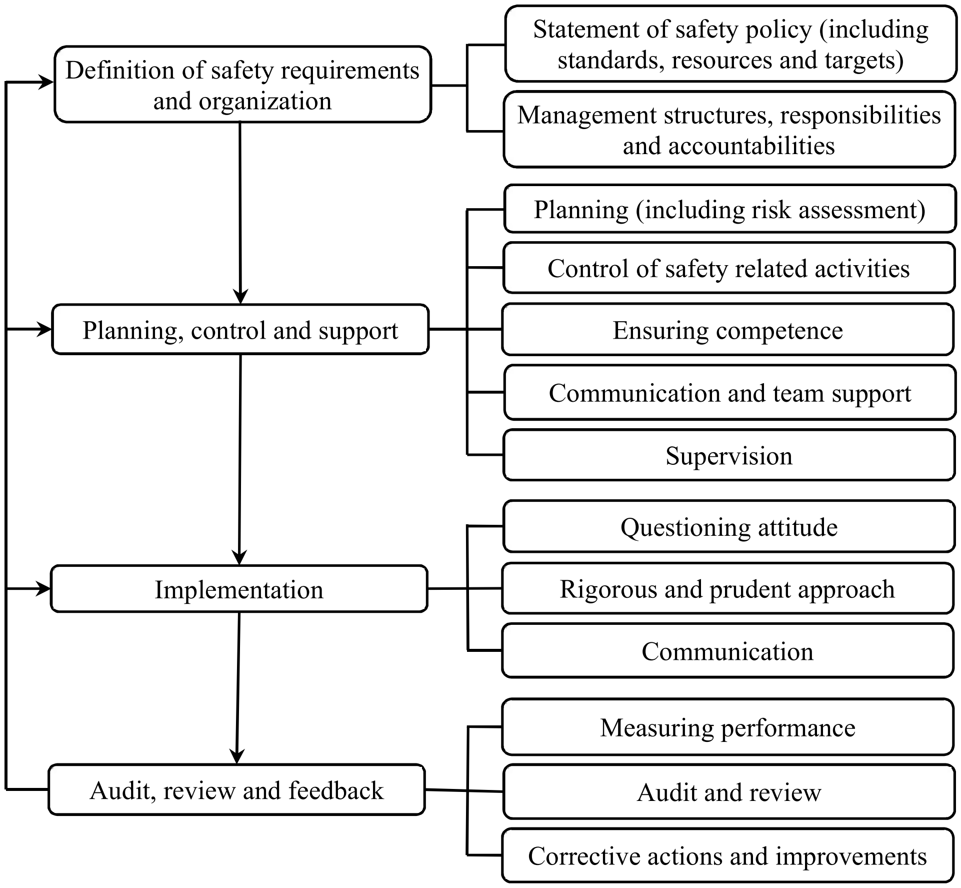

FIG. 2. Components of safety management [64].

Organizations go through a number of stages in developing their safety cultures [65]:

- Safety is compliance driven and is based mainly on rules and regulation;
- Good safety performance becomes an organizational goal;
- Safety is seen as a continuing process of improvement to which everyone can contribute.

Ref [66] presents practical advice on how to strengthen safety culture. The status of requirements for establishing, implementing, assessing and continually improving a management system for safety culture are reflected in the IAEA Safety Standards, e.g. Refs [67–70]. These include generic guidance on establishing, implementing, assessing and continually improving such a management system.

As outlined above, safety culture is a complex concept (see also Ref [71]) and there is no single indicator that can be used for determining its status. To capture both observable behaviour and people’s attitudes and basic beliefs, several methods need to be applied including interviews, focus groups, questionnaires, observations and document reviews.

When applying these assessment tools, the key safety culture characteristics and attributes described in Refs [64, 68] can be used for the identification of strengths and weaknesses in an organization’s safety culture. Annex 1 of Ref [64] sets out a series of questions for each of the major areas of concern – safety requirements and organization, planning, control and support, etc. – that are helpful in assessing the effectiveness of a safety management system and the status of an organization’s safety culture. Monitoring and measurement of the established and implemented management system effectiveness, self-assessment and performance evaluation of management at all levels, independent assessments conducted regularly, management system reviews, identification of non-conformance and establishment of corrective and preventive actions, and finally identification of improvement opportunities [67, 68] are all important elements to consider as evidence as to whether safety culture prevails.

The assessment of a safety culture can only be completed once an organization is actually operating a facility. But the necessity to inculcate a safety culture within an organization and the necessity of a safety management system need to be recognized in the planning phase for a NES. Furthermore, the proposed policies and management structures of the owner/operator can be assessed prior to operation to determine if they are consistent with safety culture.

IAEA offers a service to its Member States called ISCA (Independent Safety Culture Assessment) that can assist with evaluating the status of safety culture.

5.9. USER REQUIREMENT UR7: NECESSARY RD&D FOR ADVANCED DESIGNS

INPRO user requirement UR7 for sustainability assessment in the area of NFCF safety: The development of innovative design features of the assessed NFCF includes associated research, development and demonstration (RD&D) to bring the knowledge of facility characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for operating facilities.

INPRO user requirement UR7 discusses the necessary RD&D efforts for developing NFCFs with primarily innovative but also evolutionary[38] design features.

It is well known that intensive research may be needed to bring the level of knowledge of facility behaviour and the capability of computer codes to model phenomena and system behaviour for innovative NFCF designs to at least the same confidence level as for operating facilities.

A sound knowledge of the phenomena (e.g. chemical reaction rates, partition coefficients, solubility), and component and system behaviour, where applicable, is required to support the development of analysis tools for NFCF accidents. Hence, the more a facility differs from operating designs, the more RD&D is required. RD&D provides the basis for understanding events that threaten the integrity of barriers defined by the defence in depth concept. RD&D is also expected to provide information to reduce allowances for uncertainties in design, operating envelopes, and in estimates of accident frequencies and consequences.

For most NFCFs, it is acknowledged that the analytical tools (modelling tools) needed for completing a safety assessment comparable to the safety assessments done for nuclear reactors are currently not yet available. To promote the development of safety codes and analytical methods in the area of NFCF safety, the INPRO task group has described a situation that can hopefully be reached within the next decades.

As the development of an innovative design proceeds, RD&D is carried out to identify phenomena important to facility safety and operations and to develop and demonstrate an understanding of such phenomena. At any given point in the development process, the current understanding is incorporated into computer or analytical models that form the basis for design analysis and safety assessments. Such models are then used as tools for sensitivity analyses to identify important parameters and estimate safety margins. The results of such analyses are also

[38] An evolutionary design is an advanced design that achieves improvements over existing designs through small to moderate modifications, with a strong emphasis on maintaining proven design features to minimize technological risks. An innovative design is an advanced design that incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice.

used to identify coupled effects and interactions among systems that are important to safety. It is not unusual to obtain unexpected results, particularly in the early stages of development. The results, whether expected or not, are used to guide the RD&D program in efforts such as those to improve conceptual understanding, obtain more accurate data, confirm the extent of system interactions/independence, and adequately characterize the design. The RD&D, in turn, leads to improvements in understanding and in the analytical tools used in design and safety analyses.

The process is iterative: At the *pre-conceptual stage* of development, physical understanding, analytical models, supporting data bases, and codes may be simplistic and involve significant uncertainties. As development proceeds, understanding increases and uncertainties (both in conceptual understanding and in data) are reduced, and the validation of analytical models and codes improves. At the *time of commercialization*, all safety relevant phenomena and system interactions need to be identified and understood and the associated codes and models need to be adequately qualified and validated for use in the safety analyses, which in turn demonstrate that the facility design is safe. Complementary aspects are outlined in Ref [72].

At least the following requirements need to be met by the RD&D program of a developer for an innovative or evolutionary design:

- All significant phenomena affecting safety associated with the design and operation of an innovative NFCF have to be identified, understood, modelled and simulated (this includes the knowledge of uncertainties, and the effects of scaling and environment);
- Safety-related systems, structures and components behaviour need to be modelled with acceptable accuracy, including knowledge of all safety-relevant parameters and phenomena, and validated with a reliable database.

Figure 3 gives an overview of tasks to be performed in defining the necessary RD&D for an innovative design.

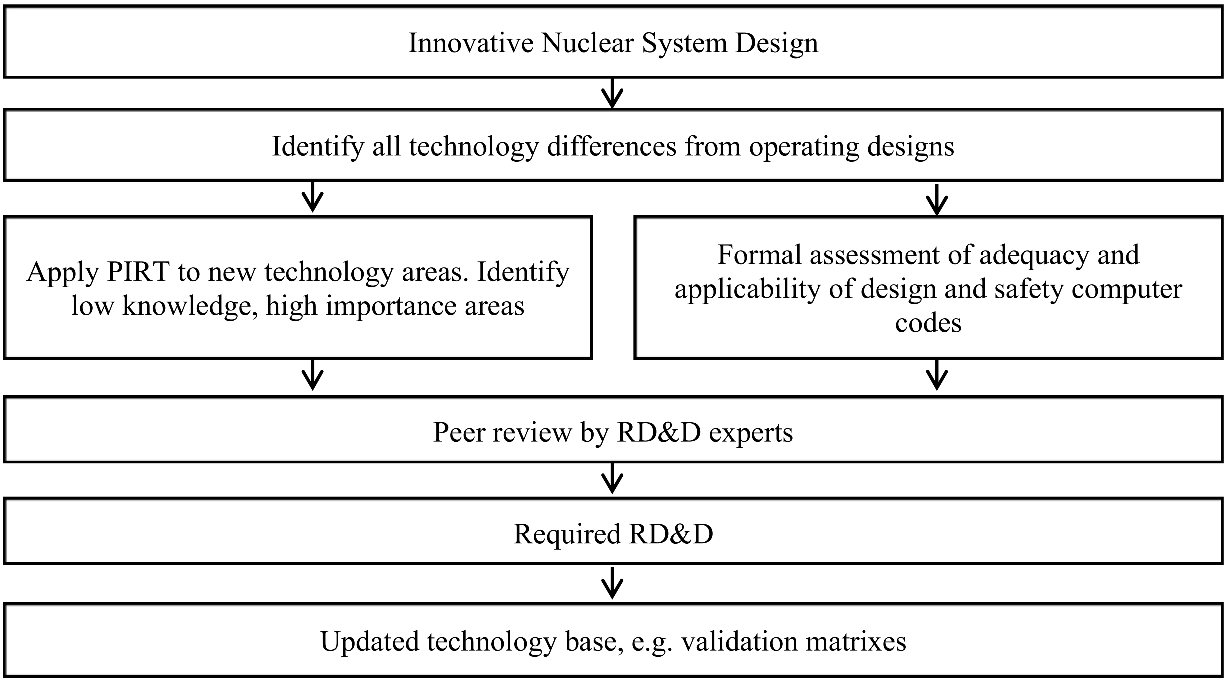

FIG. 3. Overview of the different tasks for definition of RD&D

For an innovative design, the first task is to identify all technology differences from operating designs. To identify the knowledge state and the importance of phenomena and system behaviour, an appropriate tool has to be used such as the PIRT process (Phenomena Identification and Ranking Tables), which is based mainly on engineering judgment. In addition, the adequacy and applicability of design and safety analysis computer codes have to be assessed. Both the PIRT results and the assessment of the adequacy and applicability of related computer codes inform the identification and prioritization of required RD&D efforts. An additional peer review by researchers and appropriate safety experts would strengthen the choice of the selected RD&D tasks.

Besides phenomenological data, reliability data including uncertainty bands [73] for designated components need to be evaluated to the extent possible. This is especially valid for passive safety features.

During the process of generating new and/or more detailed data (e.g. for computation fluid dynamics codes) the selected RD&D tasks are expected to be repeatedly assessed and necessary changes adopted. Qualified data need to be included in a technology base, e.g. validation matrices.

5.10. CONCLUDING REMARKS

To assess long term sustainability with regard to the safety of an NFCF to be installed after 2013, the INPRO methodology has formulated one basic principle with seven user requirements. INPRO's sustainability assessment approach in the area of NFCF safety is based on the IAEA Safety Standards and, as derived from those, the application of a DID oriented strategy for comparing the safety attributes of the assessed NFCF designs to those of reference designs. The assessment approach is supported by an increased emphasis on inherent safety characteristics and, where appropriate, passive safety features. Greater independence of the different levels of defence in depth is considered a key element for avoiding failure propagation from one DID level to the next. Using a graded approach, the number of physical barriers in a nuclear facility that are necessary to protect the environment and people depends on the potential internal and external hazards and the potential consequences of failures; therefore, the barriers will vary in number and strength depending on the type of NFCF.

The end point of the enhanced defence in depth strategy of the INPRO methodology is that, even in case of accidents, no emergency environmental releases of radioactivity and/or toxic chemicals can occur that would necessitate public evacuation. Nevertheless, effective emergency planning, preparedness and response capabilities will remain a prudent requirement.

6. MINING AND MILLING OF URANIUM AND THORIUM

This section opens with a short description of the main processes found in a facility for uranium and thorium mining and milling (or processing). With that necessary background, the section then proceeds to discuss the specific safety issues that can affect such a facility. The sustainability assessment method is then described in terms of the corresponding criteria of the INPRO methodology in the area of safety, which are adapted as necessary to the specific issues potentially affecting this type of NFCF.

6.1. MINING AND MILLING OF URANIUM

The discussion of uranium mining and milling in this section is mostly focused on the processes developed to recover uranium concentrate as the main product from the ore deposits mining, i.e. from conventional resources. Substantial amounts of uranium can be produced from unconventional resources, e.g. as a by-product of mining copper, gold, phosphates or rare earth elements. Radiation hazards in the mining and processing of unconventional uranium resources will depend on the uranium concentration in the deposit and can differ essentially from traditional mining and milling techniques. Safety issues of uranium recovery from unconventional resources are not specifically considered in the current edition of the INPRO methodology manual, however it is expected that the INPRO requirements and criteria formulated in this report will be applicable to a certain extent to the uranium production part of the process.

Uranium extraction technology, right from ore exploration to the ultimate recovery of the product, has developed rapidly over the past several decades. Significant innovations in mining, milling and leaching processes as well as in the use of equipment have also been achieved. A comprehensive description of uranium extraction technology is provided in Refs [74–76]. Open pit mining, underground mining, and in-situ leaching are the processes adopted worldwide for extraction of uranium.

6.1.1. Rock mining of uranium

Underground mining is generally economically preferable when the ore is located at greater depths (> 200 m). It consists of the following steps: identification and delineation of ore body using radiation detectors, drilling, blasting by remote control process, loose dressing and support, stowing, tramming and mucking. Very often ‘cut and fill’ and ‘open stope’ methods of underground uranium mining are used. Following identification of the area with rich uranium, a shaft is sunk in the vicinity of the ore and cross cuts are driven horizontally to the ore at various levels at an interval of 100 to 150 metres. Tunnels known as ‘drifts’ are driven along the ore from the cross cuts and tunnels known as ‘raises’ are made up and down from level to level, to reach the ore body. The ‘stope’ is the workshop for the mine, where ore extraction continues. Mining requires a network of shafts, tunnels and chambers connecting with the surface and allowing movement of workers, machine and rock within the mine, and services such as water, electric power, fresh air, exhaust and compressed air, drains and pumps to collect seeping ground water, and a communication system. Entry of miners after blasting is normally delayed till the dust and fumes disperse with ventilation air to the permissible level.

Open pit mining is usually preferable for shallower orebodies since the productivity is higher, ore recovery is better in comparison to underground mining due to lower dilution, dewatering of the open pit is easier and mining conditions are safer. The disadvantages include large-scale excavation resulting in huge overburden of rock and soil, land degradation and a chance of environmental contamination.

6.1.2. Milling of uranium after rock mining

Milling consists of the following steps (see Fig. 4): Transferring the ore by conveyor belt, crushing, and (wet) grinding the ore into sand and silt. This is followed by processing that involves leaching mostly[39] with sulphuric acid in the presence of pyrolusite as an oxidant in large containers, anion exchange separation of uranium and purification and concentration of uranyl sulphate product. Precipitation is then carried out to remove iron and other metal impurities and to recover uranium typically as ammonium diuranate (ADU) or in a peroxide form. Finally, U_3O_8 is produced by dewatering and roasting (calcining).

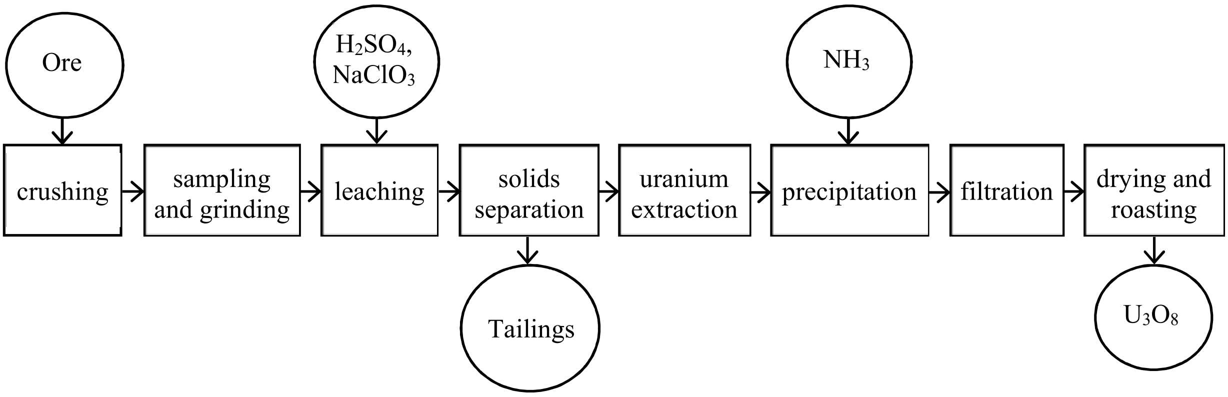

FIG. 4. Uranium milling process (example of acid leaching)

6.1.3. Uranium milling tailings management

The barren liquor from ion exchange in the milling facility generates acidic *liquid* waste (where acid has been used in the leaching process) which contains most of the radium and other radionuclides dissolved in the leaching process and traces of uranium not absorbed in the ion exchange step. The slurry form called tailings includes *solid* waste containing the un-dissolved uranium, radium and other radionuclides. When acidic, these tailings are mixed with lime for neutralization and may be sent to a hydro cyclone, where sand and slime get separated. The sand can go to the mine for backfilling and the slime to a tailings pond.

The milling waste is normally transferred from the milling facility to a tailings management facility where it consolidates and settles over a period of many years. The decanted liquid from the tailings is normally recycled whenever possible or sent to the effluent treatment plant for chemical treatment and activity removal to enable its discharge to the public domain according to the norms specified by the regulatory authority. Finally, these tailings have to be disposed of in above ground impoundments (usually with dams), open pits or underground mine voids.

The wastes produced in a (underground or open pit) mine and milling facility include, in addition to mill tailings, also waste rock and mineralized waste rock. They contain only low concentrations of radioactive material and toxic chemicals but are generated in large volumes compared to other nuclear facilities, and therefore need to be safely stored and disposed of in accordance with regulatory requirements [27].

In this manual the tailing facility is treated as an integral part of the mining and milling facility (and not as a separate waste management facility).

Overview of technologies, strategies, methods and problems related to long term management of uranium tailings are described in Ref [24].

[39] There can be also alkaline leaching depending on the ore chemistry.

6.1.4. In situ leaching uranium mines

In situ leaching (ISL) mining is defined as the leaching of uranium from a host sandstone by chemical solutions and the recovery of uranium at the surface [74, 75] (Fig. 5). ISL extraction is conducted by injecting a suitable leach solution into the ore zone below the water table; oxidizing, complexing, and mobilizing the uranium; recovering the pregnant solutions through production wells; and, finally, pumping the uranium bearing solution to the surface for further processing. ISL involves extracting the ore mineral from the deposit, with minimal disturbance of the existing natural conditions of the earth's subsurface and surface.

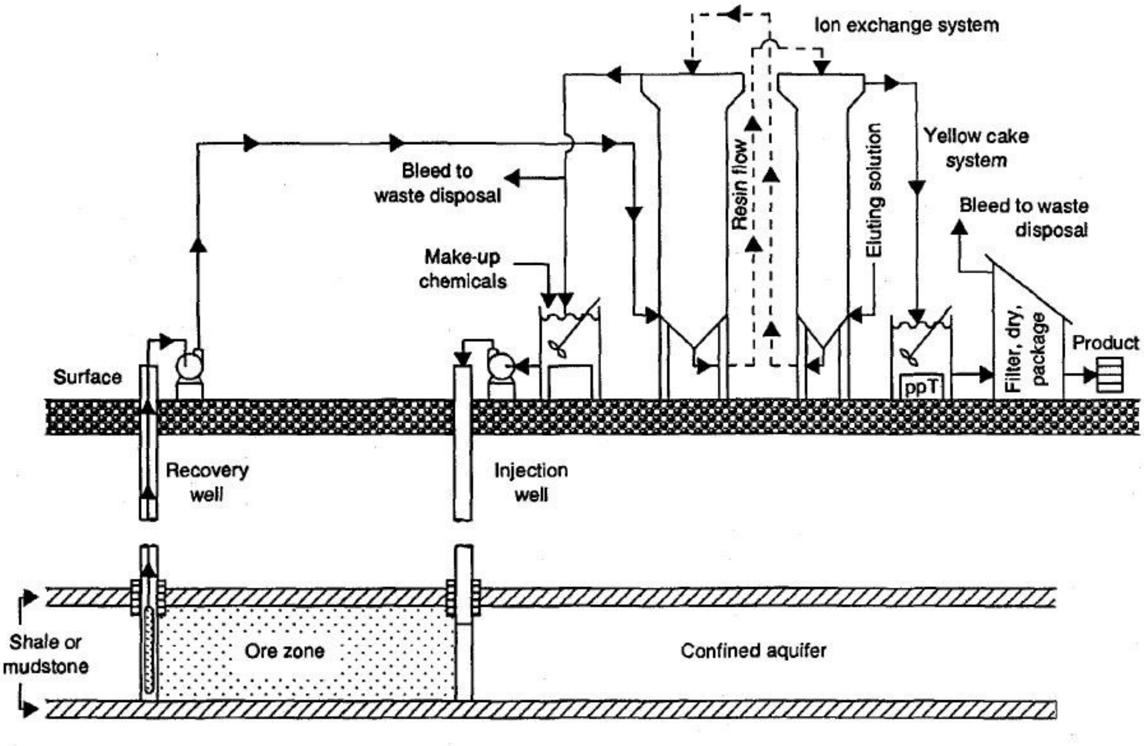

FIG. 5. In-situ leaching process [74].

Compared to underground and open pit mining, the in-situ leaching process does not introduce significant surface disturbance, eliminates the crushing, milling, grinding and leaching steps in the process, provides safer working conditions, does not generate large solid radioactive waste streams and requires less manpower. However, this process is applicable only for certain types of ores and detailed precautions are required to be taken to keep contamination from reaching water sources [75].

Thus, in contrast to underground and open pit mining, in the case of in-situ leaching (ISL) mines, there are no rock dumps and tailings storage facilities, no dewatering of aquifers, and much smaller volumes of mining and hydrometallurgical effluents that could contaminate the surface, air and water supply sources [30, 75].

6.1.5. Safety issues in uranium mining and milling

Natural uranium consists of a mix of radioactive isotopes emitting some α-, β-, and γ−radiation. The level of γ−radiation from a lump of pure natural uranium is lower than from an equivalent sized lump of granite and gloves are normally considered as sufficient protection for handling metallic natural uranium [77]. However, the mining of uranium is always associated with its progenies such as radon and radium, which have much higher radiation hazards than uranium itself.

While radiation exposure is typically the potential human health hazard most associated with NESs, including their uranium mining and milling facilities, it is by far not the only hazard that humans are exposed to in such facilities. Uranium mining and milling involves most of the same hazards associated with other mining activities and the risks associated with radiation may not necessarily be the dominant contributors to the overall risk.

Ref [78] gives an overview of risk constituents in uranium mining, including those not related to radiation factors, such as the silica exposure that cannot be avoided during the rock mining and processing of uranium ore. Silica-caused adverse health outcomes are mostly respiratory disorders such as silicosis, which is characterized as "progressive, life-threatening, fibrotic lung disease". Besides that, mining may involve traumatic injuries including fatal ones from explosions, fires, underground mine flooding, collapses of bulkheads, caving failures, rock falls, falls from height, entrapment, etc. Ref [78] provides data that can be used to estimate the rate of fatalities from traumatic injury in mining (including metal, nonmetal, stone, sand, and gravel mines) at about 20 per year per 100 000 full-time equivalent employees in the period from 1988 to 2007. Other non-radiation hazards in mining include electrical accidents, which "occur less frequently than other sources of traumatic injury, but they are disproportionately deadly" [78]; exposure to diesel fumes including carbon monoxide; nitrogen oxides in explosive gases; noise, vibration, etc.

Although normal industrial safety issues of uranium mining and milling, i.e. those not related to radiation exposure, have to be considered by the responsible national organisations as one of the most important parts of the authorisation process, these issues are deemed to be not directly linked to the sustainability of a NES and are not considered in the current edition of the INPRO methodology manual.

A large number of reports discuss the safety and environmental issues associated with uranium extraction activities [19, 23–31, 74, 79–83]. The safety issues related to the ore extraction extend through the lifetime of mining and milling facilities including their decommissioning. Though these stages are different, the hazard potential remains more or less similar. The siting of uranium mining and related components needs to consider disposal of waste and final decommissioning without unduly affecting the environment (e.g. displacement of population, change in the land use, flora and fauna, etc.). Having selected a site through a detailed survey, an extensive baseline survey of the area is to be undertaken prior to actual operation. The study of baseline data consists of the existing environmental quality (air, water and land) in terms of existing pollution load (both conventional and radioactive), land use, socioeconomic status, land use pattern, flora and fauna. A change in any of the baseline values would serve as a performance indicator of the operation of mining and milling activities and could be used for safety and environmental assessment.

As discussed above in this section, the predominant occupational and environmental hazards of mining and milling of uranium may be no different from other mineral extraction processes. However, there are additional risks involved in uranium mining due to potential exposure to radioactive materials.

The major radiological safety issues in the entire process – mining, milling, leaching, product recovery, storage and disposal of tailings – are *dust, chemical, and radiation exposure* to the workers (and to the general public). The aspect of transport of ore or the product from site to site is yet another site-specific safety related issue. The three main sources of *radiation exposure of mine workers* are:

- Radioactive dusts that are breathed in;
- Gamma radiation from the ore that irradiates the body;
- Radon gas that is breathed in (in underground mines).

Open pit and underground mining and milling operations involving uranium have a potential to generate dust that contains radioactivity in varying quantities. The hazard potential is higher if the operations are dry and dusty rather than wet operations.

The progenies of natural uranium are in equilibrium with uranium and some of them, such as ^{214}Bi and ^{214}Pb, are strong gamma emitters that pose external exposure hazards. According to data provided by India, ca. 83 % of the gamma energy is emitted from ^{214}Bi and 12 % from ^{214}Pb. The individual dose from gamma radiation from an ore body of 0.5 % grade can reach 50 mSv/a. Similarly, the gaseous uranium progeny radon (^{222}Rn) is another major source of radiation exposure to the workers in a mine and potentially also to workers in the milling and product extraction areas as well. Hence, safety in the design and operation of the process is of paramount importance and monitoring of workers' dose in accordance with national regulatory requirements is essential.

Radiation exposures to the general public from normal operations and anticipated operational occurrences (AOOs) in the uranium ore processing industry needs to be less than 1 mSv in a year. The disposal of radioactive waste and the consequent dose from all pathways of exposure to the general public residing near the industry shall not exceed the quantities derived from the prescribed limits. The safety measures for the processing operations, transport, storage and disposal of the tailings, and for the ultimate decommissioning of the facility, shall result in exposures to the public well below the regulatory limits. The radiation exposure of the public and the environment caused by a mining and milling facility, including its tailings management facility, is covered in a separate manual of the INPRO methodology on the environmental impact by stressors.

6.2. THORIUM MINING AND MILLING

Thorium mining is largely done by open-pit methods, dredging and beach sand collection.

6.2.1. Mineral sands mining and separation of thorium

The valuable minerals in mineral sands, such as ilmenite, rutile, sillimanite, garnet, zircon, monazite, etc, are mined and separated based on differences in physical properties. Thorium content in the sand is normally quite low [84]. Mining, separation and processing of these minerals involve operation of floating dredge or dry mining, gravity separation, application of high voltage and high magnetic fields, operation of dryers, and operation of material handling equipment like belt conveyors, bucket elevators, and mixer-settlers involving flammable materials.

Monazite is subjected to further processing to obtain thorium oxalate/thorium nitrate and ADU, whereas zircon frit powder can be obtained from zircon. Standard chemical processes involving digestion, solvent extraction, precipitation and filtration are used for this purpose. For example, processing of monazite, an orthophosphate of thorium and rare earth elements, is carried out by digestion of finely ground monazite with caustic soda, which results in three components, namely by-product tri-sodium phosphate, mixed hydroxides of rare earths, thorium and uranium as well as un-reacted monazite. After the majority of rare earths is first separated from the mixed hydroxide, the mixed hydroxides of thorium, uranium and residual rare earths are extracted through acid leaching. This is followed by solvent extraction to ultimately produce thorium oxalate and a crude uranyl chloride solution and to recycle the residual rare earths. The crude uranium chloride solution is subsequently refined to produce nuclear grade U_3O_8. As carried out in India [85], chemical processing of every ton of monazite containing ca. 9% of thorium to the thorium oxalate form produces approximately 0.08-0.10 ton of insolubles, 0.06-0.10 ton of Pb-Ba cakes and 0.1 ton of phosphates and other solid waste from effluent treatment. The solid wastes are buried in underground trenches. Liquid effluent is treated in the effluent treatment plant and then discharged after monitoring.

6.2.2. Safety issues in thorium mining

As the majority of thorium mining is by open-pit methods or by wet dredging, the radiological problems, particularly inhalation hazards, are relatively small compared to those in underground uranium mining. Inhalation hazards arise mainly from dust produced during the physical separation of the mineral constituents of placers or from thoron (^{220}Rn) gas. The methods used in dry operations are magnetic and electrostatic separation, and separation by wind/air tables, which produce a lot of dust. Dust is also generated during drying and conveying, etc. Thorium is present in the dust during segregation of heavy minerals. Thus, the assessment of hazards needs to include, in addition to thoron, an assessment of thorium and its long-lived daughter products in the working atmosphere. The dose delivered to the lungs from breathing in an atmosphere containing thoron and its daughters arises principally from the decay of thoron and ^{216}Po in the airways of the lung, and the deposition and subsequent decay of inhaled daughter products.

Most of the radiation exposures in the mineral sands industry come from the inhalation of airborne dust. However, if appropriate procedures are not followed, workers can also be exposed to external radiation. This external radiation may come from the emission of gamma radiation from final product storage or intermediate mineral stockpiles that have high monazite content. Most of the external radiation exposures in mineral sand processing plants can occur from being in close proximity to stored material. External radiation hazards arise from both beta and gamma radiation emitted by ^{228}Ac (1 MeV gamma radiation, and 1.2, 1.7, 1.9 and 2.2 MeV beta radiation), ^{212}Bi (2.25 MeV beta radiation) and ^{208}Tl (1.8 MeV beta and 2.6 MeV gamma radiation).

Dust deposits on surfaces depend on the operational methods used and on the wetness of the mine; normally, it is not hazardous except possibly as a source of air contamination. However, clothing contamination may be a more significant source of exposure than in uranium mines because of the more pronounced beta and gamma emitters associated with thorium. Chemical processing of monazite to extract thorium involves grinding of monazite to reduce its particle size. This operation and subsequent handling of powdered monazite can lead to air borne dust. Thorium bearing monazite usually contains a very small amount of uranium, and although the typical ratio of thorium to uranium is 25:1, ^{222}Rn and radon progenies may occur in significant air concentrations along with ^{220}Rn and thorium in the initial chemical treatment areas of the plant.

Since the hazard from thoron is predominantly attributable to ^{212}Pb, which occurs with thoron in all practical situations, it is permissible to apply the value for ^{212}Pb as the standard of control for both radionuclides. Because of the very short half-lives of thoron ^{220}Rn (55.6 s) and ^{216}Po (0.15 s) compared to ^{212}Pb (10.6 h), dilution ventilation is relatively ineffective for these radionuclides, but it can reduce the concentration of ^{212}Pb by a large factor. Thus, in some atmospheres, the concentration of thoron may exceed that of ^{212}Pb by orders of magnitude. This situation is restricted to places where clean ventilating air is continuously available at the source and therefore it could be manifested in mills. The dose from thoron itself may be comparable to that of ^{212}Pb in cases of extreme non-equilibrium. External radiation is associated with the physical treatment of monazite. In the monazite stores and filling area, however, the radiation levels could be high.

The chemical treatment of monazite gives two fractions: the thorium fraction (consisting of ^{232}Th and ^{228}Th from the thorium series, and ^{234}Th, ^{230}Th, ^{231}Th and ^{227}Th from the uranium series) and the non-thorium fraction (consisting of ^{228}Ra, ^{224}Ra and other daughters from the thorium series, and ^{226}Ra with daughters from the uranium series). The processing of monazite to extract thorium gives rise to generation of solid, liquid and gaseous wastes. The thorium ore monazite is essentially an orthophosphate of rare earths, thorium and uranium. As such, there is no significant problem of liquid waste in mining or in mineral separation plants using physical methods. However, the liquid effluents from the chemical processing of monazite contain the decay products from the uranium and thorium series. Because of suspended and total solid load in the effluents, they are allowed to pass through settling tanks, the clear overflow from which, after suitable dilution, can be released to nearby recipient water bodies.

6.3. ADAPTATION OF THE INPRO METHODOLOGY TO URANIUM AND THORIUM MINING AND MILLING

The INPRO methodology for sustainability assessment in the areas of nuclear safety was developed originally with a focus on nuclear power plants and was later adapted to NFCFs. The use of the INPRO methodology for an assessment of a uranium or thorium mining and milling facility required significant modifications of the methodology, as several user requirements and criteria are not directly applicable for such a facility. This section presents how the INPRO methodology in the area of NFCF safety was adapted to a mining and milling facility.

6.3.1. INPRO basic principle for sustainability assessment of uranium and thorium mining and milling facilities in the area of safety

INPRO basic principle for sustainability assessment of uranium or thorium mining and milling facility in the area of safety: The planned uranium or thorium mining and milling facility is safer than the reference mining and milling facility[40].

The rationale for the BP was provided in Section 5.2. The definition of the reference NFCF was discussed in Section 3.1. This definition comprises several options that can be used to determine the reference NFCF depending on the type of facility assessed and the specific technology used. In the context of uranium and thorium mining and milling, the concept of a reference design is primarily applicable to a milling facility and tailings management facility. Definition of the reference facility for the mine assessed can be fairly challenging compared to other types of NFCF because of very broad variety of technologies used in mining as stipulated by the different types of deposits and different geological/ hydrological conditions. However, when a reference facility cannot be defined for a given mine, at least the systems dealing with radiological hazards (e.g. shielding, ventilation, protection against radon and dust) can be assessed against INPRO criteria.

The INPRO methodology has defined a set of requirements for mining and milling facilities and criteria for the assessment. Several INPRO criteria defined for the sustainability assessment of mining and milling facilities in the area of safety involve consideration of 'state of the art' concept[41] as the acceptance limits. These sustainability assessment criteria are related to those specific features of the mining and milling facilities that are important to radiation protection and safety (control of radiation sources). The criteria should therefore not be interpreted as nuclear safety recommendations, industrial safety requirements or general requirements for the mining or milling technology used.

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[40] In the case of mining and milling facilities, this statement is deemed to be sufficient to cover the second part of the INPRO basic principle (see Section 5.2) formulated for NFCFs: "In the event of an accident, off-site releases of radionuclides and/or toxic chemicals are prevented or mitigated so that there will be no need for public evacuation". The necessity for public evacuation due to radiological hazards from a potential accident in a mining / milling facility that is shown to have a level of safety superior to that of a reference facility seems very unlikely.

[41] In this manual 'state of the art' means using the most recent technology and international practice.

The INPRO methodology user requirements pertaining to mining and milling facilities are displayed in Table 5.

TABLE 5. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF MINING AND MILLING FACILITIES IN THE AREA OF NFCF SAFETY

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR1: Robustness of design during normal operation: The design for the mining/ milling facility assessed is more robust than the reference design with regard to operation and systems, structures and component failures.	CR1.1: Design of normal operation systems	IN1.1: Robustness of design of normal operation systems. AL1.1: Superior to that in the reference design.
	CR1.2: Facility performance	IN1.2: Facility performance attributes. AL1.2: Superior to those in the reference design.
	CR1.3: Inspection, testing and maintenance	IN1.3: Capability to inspect, test and maintain. AL1.3: Superior to that in the reference design.
	CR1.4: Failures and deviations from normal operation	IN1.4: Expected frequency of failures and deviations from normal operation. AL1.4: Lower than that in the reference design.
	CR1.5: Occupational dose	IN1.5: Occupational dose values during normal operation and AOOs. AL1.5: Lower than the dose constraints.
UR2: Detection and interception of AOO: The mining/milling facility assessed is capable to monitor, detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions.	CR2.1: I&C systems and operator procedures	IN2.1: I&C system to monitor, detect, trigger alarms, and, together with operator actions, intercept and compensate AOOs that could lead to radiation exposure of workers. AL2.1: Availability of such systems and/or operator procedures.
	CR2.2: Grace periods for AOOs	IN2.2: Grace periods until human (operator) actions are required after detection (and alarm) of AOOs. AL2.2: Adequate grace periods are defined in the design analyses.
UR3: Accidents: The frequency of occurrence of accidents in the mining/ milling facility assessed is reduced. If an accident occurs, engineered safety features and/or operator actions are able to restore the facility assessed to a controlled state, and subsequently to a safe state, and the consequences are mitigated to ensure the confinement of nuclear and/or toxic chemical material. Reliance on human intervention is minimal, and only required after sufficient grace period.	CR3.1: Frequency of accidents	IN3.1: Calculated frequency of occurrence of accidents. AL3.1: Lower than that in the reference design.
	CR3.2: Engineered safety features and operator procedures	IN3.2: Reliability and capability of engineered safety features and/or operator procedures. AL3.2: Superior to those in the reference design.
	CR3.3: Grace periods for accidents	IN3.3: Grace periods for accidents until human intervention is necessary. AL3.3: Longer than those in the reference design.
	CR3.4: Barriers	IN3.4: Number of confinement barriers maintained (intact) after an accident. AL3.4: At least one.

TABLE 5. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF MINING AND MILLING FACILITIES IN THE AREA OF NFCF SAFETY (cont.)

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR4: Severe plant conditions	None	User requirement UR4 was found to be not directly applicable to a mining and milling facility
UR5: Inherent safety characteristics: To excel in safety and reliability, the mining/ milling facility assessed strives for elimination or minimization of some hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics.	CR5.1: Minimization of hazards	IN5.1: Examples of hazards: fire, flooding, release of radioactive material, radiation exposure, etc. AL5.1: Hazards minimized according to the state of the art.
UR6: Human factors related to safety: Safe operation of the mining/ milling facility assessed is supported by accounting for HF requirements in the design and operation of the facility, and by establishing and maintaining a strong safety culture in all organizations involved in the life cycle of the facility.	CR6.1: Human factors	IN6.1: Human factors addressed systematically over the life cycle of the mining/ milling facility assessed. AL6.1: Evidence is available.
	CR6.2: Attitude to safety	IN6.2: Prevailing safety culture. AL6.2: Evidence is provided by periodic safety reviews.
UR7: RD&D for advanced designs: The development of innovative design features of the mining/ milling facility assessed includes associated RD&D to bring the knowledge of facility characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for operating facilities.	CR7.1: RD&D	IN7.1: RD&D status. AL7.1: RD&D defined, performed and database developed.
	CR7.2: Safety assessment	IN7.2: Adequate safety assessment. AL7.2: Approved by a responsible regulatory authority.

6.3.2. User requirement UR1: Robustness of design during normal operation

The rationale for UR1 was described in Section 5.3. User requirement UR1 is focused on preventing AOOs. For a mining and milling facility, examples of AOOs that could potentially cause radiation doses to workers include the following:

- In an underground mine, a malfunction of the ventilation system (needs to be compensated by switchover to a backup system);
- In a milling facility, a malfunction of the dust prevention equipment in the crushing and grinding unit (leading to accumulation of radioactive dust);
- In a milling facility, a (small) leakage of (liquid or gaseous) radioactive material[42] in the processing unit.

It is acknowledged that an insufficient radiation protection program (RPP) or a failure by the workers to follow its (administrative) procedures (e.g. keeping distance and limiting presence, wearing of protective respiratory equipment or dose monitoring devices) and to apply (technical) measures defined in the RPP (e.g. shielding) could be also a reason for radiation

[42] Including combinations of leakages of radioactive and chemically toxic materials.

exposure of workers in a mining and milling facility. This issue of human behaviour (safety culture) is covered in user requirement UR6.

INPRO methodology selected five criteria for UR1 as displayed in Table 6.

TABLE 6. CRITERIA FOR USER REQUIREMENT UR1

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR1: Robustness of design during normal operation: The design for the mining/ milling facility assessed is more robust than the reference design with regard to operation and systems, structures and components failures.	CR1.1: Design of normal operation systems	IN1.1: Robustness of design of normal operation systems. AL1.1: Superior to that in the reference design.
	CR1.2: Facility performance	IN1.2: Facility performance attributes. AL1.2: Superior to those in the reference design.
	CR1.3: Inspection, testing and maintenance	IN1.3: Capability to inspect, test and maintain. AL1.3: Superior to that in the reference design.
	CR1.4: Failures and deviations from normal operation	IN1.4: Expected frequency of failures and deviations from normal operation. AL1.4: Lower than that in the reference design.
	CR1.5: Occupational dose	IN1.5: Occupational dose values during normal operation and AOOs. AL1.5: Lower than the dose constraints.

6.3.2.1. Criterion CR1.1: Design of normal operation systems

Indicator IN1.1: Robustness of design of normal operation systems.

Acceptance limit AL1.1: Superior to that in the reference design.

All equipment and systems relevant for safety[43] used in a mining and milling facility need to be designed against loads caused by events associated with internal and external hazards (see Section 4.2.1). The design (e.g. mechanical, thermal, electrical, etc.) of normal operating systems in a mining and milling facility can be made more robust, i.e. reducing the likelihood of failures, by increasing the design margins, improving the quality of manufacture and construction, and by the use of materials of higher quality.

As stated before, in the case of underground mining, the ventilation system and corresponding power supply system are the main operating systems relevant for radiological safety of mine workers[44]. Design of these systems is expected to be more robust in terms of reliability and needs to ensure that radon levels in the mine remain below safety limits. To minimize radon inhalation in an underground uranium mine, one could increase the robustness of the ventilation system by means of enhanced redundancy (e.g. by incorporating a standby system and an auxiliary power supply). Higher robustness of these systems could be achieved by increasing the quality of manufacture and installation and using improved materials adapted to the environment in the mine[45].

[43] In this report ‘safety’ generally means ‘radiological safety’ or ‘nuclear safety’. Conventional safety issues, including those concerning chemical toxicity, are beyond the scope of the current edition of the INPRO methodology manual.

[44] Doses to the public from releases and discharges during normal operation and anticipated occurrences are considered in the INPRO methodology manual on environmental impact of stressors.

[45] Other measures, e.g. prior testing and analysis of the rock beds, analysis of rock mechanics and incorporation of these data in the design, would result in a more robust design of an underground mine. Because these measures are not directly related to the radiological safety of mine workers and the public, they are not further discussed in this publication.

For in-situ leaching mines, the equipment (e.g. piping and pumps) used for processing the uranium solution can be made more robust through design measures similar to those described above for the ventilation system.

The tailings management facility contains large volumes of radioactive (with low activity) and chemically toxic materials as waste from the mining and milling process and these wastes are normally stored and occasionally need to be disposed of. To prevent leakage of these materials to the environment, the tailings management facility needs to be isolated appropriately. The robustness of the isolation function can be increased by such design measures as introducing a liner or even using several layers of barriers against leakage.

The **acceptance limit AL1.1** of **CR1.1** is met if evidence available to the INPRO assessor shows that the design of the facility assessed is more robust than the reference design. Alternatively, if a reference design cannot be defined, it needs to be demonstrated that the design of the facility assessed has taken available information on best international practice into account and is therefore state of the art.

6.3.2.2. Criterion CR1.2: Facility performance

Indicator IN1.2: Facility performance attributes.

Acceptance limit AL1.2: Superior to those of the reference design.

Superior facility performance can increase the robustness of mining and milling facilities.

As stated in Section 6.1.5 and 6.2.2 above, dust that contains radioactive material is one potential source of *internal radiation exposure* of workers in a uranium or thorium underground or open pit mine and milling facility. Modern operational methods (e.g. wet processes in drilling, hooded equipment in the milling facility) that minimize the generation and spreading of contaminated dust constitute an example of high quality of operation. Prevention of dust inhalation can be further improved by using higher levels of automation.

Uranium or thorium ore can also be a direct source for *external radiation exposure* for workers in an underground or open pit mine. In case of high concentrations of uranium in the ore, automation and tele-operation (e.g. raise bore method [31]) can minimize the external radiation exposure of workers.

As part of a successful radiation protection program, administrative and engineering procedures (e.g. shielding, compartmentalization, sampling of dust and radon, monitoring of workers' dose, wearing of protective respiratory equipment, etc.) need to be in place to protect the workers against external and internal radiation exposure. The workers are supposed to receive sufficient training in these administrative and engineering procedures. Worldwide operating experience in uranium and thorium mining and milling facilities is expected to be taken into account in designing the radiation protection program of a new facility.

The **acceptance limit AL1.2** of **CR1.2** is met if evidence available to the INPRO assessor shows that the quality of operation in the facility assessed is superior to that in the reference design. If a reference facility cannot be defined, it needs to be demonstrated that the operation of the facility assessed accounts for available information on best international practice and is therefore state of the art with regard to high quality of operation.

6.3.2.3. Criterion CR1.3: Inspection, testing and maintenance

Indicator IN1.3: Capability to inspect, test and maintain.

Acceptance limit AL1.3: Superior to that in the reference design.

The assessed design of mining/ milling facility is expected to permit efficient and intelligent inspection, testing and maintenance and not just require more inspections and more testing. In particular, the programs of inspection, testing and maintenance need to be driven by a sound understanding of failure mechanisms, so that the right locations are inspected and the right systems, structures and components are tested and maintained at the right time intervals.

In an underground mine, the parts of the ventilation system needing inspection, testing and maintenance (e.g. fans, motors, etc.) need to be located in fresh air so that they can be easily inspected, tested and maintained during operation [17].

In a milling facility, the equipment used to minimize dust in the air (e.g. hooding, exhaust system) and to chemically process the ore needs to be designed to enable easy inspection, testing and maintenance.

The **acceptance limit AL1.3** of **CR1.3** is met if evidence available to the INPRO assessor shows that the capability to inspect and test systems relevant to radiation protection in the facility assessed is superior to that in the reference design (or is state of the art and allows easy inspection, testing and maintenance).

6.3.2.4. Criterion CR1.4: Failures and deviations from normal operation

Indicator IN1.4: Expected frequency of failures and deviations from normal operation.

Acceptance limit AL1.4: Lower than that in the reference design.

The frequency of failures and deviations from normal operation defined (see beginning of Section 6.3.2) for a mining and milling facility needs to be derived from operational experience and supported by safety analyses (PSA, if available). For the facility assessed, the designer is expected to reduce these frequencies by increasing the robustness of the design (discussed in CR1.1 above), enabling high quality of operation (discussed in CR1.2), and ensuring efficient and intelligent inspection and maintenance (discussed in CR1.3).

The **acceptance limit AL1.4** of **CR1.4** is met if evidence available to the INPRO assessor shows that the frequencies of failures and deviations from normal operation are lower than those in the reference design. If quantitative results from operational experience and PSA are not available, a deterministic analysis needs to be developed that supports a reduction of these frequencies through increased design robustness, high quality of operation, and intelligent inspection and maintenance programs.

6.3.2.5. Criterion CR1.5: Occupational dose

Indicator IN1.5: Occupational dose values during normal operation and AOOs.

Acceptance limit AL1.5: Lower than the dose constraints.

The mining and milling facility assessed is expected to use operational experience from existing deigns to ensure efficient implementation of the concept of optimised radiation protection for workers during design, commissioning, operation, and decommissioning. Criterion CR1.5 anticipates that new mining and milling facilities will use careful layout and reliable equipment to optimise the radiation protection of workers.

Regulatory limits in the country have to comply with international standards. Ref [52] states that:

> "For occupational exposure of workers over the age of 18 years, the dose limits are:
>
> (a) An effective dose of 20 mSv per year averaged over five consecutive years (100 mSv in 5 years) and of 50 mSv in any single year;

(b) An equivalent dose to the lens of the eye of 20 mSv per year averaged over five consecutive years (100 mSv in 5 years) and of 50 mSv in any single year;

(c) An equivalent dose to the extremities (hands and feet) or to the skin of 500 mSv in a year"

Ref [52] further recommends using dose constraints "for optimization of protection and safety, the intended outcome of which is that all exposures are controlled to levels that are as low as reasonably achievable, economic, societal and environmental factors being taken into account". The role of dose constraints is explained in Refs [25, 86]. In the INPRO methodology, the dose constraints concept is discussed in more detail in the manual on environmental impact of stressors [5].

Innovative and proven techniques, such as increased automation, improved operation and maintenance techniques and effective (engineered) safety features, are required to be used in the optimization of protective measures. Ref [25] provides detailed guidance on how to achieve a successful radiation protection program for workers in a uranium (or thorium) mine and milling facility.

The **acceptance limit AL1.5** is met if evidence available to the INPRO assessor shows that the dose values of workers during normal operation and AOOs are (will be) lower than the dose constraints defined for the location of the planned facility.

6.3.3. User requirement UR2: Detection and interception of AOO

The rationale of UR2 was provided in Section 5.4. The criteria selected for user requirement UR2 are presented in Table 7.

TABLE 7. CRITERIA FOR USER REQUIREMENT UR2

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR2: Detection and interception of AOO: The mining/milling facility assessed is capable to monitor, detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions.	CR2.1: I&C systems and operator procedures	IN2.1: I&C system to monitor, detect, trigger alarms, and, together with operator actions, intercept and compensate AOOs that could lead to radiation exposure of workers. AL2.1: Availability of such systems and/or operator procedures.
	CR2.2: Grace periods for AOOs	IN2.2: Grace periods until human (operator) actions are required after detection (and alarm) of AOOs. AL2.2: Adequate grace periods are defined in the design analyses.

6.3.3.1. Criterion CR2.1: I&C systems and operator procedures

Indicator IN2.1: I&C system to monitor, detect, trigger alarms and, together with operator actions, intercept and compensate AOOs that could lead to radiation exposure of workers.

Acceptance limit AL2.1: Availability of such systems and/or operator procedures.

A mining and milling facility is expected to be designed to cope with AOOs (see beginning of Section 6.3.2) using preferably automatic operational systems, i.e. I&C systems that bring the facility back to normal operating conditions. In case automatic systems are not available, adequate operator procedures need to be. Passive and active control systems are deemed more reliable than administrative (manual) control but it is acknowledged that they are difficult to develop for mining/ milling facilities.

In an *underground mine,* an important deviation from normal operational state, i.e. an AOO, is the faulty operation of the ventilation system leading to build-up of radon in the atmosphere inside the mine. In case operational limits of radiation levels are violated, the I&C systems initiate an alarm. To take timely corrective action (e.g. switch over to a backup ventilation system or evacuate the mine), it is necessary to have continuous monitoring of radon levels in the atmosphere, and associated alarm systems. The availability of such a monitoring system is thus an acceptance criterion.

I&C systems need to be available also in the *milling facility* for controlling the air quality (radioactive dust concentration) and the radiation levels of equipment (accumulation of radioactive dust), and for detecting any leakage of radioactive (and chemically toxic) materials from the chemical processing equipment.

In the event that the barriers of the *tailings management facility* are breached and radionuclides find their way into ground water, it is important that the leakage is detected as early as possible and actions initiated to arrest further leakage [87]. This necessitates a regular system of radioactivity monitoring in nearby water bodies and bore wells. The requirements for an adequate monitoring system (and program) for a tailings facility are described in Ref [26] The availability of such a monitoring system is thus an acceptance criterion.

The **acceptance limit AL2.1** of **CR2.1** is met if evidence available to the INPRO assessor shows that I&C systems are available in the facility assessed that are capable of detecting safety-relevant deviations from normal operation, providing alarms, and initiating compensatory actions.

6.3.3.2. Criterion CR2.2: Grace periods for AOOs.

Indicator IN2.2: Grace periods until human (operator) actions are required after detection (and alarm) of AOOs.

Acceptance limit AL2.2: Adequate grace periods are defined in the design analyses.

Grace periods for AOOs are adequate if the time periods available before operator actions are required are long enough for the operator to react reasonably. The appropriate value of this grace period depends on ease of diagnosis of a failure, and the complexity of the human action to be taken. Simple failures and consecutive straightforward actions require shorter grace periods. The facility needs to have sufficient inertia to withstand transients, i.e. react slowly after AOOs.

In an *underground mine*, as stated before, after a failure of the ventilation system is detected, actions (e.g. switch over to a backup system or evacuation of the mine) are required to protect the workers against radiation exposure. Depending on the speed of build-up of radon concentrations in the mine without ventilation, an adequate grace period for necessary actions needs to be defined.

In a milling facility, after detecting excessive concentrations of dust in the air, or after detecting excessive radiation from equipment due to an accumulation of radioactive dust, or due to a leakage of radioactive material from the chemical processing unit, alarms are expected to be given and timely corrective actions have to be initiated by the operator. Such corrective actions may require the shutdown of the facility (including evacuation of workers) to minimize radiation exposure.

The grace period available to the operator for each AOO needs to be defined within the design analysis.

The **acceptance limit AL2.2** of **CR2.2** is met if evidence available to the INPRO assessor shows that an adequate grace period for each AOO has been determined in the design analysis for the facility assessed.

6.3.4. User requirement UR3: Accidents

The rationale of UR3 was provided in Section 5.5. UR3 for mining and milling facilities deals with accidents[46]. Examples of accidents for mining and milling facilities include:

- In an underground mine, a complete failure of the ventilation system;
- In a milling facility, a rupture of components (pipes, vessels) in the chemical processing unit of the milling facility with subsequent (large) spillage of radioactive and/or chemically toxic material;
- A fire in an underground mining facility;
- A fire in a milling facility; and
- A loss of the integrity of the tailings (storage and disposal) facility due to external hazards such as flooding (or dam break) with a significant release of solid and/or liquid radioactive and chemically toxic material to the environment[47].

Other external hazards (defined in Sections 4.2.1 and 4.2.6), such as earthquakes, flooding, etc, can also lead to accidents in all types of mines and milling facilities. As stated before, the facilities need to be designed against both external and internal hazards.

The criteria selected for user requirement UR3 are presented in Table 8.

TABLE 8. CRITERIA FOR USER REQUIREMENT UR3

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR3: Accidents: The frequency of occurrence of accidents in the mining/ milling facility assessed is reduced. If an accident occurs, engineered safety features and/or operator actions are able to restore the facility assessed to a controlled state, and subsequently to a safe state, and the consequences are mitigated to ensure the confinement of nuclear and/or toxic chemical material. Reliance on human intervention is minimal, and only required after sufficient grace period.	CR3.1: Frequency of accidents	IN3.1: Calculated frequency of occurrence of accidents. AL3.1: Lower than that in the reference design.
	CR3.2: Engineered safety features and operator procedures	IN3.2: Reliability and capability of engineered safety features and/or operator procedures. AL3.2: Superior to those in the reference design.
	CR3.3: Grace periods for accidents	IN3.3: Grace periods for accidents until human intervention is necessary. AL3.3: Longer than those in the reference design.
	CR3.4: Barriers	IN3.4: Number of confinement barriers maintained (intact) after an accident. AL3.4: At least one.

6.3.4.1. Criterion CR3.1: Frequency of accidents.

Indicator IN3.1: Calculated frequency of occurrence of accidents.

Acceptance limit AL3.1: Lower than that in the reference design.

Examples of mining and milling facility accidents with potential radiological and chemical hazards, including their tailings storage and disposal facilities, have been presented above

[46] Design basis accidents have not been defined for some mining and milling facilities.

[47] Large releases of radioactive materials (together with toxic chemicals) to the environment are normally considered in Levels 4 and 5 of the defence in depth concept. Due to the low radioactivity of tailing materials and the slow movement of the radioactive materials (and toxic chemicals), this situation is considered here as a DBA whose consequences are assumed to be kept within regulatory limits in view of the long time available for remedial actions.

(beginning of Section 6.3.4). Deterministic considerations[48] of potential accidents in mining and milling facilities have to be complemented by probabilistic analysis. It is expected that further development of probabilistic methods and tools applicable to mining and milling facilities will enable an expansion of the scope of probabilistic safety assessment to eventually cover all major hazards, initiating events and scenarios.

Accidents are expected to have very low frequencies (similar to the DBA frequencies in a modern NPP) and the value of the frequency needs to be confirmed by a probabilistic safety analysis covering both internal and external hazards. The calculated frequency of accidents caused by external hazards can be influenced by the designer, e.g. by increasing the robustness of the confinement wall (building for milling etc.), and by the future owner/operator of the facility by selecting an appropriate site (see UR5).

The **acceptance limit AL3.1** of **CR3.1** is met if evidence available to the INPRO assessor shows that based on probabilistic analyses the frequencies of accidents in the facility assessed is lower than for the reference facility. If quantitative results are not available, deterministic analysis needs to be developed that supports low frequencies based on an increase of design robustness, high quality of operation, intelligent inspection and maintenance programs, and advanced I&C systems.

6.3.4.2. Criterion CR3.2: Engineered safety features and operator procedures

Indicator IN3.2: Reliability and capability of engineered safety features and/or operator procedures.

Acceptance limit AL3.2: Superior to those in the reference design.

Engineered safety features (automatic) are expected to be designed and installed in the facility. After detection of the accident these features need to be capable of controlling the accident, restoring the facility to a controlled state, and keeping the radiological consequences of the accident within authorized limits. To assure necessary reliability, these features need to be designed with sufficient levels of redundancy, diversity and independence.

In case automatic systems are not available, adequate operator procedures are necessary. Redundant, diverse and independent passive and active systems are deemed more reliable than administrative controls (operator interventions) but it is acknowledged that they are difficult to develop for mining/ milling facilities.

Examples of safety features in mining/ milling facilities are as follows. In a milling facility the safety systems can be available that detect a rupture of equipment with subsequent large spillage of radioactive and chemically toxic material and thereupon provide an alarm to initiate the necessary corrective actions. Operator procedures and corresponding equipment (e.g. an emergency exhaust system) need to be available to mitigate the consequences of this kind of accident.

In case of a fire in a mining/ milling facility, (automatic) firefighting systems (e.g. spray systems) can be available that can extinguish the fire. In case of detection of a large leakage from the tailings storage and disposal facility, a program needs to be initiated to stop the leakage and perform remediation of the environment [88].

[48] Deterministic consideration in operating mining and milling facilities are normally based on a hazard and operability study (HazOp), which is a structured and systematic evaluation that involves the use of expert judgements on the design, processes and procedures to reveal potential hazards and operability issues.

As mentioned above, the facility is also expected to have engineered safety features protecting against external hazards (see Section 4.2.1 and 4.2.6), e.g. shock absorbers and dampers for safety related equipment to mitigate the effects of an earthquake.

The **acceptance limit AL3.2** of **CR3.2** is met if evidence available to the INPRO assessor shows that the reliability and capability of engineered safety features (automatic systems) and/or operator procedures in the facility assessed is superior to that in the reference design and assure that after the beginning of an accident the necessary actions to mitigate the accident consequences will be initiated in time to prevent an accidental release of nuclear material and/or toxic chemicals from the facility. Alternatively, if a reference facility cannot be found, it needs to be demonstrated that the design of the facility assessed involves best international practice and is therefore state of the art.

6.3.4.3. Criterion CR3.3: Grace periods for accidents

Indicator IN3.3: Grace periods for accidents until human intervention is necessary.

Acceptance limit AL3.3: Longer than those in the reference design.

An explanation of 'adequate grace period' is provided in section 6.3.3.2 as introduced earlier for control of AOOs (see CR2.2) in Level 2 of DID. For accidents (caused by events associated with internal or / and external hazards) the criterion requires that the system response and/or automatic actions of active and/or passive safety systems provide an adequate grace period for the operator to intervene.

In case of a complete failure of the ventilation system, the workers in an underground mine have to be evacuated to the surface to avoid excessive inhalation of radon[49]. This action has to be initiated by automatic systems or the mine operators before excessive inhalation can occur in view of the speed of radon build-up in the mine.

After detection of large spills of radioactive (and chemically toxic) material in the chemical processing unit of the milling facility, automatic systems (ventilation, exhaust system) are expected to mitigate this accident before corrective actions initiated by the facility operator. The operator intervention needs to start after the detection of this accident within a grace period defined for this accident in the design documentation.

After detection of a fire in the facility, within a grace period defined for this accident in the design documentation in addition to the automatic systems the operator is expected to perform the necessary actions to protect the workers.

The potential migration of radionuclides (and toxic chemicals) from a tailing storage and disposal facility through the surrounding soil after a leakage event needs to be analysed within an environmental impact assessment (EIA) during the site licensing process. Usually, it takes a long time (e.g. a few months to several years) for radioactive (and chemically toxic) material to reach the public domain. It is recognized that the migration time would depend upon the type of soil and ground conditions (water table, etc.) at a given tailings facility site, and will vary from site to site. A uranium mining and milling facility site is supposed to have a system for detecting leakages by monitoring the radioactivity in bore wells and water bodies in the vicinity of the tailing facility. Thus, after detection of a large leakage, a program for corrective (remedial) actions has to be initiated by the operator within a grace period that is evaluated in

[49] Other factors, e.g. non-radioactive gases, dust, aerosols, etc, can be more dangerous; however, the INPRO methodology is focused primarily on nuclear safety and radiation protection.

the facility design analysis and shown to be sufficient to avoid excessive contamination of the environment.

The grace periods have to be defined for each accident within the design analyses.

The **acceptance limit AL3.3** of **CR3.3** is met if evidence available to the INPRO assessor shows that in the facility assessed has longer accident grace periods than the reference design. Alternatively, if a reference design is not available, it needs to be demonstrated that the design of the facility assessed took available information on best international practice into account and is therefore state of the art.

6.3.4.4. Criterion CR3.4: Barriers

Indicator IN3.4: Number of confinement barriers maintained (intact) after an accident.

Acceptance limit AL3.4: At least one.

An accident with a loss of integrity of the tailing pond dam of a tailings storage and disposal facility would result in a loss of the single confinement barrier and lead to a large release of low radioactive and chemically toxic material into the environment. Geotechnical monitoring to detect the movements may allow sufficient time for correcting them before dam failure or for repairing the dam of the tailing storage and disposal facility and for remedial actions to avoid further contamination. Ideally, it might be worthwhile to consider the design of a double barrier for a tailings storage and disposal facility. This system needs to have monitoring systems in between the two barriers which would cause an alarm if radioactivity (and/or toxic chemicals) were to penetrate the first barrier. Such a double barrier would ensure that always one barrier is intact as required by Level 3 of the defence in depth concept.

For the chemical processing unit in a milling facility, the building needs to be designed as a confinement that prevents accidental releases of spilled radioactive (and chemically toxic) materials to the outside.

The **acceptance limit AL3.4** of **CR3.4** is met if evidence available to the INPRO assessor shows that engineered safety features and/or operator procedures are adequately defined and able to keep the accident consequences within design limits.

Examples: (1) The assessed mine tailings storage and disposal facility is monitored and potential breaches are effectively isolated in time to prevent excessive releases to the environment. (2) In a milling facility, the chemical processing unit building is able to contain major spills of radioactive (and chemically toxic) materials under the accident conditions and prevent their release to the environment.

6.3.5. User requirement UR4: Severe plant conditions

User requirement UR4 was found to be not directly applicable to a mining and milling facility.

6.3.6. User requirement UR5: Inherent safety characteristics

Rationale of UR5 was provided in Section 5.7. The criterion selected for user requirement UR5 is presented in Table 9.

TABLE 9. CRITERION FOR USER REQUIREMENT UR5

User requirement	Criterion	Indicator (IN) and Acceptance Limit (AL)
UR5: Inherent safety characteristics: To excel in safety and reliability, the mining/ milling facility assessed strives for elimination or minimization of some hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics.	CR5.1: Minimization of hazards	IN5.1: Examples of hazards: fire, flooding, release of radioactive material, radiation exposure, etc. AL5.1: Hazards minimized according to the state of the art.

6.3.6.1. Criterion CR5.1: Minimization of hazards

Indicator IN5.1: Examples of hazards: fire, flooding, release of radioactive material, radiation exposure, etc.

Acceptance limit AL5.1: Hazards minimized according to the state of the art.

A mining/ milling facility and its tailings facility may have to define a design basis event with flooding and consequent release of radioactive (and chemically toxic) material to the environment. Thus, if the facility site has no upstream dams and no catchment areas, the facility would be superior from a safety point of view in this particular aspect (flooding), because a potential hazard would be eliminated. Similarly, other external hazards can be reduced for new facilities by appropriate siting.

Using fire resistant materials and reducing the amount of burnable material in a mine and milling facility would reduce the hazard of a fire. Eliminating specific dangerous chemicals from the ore processing technology would eliminate the hazard of their release.

The hazard of a release of radioactive (and chemically toxic) material could be reduced in the tailings storage and disposal facility by increasing the robustness of the barriers and applying a passive approach such as earthen covers or a permanent water pond over the tailings[50].

The **acceptance limit AL5.1** of **CR5.1** is met if evidence available to the INPRO assessor shows that hazards in the NFCF assessed have been minimized by applying the state of the art technology.

6.3.7. User requirement UR6: Human factors related to safety

Descriptions of the user requirement UR6 and corresponding criteria are common for all NFCFs discussed in this report (i.e. mining/ milling, conversion, enrichment, fuel fabrication, spent fuel storage and reprocessing). The rationale of UR6 was provided in Section 5.8. There are two aspects of safety covered in this user requirement. The first one is focused on the design of equipment related to safety to minimize human errors, and the second one covers the attitude to safety of people in nuclear facilities and related organizations.

The criteria selected for user requirement UR6 are presented in Table 10.

[50] The approach using permanent water ponds may be applicable only in some specific climate conditions and locations.

TABLE 10. CRITERIA FOR USER REQUIREMENT UR6

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR6 Human factors related to safety: Safe operation of the mining/ milling facility assessed is supported by accounting for HF requirements in the design and operation of the facility, and by establishing and maintaining a strong safety culture in all organizations involved in the life cycle of the facility.	CR6.1: Human factors	IN6.1: Human factors addressed systematically over the life cycle of the mining/ milling facility assessed. AL6.1: Evidence is available.
	CR6.2: Attitude to safety	IN6.2: Prevailing safety culture. AL6.2: Evidence is provided by periodic safety reviews.

6.3.7.1. Criterion CR6.1: Human factors

Indicator IN6.1: Human factors addressed systematically over the life cycle of the mining/ milling facility assessed.

Acceptance limit AL6.1: Evidence is available.

In the lifecycle of mining and milling facilities and other NFCFs, humans are considered as a valuable resource that plays important roles in the design, construction, commissioning, operation, testing, maintenance and inspections, and decommissioning of the facilities. However human interventions have limited reliability and may create unnecessary disturbances that have to be analysed in the facility design to achieve a sufficient level of safety.

Human factors are important for safe and reliable operation of mining and milling facilities and the designer of a new facility is expected to place increased emphasis on them to minimize the possibilities for the human errors during plant normal operation to initiate an incident or accident or contribute to the failure of backup (safety) systems. The possibilities for human errors committed during incident or accident scenarios to aggravate the scenarios and their consequences also need to be minimized. As a common principle it has to be ensured that:

- The functions assigned to personnel constitute consistent tasks and correspond to the abilities and strengths of the personnel (e.g. appropriate number of tasks and sharing among centralized and local operating actions);
- The human-systems interface (i.e. control means, processing of information to be presented to the operators) supports the tasks of personnel and minimizes the potential for error.

Addressing human factors in the design of safety related equipment and the radiation protection program (RPP) in mining and milling facilities and other NFCFs will increase the level of safety. Human errors during the facility operation, including maintenance, inspections and tests, and decommissioning need to be considered in the facility safety analysis.

The training programmes that have to be developed and implemented in the mining and milling facility are discussed in the criterion CR1.2.

The **acceptance limit AL6.1** of **CR6.1** is met if evidence available to the INPRO assessor shows that human factors were addressed in the design and the RPP of the mining/milling facility assessed.

6.3.7.2. Criterion CR6.2: Attitude to safety

Indicator IN6.2: Prevailing safety culture.

Acceptance limit AL6.2: Evidence is provided by periodic safety reviews.

Safety culture is discussed in this report in Section 5.7.

The periodic reviews concerning safety culture are expected to cover not only the operating organization but also regulatory and other responsible government authorities as well as industrial entities. The assessment of this criterion CR6.2 is based on the outcome of safety culture reviews of at least the following organisations: operating organisation, facility / installation developer and supplier, and regulatory authority.

The assessment of CR6.2 regarding safety culture of an operating organisation can only be performed once an organization is actually operating a facility. But the need to inculcate a safety culture within an organization and the need for a safety management system needs to be recognized in the planning phase for an NFCF. Furthermore, the proposed policies and management structure of the owner/operator can be assessed, prior to operation, to determine if they are consistent with safety culture.

The **acceptance limit AL6.2** (*evidence that a safety culture prevails*) of **CR6.2** is met for the NFCF assessed if evidence available to the INPRO assessor shows that such reviews are being (planned to be) performed at appropriate intervals. The INPRO methodology recommends using the support of experienced organizations for such reviews.

6.3.8. User requirement UR7: RD&D for advanced designs

A description of the user requirement UR7 and corresponding criteria are common for all NFCFs discussed in this report (i.e. mining/ milling, conversion, enrichment, fuel fabrication, spent fuel storage and reprocessing). The rationale of UR7 was provided in Section 5.9. The user requirement UR7 discusses the necessary RD&D effort for developing a facility with primarily innovative[51] but also evolutionary[52] design features.

The criteria selected for user requirement UR7 are presented in Table 11.

TABLE 11. CRITERIA FOR USER REQUIREMENT UR7

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR7: RD&D for advanced designs: The development of innovative design features of the mining/ milling facility assessed includes associated RD&D to bring the knowledge of facility characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for operating facilities.	CR7.1: RD&D	IN7.1: RD&D status. AL7.1: RD&D defined, performed and database developed.
	CR7.2: Safety assessment	IN7.2: Adequate safety assessment. AL7.2: Approved by a responsible regulatory authority.

6.3.8.1. Criterion CR7.1: RD&D

Indicator IN7.1: RD&D status.

Acceptance limit AL7.1: RD&D defined, performed and database developed.

RD&D on the reliability of innovative components and systems of an NFCF needs to be performed to achieve a thorough understanding of all relevant physical and engineering phenomena required to support the safety assessment.

[51] An innovative design is an advanced design that incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice.
[52] An evolutionary design is an advanced design that achieves improvements over existing designs through small to moderate modifications, with a strong emphasis on maintaining proven design elements to minimize technological risks.

At least the following criteria are expected be met by the RD&D program of a developer for an innovative design (but also for an evolutionary design):

- Significant phenomena associated with the innovative technologies used in NFCF and affecting safety were identified, understood, modelled and simulated (this includes the knowledge of uncertainties, and the effect of scaling and environment);
- Safety-related system or component behaviour was modelled with acceptable accuracy, including knowledge of all safety-relevant parameters and phenomena, and validated with a reliable database.
- The necessity of using a pilot facility in the development process was clarified.

The **acceptance limit AL7.1** of **CR7.1** is met if evidence available to the INPRO assessor shows that for an NFCF with innovative (or evolutionary) design features relevant to safety sufficient RD&D has been performed prior to start-up of the facility.

6.3.8.2. Criterion CR7.2: Safety assessment

Indicator IN7.2: Adequate safety assessment.

Acceptance limit AL7.2: Approved by a responsible regulatory authority.

A safety case of a facility with innovative or evolutionary features needs to be established based on a comprehensive safety assessment that meets national and international standards [27, 79, 89–91]. Where appropriate, a risk informed approach is expected to be adopted in the design, construction and operation of NFCFs. In line with the risks involved, the emphasis needs to be more on long term effects on environment and public [73, 90, 92–94].

The **acceptance limit AL7.2** of **CR7.2** is met if evidence available to the INPRO assessor shows that an adequate safety assessment has been performed for the facility assessed and was accepted by the responsible regulatory authority in the country of origin.

7. URANIUM REFINING/CONVERSION AND ENRICHMENT

In this section, firstly, a short description of the main processes in U refining/ conversion and uranium enrichment facilities is given and the corresponding specific safety issues are discussed. Secondly, the assessment method is presented based on the corresponding criteria of the INPRO methodology in the area of safety, which have been, where necessary, adapted to the specific issues of these two kinds of NFCFs.

As of end of 2016 the following countries have uranium refining/ conversion facilities in operation: Argentina (Pilcaniyeu), Canada (Port Hope), China (Lanzhou), France (Malvesi, Pierrelatte), Iran (Isfahan), Russian Federation (Glazov; Seversk, Angarsk), USA (Metropolis).

Countries with uranium enrichment facilities operating (or under construction) at the end of 2016 are: Argentina (Pilcaniyeu), Brazil (Resende), China (Lanzhou, Hanzhong), France (Tricastin), Germany (Gronau), India (Karnataka), Iran (Natanz, Qom), Japan (Rokkasho), Netherlands (Almelo), Pakistan (Kahura), Russian Federation (Novouralsk, Seversk, Zelenogorsk, Angarsk), United Kingdom (Capenhurst), USA (Lea County).

7.1. URANIUM REFINING AND CONVERSION TO HEXAFLUORIDE UF_6

The end product from the mining and milling stage of the fuel cycle is called 'yellow cake', which is essentially an impure uranium compound.

Refining or purification processes are required to bring the uranium to the standard of purity necessary for nuclear reactor fuel element manufacture. The various stages in the purification process are dissolution, solvent extraction, concentration and thermal de-nitration to uranium trioxide (Fig.6). The processes are typical for those of a chemical industry handling a chemically toxic rather than a radioactive material, the toxicity of natural uranium being about the same as that of lead.

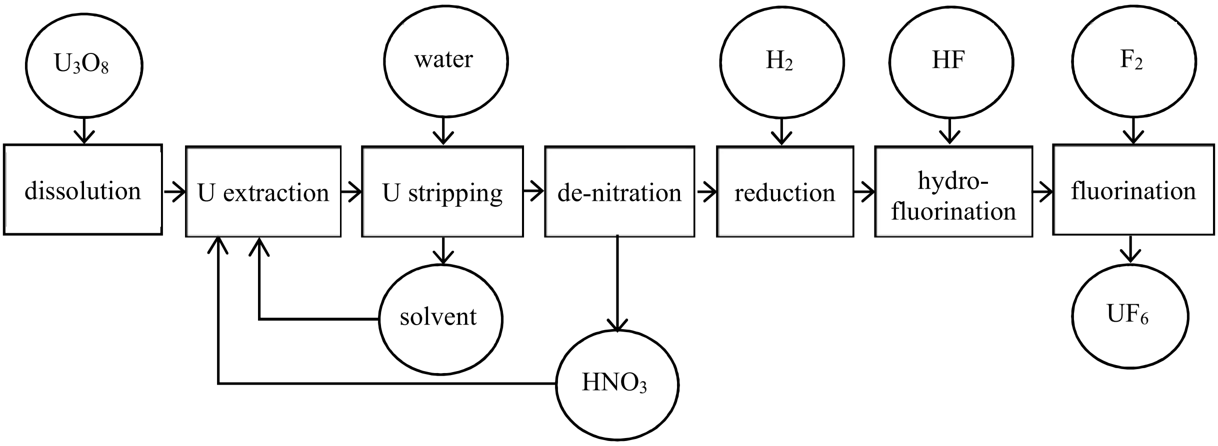

FIG. 6. Flowchart of uranium refining and conversion.

For the *conversion* process, i.e. production of uranium hexafluoride, there are several methods: a batch process using chlorine tri-fluoride; direct fluorination; and a modern method using pure uranium tri-oxide, UO_3, which is converted to UF_4, uranium tetra-fluoride, then to uranium hexafluoride, UF_6, prior to enrichment. The chemical reactions [31] are:

$$UO_3 \xrightarrow{H_2} UO_2 \xrightarrow{HF} UF_4 \xrightarrow{F_2} UF_6 \quad (1)$$

The reactions are generally carried out using fluidized bed technology. A typical flowchart of uranium refining and conversion is shown in Fig. 6.

For details on the conversion processes see Ref [95].

7.2. URANIUM ENRICHMENT

Natural uranium primarily contains two isotopes, ^{238}U (99.3 %) and ^{235}U (0.7 %). The concentration of ^{235}U, the fissionable isotope in uranium, needs to be increased to 3 to 5 % for use as a nuclear fuel in LWRs.

The uranium enrichment can be performed in several ways:

- Electromagnetic isotope separation (EMIS);
- Thermal diffusion;
- Aerodynamic uranium enrichment process;
- Chemical exchange isotope separation;
- Ion exchange process;
- Plasma separation process;
- Gaseous diffusion process;
- Gas centrifuge process, and
- Laser isotope separation.

Of these, gaseous diffusion processes and gas centrifuge processes are currently used in commercial industries[53]. However, no new diffusion plants are being built because the centrifuge plants are more economical. For details on the enrichment processes see Refs [96–98].

7.2.1. Gaseous diffusion process

Uranium arrives at the plant in the form of solid UF_6. It is vaporized and advantage is taken of the difference in the molar masses of the isotopes to separate them selectively by passage of UF_6 through a porous wall, a 'barrier'. The lighter isotope ^{235}U passes more easily through the wall than ^{238}U. Because enrichment by means of a single barrier is very small, it is necessary to repeat the operation a great number of times. The elementary unit in enrichment is the stage, which is composed of a diffuser containing barriers; a compressor which forces the UF_6 to pass through the barriers and an exchanger, which removes the heat generated by the compressor. The stages are placed in a series. The part of the flux that passes through the barrier goes to the following stage; the part that does not pass is directed towards the lower stage. The stages are linked together of ten to twenty units that constitute a group. Several groups constitute a cascade. UF_6 is introduced into the centre of the cascade. The UF_6 that has been enriched in uranium ^{235}U is withdrawn at one end and the depleted UF_6 at the other. One of the disadvantages in gaseous diffusion process is the heavy use of electricity.

7.2.2. Gas centrifuge process

In the gas centrifuge uranium-enrichment process, gaseous UF_6 is fed into a cylindrical rotor that spins at high speed inside an evacuated casing. Because the rotor spins so rapidly, centrifugal force results in the gas occupying only a thin layer next to the rotor wall, with the gas moving at approximately the speed of the wall. Centrifugal force also causes the heavier $^{238}UF_6$ molecules to tend to move closer to the wall than the lighter $^{235}UF_6$ molecules, thus partially separating the uranium isotopes. This separation is increased by a relatively slow axial counter-current flow of gas within the centrifuge that concentrates enriched gas at one end and

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[53] A commercial laser isotope separation facility is currently (2013) under construction in the USA.

depleted gas at the other. This flow can be driven mechanically by scoops and baffles or thermally by heating one of the end caps. A schematic diagram of the gas centrifuge is shown in Fig. 7.

The main subsystems of the centrifuge are rotor and end caps; top and bottom bearing/suspension system; electric motor and power supply (frequency changer); centre post, scoops and baffles; vacuum system; and casing. Because of the corrosive nature of UF_6, all components that come in direct contact with UF_6 need to be fabricated from, or lined with, corrosion-resistant materials. The separative capacity of a single centrifuge increases with the length of the rotor and the rotor wall speed. The primary limitation on rotor wall speed is the strength-to-weight ratio of the rotor material. Another limitation on rotor speed is the lifetime of the bearings at either end of the rotor. Balancing of rotors to minimize their vibrations is especially critical to avoid early failure of the bearing and suspension systems. Because perfect balancing is not possible, the suspension system needs to be capable of damping some amount of vibration.

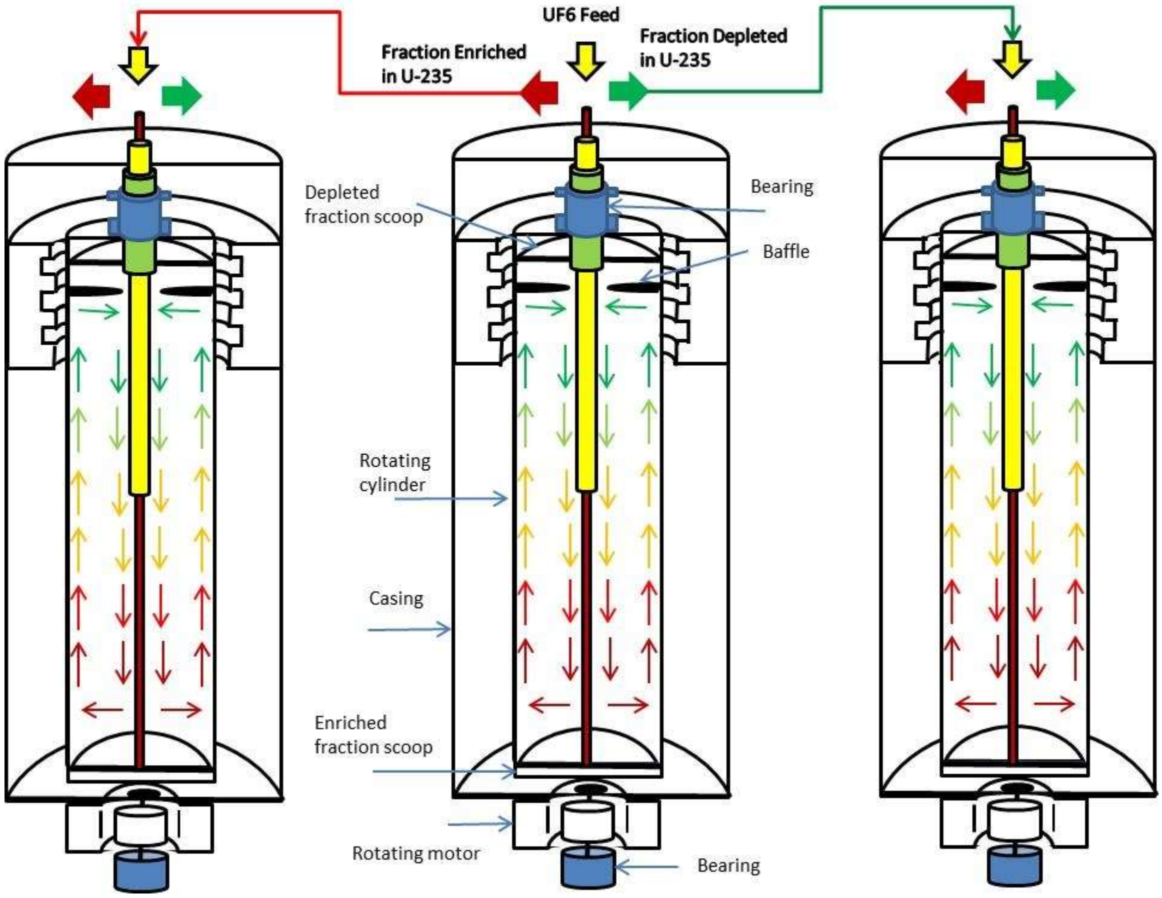

FIG. 7. Gas centrifuge process.

One of the key components of a gas centrifuge enrichment plant is the power supply (frequency converter) for the gas centrifuge machines. The power supply is supposed to accept alternating current (AC) input at the 50 or 60 Hz line frequency available from the electric power grid and provide an AC output at a much higher frequency (typically 600 Hz or more). The high-frequency output from the frequency changer is fed to the high-speed gas centrifuge drive motors (the speed of an AC motor is proportional to the frequency of the supplied current). The centrifuge power supplies are expected to operate at high efficiency, provide low harmonic distortion, and provide precise control of the output frequency.

The casing is needed both to maintain a vacuum and to contain the rapidly spinning components in the event of a failure. If the shrapnel from a single centrifuge failure is not contained, a 'domino effect' may result and destroy adjacent centrifuges. A single casing may enclose one or several rotors. The enrichment effect of a single centrifuge is small, so they are linked together by pipes into cascades, to obtain the required enrichment. Once started, a modern centrifuge runs for more than 10 years with no maintenance.

7.2.3. Enrichment of UF_6 resulting from uranium recovered after reprocessing

The enrichment of ^{235}U in reprocessed uranium (RepU) fuel has to be higher than in standard fuel, in order to compensate for the decrease in reactivity due to the presence of ^{234}U and ^{236}U. The use of RepU has a major impact on the choice of an enrichment process. If fuel is fabricated using only reprocessed uranium, the gas centrifuge process is better suited than the gaseous diffusion for enrichment of uranium, particularly because of the modular installations with relatively small capacity in gas centrifuge process. In addition, the modules are easier to cleanse of ^{234}U and ^{236}U than those of a gaseous diffusion plant.

7.2.4. Laser separation process

Isotopic separation of uranium can be achieved based on photo excitation principles (exciting the molecules using laser light). Such technologies have been named Atomic Vapor Laser Isotope Separation (AVLIS), Molecular Laser Isotope Separation (MLIS), and Separation of Isotopes by Laser Excitation (SILEX). They are based on the difference in the ionization energies of the isotopes of a given element. A laser beam illuminates vapour of uranium metal or uranium metal alloy and selectively ionizes the atoms of ^{235}U, removing an electron from each and leaving them with a positive charge. ^{235}U is then collected on negatively charged plates. Neutral ^{238}U condenses on collectors on the roof of the separator.

7.3. SAFETY ISSUES IN REFINING/ CONVERSION AND ENRICHMENT FACILITIES

The safety issues related to refining/ conversion and enrichment facilities are documented in detail in the IAEA safety standards [17, 32].

Generally in a *refining/ conversion facility*, only natural (or slightly enriched uranium from reprocessing) is processed. The radiotoxicity of this material is low, and thus the expected off-site radiological consequences following potential accidents are limited. However, in conversion facilities that process uranium with a ^{235}U concentration of more than 1 % (using reprocessed uranium or scrap uranium pellets as feed material), criticality can also be a hazard.

The existing processes for natural uranium refining and conversion to hexafluoride usually give rise to no significant radiological hazards, and the safety problems associated with the handling of this material are essentially those of a conventional chemical industry dealing with toxic, corrosive, combustible and/or explosive chemicals such as hydrofluoric acid (HF) and fluorine (F_2) [32]. Hazards of uranium hexafluoride (UF_6) handling, storage and transportation are common to several stages of the nuclear fuel cycle, such as conversion, enrichment and fuel fabrication [99].

The conversion to UF_6 of uranium recovered from the reprocessing of spent fuel from power reactors could give rise to an increase in radiological hazards (external exposure) associated with UF_6 handling. This uranium product contains small quantities of plutonium, other actinides (^{232}U) and fission products [32]. The latter may accumulate in some parts of the process, particularly in the hexafluoride production stage. Assessment and control of these hazards can be made by continuous monitoring of the radioactivity in the process vessels, although there is clearly no potential for a rapid increase in the activity levels.

The main credible hazard in an enrichment facility is a potential release of UF_6. There is also a criticality hazard due to the handling of U with more than 1 % of ^{235}U. As in conversion facilities there is an increased hazard of external radiation exposure in situations involving reprocessed U [32].

The refining/ conversion and enrichment process relies to a large extent on operator intervention and administrative controls to ensure safety, in addition to active and passive engineered safety measures.

In the following the main hazards, i.e. UF_6 release, internal and external exposure, criticality, leaks of toxic chemicals, fire and explosions, and external hazards in a refining/ conversion and enrichment facility are discussed in more detail.

7.3.1. UF_6 release

Failures of vessels, gaskets, valves, instruments or lines can result in either liquid or gaseous UF_6 release. These failures can be caused by corrosion, mechanical damage, mal-operation of the system, or overheating of all or part of the system.

Four types of cylinders with capacities ranging from 2 to 12 t are commonly used for storage and transport of UF_6, which is then in solid form. These cylinders need to be heated up for the transfer of UF_6 into its gaseous form. Though UF_6 is not in itself flammable, if a container were present in a fire, the container could explode by virtue of the internal stresses built up in the container and spread its contents over a wide area. The presence of oil or other (organic) impurities in storage cylinders or process equipment can lead to an exothermic reaction, which might give rise to a UF_6 release [99]. The chemical toxicity of uranium in soluble form such as UF_6 is more significant than its radiotoxicity.

At room temperature, at which it is handled and stored, UF_6 is a colourless, crystalline solid with a significant but low vapour pressure. When heated at atmospheric pressure to facilitate transfer, the crystals sublime without melting and the vapour pressure reaches 760 mm Hg at a temperature of about 56°C. At higher pressures, the crystals will melt, at a temperature of about 64°C and this melting is accompanied by a very substantial increase in specific volume.

Uranium hexafluoride is a highly reactive substance. It reacts chemically with water, forming soluble reaction products with most organic compounds and with many metals. Its reactivity with most saturated fluorocarbons is very low. It does not react with oxygen, nitrogen, or dry air.

The prime hazard following a UF_6 release arises from the reaction between UF_6 and moisture, which is normally present in the atmosphere producing two toxic substances hydrofluoric acid (HF) and uranyl fluoride (UO_2F_2) according to the equation:

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF \tag{2}$$

With gaseous UF_6, this reaction proceeds rapidly liberating some heat and is accompanied by a substantial volume increase at atmospheric pressure. Both UO_2F_2 and HF are toxic. Deposition of UO_2F_2 from a cloud formed following a release to the outside of the facility could result in the contamination of agricultural crops and grassland. The rate at which deposition will occur and hence the contamination contours will be very dependent on atmospheric conditions at the time of release. Calculation of the dispersion of toxic material following a UF_6 release is complicated by virtue of the high density levels of some of the products and the chemical reactions which occur.

UF_6 leakage needs to be restricted to less than 0.2 mg/m^3 (= chemical toxicity limit for natural U and up to 2.5 % enrichment).

There have been several accidents involving UF_6. As early as 1944, in the United States, a weld ruptured on an 8 ft (~240 cm) long cylinder containing gaseous natural UF_6 that was being heated by steam. An estimated 400 lb (~181 kg) of UF_6 was released, which reacted with steam from the process and created HF and uranyl fluoride. This accident resulted in two deaths from HF inhalation and three individuals seriously injured from both HF inhalation and uranium toxicity. Another UF_6 accident involving a cylinder rupture occurred at a commercial uranium conversion facility (Sequoya Fuels Corp., USA) in 1986. The accident occurred when an over-loaded shipping cylinder was reheated to remove an excess of UF_6. The cylinder ruptured, releasing a dense cloud of UF_6 and its reaction products. This accident resulted in the death of one individual from HF inhalation.

7.3.2. Internal and external radiation exposure

Inhalation of uranium compounds leads to internal radiation exposure. Depending on the solubility and diameters of particulates, inhalations of different uranium compounds give different doses. Ref [100] provides effective dose coefficients for inhaled particulates for workers which can be used to estimate the activity and mass of uranium that give 20 mSv on inhalation. The result of such estimations for the case of activity median aerodynamic diameters of particulates equal to 5 µm is presented in Table 12.

TABLE 12. ACTIVITY AND MASS OF URANIUM WHICH GIVE 20 mSv ON INHALATION

Class (depending on solubility)	Activity[1] (Bq)	Mass (mg)	
		natural U	3.5 % enriched U
S Class – UO_2, U_3O_8, U	3509	140	33
M Class – UO_3, UF_4, MDU, ADU	12500	503	115
F Class – UF_6, UO_2F_2, $UO_2(NO_3)_2$	34482	1379	316

Note:
[1] – activity is subjected to small variations with respect to enrichment but for the purpose of INPRO assessment can be considered as independent of enrichment.

The natural and enriched uranium but also the depleted uranium in a UF_6 cylinder emits neutron and gamma radiation from thorium and its progenies. In case of reprocessed uranium in addition to the presence of Pu and fission products the build-up of ^{232}U can lead to radiation exposure.

In the proximity of cylinders carrying depleted uranium, up to 20 % of the radiation exposure can be due to the neutron radiation. In the proximity of cylinders carrying enriched uranium, up to 70 % of the radiation exposure can be due to neutron radiation.

7.3.3. Criticality

Depending upon the concentration, mass and enrichment, degree of moderation and reflection, and presence of appropriate neutron absorbers, material with enriched U can attain criticality in some geometries. Hence safe geometries need to be ensured in the facility based on specific criticality analyses. Criticality is not possible with gaseous UF_6. At low enrichments of less than 1 % of ^{235}U, even liquids with moderation do not go critical. For higher enrichments, moderation is important. Typically at 7 % enrichment, the H/U atom ratio needs to be kept below 0.38 (see also Section 4.2.2). Criticality monitoring and alarm systems are mandatory in such an NFCF.

7.3.4. Containment of radioactive material and/or toxic chemicals

Inside the facility, leaks of radioactive material [32] such as uranium solutions and powders, gaseous or liquid UF_6 and toxic chemicals such as HF, F_2, NH_3, ClF_3 from systems consisting of vessels, pumps, valves and pipes can lead to dispersion (and unnecessary generation of waste). Leaks of hydrogenous fluids (water, oil, etc.) leading to flooding can adversely affect criticality safety parameters such as reflection and moderation. Leaks of flammable gases (e.g. H_2) or liquids can lead to explosions and/or fires.

Leak detection systems are expected to be deployed in locations where leaks could occur. Vessels containing significant amounts of nuclear material in solution form need to be equipped with level detectors and alarms to prevent overfilling and with secondary containment features such as bunds or drip trays of appropriate capacity and configuration to ensure criticality safety (see also Section 4.2.2).

To prevent a release of radioactive material and/or toxic chemical to the outside of the plant several barriers (combinations of 'static' and 'dynamic' containments) are necessary: The first barrier is the casing of the equipment (e.g. wall of a cylinder, vessel or pipe), and the second barrier is the building wall of the facility. Additionally, dynamic containments are to be provided by ventilation systems in process equipment and in the working area of the facility[54].

7.3.5. Fire and explosions

Detailed recommendations on the consideration of fire and explosions in refining/ conversion facilities and enrichment facilities design are provided in Ref [32]. Fire in a refining/ conversion and enrichment facility can occur due to the existence of materials such as anhydrous ammonia (explosive and flammable), nitric acid (ignition if in contact with organic materials) and hydrogen. A fire can lead to the dispersion of radioactive and/or toxic material by breaching the containment barriers or may even cause a criticality accident (see also Section 4.2.4).

7.3.6. External hazards

Detailed recommendations on the consideration of initiating events associated with external hazards are provided in Ref [32]. Conversion facilities and enrichment facilities need to be designed for all credible external hazards (see Section 4.2.1), e.g. the design needs to consider an earthquake to ensure that the integrity of the confinement is assured (especially for UF_6 and HF) and no change in criticality parameters such as geometry and moderation is induced (see also Section 4.2).

7.4. ADAPTATION OF THE INPRO METHODOLOGY TO URANIUM REFINING/ CONVERSION AND ENRICHMENT FACILITIES

Adapting the INPRO methodology for use in assessing uranium refining/ conversion and enrichment facilities entails more significant modifications and adjustments than for other types of NFCFs. Although significant technical differences exist between refining/ conversion and enrichment facilities, it was nevertheless found that applying the INPRO methodology to these diverse facilities does not require a separate treatment. The following sections describe how the INPRO methodology in the area of safety is adapted to facilities for U refining/ conversion and enrichment.

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[54] Although ventilation is considered a dynamic barrier in NFCFs and can perform certain containment functions against airborne radionuclides, the effectiveness of ventilation depends on the tightness of static barriers. In the case of significant damage of a static barrier (e.g. building wall collapse), ventilation alone cannot be considered a sufficient protection measure. Moreover, ventilation cannot be considered a barrier for protecting against liquid releases, e.g. in the case of flooding.

7.4.1. INPRO basic principle for sustainability assessment of uranium refining/ conversion or enrichment facility in the area of safety

INPRO basic principle for sustainability assessment of uranium refining/ conversion or enrichment facility in the area of safety: The planned refining/ conversion or enrichment facilities are safer than the respective reference facilities. In the event of an accident, off-site releases of radionuclides and/or toxic chemicals are prevented or mitigated so that there will be no need for public evacuation[55].

The rationale of the BP was provided in Section 5.2. An explanation of the requirement for superiority in the INPRO methodology area of NFCF safety is provided in section 6.3.1. The INPRO methodology has defined a set of requirements for uranium refining/ conversion and enrichment facilities as displayed in Table 13.

TABLE 13. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF URANIUM REFINING/ CONVERSION AND ENRICHMENT FACILITIES IN THE AREA OF NFCF SAFETY

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR1: Robustness of design during normal operation: The uranium refining, conversion or enrichment facility assessed is more robust than the reference design with regard to operation and systems, structures and components failures.	CR1.1: Design of normal operation systems	IN1.1: Robustness of design of normal operation systems. AL1.1: Superior to that in the reference design.
	CR1.2: Subcriticality	IN1.2: Subcriticality margins. AL1.2: Sufficient to cover uncertainties and avoid criticality.
	CR1.3: Facility performance	IN1.3: Facility performance attributes. AL1.3: Superior to those in the reference design.
	CR1.4: Inspection, testing and maintenance	IN1.4: Capability to inspect, test and maintain. AL1.4: Superior to that in the reference design.
	CR1.5: Failures and deviations from normal operation	IN1.5: Expected frequency of failures and deviations from normal operation. AL1.5: Lower than that in the reference design.
	CR1.6: Occupational dose	IN1.6: Occupational dose values during normal operation and AOOs. AL1.6: Lower than the dose constraints.
UR2: Detection and interception of AOOs: The uranium refining, conversion or enrichment facility assessed has improved capabilities to detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions.	CR2.1: I&C systems and operator procedures	IN2.1: I&C system to monitor, detect, trigger alarms and, together with operator actions, intercept and compensate AOOs. AL2.1: Availability of such systems and operator procedures.
	CR2.2: Grace periods for AOOs	IN2.2: Grace periods until human actions are required after AOOs. AL2.2: Adequate grace periods are defined in design analyses.

[55] Other protective measures still may be needed. Effective emergency planning, preparedness and response capabilities will remain a prudent requirement as discussed in the INPRO methodology area of Infrastructure.

TABLE 13. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF URANIUM REFINING/ CONVERSION AND ENRICHMENT FACILITIES IN THE AREA OF NFCF SAFETY (cont.)

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR3: Design basis accidents: The frequency of occurrence of DBAs in the uranium refining, conversion or enrichment facility assessed is reduced. If an accident occurs, engineered safety features and/or operator actions are able to restore the assessed facility to a controlled state and subsequently to a safe state, and the consequences are mitigated to ensure the confinement of nuclear and/or toxic chemical material. Reliance on human intervention is minimal, and only required after sufficient grace period.	CR3.1: Frequency of DBAs	IN3.1: Calculated frequency of occurrence of DBAs. AL3.1: Lower than that in the reference design.
	CR3.2: Engineered safety features and operator procedures	IN3.2: Reliability and capability of engineered safety features and/or operator procedures. AL3.2: Superior to those in the reference design.
	CR3.3: Grace periods for DBAs	IN3.3: Grace periods for DBAs until human intervention is necessary. AL3.3: Longer than those in the reference design.
	CR3.4: Barriers	IN3.4: Number of confinement barriers maintained (intact) after DBAs. AL3.4: At least one.
	CR3.5: Robustness of containment design	IN3.5: Containment loads covered by design of the facility assessed. AL3.5: Greater than those in the reference design.
UR4: Severe plant conditions: The frequency of an accidental release of radioactivity into the environment is reduced. The source term of accidental release into the environment remains well within the envelope of the reference facility source term and is so low that calculated consequences would not require public evacuation.	CR4.1: In-facility severe accident management	IN4.1: Natural or engineered processes, equipment, and AM procedures and training to prevent an accidental release to the environment in the case of accident. AL4.1: Sufficient to prevent an accidental release to the environment and regain control of the facility.
	CR4.2: Frequency of accidental release into environment	IN4.2: Calculated frequency of an accidental release of radioactive materials and/or toxic chemicals into the environment. AL4.2: Lower than that in the reference facility.
	CR4.3: Source term of accidental release into environment	IN4.3: Calculated inventory and characteristics (release height, pressure, temperature, liquids/gas/aerosols, etc) of an accidental release. AL4.3: Remains well within the inventory and characteristics envelope of the reference facility source term and is so low that calculated consequences would not require evacuation of population.

TABLE 13. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF URANIUM REFINING/ CONVERSION AND ENRICHMENT FACILITIES IN THE AREA OF NFCF SAFETY (cont.)

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR5: Inherent safety characteristics: To excel in safety and reliability, the refining, conversion or enrichment facility assessed strives for elimination or minimization of some hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics when appropriate.	CR5.1: Minimization of hazards	IN5.1: Examples of hazards: fire, flooding, release of radioactive material, radiation exposure, etc. AL5.1: Hazards are reduced in relation to those in the reference facility.
UR6: Human factors related to safety: Safe operation of the refining, conversion or enrichment facility assessed is supported by accounting for HF requirements in the design and operation of the facility, and by establishing and maintaining a strong safety culture in all organizations involved in the life cycle of the facility.	CR6.1: Human factors	IN6.1: Human factors addressed systematically over the life cycle of the refining, conversion or enrichment facility. AL6.1: Evidence is available.
	CR6.2: Attitude to safety	IN6.2: Prevailing safety culture. AL6.2: Evidence is provided by periodic safety reviews.
UR7: RD&D for advanced designs: The development of innovative design features of the refining, conversion or enrichment facility assessed includes associated RD&D to bring the knowledge of facility characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for operating facilities.	CR7.1: RD&D	IN7.1: RD&D status. AL7.1: RD&D defined, performed and database developed.
	CR7.2: Safety assessment	IN7.2: Adequate safety assessment. AL7.2: Approved by a responsible regulatory authority.

7.4.2. User requirement UR1: Robustness of design during normal operation

The rationale of UR1 was provided in Section 5.3. UR1 deals with prevention of AOOs. For refining/ conversion and enrichment facilities, examples of AOOs are [32]:

- Leakage (e.g. due to corrosion) of flammable (explosive) gases such as H_2 leading to explosive mixtures in air;
- Leakage of radioactive and/or toxic chemicals such as UF_6, HF, and NH_3; and
- Fire in a room with significant amount of fissile or toxic chemical material.
- Temporary loss of utilities such as electrical power, pressurized air, coolant, ventilation.

Criteria selected for user requirement UR1 are presented in Table 14.

TABLE 14. CRITERIA FOR USER REQUIREMENT UR1

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR1: Robustness of design during normal operation: The uranium refining, conversion or enrichment facility assessed is more robust than the reference design with regard to operation and systems, structures and components failures.	CR1.1: Design of normal operation systems	IN1.1: Robustness of design of normal operation systems. AL1.1: Superior to that in the reference design.
	CR1.2: Subcriticality	IN1.2: Subcriticality margins. AL1.2: Sufficient to cover uncertainties and avoid criticality.
	CR1.3: Facility performance	IN1.3: Facility performance attributes. AL1.3: Superior to those in the reference design.
	CR1.4: Inspection, testing and maintenance	IN1.4: Capability to inspect, test and maintain. AL1.4: Superior to that in the reference design.
	CR1.5: Failures and deviations from normal operation	IN1.5: Expected frequency of failures and deviations from normal operation. AL1.5: Lower than that in the reference design.
	CR1.6: Occupational dose	IN1.6: Occupational dose values during normal operation and AOOs. AL1.6: Lower than the dose constraints.

7.4.2.1. *Criterion CR1.1: Design of normal operation systems*

Indicator IN1.1: Robustness of design of normal operation systems.

Acceptance limit AL1.1: Superior to that in the reference design.

The design (e.g. mechanical, thermal, electrical, etc.) of normal operating systems in a uranium refining/ conversion or enrichment facility can be made more robust, i.e. reducing the likelihood of failures, by increasing the design margins, improving the quality of manufacture and construction, and by use of materials of higher quality.

For an enrichment facility using centrifuges, the separating element is expected to be designed with a lesser number of probable leakage points. The provision of secondary seals in the centrifuges would lessen the probability of leakage and make the system more robust. Passive safety through low pressure operations and a hermetically sealed design would ensure increased robustness. Vessels can be designed for preventing criticality, considering the maximum enrichment targeted. Isolation of the cascade hall and handling area, clear operation limits for critical parameters and adequate factors of safety in containment are other measures towards increasing robustness. A stable power supply is considered as an important requirement of enrichment processes based on centrifuges. Thus, the power supply needs to be of a high standard (including a backup power supply).

The use of corrosion resistant materials in a refining and conversion facility can reduce the probability of leaks in equipment containing corrosive material (e.g. HF).

The **acceptance limit AL1.1** of **CR1.1** is met if evidence available to the INPRO assessor shows that the normal operation system design of the facility assessed is superior to that of the reference facility design (e.g. has increased design margins, improved quality of manufacture and construction, or uses materials of higher quality), or, if no reference plant can be defined, took best international practice into account and is therefore state of the art technology.

7.4.2.2. *Criterion CR1.2: Subcriticality*

Indicator IN1.2: Subcriticality margins.

Acceptance limit AL1.2: Sufficient to cover uncertainties and avoid criticality.

Ref [101] introduces the effective neutron multiplication factor (k_{eff}) as "the ratio of the total number of neutrons produced by a fission chain reaction to the total number of neutrons lost by absorption and leakage", and subcriticality as the state characterised by $k_{eff}<1$ which can be maintained by appropriate combination of the control parameters, such as isotopic composition, geometry, mass, volume, concentration / density, characteristics of neutron absorption and moderation. Ref [101] further requires that "safety margins should be applied to determine the safety limits" and in applying safety margins to k_{eff} "consideration should be given to uncertainty in the calculation" including the possibility of any code bias.

The INPRO task group for the area of NCFC safety has proposed that, for a new NFCF that handles uranium with the enrichments above 1 % ^{235}U, a criticality analysis needs to be performed that demonstrates ample design margins by showing that a $k_{eff} < 0.90$ characterizes all possible configurations of fissile material and thereby provides high confidence that potential criticality accidents are avoided. In this analysis, all parameters influencing k_{eff}, such as mass, concentration, shape, moderation, etc, have to be considered. All process equipment in the material handling areas needs to be designed to ensure subcriticality under submerged and water filled conditions.

The **acceptance limit AL1.2** is met if evidence available to the INPRO assessor shows that in the facility assessed no critical configuration can occur, taking uncertainties into account.

7.4.2.3. Criterion CR1.3: Facility performance

Indicator IN1.3: Facility performance attributes.

Acceptance limit AL1.3: Superior to those in the reference design.

Superior facility performance can reduce the frequency of AOOs and accidents in a uranium refining/ conversion or enrichment facility.

The clear definition of roles and responsibilities, appropriate surveillance and the training of personnel in the handling of UF_6 gas cylinders and the actions to be taken in the event of leakage of UF_6 gas, etc, complemented by instructions based upon learning from experience where available, will ensure that facilities for refining/ conversion and enrichment can operate in a safe regime.

The strategy of ageing management is expected to cover all relevant stages in the NFCF lifecycle, including design, manufacture, construction, commissioning, operation and decommissioning, all normal operation states, AOOs and accidents influencing a given system, and all relevant mechanisms of ageing, including but not limited to corrosion, deposits, irradiation, fatigue and wear. The NFCF designer has to determine the design life of safety related equipment, to provide appropriate design margins to take due account of age related degradation and to provide methods and tools for assessing ageing during the NFCF operation. The NFCF operating organization has to develop a plan for preparing, coordinating, maintaining and improving activities for ageing management implementation at the different stages of the NFCF lifecycle. Implementation of this plan needs to involve activities for managing ageing mechanisms, detecting and assessing ageing effects, and managing ageing effects.

Acceptance criteria for the quality of operation can be taken to be:

- High(er) degree of remote control;
- Availability of operations manuals and emergency instructions manuals;
- Availability of procedures for feedback on the application of operations manuals;

- Availability of surveillance requirements including periodic tests to verify the performance level for safe operation;
- Consideration of ageing management in the design documentation;
- Availability of plan for implementation of ageing management;
- Periodic and intensive training of operators;
- Periodic mock-ups to ensure readiness of operators to handle emergencies.

The **acceptance limit AL1.3** of **CR1.3** is met if evidence available to the INPRO assessor shows that the design of the facility assessed is superior to the reference facility design or, when no reference facility can be defined, at least took best international practice into account and is therefore state of the art technology.

7.4.2.4. Criterion CR1.4: Inspection, testing and maintenance

Indicator IN1.4: Capability to inspect, test and maintain.

Acceptance limit AL1.4: Superior to that in the reference design.

To achieve an improved capability to inspect, test and maintain, the design of uranium refining/ conversion or enrichment facility assessed is expected to permit efficient and intelligent inspection, testing and maintenance and not just require more inspections and more testing. In particular, the programs of inspection, testing and maintenance need to be driven by a sound understanding of failure mechanisms, so that the right locations are inspected and the right systems, structures and components are tested and maintained at the right time intervals.

The **acceptance limit AL1.4** of **CR1.4** is met if evidence available to the INPRO assessor shows that the capability to inspect, test and maintain the systems relevant to safety in the facility assessed is superior to that in the reference design, or is state of the art, and allows easy inspection, testing and maintenance.

7.4.2.5. Criterion CR1.5: Failures and deviations from normal operation

Indicator IN1.5: Expected frequency of failures and deviations from normal operation.

Acceptance limit AL1.5: Lower than that in the reference design.

The frequency of failures and deviations from normal operation (see examples in the beginning of Section 7.4.2) in a refining/ conversion and enrichment facility needs to be derived from operational experience and supported by PSA. For the design assessed, these frequencies can be reduced through increased robustness of the design, high quality of operation, and efficient and intelligent inspection.

The **acceptance limit AL1.5** of **CR1.5** is met if evidence available to the INPRO assessor shows that in the facility assessed the frequencies of failures and deviations from normal operation are lower than those in the reference design, or, if a reference facility cannot be defined, that the facility assessed took best international practice into account and is therefore state of the art technology. If quantitative results from operational experience and PSA are not available, alternatively, deterministic analysis can be developed that supports a reduction of the probability of occurrence for deviations from normal operation and failures in the facility assessed.

7.4.2.6. Criterion CR1.6: Occupational dose

Indicator IN1.6: Occupational dose values during normal operation and AOOs.

Acceptance limit AL1.6: Lower than the dose constraints.

The limit (effective dose) and dose constraints for occupational workers were discussed in section 6.3.2.5. Innovative and proven techniques such as increased automation, improved

O&M techniques and effective (engineered) safety features can be used to further reduce occupational exposure in refining/ conversion and enrichment facilities.

In refining/ conversion and enrichment facilities, the radiological hazard from radium and radon is much lower than in the mining and milling facilities discussed above; however, the radiological hazard cannot be neglected. Both radiological and chemical toxicity limits are applicable to the working environment in the refining/ conversion and enrichment facilities. The radiological limit for UF_6 concentration in air can be derived from annual limits on intake (ALI) values introduced in Ref [102] at the level of 13 Bq/m^3, subject to small variations with respect to enrichment. The uranium air concentration chemical limit is normally stated as 0.2 mg/m^3 [103]. Table 15 gives the uranium concentrations in air that correspond to the radiological limit as well as the uranium activity levels in air that correspond to the chemical toxicity limit.

TABLE 15. RADIOLOGICAL AND CHEMICAL TOXICITY LIMITS FOR UF_6 AS URANIUM IN AIR

Enrichment, %	Radiological limit, Bq/m^3	Concentration of U in air corresponding to the radiological limit mg/m^3	Chemical toxicity limit, mg/m^3	Activity of U in air corresponding to the chemical toxicity limit Bq/m^3
0.7	13	0.52	0.2	5
1		0.42		6
2		0.22		12
2.3		0.20		13
3		0.14		18
3.5		0.12		22
5		0.08		33

Comparing the activities and concentrations in Table 15 against the corresponding limits shows that the chemical toxicity limit (0.2 mg/m^3) is more restrictive than the radiological limit (13 Bq/m^3) up to the enrichment value of 2.3%. For enrichments higher than 2.3%, the radiological limit becomes more important and for the enrichment of 5% the maximum permitted concentration of uranium in air due to its radiological properties is less than half of chemical toxicity limit.

A detailed guide on how to achieve a successful radiation protection program for workers in a refining/ conversion and enrichment facility is provided in Ref [32].

The **acceptance limit AL1.6** of **CR1.6** is met if evidence available to the INPRO assessor shows that the dose values of workers during normal operation and AOOs will be lower than the dose constraints defined for the location of the planned facility.

7.4.3. User requirement UR2: Detection and interception of AOOs

The rationale of UR2 was provided in Section 5.4. The criteria selected for user requirement UR2 are presented in Table 16.

TABLE 16. CRITERIA FOR USER REQUIREMENT UR2

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR2: Detection and interception of AOOs: The uranium refining, conversion or enrichment facility assessed has improved capabilities to detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions.	CR2.1: I&C systems and operator procedures	IN2.1: I&C system to monitor, detect, trigger alarms and, together with operator actions, intercept and compensate AOOs. AL2.1: Availability of such systems and operator procedures.
	CR2.2: Grace periods for AOOs	IN2.2: Grace periods until human actions are required after AOOs. AL2.2: Adequate grace periods are defined in design analyses.

7.4.3.1. Criterion CR2.1: I&C systems and operator procedures

Indicator IN2.1: I&C system to monitor, detect, trigger alarms, and, together with operator actions, intercept and compensate AOOs.

Acceptance limit AL2.1: Availability of such systems and operator procedures.

Refining/ conversion and enrichment facilities are expected to be designed to cope with AOOs (see beginning of Section 7.4.2) by using automatic operational systems, i.e. I&C systems that bring the facility back to normal operating conditions. In case automatic systems are not available, adequate operator procedures need to be. Passive and automatic active control systems are deemed more reliable than administrative (manual) control. The operator needs to get appropriate information in a control room about automatic actions during normal operation and AOOs and the status and performance of the facility.

Monitoring of operational data is important for early detection of the onset of integrity loss of system components in uranium refining/ conversion and enrichment facilities and to avoid complete failures of components. Such systems for monitoring component health might include, e.g. a diagnostic system of the rotating machinery for fans, pumps, and turbines. The basic monitoring of pumps may be done by monitoring the pump house vibrations.

Provision of an on-line digital I&C system with an intelligent controller and sufficient capability to monitor would ensure that an enrichment facility could be safely operated. Redundancy in devices for detecting overloading of the separation system and measurement of a parameter based on different principles wherever applicable, would provide enhanced safety. For example, use of two independent parameters to indicate faulty operation of centrifuges (e.g. current drawn by motor and vibration) would enable prompt correcting action. A strategy to isolate and limit damage to the separation system needs to be available.

For mitigating a leakage of toxic or explosive gases, a detection and exhaust scrubbing system needs to be available that automatically removes such gases from the air in the building and thereby avoids toxic or explosive concentrations. To fight fires, a detection and, preferably, an automatic fire extinguishing system (e.g. a spray system) needs to be available and related criticality considerations taken into account (e.g. exclusion of water).

An emergency power supply system is expected to be available for systems relevant to safety, such as monitoring, detection and alarm systems for radiation protection and criticality, detection and alarm systems for fires and leaks of hazardous materials, ventilation systems, etc. A loss of external power needs to be compensated by a back-up power system available at the site of the facility.

Safe operating conditions of all systems are expected to be clearly defined in the design analysis and different limits for alarm (and shutdown) conditions (e.g. pressure, temperature and overloading) need to be determined. For the operational I&C systems to be acceptable, the results of the analyses need to demonstrate that all limits for alarm (and actions including shutdown) are met in case of assumed deviations from normal operation. In addition to automatic systems the systems and clearly defined procedures for the operator on how to restore the facility after an AOO to normal operational state need to be available.

The **acceptance limit AL2.1** of **CR2.1** is met if evidence available to the INPRO assessor shows that I&C systems are available in the facility assessed that are capable of detecting failures and deviations from normal operation of systems relevant for safety, providing alarm, and initiate automatic or manual actions that bring the facility back to normal operation.

7.4.3.2. Criterion CR2.2: Grace periods for AOOs.

Indicator IN2.2: Grace periods until human actions are required after AOOs.

Acceptance limit AL2.2: Adequate grace periods are defined in design analyses.

An explanation of the 'adequate grace period' is provided in section 6.3.3.2. The grace period available for the operator for each AOO needs to be defined in the safety analysis of the facility design. In addition to the automatic actions of the normal operation systems a refining/ conversion or enrichment facility is expected to have sufficient inertia to withstand transients, i.e. react slowly after AOO.

After detection of an AOO (see beginning of Section 7.4.2) in a refining/ conversion or enrichment facility, automatic operational systems (presented in Section 7.4.3.1 above) need to mitigate these incidents before the operator intervention. For example, 30 minutes are deemed sufficient in case of a leak of UF6 gas during normal operation. Efficient automatic measures can facilitate longer grace periods.

In an enrichment facility with centrifuges, sufficient grace periods for operator actions necessary for keeping an AOO from progressing into an accident can be assured by providing surge suppression limiters, a fly wheel in the driving system of the centrifuge machine in case of electricity fault, adequate thermal inertia of the heating furnace, and multi-stage control for limiting transients.

The **acceptance limit AL2.2** of **CR2.2** is met if evidence available to the INPRO assessor shows that adequate grace periods have been determined for all AOOs in the design analysis for the facility assessed.

7.4.4. User requirement UR3: Design basis accidents

Rationale of UR3 was provided in Section 5.5. Ref [32] admits that specification of DBA will depend on the facility design and national requirements. However, it recommends that [32]:

> "… particular consideration should be given to the following hazards in the specification of design basis accidents for conversion facilities:
> (a) A release of HF or ammonia (NH3) due to the rupture of a storage tank;
> (b) A release of UF_6 due to the rupture of a storage tank, piping or a hot cylinder;
> (c) A large fire originating from H_2 or solvents;
> (d) An explosion of a reduction furnace (release of H_2);
> (e) Natural phenomena such as earthquakes, flooding or tornadoes1;
> (f) An aircraft crash;
> (g) Nuclear criticality accidents, e.g. in a wet process area with a ^{235}U content of more than 1% (reprocessed uranium or unirradiated LEU)."

The following recommendation is provided for DBA consideration in enrichment facility [32]:

> "… particular consideration should be given to the following hazards in the specification of design basis accidents for enrichment facilities:
> (a) The rupture of an overfilled cylinder during heating (input area);
> (b) The rupture of a cylinder containing liquid UF_6 or the rupture of piping containing liquid UF_6 (depending on the facility design for product take-off);
> (c) A large fire, especially for diffusion facilities;
> (d) Natural phenomena such as earthquakes, flooding or tornadoes (…);
> (e) An aircraft crash;
> (f) A nuclear criticality accident."

Criteria selected for user requirement UR3 are presented in Table 17.

TABLE 17. CRITERIA FOR USER REQUIREMENT UR3

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR3: Design basis accidents: The frequency of occurrence of DBAs in the uranium refining, conversion or enrichment facility assessed is reduced. If an accident occurs, engineered safety features and/or operator actions are able to restore the assessed facility to a controlled state and subsequently to a safe state, and the consequences are mitigated to ensure the confinement of nuclear and/or toxic chemical material. Reliance on human intervention is minimal, and only required after sufficient grace period.	CR3.1: Frequency of DBAs	IN3.1: Calculated frequency of occurrence of DBAs. AL3.1: Lower than that in the reference design.
	CR3.2: Engineered safety features and operator procedures	IN3.2: Reliability and capability of engineered safety features and/or operator procedures. AL3.2: Superior to those in the reference design.
	CR3.3: Grace periods for DBAs	IN3.3: Grace periods for DBAs until human intervention is necessary. AL3.3: Longer than those in the reference design.
	CR3.4: Barriers	IN3.4: Number of confinement barriers maintained (intact) after DBAs. AL3.4: At least one.
	CR3.5: Robustness of containment design	IN3.5: Containment loads covered by design of the facility assessed. AL3.5: Greater than those in the reference design.

7.4.4.1. Criterion CR3.1: Frequency of DBAs.

Indicator IN3.1: Calculated frequency of occurrence of DBAs.

Acceptance limit AL3.1: Lower than that in the reference design.

The DBAs to be considered in a refining/ conversion or enrichment facility have been listed in the beginning of Section 7.4.4. The frequency of occurrence of a DBA in the facility assessed is to be determined via a probabilistic risk assessment.

The calculated frequency of DBAs caused by external hazards can be influenced by the designer primarily by increasing the robustness of the confinement wall, and by the owner/ operator by selecting an appropriate site (see UR7).

The **acceptance limit AL3.1** of **CR3.1** is met if evidence available to the INPRO assessor shows the use of probabilistic analyses to determine that DBAs in the assessed facility are less probable than in the reference design. If quantitative results of probabilistic analyses are not available, a deterministic analysis needs to be available that supports a reduction of these frequencies based on an increase of design robustness, high quality of operation, an intelligent inspection and maintenance programs, advanced I&C systems and/or operator procedures, increased grace time and inertia.

7.4.4.2. Criterion CR3.2: Engineered safety features and operator procedures

Indicator IN3.2: Reliability and capability of engineered safety features and/or operator procedures.

Acceptance limit AL3.2: Superior to those in the reference design.

Engineered safety features (automatic) are expected to be designed and installed in the facility. After detection of the accident these features need to be capable of controlling the accident, restoring the facility to a controlled state, and keeping the radiological consequences of the accident within authorized limits. To assure necessary reliability, these features need to be designed with sufficient levels of redundancy, diversity and independence. Redundant, diversified and independent passive and automatic active systems are deemed more reliable than administrative control (operator intervention) but it is acknowledged that passive systems are difficult to be designed for refining/ conversion or enrichment facility.

The engineered safety features of a refining/conversion facility can be essentially different from an enrichment facility and in the following several examples they are discussed separately.

In refining/ conversion facility, a release of gaseous or liquid radioactive and/or chemically toxic material (UF_6, HF, and NH_3) is expected to be timely detected, an alarm started (to initiate evacuation of the facility) and automatic systems (e.g. exhaust scrubbers, shut down of gas supply) need to be available to mitigate the consequences of these DBA, i.e. limit exposure of the workers to chemicals and/or radioactive material. The release in the working area needs to be contained within the process area itself. Process specific, sub-atmospheric pressure operation is likely to ensure that this can be achieved.

In case of a fire in refining/ conversion facility, e.g. originating from release of H_2 or solvents, alarm needs to be initiated, and automatic fire fighting systems (spray systems) start in rooms with flammable chemicals that are capable of extinguishing the fire taking criticality considerations into account (e.g. exclusion of water). Alternatively or additionally, equipment needs to be available for the operator to fight the fire manually.

In an enrichment facility based on centrifuges, in the event of a beginning failure, automatic provisions can be available in the form of suitable brakes, to absorb the momentum of a failing centrifuge. This would keep the damage localized and prevent the failed centrifuge from becoming a missile. Safety interlocks need to be provided for addressing the instability and vibration in motors for the centrifuges.

As mentioned above, refining/ conversion and enrichment facilities are expected to have engineered safety features to protect against DBAs caused by external hazards (see Section 4.2.1 and 4.2.6). For example, to mitigate an earthquake [47], equipment in the facility – that if failed would create a radiological and/or chemical hazard – needs to be protected by shock absorbers, dampers, etc.

The **acceptance limit AL3.2** of **CR3.2** is met if evidence available to the INPRO assessor shows that the assessed facility's engineered safety features (automatic systems) and/or operator procedures are superior to those in the reference facility and assure that after the beginning of a DBA the necessary actions to mitigate the accident consequences will be initiated in a timely manner and successfully completed. The INPRO assessor's judgement of the superiority of the new design has to be supported by the results of equipment tests and/or deterministic and probabilistic analyses described in the facility design information. Alternatively, if a reference facility cannot be found, it could be demonstrated that the design of the facility assessed took available information on best international practice into account and is therefore state of the art.

7.4.4.3. Criterion CR3.3: Grace periods for DBAs

Indicator IN3.3: Grace periods for DBAs until human intervention is necessary.

Acceptance limit AL3.3: Longer than those in the reference design.

An explanation of 'adequate grace periods' is provided in section 6.3.3.2 for the control of AOOs (see CR2.2) in Level 2 of DID. The criterion CR3.3 'grace periods for DBAs' implies a similar concept. For DBAs (caused by events associated with internal or / and external hazards) the criterion requires that the system response (inertia) and/or automatic actions of active (and/or passive) safety features provide an adequate grace period for the operator to intervene. Adequate grace periods are also assumed to be longer than those for the reference design.

Since a large-scale gas leak has a potential to propagate outside the facility, a grace time of 15 minutes is expected to be provided for mitigating the gas leak, by for example, starting an emergency exhaust scrubber/ventilation system.

For a criticality accident, a grace period of a few minutes can be achieved by providing shielded enclosures wherever concentrations of uranium are expected to be high. Lower pressure in the process handling area and criticality monitors are normally provided. Risk to humans is expected to be limited to the material handling area only.

The grace periods have to be determined for each DBA in the design analyses.

The **acceptance limit AL3.3** of **CR3.3** is met if evidence available to the INPRO assessor shows that in the assessed facility's grace periods for DBAs are longer than those of the reference design. Alternatively, it may be demonstrated that the design of the facility assessed took available information on best international practice into account and is therefore state of the art.

7.4.4.4. Criterion CR3.4: Barriers

Indicator IN3.4: Number of confinement barriers maintained (intact) after DBAs.

Acceptance limit AL3.4: At least one.

The design of engineered safety features and/or operator procedures are expected to provide deterministically for continued integrity at least of one barrier containing the radioactive and chemically toxic material following any DBA caused by events associated with internal or external hazards. Alternatively, the probability of losing all barriers may be used as an INPRO methodology indicator with a sufficiently low value (e.g. less than 10^{-6} per year) as its acceptance limit.

Examples of barriers in refining/conversion and enrichment facilities are the casing of machinery (pumps, valves, centrifuge) and equipment (vessels, piping), and a building structure with isolated compartments. The ventilation system including a cleaning system such as wet scrubbers or cold traps and a stack could also be regarded as a dynamic confinement. The design analysis needs to confirm that at least one barrier against an accidental release of radioactive and/ chemically toxic material into the outside of the plant will remain intact after a DBA.

The **acceptance limit AL3.4** of **CR3.4** is met if evidence available to the INPRO assessor shows that after a DBA at least one barrier remains intact in the facility assessed avoiding an accidental release of radioactivity and/or toxic chemicals to the outside of the facility that would require evacuation.

7.4.4.5. Criterion CR3.5: Robustness of containment design

Indicator IN3.5: Containment loads covered by design of the facility assessed.

Acceptance limit AL3.5: Greater than those in the reference design.

To avoid a loss of containment/confinement integrity due to for example overpressure and high temperatures – compared to operating refining/ conversion or enrichment facility – the containment of new facility is expected to be designed against higher loads caused by an accident with an accidental release of radioactive material and/or toxic chemicals into the containment.

The containment, i.e. the building structure of the facility needs also to be designed for external hazards challenging the integrity of the structure with a higher margin.

The **acceptance limit AL3.5** of **CR3.5** is met if evidence available to the INPRO assessor shows that the confinement/containment of the refining/ conversion or enrichment facility assessed has been designed against higher loads and with higher reliability compared to a reference design. Alternatively, if a reference design is not available, it could be demonstrated that the design of the facility assessed took available information on best international practice into account and is therefore state of the art.

7.4.5. User requirement UR4: Severe plant conditions

Rationale of UR4 was provided in Section 5.6. Criteria selected for user requirement UR4 are presented in Table 18.

TABLE 18. CRITERIA FOR USER REQUIREMENT UR4

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR4: Severe plant conditions: The frequency of an accidental release of radioactivity into the environment is reduced. The source term of accidental release into the environment remains well within the envelope of the reference facility source term and is so low that calculated consequences would not require public evacuation.	CR4.1: In-facility severe accident management	IN4.1: Natural or engineered processes, equipment, and AM procedures and training to prevent an accidental release to the environment in the case of accident. AL4.1: Sufficient to prevent an accidental release to the environment and regain control of the facility.
	CR4.2: Frequency of accidental release into environment	IN4.2: Calculated frequency of an accidental release of radioactive materials and/or toxic chemicals into the environment. AL4.2: Lower than that in the reference facility.
	CR4.3: Source term of accidental release into environment	IN4.3: Calculated inventory and characteristics (release height, pressure, temperature, liquids/gas/aerosols, etc) of an accidental release. AL4.3: Remains well within the inventory and characteristics envelope of the reference facility source term and is so low that calculated consequences would not require evacuation of population.

7.4.5.1. Criterion CR4.1: In-facility severe accident management

Indicator IN4.1: Natural or engineered processes, equipment, and AM procedures and training to prevent an accidental release to the environment in the case of accident.

Acceptance limit AL4.1: Sufficient to prevent an accidental release to the environment and regain control of the facility.

Examples of relevant system parameters are concentrations of UF_6 and other radioactive and/or toxic chemicals in air, and activity, temperature and pressure inside the confinement/ containment. An emergency ventilation system is expected to be capable of reducing these system parameters to acceptable levels enabling mitigating measures by operators.

In an enrichment facility with centrifuges cascade segment isolation and cascade isolation based on pressure rise are processes to limit the consequences of accidents with a large release of UF_6. Emergency exhaust scrubber with alkali washing needs to be provided to bring down concentration of UF_6 to less than 0.2 mg/m^3 within 30 minutes. Failure of one system normally does not lead to the failure of other systems by preventing transmission of shock or vibration to other cascades. Each cascade and handling system need to be made as independent modules. Reliability of secondary back-up seals in the centrifuges is expected to be excellent, with a failure rate better than 10^{-4} per operation year. This needs to be confirmed by accelerated tests under simulated conditions.

If a large release of fissile material into the confinement (the building of the facility) leads to a critical configuration, this needs to be automatically detected (neutron flux increase) and lead to initiation of measures to end the criticality (injection of neutron absorbers).

In case automatic systems alone are not sufficient to prevent an accidental release to the environment and regain control of the NFCF, adequate operator procedures need to be established to handle a severe accident. For example, after detection of a large release of toxic and/or radioactive material into the confinement/ containment, the operator cuts off the source, activates the isolation of the process and the area, followed by evacuation/scrubbing. Next step would be activation of an on-site emergency plan documented in a safety manual to prevent spread of toxic and/or radioactive material into uncontrolled areas. Periodic mock-up drills and training programs are necessary to ensure that operators are in readiness to handle such emergencies.

The **acceptance limit AL4.1** of **CR4.1** is met if evidence available to the INPRO assessor shows that in the facility assessed processes and equipment are available to control relevant parameters (e.g. temperature, activity, concentrations of chemicals) and AM measures have been prepared that are sufficient to prevent an accidental release to the outside of the facility.

7.4.5.2. Criterion CR4.2: Frequency of accidental release into environment

Indicator IN4.2: Calculated frequency of an accidental release of radioactive materials and/or toxic chemicals into the environment.

Acceptance limit AL4.2: Lower than that in the reference facility.

An accidental release of radioactivity and/or toxic chemicals from the refining/ conversion or enrichment facility into the environment can occur only if the containment loses its integrity during an accident with severe damage. An example for a cause of containment failure is overpressure due to a hydrogen explosion. Via a probabilistic safety analysis the frequency of a containment failure including uncertainties needs to be determined covering all plant states (normal operation, shut down) and internal as well as external hazards leading to accidents; the probabilistic analyses is supposed to use best estimate methods and consider the associated uncertainties.

INPRO suggests that calculated frequency of accidental release outside the facility assessed needs to be reduced as compared against reference facility, e.g. by increasing the design pressure of the containment. Where PSA data for reference facilities are not available, INPRO suggests using limit of $<10^{-6}$ per facility-year as the target value for calculated frequency of accidental release to the environment.

When the frequency of accidental release of radioactivity cannot be calculated with a high level of confidence the new NFCF design needs to demonstrate deterministically that probability of an accidental release of radioactivity and/or toxic chemicals into the environment due to a failure of the containment/ confinement has been reduced compared against reference facility, e.g. through improved engineered safety features, prescribed advanced operator actions, and increased use of inherent safety characteristics or by additional minimization of hazards, and that the consequences (dose, concentration of toxic chemicals) of an accident do not require the evacuation of population except as a short time precautionary measure.

The **acceptance limit AL4.2** of **CR4.2** is met if evidence available to the INPRO assessor shows that in the facility assessed the calculated (best estimate) frequency for an accidental release of radioactivity and/or toxic chemicals into the environment due to a failure of the containment is lower than in reference facility. Alternatively, if PSA data for a reference design is not available, it could be demonstrated that frequency for an accidental release of radioactivity from NFCF is well below 10^{-6} per unit-year or that the design of the NFCF took available information on best international practice into account and is therefore state of the art.

7.4.5.3. Criterion CR4.3: Source term of accidental release into environment

Indicator IN4.3: Calculated inventory and characteristics (release height, pressure, temperature, liquids/gas/aerosols, etc) of an accidental release.

Acceptance limit AL4.3: Remains well within the inventory and characteristics envelope of the reference facility source term and is so low that calculated consequences would not require evacuation of population.

Evacuation of population is the protective action in an emergency which can reduce the risk of stochastic effects, i.e. reduce consequences of the accident. Radiological criteria for evacuation of populations are normally formulated in terms of projected dose [58].

Estimation of the consequence of the emergency external release can be divided into two major parts. First part is focused on the definition of the characteristics of the release source term. These characteristics can be calculated as the result of the accident consequence modelling within the NFCF either deterministically or as a part of PSA Level 2 analysis. Second part models the transportation of the radionuclides to the population outside of the NFCF through different potential routes and scenarios (PSA Level 3).

The definition of source term of an accidental release to the environment involves the inventory of radioactive materials released, the description of physical and chemical forms of release and other release characteristics such as the height of damaged zone of the confinement, pressure and temperature of the released gas (including potential explosions).

Since the results of modelling of radionuclide transport in the environment may heavily depend on a series of assumptions such as weather conditions (wind directions in different altitudes, humidity etc) the first part of acceptance limit in this INPRO criterion states that source term characteristics in the new NFCF including the inventory of released radionuclides remains well within the envelope of reference facility source term. In this context ‘well within the envelope’ means that in the new NFCF source term all characteristics will be equal or lower compared against reference design and at least some of them will be lower by the level of uncertainties associated with the accident consequence modelling within the confinement.

For new NFCF the capability and reliability of natural and/or engineered processes for controlling of the complex accident sequences with severe damage are expected to be increased, including their instrumentation, control and diagnostic systems, and appropriate severe accident management procedures need to be developed. By these measures, the frequency of accidental release of radioactivity can be reduced and the inventory and conditions of release are expected to be restrained to avoid the evacuation of population.

It is noted that to meet the objective of Level 5 of defence in depth an emergency protection and response has to be planned around the NFCF [2] commensurate with the hazard of the accidental release of radioactive and chemically toxic material into the environment.

The **acceptance limit AL4.3** of **CR4.3** is met if evidence available to the INPRO assessor shows that in the NFCF assessed the calculated inventory and characteristics of an accidental release remain well within the inventory and characteristics envelope of reference facility source term and low enough so that calculated consequences would not require evacuation of population.

7.4.6. User requirement UR5: Inherent safety characteristics

INPRO methodology requirement on the independence of DID levels has been found not to be fully applicable for a uranium refining/conversion and enrichment facility. Rationale of UR5 was provided in Section 5.7. Criterion selected for user requirement UR5 is presented in Table 19.

TABLE 19. CRITERION FOR USER REQUIREMENT UR5

User requirement	Criterion	Indicator (IN) and Acceptance Limit (AL)
UR5: Inherent safety characteristics: To excel in safety and reliability, the refining, conversion or enrichment facility assessed strives for elimination or minimization of some hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics when appropriate.	CR5.1: Minimization of hazards	IN5.1: Examples of hazards: fire, flooding, release of radioactive material, radiation exposure, etc. AL5.1: Hazards are reduced in relation to those in the reference facility.

7.4.6.1. Criterion CR5.1: Minimization of hazards

Indicator IN5.1: Examples of hazards: fire, flooding, release of radioactive material, criticality, radiation exposure, etc.

Acceptance limit AL5.1: Hazards are reduced in relation to those in the reference facility.

To minimize the fire hazard a specific safety (fire) analysis is required [32]. Using of fire resistant material and reduction of the amount of burnable material in a refining/ conversion or enrichment facility would reduce the hazard of a fire. In a conversion facility there are the following chemicals causing fire hazards: anhydrous ammonia (explosive and flammable), nitric acid (ignition if in contact with organic materials) and hydrogen (explosive and flammable). Compartmentalizing of buildings and ventilation ducts needs to be performed to prevent spreading of fires. Ventilation ducts need to be equipped with fire dampers and be made of fire resistant material. Buildings are normally divided into separate fire areas to make sure that a fire breaking out within a given fire area would not be able to spread beyond this sector. The higher the fire risk, the greater the number of areas in a building. For example, damage to the separation system in an enrichment facility and process handling system in a refining/ conversion facility needs to be confined within the given area and not to spread to other areas. The design of ventilation systems is expected to be given particular consideration with regard to fire prevention.

The hazard of release of radioactive and/or chemically toxic material is normally minimized by establishing several barriers, such as glove box or hooding of equipment, compartmentalized building, and a dynamic confinement by a ventilation system.

The hazard of radiation exposure of workers in the facility can be minimized by establishing and maintaining an adequate radiation protection program in accordance with national and international standards [25]. An adequately sized ventilation system can minimize the hazard of radiation exposure of workers.

To reduce the hazard of a criticality accident, control of the inventory of radioactive materials is the first step. This can be achieved not merely through administrative measures but also through monitoring systems that will give a warning if set limits of inventories are exceeded. Sub-atmospheric pressure operation would also minimize releases from equipment containing fissile material.

The external hazards can be reduced for new facilities by appropriate selection of their site. For example, to minimize the hazard of flooding the facility needs to be located at sufficient elevation.

The **acceptance limit AL5.1** of **CR5.1** is met if evidence available to the INPRO assessor shows that in the refining/ conversion or enrichment facility assessed hazards have been reduced compared to a reference facility. Alternatively, if a reference facility cannot be found, it needs to be demonstrated that the design of the facility assessed took available information on best international practice into account and is therefore state of the art.

7.4.7. User requirement UR6 and UR7

Rationale for UR6 and UR7 are provided in Section 5.8 and 5.9, respectively. Assessment of user requirement UR6 (human factors related to safety) and UR7 (RD&D for advanced designs) for the refining / conversion or enrichment facility is deemed to be sufficiently similar to the assessment method of UR6 and UR7 described in Sections 6.3.7 and 6.3.8 for mining and milling facilities (including criteria, indicators and acceptance limits).

A number of areas for RD&D exist with regard to stable and safe operation of centrifugation, including development of frictionless bearings, avoiding external drives for gas transport, etc. Use of non-hydrogenous coolants can contribute to safety with regard to criticality. Development of materials to withstand corrosion by UF6 is another area for RD&D. The existence of a robust RD&D programme on the above areas and other such areas would be a necessary step for enhancing safety.

8. URANIUM OXIDE AND MOX FUEL FABRICATION

In this section, firstly, a short description of the main processes in a U oxide and in a U-Pu mixed oxide (MOX) fuel production facility is given and the corresponding specific safety issues are discussed. Secondly, the assessment method is discussed based on the corresponding criteria of the INPRO methodology in the area of safety, which have been, where necessary, adapted to the specific issues of these NFCF.

As there are various types of nuclear fuel cycles, different kinds of fuel are fabricated in different facilities for different reactors. Light water reactors (LWRs) and PHWRs most often use uranium oxide fuel. For LWR, the uranium is enriched in a range from 1.2 % to about 5 % of ^{235}U. In PHWR, mostly natural uranium is used. In some LWR and fast reactors (FR), MOX fuel is being used. The majority of the research reactors uses metallic/alloy fuel in plate form. Several countries are planning to use metal fuels or nitride fuels in FR to achieve higher breeding ratios. A comprehensive description of fabrication technology of these and several other types of nuclear fuel is documented in Ref [104].

As of end of 2015 the following countries have commercial fuel fabrication facilities (UO_2 and/or MOX) in operation: Argentina (Cordoba, Ezeiza), Belgium (Dessel), Brazil (Resende), Canada (Port Hope, Peterborough), China (Yibin, Baotou), France (Romans, Marcoule), Germany (Lingen), India (Hyderabad, Tarapur, Trombay), Iran (Isfahan), Japan (Yokosuka, Tokai-Mura, Kumatori), Kazakhstan (Ust-Kamenogorsk), Republic of Korea (Taejon), Pakistan (Chashma), Romania (Mioveni-Arges), Russian Federation (Elektrostal, Novosibirsk), Spain (Juzbado), Sweden (Västeras), United Kingtom (Springfields), USA (Richland, Colombia, Wilmington). In this manual the discussion is restricted to the fabrication of fuels most commonly used in power reactors, i.e. natural and enriched UO_2 fuel and UO_2-PuO_2 (MOX) fuel.

Fresh nuclear fuel transportation has not been considered in this manual as an independent stage of nuclear fuel cycle. INPRO methodology implies that safety of fresh fuel transportation is to be considered as part of the INPRO assessment of fuel fabrication facility producing a given type of fuel. Commonly the fuel fabrication facility sets up the requirements[56] to the packaging design, performs fuel package, and also participates in licensing of transportation and organising the transportation.

8.1. NATURAL UO_2 FUEL FOR PHWR

For PHWR starting with uranium concentrate in the form of impure ADU or magnesium di-uranate (MDU) or uranium peroxide or UO_3, high-density UO_2 pellets are prepared by a 'powder-pellet' route [42]. The major process steps for fabricating PHWR fuel pellets are as follows:

- Dissolution in nitric acid followed by solvent extraction purification of impure uranium nitrate solution by using tri-butyl phosphate (TBP) in kerosene as solvent;
- Addition of ammonium hydroxide to pure uranium nitrate solution to precipitate pure ADU or addition of NH_3 and CO_2 gases to uranium nitrate solution to precipitate pure ammonium uranyl carbonate (AUC);
- Controlled air-calcination followed by hydrogen reduction and stabilization of ADU or AUC to obtain sinterable grade UO_2 powder; and

[56] These requirements have to be consistent with national and when appropriate international safety regulations.

- Cold-pelletisation of powder followed by high temperature sintering (1700 to 1725 ºC) in hydrogen atmosphere and centreless grinding to achieve the desired diameter.

In most countries, sinterable grade UO_2 powder for PHWR is obtained by adapting the ADU route. This ex-ADU uranium dioxide powder is extremely fine with average particle size < 1.0 μ with specific surface area in the range of 2.5 – 3.5 m^2/g and requires a granulation step for making free-flowing press-feed granules. UO_2 granules (1 to 2 mm) are obtained by either 'roll-compaction-granulation' or 'pre-compaction-granulation'.

On the other hand, the ex-AUC uranium dioxide powder is free-flowing, relatively coarse (~10 μ) and porous with specific surface area in the range of 5 m^2/g and suitable for direct pelletisation, avoiding the granulation step. In the AUC route, calcination, reduction and stabilization are simultaneously carried out in a vertical fluidized bed reactor.

In the beginning of nuclear power in most countries, cold-pelletisation was carried out by employing conventional hydraulic press with multiple die-punch sets of tungsten carbide or die steel. However, in recent years the high-speed `rotary compaction' press has been selected. For densification of green pellets, high temperature sintering is carried out at ~1700 ºC in continuous sintering furnace.

For example, in India, natural UO_2 powder is produced from MDU. The UO_2 powder is subjected to either `roll compaction-granulation' or `pre-compaction-granulation' to obtain free-flowing granules. Lubricant such as zinc stearate is admixed to the granules in a blender. The granules are subjected to final compaction in a double acting hydraulic press with multiple die-punch sets. This is followed by high temperature sintering at ~1700 ºC in cracked ammonia in pusher type continuous sintering furnace. The sintered pellets are finally subjected to wet centreless-grinding to obtain UO_2 pellets with designed geometry. The pellets are then loaded into a clad tube filled with a inert gas (e.g. helium) which is subsequently sealed by welding it with an end plug on both sides to become a fuel rod. Several fuel rods are inserted into a fuel element structure (with spacers, upper and lower tie plate, etc.) to produce a fuel assembly (or fuel bundle).

8.2. ENRICHED URANIUM OXIDE FUEL FOR LWR

The LWR mostly use Zircaloy cladded LEU oxide fuel assemblies with a ^{235}U content in the range of about 1.2 to 5 %. The enriched UO_2 pellets are fabricated by 'powder-pellet' route involving preparation of enriched UO_2 powder using UF_6 as starting material.

There are dry and wet conversion processes to produce UO_2 from UF_6. In wet processes UF_6 is injected into water to form a UO_2F_2 particulate slurry. Either ammonia (NH_3) is added to this slurry and reacts to produce ADU, or ammonium carbonate ($(NH_3)_2CO_3$) is added to this slurry and reacts to produce ammonium uranyl carbonate (AUC). In both cases the slurry is filtered, dried and heated in a reducing atmosphere to produce pure UO_2 powder.

In most countries the integrated dry route (IDR) is followed for preparation of fine UO_2 powder by reacting UF_6 vapour with a mixture of super-heated dry steam and hydrogen at ~600 – 700 ºC. The chemical reaction is as follows:

$$UF_6 + 2H_2O + H_2 = UO_2 + 6HF \tag{3}$$

The process does not generate any liquid effluent and the only by-product is high purity HF, which could be recovered and reutilized or sold. The specific surface area of the IDR-derived UO_2 powder is low (about 2 m^2/g) compared to the powder produced by the wet chemical route. The IDR powder is extremely fine (~0.2μ) and requires granulation. The powder is usually transferred into orbital screw blenders for homogenization. Pore formers such as polyvinyl alcohol, methyl cellulose or U_3O_8 are added at the blending stage. The UO_2 powder is subjected to cold-pelletisation and high temperature sintering in hydrogen atmosphere. Fuel rod production and assembling is similar as for natural uranium fuel.

8.3. MIXED OXIDE FUEL

Nuclear fuel containing in addition to U also Pu can be in the form of mixed oxide, carbide or nitride. As plutonium is highly radiotoxic, all operations for fuel fabrication involving Pu have to be carried out in glove boxes or hot cells. Containment and ventilation systems in such a facility need to be very reliable.

Fabrication of Th-Pu mixed oxide fuel can be done in a similar manner. Th-Pu MOX can be sintered in air, which adds to economy and convenience. (Th-^{233}U) mixed oxide fuel fabrication calls for development of automated and remote fabrication technology due to the presence of ^{232}U.

8.4. SAFETY ISSUES IN URANIUM AND MOX FUEL FABRICATION FACILITIES

Safety issues in uranium and MOX fuel fabrication facilities are documented in the IAEA Safety Standards [17, 33, 34] (for supplementary information, see also Refs [99, 105–111]). Criticality accidents and the accidental release of hazardous materials may be the major safety issues in these facilities [112].

The facility needs to be designed to restrict exposure from normal operations to authorised limits. In case of enriched uranium/MOX fuel fabrication, special care is required to minimize contamination. Shielding may be needed for protection of the workers due to higher gamma dose rates compared to natural U fuel production.

Examples of the design principles are prevention of criticality by design (the double contingency principle is the preferred approach, see Annex II of Ref [17]) and confinement of toxic and radioactive chemicals (includes the control of any route into the workplace and the environment).

The main differences regarding hazards between a U and a MOX fuel fabrication facility are:

- The radiotoxicity of plutonium, higher than that of uranium;
- The dry process fabrication method that is preferably used in current industrial-size MOX facilities, which has a higher potential for criticality and for dispersion of radioactive material;
- The thermal power of the plutonium requires that the release of heat be taken into account.

8.4.1. Criticality

Ref [17] requires that criticality safety (see also Section 4.2.2) needs to be achieved by preventive measures (double contingency principle) preferably envisaged in the design but could be also supported by administrative procedures.

In a U and MOX fuel fabrication facility the following parameters [17] need to be kept within subcritical limits during normal operation, AOO and DBA conditions: mass and enrichment of fissile material, geometry characteristics of processing equipment, concentration of fissile material in solutions, characteristics of neutron absorbers, reflectors and moderators. Criticality analysis is supposed to involve potential flooding, mechanical failures, load drops, operator errors, etc. Specific recommendations on how to avoid criticality and how to perform a criticality analysis in a U and MOX fuel fabrication facility are provided in Refs [33, 34].

8.4.2. Internal exposure to radioactive and toxic chemicals

Consideration needs to be given to protecting the workers, public and environment from releases of hazardous material in both normal operational states, anticipated operational occurrences and accident conditions (design basis accidents and design extension conditions) by for example keeping the inventory of liquid UF_6 in the facility to a minimum [33].

Containment is the primary method for protection against the spreading of dust contamination, e.g. from areas where uranium or hazardous substances are held or processed in a powder or gaseous form. In MOX and enriched U fuel fabrication facilities, in addition of a static containment system (physical barriers such as a glove box) also a dynamic containment system (ventilation) needs to be used to cause a flow of air towards parts of equipment or areas that are more contaminated [33].

Ref [33] recommends that in the design of the ventilation and containment systems in the U fuel fabrication facility "account should be taken of criteria such as: (i) the desired pressure difference between different parts of the premises; (ii) the air replacement ratio in the facility; (iii) the types of filters to be used; (iv) the maximum differential pressure across filters; (v) the appropriate flow velocity at the openings in the ventilation and containment systems (e.g. the acceptable range of air speeds at the opening of a hood); and (vi) the dose rate at the filters."

Guidance on the design of ventilation and containment systems in the MOX fuel fabrication facility is provided in Ref [34]. It involves additional recommendations on the contamination monitoring, gloveboxes ventilation monitoring, buildings compartmentalisation, filters locations etc.

Recommendations on the protection of workers against internal exposure and chemical hazards are provided in Refs [33, 34] and include the following:

- Ventilation systems help to minimize the workers exposure to airborne hazardous material;
- The need for use of respiratory protective equipment can be minimized through careful design of containment and ventilation systems, and the installation as necessary of the monitoring and alarm equipment;
- Location of primary filters as close to the source of contamination as practicable helps to minimize the build-up of uranium oxide and/or MOX powder in the ventilation ducts. Having multiple filters in series is preferable since it avoids reliance on a single barrier;
- Nonporous and easy to clean walls, floors and ceilings in areas of the facility where contamination may occur facilitate the decontamination.

The design of the fuel fabrication facility is expected to provide for the monitoring of the environment of the facility and detection of breaches in the confinement/containment barriers. Uncontrolled dispersion of radioactive substances to the environment from accidents could occur if the containment barrier(s) were impaired. In addition, ventilation of these containment systems, with discharge of exhaust gases through stack via a gas cleaning process such as filtering, can reduce environmental discharges of radioactive materials to very low levels [33, 34]. Efficiency and resistance of these filters to chemicals (e.g. HF), high temperatures in the exhaust gases and fire conditions need to be taken into consideration. There needs to be uninterrupted monitoring and control of the stack exhaust.

Radiation exposure of the public and the environment during normal operation is discussed further in the INPRO methodology manual on environmental impact of stressors [5].

8.4.3. External radiation exposure

External radiation exposure of personnel can be minimized by means of sufficient distance to a radiation source, shielding of it, optimisation of time and quantity of radioactive material storage and processing and restrictions of occupancy near it (see also Section 4.2.5). Personnel radiation monitoring instruments need to be provided for radiation protection.

In U fuel production facilities, a hazard of external radiation exposure exists in areas used for storing UF_6 cylinders, in particular empty ones that have contained reprocessed uranium, and in areas where significant amounts of uranium with a high specific density are present (e.g. in storage areas for pellets and fuels). Shielding provided by vessels and pipework is usually sufficient to control exposure in case of low density UO_2 (used in a conversion or blending unit); however, if reprocessed U is used additional precautions are necessary to limit the exposure of workers to ^{232}U decay products such as ^{208}Tl and ^{212}Bi.

External exposure in a MOX fuel fabrication facility is possible from ^{238}Pu and ^{240}Pu isotopes (neutron emission) and ^{241}Am (gamma emission) generated by the decay of ^{241}Pu during storage. Due to the higher activity of plutonium, shielding provided by vessels and/or glove boxes may not be sufficient to limit exposure adequately. Thus, additional measures can be taken such as limitation of occupancy and proximity, installation of additional shielding, and remote operation of process equipment [34].

8.4.4. Fire and explosion

The facility design needs to account for fire safety (see Section 4.2.4) on the basis of a fire safety analysis and implementation of defence in depth (prevention, detection, control and mitigation). As in all industrial facilities, facilities have to be designed to control fire hazards in order to protect the workers and the public. Fire in these facilities can lead to dispersion of radioactive or toxic materials by destroying the containment barriers or cause a criticality accident by modifying the safe conditions [33].

Ref [33] further recommends:

> "Special fire hazard analyses should be carried out for:
> (a) Processes involving hydrogen, such as conversion, sintering and reduction of uranium oxide;
> (b) Processes involving zirconium in powder form or the mechanical treatment of zirconium metal;
> (c) Workshops such as the recycling shop and laboratories where flammable liquids and/or combustible liquids are used in processes such as solvent extraction;
> (d) The storage of reactive chemicals (e.g. NH_3, H_2SO_4, HNO_3, H_2O_2, pore formers and lubricants);
> (e) Areas with high fire loads, such as waste storage areas;
> (f) Waste treatment areas, especially those where incineration is carried out;
> (g) Rooms housing safety related equipment, e.g. items such as air filtering systems, whose degradation may lead to radiological consequences that are considered to be unacceptable;
> (h) Control rooms."

Explosions can occur due to gases (H_2 used in conversion process and sintering furnaces, etc.) and chemical compounds (ammonium nitrate in recycling processes) [113]. In some cases, explosion can be prevented by using inert gas atmosphere or dilution systems. Recycling systems need to be regularly monitored to prevent ammonium nitrate deposits. In areas with potentially explosive atmospheres, the electrical network and equipment are expected to be protected according to corresponding industrial safety regulations [33].

8.4.5. Containment of radioactive material and/or hazardous chemicals

Leaks may create several hazards [33]:

- Dispersion of radioactive material (e.g. UO_2, U_3O_8 powder and UF_6) and/or toxic chemicals (e.g. HF) leaking from equipment and components like pumps, valves and pipes;
- Flooding by hydrogenous fluids (water, oil, etc.) which can change the moderation in fissile materials and reduce criticality safety.
- Explosions and/or fire caused by leaks of flammable gases (H_2, natural gas, propane) or liquids.

Ref [33] recommends deploying leak detection systems where leaks could occur and equipping vessels containing significant amounts of nuclear material in liquid form with alarms to prevent overfilling and with secondary containment features to prevent criticality.

As discussed for U conversion and enrichment facilities, UF_6 leakage in a fuel production facility needs to be restricted to less than 0.2 mg/m^3 (chemical toxicity limit for natural 'U' and up to 2.5 % enrichment) (see Table 15 in Section 7.4.2.6). The radiation limit is 13 Bq/m^3; this implies 80 μg/m^3 for U with 5 % enrichment and 32 μg/m^3 for U with 10 % enrichment.

To prevent a release of radioactive material and/or toxic chemical to the outside of the plant several barriers (combinations of static and dynamic containments) are necessary: The first barrier is the casing of the equipment (e.g. wall of a vessel or pipe), the second barrier could be a glove box, and the last one is the building of the facility. Additionally, dynamic containments are to be provided by ventilation systems in process equipment and glove boxes but also in the working area of the facility.

8.4.6. Decay heat from MOX fuel material

Isotopes of Pu processed in the MOX fuel fabrication facilities generate heat due to the radioactive decay. Most of the heat is produced from ^{238}Pu decay and can potentially create essential heat loads on the fuel being produced and on the facility systems, structures and components. Appropriate ventilation of the gloveboxes, storage rooms and other production units containing MOX fuel materials is generally sufficient for the temperature control. Ventilation systems have to be monitored continuously and in case of their unavailability the time interval before damage occurs needs to be adequate for repairing the failure or for taking alternative actions.

8.4.7. External hazards

Fuel production facilities are expected to be designed against all credible external hazards (see Section 4.2.1 and 4.2.6).

8.5. ADAPTATION OF THE INPRO METHODOLOGY TO A URANIUM AND MOX FUEL PRODUCTION FACILITY

The use of the INPRO methodology for an assessment of a uranium and MOX fuel fabrication facility required significant modifications and adjustments compared to other types of NFCF. The significant technical differences between the uranium and MOX fuel fabrication facilities are acknowledged but it was found that the application of the INPRO methodology does not require a separate treatment.

In this section the INPRO methodology in the area of safety adapted to these NFCF is presented.

8.5.1. INPRO basic principle for sustainability assessment of fuel fabrication facility in the area of safety

INPRO basic principle for sustainability assessment of fuel fabrication facility in the area of safety: The planned uranium or MOX fuel fabrication facility is safer than the reference fuel fabrication facility. In the event of an accident, off-site releases of radionuclides and/or toxic chemicals are prevented or mitigated so that there will be no need for public evacuation[57].

Rationale of the BP was provided in Section 5.2. Explanation on the requirement of superiority in the INPRO methodology area of NFCF safety is provided in section 6.3.1. INPRO methodology defined a set of requirements to fuel fabrication facilities as displayed in Table 20.

TABLE 20. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF FUEL FABRICATION FACILITY IN THE AREA OF NFCF SAFETY

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR1: Robustness of design during normal operation: The uranium or MOX fuel fabrication facility assessed is more robust than the reference design with regard to operation and systems, structures and components failures.	CR1.1: Design of normal operation systems	IN1.1: Robustness of design of normal operation systems. AL1.1: Superior to that in the reference design.
	CR1.2: Subcriticality	IN1.2: Subcriticality margins. AL1.2: Sufficient to cover uncertainties and avoid criticality.
	CR1.3: Facility performance	IN1.3: Facility performance attributes. AL1.3: Superior to those in the reference design.
	CR1.4: Inspection, testing and maintenance	IN1.4: Capability to inspect, test and maintain. AL1.4: Superior to that in the reference design.
	CR1.5: Failures and deviations from normal operation	IN1.5: Expected frequency of failures and deviations from normal operation. AL1.5: Lower than that in the reference design.
	CR1.6: Occupational dose	IN1.6: Occupational dose values during normal operation and AOOs. AL1.6: Lower than the dose constraints.

[57] Other protective measures still may be needed. Effective emergency planning, preparedness and response capabilities will remain a prudent requirement as discussed in the INPRO methodology area of Infrastructure.

TABLE 20. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF FUEL FABRICATION FACILITY IN THE AREA OF NFCF SAFETY (cont.)

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR2: Detection and interception of AOOs: The uranium or MOX fuel fabrication facility assessed has improved capabilities to detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions.	CR2.1: I&C systems and operator procedures	IN2.1: I&C system to monitor, detect, trigger alarms and, together with operator actions, intercept and compensate AOOs. AL2.1: Availability of such systems and operator procedures.
	CR2.2: Grace periods for AOOs	IN2.2: Grace periods until human actions are required after AOOs. AL2.2: Adequate grace periods are defined in design analyses.
UR3: Design basis accidents: The frequency of occurrence of DBAs in the uranium or MOX fuel fabrication facility assessed is reduced. If an accident occurs, engineered safety features and/or operator actions are able to restore the assessed facility to a controlled state and subsequently to a safe state, and the consequences are mitigated to ensure the confinement of nuclear and/or toxic chemical material. Reliance on human intervention is minimal, and only required after sufficient grace period.	CR3.1: Frequency of DBAs	IN3.1: Calculated frequency of occurrence of DBAs. AL3.1: Lower than that in the reference design.
	CR3.2: Engineered safety features and operator procedures	IN3.2: Reliability and capability of engineered safety features and/or operator procedures. AL3.2: Superior to those in the reference design.
	CR3.3: Grace periods for DBAs	IN3.3: Grace periods for DBAs until human intervention is necessary. AL3.3: Longer than those in the reference design.
	CR3.4: Barriers	IN3.4: Number of confinement barriers maintained (intact) after DBAs. AL3.4: At least one.
	CR3.5: Robustness of containment design	IN3.5: Containment loads covered by design of the facility assessed. AL3.5: Greater than those in the reference design.
UR4: Severe plant conditions: The frequency of an accidental release of radioactivity into the environment is reduced. The source term of accidental release into the environment remains well within the envelope of the reference facility source term and is so low that calculated consequences would not require public evacuation.	CR4.1: In-facility severe accident management	IN4.1: Natural or engineered processes, equipment, and AM procedures and training to prevent an accidental release to the environment in the case of accident. AL4.1: Sufficient to prevent an accidental release to the environment and regain control of the facility.
	CR4.2: Frequency of accidental release into environment	IN4.2: Calculated frequency of an accidental release of radioactive materials and/or toxic chemicals into the environment. AL4.2: Lower than that in the reference facility.
	CR4.3: Source term of accidental release into environment	IN4.3: Calculated inventory and characteristics (release height, pressure, temperature, liquids/gas/aerosols, etc) of an accidental release. AL4.3: Remains well within the inventory and characteristics envelope of the reference facility source term and is so low that calculated consequences would not require evacuation of population.

TABLE 20. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF FUEL FABRICATION FACILITY IN THE AREA OF NFCF SAFETY (cont.)

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR5: Independence of DID levels and inherent safety characteristics: An assessment is performed for the uranium or MOX fuel fabrication facility to demonstrate that the DID levels are more independent from each other than in the reference design. To excel in safety and reliability, the assessed facility strives for better elimination or minimization of hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics.	CR5.1: Independence of DID levels	IN5.1: Independence of different levels of DID in the assessed fuel fabrication facility. AL5.1: More independence of the DID levels is demonstrated compared to that in the reference design, e.g. through deterministic and probabilistic means, hazards analysis, etc.
	CR5.2: Minimization of hazards	IN5.2: Examples of hazards: fire, flooding, release of radioactive material, radiation exposure, etc. AL5.2: Hazards are reduced in relation to those in the reference facility.
UR6: Human factors related to safety: Safe operation of the assessed fuel fabrication facility is supported by accounting for HF requirements in the design and operation of the facility, and by establishing and maintaining a strong safety culture in all organizations involved in the life cycle of the facility.	CR6.1: Human factors	IN6.1: Human factors addressed systematically over the life cycle of the fuel fabrication facility. AL6.1: Evidence is available.
	CR6.2: Attitude to safety	IN6.2: Prevailing safety culture. AL6.2: Evidence is provided by periodic safety reviews.
UR7: RD&D for advanced designs: The development of innovative design features of the assessed fuel fabrication facility includes associated RD&D to bring the knowledge of facility characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for operating facilities.	CR7.1: RD&D	IN7.1: RD&D status. AL7.1: RD&D defined, performed and database developed.
	CR7.2: Safety assessment	IN7.2: Adequate safety assessment. AL7.2: Approved by a responsible regulatory authority.

8.5.2. User requirement UR1: Robustness of design during normal operation

The rationale of UR1 was provided in Section 5.3. UR1 is focused on prevention of abnormal operation and failures. For a U or MOX fuel fabrication facility, the following examples[58] of AOOs to be prevented are similar to those presented in Section 7.4.2 for refining/ conversion and enrichment facilities [33, 34]:

- Leakage (e.g. due to corrosion) of flammable (explosive) gases such as H_2;
- Leakage of radioactive and/or toxic chemicals such as U and U-Pu compounds, UF_6, HF, and NH_3;
- Fire in a room with significant amounts of fissile or toxic chemical material;

[58] Assuming that these occurrences have moderate consequences and cannot be qualified as accident conditions.

- Loss of utilities such as electrical power, pressurized air, coolant, ventilation.

The criteria selected for user requirement UR1 are presented in Table 21.

TABLE 21. CRITERIA FOR USER REQUIREMENT UR1

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR1: Robustness of design during normal operation: The uranium or MOX fuel fabrication facility assessed is more robust than the reference design with regard to operation and systems, structures and components failures.	CR1.1: Design of normal operation systems	IN1.1: Robustness of design of normal operation systems. AL1.1: Superior to that in the reference design.
	CR1.2: Subcriticality	IN1.2: Subcriticality margins. AL1.2: Sufficient to cover uncertainties and avoid criticality.
	CR1.3: Facility performance	IN1.3: Facility performance attributes. AL1.3: Superior to those in the reference design.
	CR1.4: Inspection, testing and maintenance	IN1.4: Capability to inspect, test and maintain. AL1.4: Superior to that in the reference design.
	CR1.5: Failures and deviations from normal operation	IN1.5: Expected frequency of failures and deviations from normal operation. AL1.5: Lower than that in the reference design.
	CR1.6: Occupational dose	IN1.6: Occupational dose values during normal operation and AOOs. AL1.6: Lower than the dose constraints.

8.5.2.1. *Criterion CR1.1: Design of normal operation systems*

Indicator IN1.1: Robustness of design of normal operation systems.

Acceptance limit AL1.1: Superior to that in the reference design.

Normal operation systems and equipment relevant for safety used in a fuel production facility need to be designed against loads caused by postulated initiating events including events associated with external hazards (see Section 4.2.1). The design (e.g. mechanical, thermal, electrical, etc.) of normal operation systems in a fuel production facility can be made more robust, i.e. reducing the likelihood of failures, by increasing the design margins, improving the quality of manufacture and construction, and by use of materials of higher quality. It is acknowledged that increasing the robustness of a facility design is a challenging task for a designer because enhancing one aspect could have a negative influence on other aspects. Thus, an optimised combination of design measures is necessary to increase the overall robustness of a design.

The **acceptance limit AL1.1** of **CR1.1** is met if evidence available to the INPRO assessor shows that the design of the facility assessed is superior in this respect to the reference design (e.g. has increased design margins, improved quality of manufacture and construction, or uses materials of higher quality), or, in case a reference facility could not be defined, took best international practice into account and is therefore state of the art technology.

8.5.2.2. *Criterion CR1.2: Subcriticality*

Indicator IN1.2: Subcriticality margins.

Acceptance limit AL1.2: Sufficient to cover uncertainties and avoid criticality.

Criticality control in fuel production facilities necessitates the mass control of fissile material, the use of safe geometry (with respect to criticality) in equipment layout to provide safe separation between equipment as well as storage systems, the minimization of hydrogenous materials in process and the use of neutron absorbing materials.

As proposed by the INPRO task group in this area and previously discussed in section 7.4.2.2 for uranium refining/ conversion and enrichment facilities, the adequate avoidance of criticality in facilities that handle MOX, Pu or U enriched above 1 % ^{235}U is expected to be shown by a criticality analysis that demonstrates a design margin of k_{eff} < 0.90 for all possible configurations of fissile material. In this analysis, all parameters relevant to criticality, such as mass concentration, shape, moderation, etc, have to be considered. All process equipment in the material handling area needs to be designed to remain subcritical under submerged and water filled conditions.

The **acceptance limit AL1.2** of **CR1.2** is met if evidence available to the INPRO assessor shows that in the facility assessed no critical configuration can occur taking uncertainties into account.

8.5.2.3. Criterion CR1.3: Facility performance

Indicator IN1.3: Facility performance attributes.

Acceptance limit AL1.3: Superior to those in the reference design.

Superior performance attributes can increase the robustness of a uranium or MOX fuel fabrication facility. A distinctive feature of fuel fabrication facilities is the presence of large inventories of powders of uranium oxide, plutonium oxide or mixed oxide. These are usually in finely divided form, and unless a high quality of operation is ensured, spillage of these fuel materials inside the enclosures could lead to long term accumulation in various difficult-to-access areas and in glass panels of glove boxes. This could ultimately lead to increased dosage to the operator.

High quality of operation, by way of intensive training of operators, is also essential to ensure that human factors do not lead to unexpected accumulations of fissile material in any part of the plant and thus lead to criticality: Strict adherence to administrative procedures is an indication of high quality of training. An inappropriate response to an alarm indicating an emergency could also be a result of inadequate operator training.

The strategy of ageing management is expected to cover all relevant stages in the fuel production facility lifecycle, including design, manufacture, construction, commissioning, operation and decommissioning, and needs to address all relevant mechanisms of ageing for the operational states and accident conditions influencing a given system. The designer of a fuel production facility has to determine the design life of SSCs important to safety, provide appropriate design margins to take due account of age related degradation and provide methods and tools for assessing ageing during the fuel production facility operation. The operating organization has to develop a plan for preparing, coordinating, maintaining and improving activities for ageing management implementation at the different stages of the fuel production facility lifecycle. Implementation of this plan will involve activities for managing ageing mechanisms, detecting and assessing ageing effects, and managing ageing effects.

A high degree of automation/remote control/robotics would lead to reduction of dose received by the operators. Typical items that are taken into account for establishing acceptance criteria for facility performance include:

- High(er) degree of remote control;
- Availability of operations manuals and emergency instructions manuals;
- Availability of procedure for the feedback on application of operations manuals;
- Availability of surveillance requirements including periodic tests to verify the performance level for safe operation;
- Consideration of ageing management in the design documentation;
- Availability of plan for implementation of ageing management;

- Periodic and intensive training of operators;
- Periodic mock-ups to ensure readiness of operators to handle emergencies.

The **acceptance limit AL1.3** of **CR1.3** is met if evidence available to the INPRO assessor shows that the design of the facility assessed is superior to a reference design or, in case a reference facility could not be defined, took best international practice into account and is therefore state of the art technology.

8.5.2.4. Criterion CR1.4: Inspection, testing and maintenance

Indicator IN1.4: Capability to inspect, test and maintain.

Acceptance limit AL1.4: Superior to that in the reference design.

To achieve an improved capability to inspect, test and maintain, the design of fuel fabrication facility assessed is expected to permit efficient and intelligent inspection, testing and maintenance and not just require more inspections and more testing. In particular, the programs of inspection, testing and maintenance need to be driven by a sound understanding of failure mechanisms (corrosion, erosion, fatigue, etc.), so that the right locations are inspected and the right systems, structures and components are tested and maintained at the right time intervals.

The **acceptance limit AL1.4** of **CR1.4** is met if evidence available to the INPRO assessor shows that the capability to inspect, test and maintain systems relevant to safety in the facility assessed is superior to that in the reference design or, in case a reference facility could not be defined, is state of the art and allows easy inspection, testing and maintenance.

8.5.2.5. Criterion CR1.5: Failures and deviations from normal operation

Indicator IN1.5: Expected frequency of failures and deviations from normal operation.

Acceptance limit AL1.5: Lower than that in the reference design.

The estimated frequencies of the AOOs selected (see beginning of Section 8.5.2) for a fuel production facility need to be derived from operational experience and supported by PSA. For the design assessed, theses frequencies can be reduced through achieving increased robustness of the design (discussed in CR1.1 above), high quality of operation (discussed in CR1.2), and efficient and intelligent inspection and maintenance (discussed in CR1.3).

The **acceptance limit AL1.5** of **CR1.5** is met if evidence available to the INPRO assessor shows that in the facility assessed the frequencies of AOOs are lower than those in the reference design, or, in case a reference facility could not be defined, that the facility assessed took best international practice into account and is therefore state of the art technology. If quantitative results from operational experience and PSA are not available, alternatively, deterministic analysis needs to be developed that indicates the reduction of probability of occurrence for AOOs.

8.5.2.6. Criterion CR1.6: Occupational dose

Indicator IN1.6: Occupational dose values during normal operation and AOOs.

Acceptance limit AL1.6: Lower than the dose constraints.

Fuel production facilities may control contamination using such independent strategies as maintaining differential pressure in process enclosures and operating areas, providing easy access to equipment in operating areas, using automation/robotics for handling radioactive materials, zoning the layout of the plant for hazardous operations, providing single port entry and exit for personnel and equipment and employing multiple levels of filtration.

The assessment of CR1.6 for a conversion and enrichment facility was presented in Section 7.4.2.6 and is deemed substantially similar to the corresponding assessment for a fuel production facility (U, Pu or MOX). Therefore, the assessor is requested to use the assessment approach described for a conversion and enrichment facility also for a fuel production facility.

8.5.3. User requirement UR2: Detection and interception of AOO

Rationale of UR2 was provided in Section 5.4. Criteria selected for user requirement UR2 are presented in Table 22.

TABLE 22. CRITERIA FOR USER REQUIREMENT UR2.

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR2: Detection and interception of AOOs: The uranium or MOX fuel fabrication facility assessed has improved capabilities to detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions.	CR2.1: I&C systems and operator procedures	IN2.1: I&C system to monitor, detect, trigger alarms and, together with operator actions, intercept and compensate AOOs. AL2.1: Availability of such systems and operator procedures.
	CR2.2: Grace periods for AOOs	IN2.2: Grace periods until human actions are required after AOOs. AL2.2: Adequate grace periods are defined in design analyses.

8.5.3.1. Criterion CR2.1: I&C systems and operator procedures

Indicator IN2.1: I&C system to monitor, detect, trigger alarms and, together with operator actions, intercept and compensate AOOs.

Acceptance limit AL2.1: Availability of such systems and operator procedures.

A fuel production facility is expected to be designed to cope with AOOs (see beginning of Section 8.5.2) using automatic operational systems, i.e. I&C systems that bring the facility back to normal operating conditions. In case automatic systems are not available, adequate operator procedures need to be. Passive and active control systems are deemed more reliable than administrative (manual) control. The operator needs to get appropriate information in a control room about automatic actions during normal operation and AOOs and the status and performance of the facility.

Fuel fabrication facilities involve many safety critical systems such as glove boxes, furnaces, vacuum systems etc, thus, instrumentation and control systems play an important role in ensuring healthiness and safety of various systems and ensuring that they operate in safe regimes of parameters. The design analysis is expected to define safe operating conditions for every system, and different limits for alarm and shutdown conditions need to be indicated. For example, furnaces need to be equipped with temperature control systems to shut down the power supply to prevent escalation of temperature in case of loss of cooling water. Pressure control systems in glove boxes need to be able to detect loss of negative pressure (e.g. through a puncture in a glove) and actuate additional exhaust systems to ensure that the glove box pressure remains below the one in the operating area. Measurement of these parameters based on different principles wherever applicable and by more than one device for measurement would provide enhanced safety.

Online monitoring systems, with accessibility to inspect and more than one way to measure the same parameter, are necessary requirements. Access has to be provided for condition monitoring parameters and trending to predict incipient failures. In the ventilation systems, continuous monitoring of pressure drops across HEPA filters would ensure an adequate number of air changes in operating areas. Similarly, on-line monitoring is required to ensure adequate cooling water supply to sintering furnaces and ensure that the furnace is shut down when water flow is reduced below a certain level.

The **acceptance limit AL2.1** of **CR2.1** is met if evidence available to the INPRO assessor shows that I&C systems are available in the facility assessed that are capable of detecting failures and deviations from normal operation of systems relevant for safety, providing alarm, initiate automatic (and manual actions), and bring the facility back to normal operation.

8.5.3.2. Criterion CR2.2: Grace periods for AOOs

Indicator IN2.2: Grace periods until human actions are required after AOOs.

Acceptance limit AL2.2: Adequate grace periods are defined in design analyses.

An explanation of 'adequate grace period' is provided in section 6.3.3.2. The grace period available for the operator for each AOO needs to be defined in the safety analysis of the facility design. After detection of an AOO (see beginning of Section 8.5.2) in a fuel production facility, the automatic operational systems (presented in Section 8.5.3.1 above) needs to control these incidents before the operator intervention. The operation manual is expected to list all anticipated incidents, a corresponding action plan and the time until the actions have to be completed by the workers. For example, the design of glove boxes in MOX fabrication facilities needs to ensure that, in the event of a ventilation failure, radioactivity levels in the operating areas do not exceed regulatory limits for at least one hour, so that operators can safely shut down furnaces and other systems before evacuating the laboratory.

In addition to the automatic actions of the normal operation systems a fuel fabrication facility is expected to have sufficient inertia to withstand transients, i.e. react slowly after AOOs. For example, design of furnaces and (redundant) cooling systems needs to ensure that in the event of a temporary loss of cooling water supply, the furnace casing temperature will not exceed design limits within a reasonable time frame to enable the operator to bring the furnaces to a safe shut down state if necessary or continue to operate if he can restore water supply in time.

The **acceptance limit AL2.2** of **CR2.2** is met if evidence available to the INPRO assessor shows that adequate grace periods have been determined for all AOOs in the design analysis for the facility assessed.

8.5.4. User requirement UR3: Design basis accidents

The rationale of UR3 was provided in Section 5.5. Refs [33, 34] recognise that specification of DBAs will depend on the facility design and national requirements. However, they recommend that particular consideration needs to be given to the following hazards in the specification of DBAs at fuel fabrication facilities [33, 34]:

- A nuclear criticality accident;
- A release of uranium, e.g. in the explosion of a reaction vessel during the conversion of UF_6 to UO_2;
- A hydrogen explosion, e.g. in the pellet sintering equipment;
- A release of UF_6 due to the rupture of a hot cylinder;
- A release of HF due to the rupture of a storage tank;
- A fire;
- Natural phenomena such as earthquakes, flooding, or tornadoes;
- An aircraft crash.

The criteria selected for user requirement UR3 are presented in Table 23.

TABLE 23. CRITERIA FOR USER REQUIREMENT UR3

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR3: Design basis accidents: The frequency of occurrence of DBAs in the uranium or MOX fuel fabrication facility assessed is reduced. If an accident occurs, engineered safety features and/or operator actions are able to restore the assessed facility to a controlled state and subsequently to a safe state, and the consequences are mitigated to ensure the confinement of nuclear and/or toxic chemical material. Reliance on human intervention is minimal, and only required after sufficient grace period.	CR3.1: Frequency of DBAs	IN3.1: Calculated frequency of occurrence of DBAs. AL3.1: Lower than that in the reference design.
	CR3.2: Engineered safety features and operator procedures	IN3.2: Reliability and capability of engineered safety features and/or operator procedures. AL3.2: Superior to those in the reference design.
	CR3.3: Grace periods for DBAs	IN3.3: Grace periods for DBAs until human intervention is necessary. AL3.3: Longer than those in the reference design.
	CR3.4: Barriers	IN3.4: Number of confinement barriers maintained (intact) after DBAs. AL3.4: At least one.
	CR3.5: Robustness of containment design	IN3.5: Containment loads covered by design of the facility assessed. AL3.5: Greater than those in the reference design.

8.5.4.1. Criterion CR3.1: Frequency of DBAs

Indicator IN3.1: Calculated frequency of occurrence of DBAs.

Acceptance limit AL3.1: Lower than that in the reference design.

Examples of the DBAs to be considered in a fuel fabrication facility have been provided above in the beginning of Section 8.5.4. The frequency of occurrence of a DBA in the facility assessed is to be determined via a probabilistic risk assessment. Ref [18] gives an overview of the methods used for probabilistic evaluations of NFCFs, such as layer of protection analysis and the index method, and the areas of their application. Several examples of probabilistic studies of NFCFs and an overview of the regulatory requirements in different countries can be found in Ref [114].

The frequency of DBA caused by external hazards can be influenced by the designer, e.g. via an increase of robustness of the confinement wall, and by the owner/ operator of the facility by selecting an appropriate site (see UR7).

When the probabilistic risk assessment results are not available for the NFCF assessed, the superiority of the new design, i.e. improvements to reduce frequency of initiating events, can be demonstrated deterministically.

The **acceptance limit AL3.1** of **CR3.1** is met if evidence available to the INPRO assessor shows that in the facility assessed based on probabilistic analyses the frequency for the defined DBAs is superior to a reference design. If quantitative results are not available a deterministic analysis needs to support a reduction of these frequencies based on an increase of design robustness, high quality of operation, an intelligent inspection and maintenance programs, advanced I&C systems and increased inertia.

8.5.4.2. Criterion CR3.2: Engineered safety features and operator procedures

Indicator IN3.2: Reliability and capability of engineered safety features and/or operator procedures.

Acceptance limit AL3.2: Superior to those in the reference design.

In case of a DBA (see beginning of Section 8.5.4) there need to be automatic reliable engineered safety features available that after detection of an accident are capable of controlling the accident, restoring the facility to a controlled state, and keeping the consequences within authorized limits. To assure necessary reliability these features have to be designed with sufficient level of redundancy, diversity and independence.

In case automatic systems are not available, adequate operator procedures are necessary. Redundant, diversified and independent passive and automatic active systems are deemed to be more reliable than administrative control (operator intervention) however it is acknowledged that they are difficult to be designed for fuel fabrication facility.

As mentioned above the facility is expected to have engineered safety features protecting against DBA caused by (credible) external hazards (see Section 4.2.1 and 4.2.6).

The **acceptance limit AL3.2** of **CR3.2** is met if evidence available to the INPRO assessor shows that the reliability and capability of engineered safety features in the facility assessed is superior to a reference design and assure that after the beginning of a DBA the necessary actions to mitigate the consequences of the accidents will be timely initiated. Alternatively, if a reference facility cannot be found, it could be demonstrated that the design of the facility assessed took available information on best international practice into account and is therefore state of the art.

8.5.4.3. Criterion CR3.3: Grace periods for DBAs

Indicator IN3.3: Grace periods for DBAs until human intervention is necessary.

Acceptance limit AL3.3: Longer than those in the reference design.

An explanation of 'adequate grace period' is provided in section 6.3.3.2 as introduced earlier for control of AOOs (see CR2.2) in Level 2 of DID. The criterion CR3.3 'grace period for DBA' implies a similar concept. For DBA (caused by events associated with internal and external hazards) the criterion requires that the system response (inertia) and/or automatic actions of active (and/or passive) safety features provide an adequate grace period for the operator to intervene. Adequate grace periods in the new facility are also assumed to be longer than those in the reference design.

For example, a criticality accident in a fuel fabrication plant could be caused by human errors such as double batching or by flooding of glove boxes containing large inventories of fissile material. Provision of a criticality monitor (e.g. neutron counter, liquid level monitor in a glove box) is essential[59]. In the event of criticality, a grace time of a few minutes only may be available to take necessary protective measures, e.g. halt flow of liquid, close valve. In the event of flooding of glove boxes due to a coolant pipe rupture, and unavailability of automatic safety features, the grace time available for the operator to avoid criticality or release of radioactive material would depend on the design of the box and the flow rate of water. The safety analysis needs to take into account these factors and define the time limits sufficient for human action. The grace periods have to be provided for each DBA by the design[60].

The **acceptance limit AL3.3** of **CR3.3** is met if evidence available to the INPRO assessor shows that in the facility assessed the grace periods are superior to a reference design. Alternatively, if a reference facility cannot be found, it could be demonstrated that the design

[59] In a defence in depth approach the level monitor and alarm/display are expected to be completely independent of the automatic system and its detector including, where possible, measure a different physical parameter or and/or use a different technology, e.g. ultrasonic and pneumatic level detection.
[60] The design/safety analysis process needs to be iterative to ensure all safety requirement are satisfied.

of the facility assessed took available information on best international practice into account and is therefore state of the art.

8.5.4.4. Criterion CR3.4: Barriers

Indicator IN3.4: Number of confinement barriers maintained (intact) after DBAs.

Acceptance limit AL3.4: At least one.

The design of engineered safety features is expected to provide deterministically for continued integrity at least of one barrier containing the radioactive and chemically toxic material following any DBA caused by events associated with internal or external hazards. Alternatively, the probability of losing all barriers could be used as an INPRO methodology indicator with a sufficient low value of it as acceptance limit.

The most important engineered safety features of a fuel fabrication facility are the barriers against a release of radioactive material into the environment. At present, all Pu (but also some U) based materials are handled in glove boxes, whose panels and gloves constitute one barrier (another barrier is the building wall). However, it is important to ensure that a glove box is designed as a second barrier and larger inventories of fuel materials are always maintained in another suitable enclosure which would constitute the first barrier. For example, in glove boxes containing equipment with moving parts such as a press or grinder, this equipment needs to be surrounded by a safe enclosure which would ensure that any flying object from the equipment would not damage the glass panel of the box.

It is apparent that the higher the number of such barriers, the safer the system with respect to release of radioactivity and thus would meet the requirement of defence in depth concept.

The **acceptance limit AL3.4** of **CR3.4** is met if evidence available to the INPRO assessor shows that after a DBA at least one barrier remains intact in the facility assessed avoiding a large release of radioactivity and/or toxic chemicals to the outside of the facility.

8.5.4.5. Criteria CR3.5: Robustness of containment design

The assessment of CR3.5 presented for a U conversion and enrichment facility in Sections 7.4.4.5 is deemed to be sufficient similar to a fuel fabrication facility. Thus, this approach can be used by the assessor also for the fuel fabrication facility.

8.5.5. User requirement UR4: Severe plant conditions

Rationale of UR4 was provided in Section 5.6. INPRO methodology has defined the three criteria for UR4: in-facility severe accident management, frequency of accidental release into environment, source term of accidental release into environment.

It is noted that a fuel production facility using enriched uranium (> 1 % of ^{235}U) or plutonium has a higher probability of a criticality accident due to the existence of high density fissile material (pellets) than an enrichment plant where fissile material is mostly in volatile form (UF_6). However, the INPRO assessment of a fuel production facility against user requirement UR4 (Severe plant conditions) is deemed to be sufficiently similar to the assessment of an enrichment facility. Therefore, the assessor is requested to use the assessment method of UR4 described in Section 7.4.5 for an enrichment facility (including criteria, indicators and acceptance limits) also for a fuel production facility.

8.5.6. User requirement UR5: Independence of DID levels and inherent safety characteristics

Rationale of UR5 was provided in Section 5.7. Criteria selected for user requirement UR5 are presented in Table 24.

TABLE 24. CRITERIA FOR USER REQUIREMENT UR5

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR5: Independence of DID levels and inherent safety characteristics: An assessment is performed for the uranium or MOX fuel fabrication facility to demonstrate that the DID levels are more independent from each other than in the reference design. To excel in safety and reliability, the assessed facility strives for better elimination or minimization of hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics.	CR5.1: Independence of DID levels	IN5.1: Independence of different levels of DID in the assessed fuel fabrication facility. AL5.1: More independence of the DID levels is demonstrated compared to that in the reference design, e.g. through deterministic and probabilistic means, hazards analysis, etc.
	CR5.2: Minimization of hazards	IN5.2: Examples of hazards: fire, flooding, release of radioactive material, radiation exposure, etc. AL5.2: Hazards are reduced in relation to those in the reference facility.

8.5.6.1. Criterion CR5.1: Independence of DID levels

Indicator IN5.1: Independence of different levels of DID in the assessed fuel fabrication facility.

Acceptance limit AL5.1: More independence of the DID levels is demonstrated compared to that in the reference design, e.g. through deterministic and probabilistic means, hazards analysis, etc.

Systems that provide for different levels of defence in depth may be either dependent or independent. Independent systems can provide protection from potential hazards with higher reliability. Using the same system or several dependant systems in different levels of defence in depth can make these levels vulnerable to the common cause failure. Ref [18] states:

> "To qualify as independent, the failure of one item relied on for safety (IROFS) should neither cause the failure nor increase the likelihood of failure of another IROFS. No single credible event should be able to defeat the system of IROFS such that an accident is possible. A systematic method of hazard identification should thus be used to provide a high degree of assurance that all credible failure mechanisms that could contribute to (i.e. by initiating or failing to prevent or mitigate) an accident have been identified."

Ref [18] further provides an exemplary list of factors undermining independence of the systems, structures and components, and therefore having significant effect on the likelihood of an accident sequence:

> "A partial list of conditions that will almost always lead to two or more IROFS not being independent follows:
>
> - The same individual performs administrative actions.
> - Two different individuals perform administrative actions but use the same equipment and/or procedures.
> - Two engineered controls share a common hardware component or common software.
> - Two engineered controls measure the same physical variable using the same model or type of hardware.
> - Two engineered controls rely on the same source of essential utilities (e.g. electricity, instrument air, compressed nitrogen, water).
> - Two engineered controls are collocated such that credible internal or external events (e.g. structural failure, forklift impacts, fires, explosions, chemical releases) can cause both to fail.
> - Administrative or engineered controls are susceptible to failure because of the presence of credible environmental conditions (e.g. two operator actions defeated by corrosive atmosphere, sensors rendered inoperable because of high temperature)."

The analysis of independence of systems, structures and components in NFCF is normally part of the application of the ‘double contingency principle’ defined in Ref [115]. This principle states that “process designs should, in general, incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.”

It is expected that the deterministic method for assessing the DID capabilities of a nuclear reactor design described in Ref [116] will be adapted to fuel fabrication facility. This method is based on objective trees for each level of DID defining the following elements from top to bottom: the objective of the DID level, the relevant safety functions to be met, identified general challenges to the safety functions based on specific root mechanisms for each of these challenges and a list of provisions in design and operation for preventing the mechanism from occurring.

Special attention is expected to be demonstrated in the design to such hazards as fire, flooding or earthquakes which could potentially impair several levels of DID; for example, they could bring about accident situations and, at the same time, inhibit the means of coping with such situations [39].

The safety analysis report of a fuel fabrication facility needs to demonstrate clearly the independence of the levels of defence. A probabilistic safety analysis [117], if done carefully, would highlight systems and elements which are not sufficiently independent, and identify cross-links which compromise the independence of the levels of DID. A fuel fabrication facility assessed is expected to demonstrate calculated frequency ranges of reaching the different levels of DID after an initiating event below (superior to) those of a reference facility.

The **acceptance limit AL5.1** (*independence of DID levels*) is met for the fuel fabrication facility assessed if evidence available to the INPRO assessor shows that demonstrates improved independence of the different levels of DID in comparison to a reference plant based on a deterministic and probabilistic analyses. Alternatively, if a reference facility cannot be found, it could be demonstrated that the design of the facility assessed took available information on best international practice into account and is therefore state of the art.

8.5.5.2. Criterion CR5.2: Minimization of hazards

The assessment of CR5.1 (minimisation of hazards) presented for a uranium conversion and enrichment facility in Section 7.4.6.1 is deemed to be sufficient similar to a fuel fabrication facility. Thus, this approach can be used by the assessor also for the fuel fabrication facility.

8.5.7. User requirement UR6 and UR7

Rationale for UR6 and UR7 is provided in Section 5.8 and 5.9. Assessment of user requirement UR6 (human factors related to safety) and UR7 (RD&D for advanced designs) for fuel fabrication facilities (U, Pu, MOX) is deemed to be sufficiently similar to the assessment method of UR6 and UR7 described in Sections 6.3.7 and 6.3.8 for mining and milling facilities (including criteria, indicators and acceptance limits).

9. REPROCESSING OF SPENT NUCLEAR FUEL

In this section, firstly, a short description of the main processes in a reprocessing facility is given and the corresponding specific safety issues are discussed. Secondly, the assessment method will be discussed based on the corresponding criteria of the INPRO methodology in the area of safety, which have been, where necessary, adapted to the specific issues of this type of facility.

Fuel reprocessing is the technology for recovery and recycle of nuclear fuel including bred fissile material from spent fuel assemblies (or fuel elements or bundles) discharged from a nuclear reactor after irradiation. As of beginning of 2017 the following five countries have commercial reprocessing facilities in operation: France (La Hague), Great Britain (Sellafield), India (Trombay, Tarapur, and Kalpakkam), Japan (Rokkasho, Tokai), and the Russian Federation (Tscheljablinsk). These facilities all use variants of the so-called PUREX process to be briefly described in the following.

In recognition of the importance of spent fuel reprocessing in the back end of the fuel cycle, the IAEA has provided a forum for exchange of information on the status and trends in spent fuel reprocessing since the 1970s, from which several publications have been issued [45, 118–120] (see also Refs [121–123]).

Spent nuclear fuel transportation has not been considered in this manual as an independent stage of nuclear fuel cycle. INPRO methodology implies that safety of spent fuel transportation is to be considered as part of the INPRO assessment either of the spent fuel storage facility or the spent fuel reprocessing facility.

9.1. PUREX PROCESS

Historically, several extraction systems were explored for reprocessing of nuclear fuel, before an efficient method was identified. The combination known generically as PUREX (plutonium uranium redox extraction), which utilizes as extractant tri-butyl phosphate (TBP) mixed in a largely inert hydrocarbon diluent, has now replaced all earlier solvent extraction media. The PUREX process has a number of advantages in comparison to earlier versions including lower solvent volatility and flammability, higher chemical and radiation stability of the solvent and lower operating costs. Since the opening of the first PUREX plant at Savannah River in 1954, the PUREX process has been utilized in a variety of flow sheets, and, as stated before, is still being used in all commercial reprocessing plants currently operating.

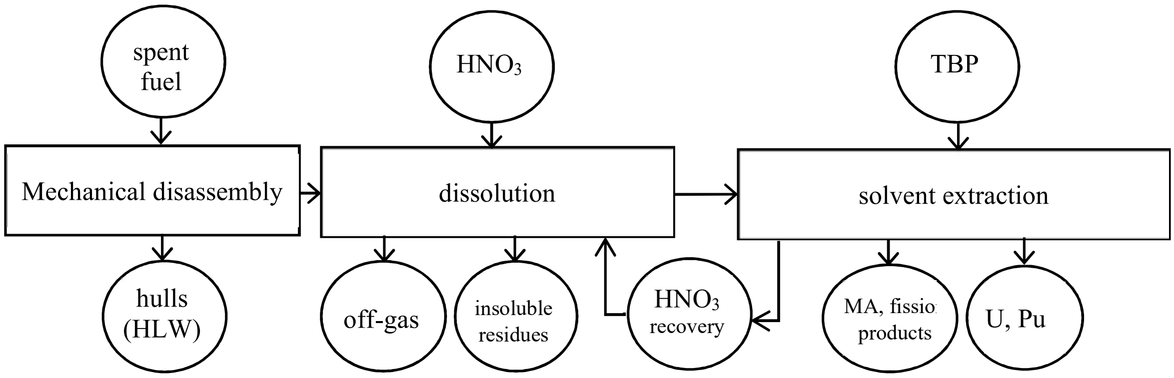

FIG. 8. Conventional PUREX process.

Commercial reprocessing relies on a series of four main technological operations: fuel handling and shearing, fuel dissolution, materials separation and purification, and finally, waste treatment and conditioning. The aqueous process route is assumed to be viable up to a burn-up of at least 100 GWd/t; radiation and chemical degradation of the extractant being the limitation for processing fuels of higher burn-up. A typical schematic of a conventional PUREX process is shown in Fig. 8.

The major steps in the PUREX process are:

(1) Head end process: the fuel is received, chopped and prepared for dissolution;
(2) The U, Pu, minor actinides and fission products go into solution in nitric acid;
(3) The hull is monitored for residual fissile material content and treated as solid waste;
(4) Off gas is treated to remove radioactive isotopes such as iodine, krypton and xenon:
(5) Separation of U and Pu from fission products and minor actinides, using aqueous organic solvents. In the PUREX process, aqueous phase is nitric acid and organic phase is TBP in hydro-carbon solution. Aqueous/ organic phase contact is achieved by centrifugal or other types of contactors such as pulsed columns and mixer-settlers;
(6) Purification and concentration of U and Pu and storage; and
(7) Recovery of minor actinides and long lived fission products to be managed as high level waste.

9.2. OTHER REPROCESSING METHODS

There is a variety of other reprocessing methods under different state of development. They can be grouped into the following categories [120]:

- Evolutionary technologies based on aqueous separation methods using TBP as extractant (derived from the PUREX process). Examples are: COEX (France), NUEX (UK), Simplified PUREX and REPA (Russian Federation), THOREX (India), NEXT (Japan);
- Innovative aqueous processes using new extractants. Examples are: DIAMEX-SANEX and GANEX (France), UREX+3a and UREX+1a (USA), and PARC and ARTIST (Japan);
- Non aqueous technologies (dry route) – pyro chemical processes. Examples are: Pyro-chemical-(liq-liq) process (France), DDP (Russian Federation), Electro Metallurgical process (USA);
- Hybrid methods combining hydro and pyro chemical processes. Examples are: FLUOREX (Japan), Combined process including gas fluoride and extraction technologies (Russian Federation); and
- Innovative processes using fluid extraction or precipitation methods. Examples are: Fluid extraction (Russian Federation, Japan), Ion-exchange processes (Japan, Belgium), Sedimentation (Japan).

9.3. SAFETY ISSUES IN A REPROCESSING FACILITY

Among the NFCFs other than nuclear reactors, reprocessing facilities are the most complex with respect to safety. The potential safety hazards to which fuel reprocessing facilities are prone include criticality excursions, radiation exposure, chemical reactions, fire and explosion.

Safety issues related to the storage of spent nuclear fuel at a reprocessing facility before the reprocessing starts are discussed in the following Section 10.

9.3.1. Criticality

Criticality control is a dominant safety issue for reprocessing plants due to the large amount of fissile materials treated and the presence of water, a moderator, in many parts of the plant. To prevent criticality relevant parameters such as the mass, volume, concentration of fissile material have to be controlled and geometry of components that contain fissile material has to be designed accordingly.

The following areas in a reprocessing facility have a significant risk of criticality [31]: Shear pack (due to accumulation of spent fuel powder), storage of hulls (due to residual fuel), dissolver, solvent extraction process, and purification of U and Pu, and conversion to oxide. Use of neutron poisons (in liquids or in casings of equipment) can allow larger sized components and reduce the possibility of criticality.

9.3.2. Radiation exposure

Dispersion of (highly) radioactive material may result in exposure of personnel. Thus, several barriers are necessary to be installed to avoid dispersion leading to contact of radioactive material with the workers. Two types of containment are used as barriers against dispersion: static and dynamic containments. The first static containment (or barrier) is composed of the walls of process equipment, i.e. such as tanks and pipes. Almost all process equipment is contained within (hot) cells. Thus, the second one consists of the enclosures of the (hot) cells, i.e. typically thick concrete walls. The third containment is formed of the external walls of the building. An additional dynamic containment function is achieved by adequate ventilation of the process equipment, vessels and cells, plus the working areas. The IAEA safety standard [36] states:

> "4.28. In a reprocessing facility (for most areas), three barriers (or more, as required by the safety analysis) should be provided, in accordance with a graded approach. The first static barrier normally consists of process equipment, vessels and pipes, or gloveboxes. The second static barrier normally consists of cells around process equipment or, when gloveboxes are the first containment barrier, the rooms around the glovebox(es). The final static barrier is the building itself. The design of the static containment system should take into account openings between different confinement zones (e.g. doors, mechanisms, instruments and pipe penetrations)".

To minimize external radiation exposure, the working areas need to be far away from radioactive material, adequate shielding can be established adapted to nature and energy of the radiation, and the exposure time of workers needs to be limited.

Radiation exposure of the public and the environment is discussed in the INPRO methodology manual on environmental impact of stressors [5].

9.3.3. Fire and explosion

A large fire spreading through a reprocessing plant or an explosion might be one of the main causes of dispersion of radioactive material into the environment, notably in the event of ventilation system failure, and damage to engineered controls. Reprocessing plants use flammable solvents (e.g. kerosene or dodecane), certain pyrophoric materials (e.g. zirconium powder, TBP), and agents (e.g. hydrogen, hydrazine) with oxidising and reducing properties. For example, the fragmentation of LWR fuel assemblies by shearing during head-end processing produces powder of Zircaloy that present a potential for ignition or explosion. Hydrogen gas probably represents the greatest potential cause of explosion, due to its rapid rate of diffusion, its low ignition energy input, and the wide range of concentration limits that may give rise to an explosion.

The risk of fire can be reduced by eliminating sources of ignition and hot points and by installing fire detection and extinction systems.

In the following an example is presented of an explosion in a reprocessing facility, namely, the so-called 'red oil' explosion. Red oil is the name of a substance of non-specific composition formed when an organic phase consisting of tri-butyl-phosphate (TBP) and diluents in contact with concentrated nitric acid is heated above 120°C under reflux. At temperatures above 130°C, the degradation of TBP, diluents, and nitric acid proceeds at rates fast enough to generate heat and voluminous amounts of detonable vapour. The generated heat further increases the temperature of the liquid, which in turn increases the rate of reaction (i.e. a runaway or autocatalytic reaction). For a comprehensive review, see Refs [124–126].

Three red oil events have occurred in the U.S. Department of Energy's defence nuclear facilities complex at the Hanford Site in 1953, and at the Savannah River Site in 1953 and 1975 [127, 128]. A red oil explosion also occurred in 1993 at the Tomsk-7 facility in Seversk, Russian Federation. More recent fire and explosion incident occurred in 1997 at the Bituminization Demonstration Facility of the Tokai reprocessing plant in Japan.

A report by U.S. Defence Nuclear Facilities Safety Board highlights the conditions under which red oil formation and explosion can take place and means to avoid such incidents [129].

9.3.4. Containment of radioactive material and/or toxic chemicals

Due to the use of highly aggressive acids and other chemicals in a reprocessing facility corrosion in process equipment can cause leaks of radioactive material and/or toxic chemicals.

In the following an example is presented of a leakage of radioactive material that occurred on 21 April 2005 in the feed clarification cell of the THORP plant at Sellafield in Great Britain. The THORP plant is designed to reprocess irradiated fuels produced by advanced gas-cooled reactors (AGR) and light water reactors. Reprocessing campaigns have been carried out with uranium oxide spent fuel originally enriched in ^{235}U by up to 4.8 %. The plant had already reprocessed around 5700 tonnes since its commissioning in 1994. During the incident about 83 m^3 of clarified radioactive fluid leaked into one of the recovery pans and was discovered during a camera inspection of the main feed clarification cell. This cell is closed off to personnel at all times and its walls guarantee the radiological protection of the adjacent premises. The toxic fluid present in the recovery pan contained uranium and plutonium that were yet to be separated from the fission products and estimated to represent about 20 tonnes and 200 kg, respectively. The plant was shut down as soon as the incident was discovered. The main cause of the leakage was from a failed pipe running between two accountancy tanks. Only the first barrier consisting of the transfer pipe failed during this incident. The static and dynamic integrity of the two remaining containment barriers remained intact. The operator emphasized that the leak posed no danger to the workers or the environment. In particular, no abnormal activity around the plant's stack has been detected. The operator also underlined the absence of any risk of criticality, which was corroborated by the British safety regulator.

A leak in the pipes and tanks of reprocessing facilities, if undetected, can create a hazard of internal flooding. Examples of fluids causing flooding are: cooling water, heating water, treated water, chemical solutions, fire-fighting water, etc. In a reprocessing plant, the most important hazardous consequences generated by flooding are: criticality, damage to equipment fulfilling safety functions, and dispersion into the environment of radioactive material transported by the fluid involved.

To prevent a release of radioactive material and/or toxic chemical to the outside of the plant several barriers (combinations of static and dynamic containments) are necessary: The first barrier is the casing of the equipment (e.g. wall of a vessel or pipe), the second barrier is a glove box or hot cell, and the last one is the building of the facility. Additionally, dynamic containments are to be provided by ventilation systems in process equipment, hot cells and glove boxes but also in the working area of the facility.

9.3.5. External hazards

Reprocessing facilities are expected to be designed against all credible external hazards (see Section 4.2.1 and 4.2.6). The IAEA Safety Standard [17] provides a list of selected external postulated initiating events including natural phenomena and human induced phenomena:

"(a) Earthquakes, volcanoes and surface faulting;
(b) Meteorological events, including extreme values of meteorological phenomena and rare events such as lightning, tornadoes and tropical cyclones;
(c) Floods, including water waves induced by earthquakes or other geological phenomena or floods and waves caused by failure of water control structures;
(d) Geotechnical hazards, including slope instability, collapse, subsidence or uplift of the site surface, and soil liquefaction;
(e) External human induced events, including transport events such as aircraft crashes and accidents at surrounding activities such as chemical explosions".

9.4. ADAPTATION OF THE INPRO METHODOLOGY TO A REPROCESSING FACILITY

The use of the INPRO methodology for an assessment of a reprocessing facility required significant modifications and adjustments compared to other types of NFCFs.

The following sections present the INPRO methodology in the area of safety adapted to a reprocessing facility.

9.4.1. INPRO basic principle for sustainability assessment of reprocessing facility in the area of safety

INPRO basic principle for sustainability assessment of reprocessing facility in the area of safety: The planned reprocessing facility is safer than the reference reprocessing facility. In the event of an accident, off-site releases of radionuclides and/or toxic chemicals are prevented or mitigated so that there will be no need for public evacuation[61].

The rationale of the BP was provided in Section 5.2. An explanation on the requirement of superiority in the INPRO methodology area of NFCF safety is provided in section 6.3.1. The INPRO methodology has defined a set of requirements to reprocessing facilities as displayed in Table 25.

[61] Other protective measures still may be needed. Effective emergency planning, preparedness and response capabilities will remain a prudent requirement as discussed in the INPRO methodology area of Infrastructure.

TABLE 25. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF SPENT FUEL REPROCESSING FACILITY IN THE AREA OF NFCF SAFETY

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR1: Robustness of design during normal operation: The assessed reprocessing facility is more robust than the reference design with regard to operation and systems, structures and components failures.	CR1.1: Design of normal operation systems	IN1.1: Robustness of design of normal operation systems. AL1.1: Superior to that in the reference design.
	CR1.2: Subcriticality	IN1.2: Subcriticality margins. AL1.2: Sufficient to cover uncertainties and avoid criticality.
	CR1.3: Facility performance	IN1.3: Facility performance attributes. AL1.3: Superior to those in the reference design.
	CR1.4: Inspection, testing and maintenance	IN1.4: Capability to inspect, test and maintain. AL1.4: Superior to that in the reference design.
	CR1.5: Failures and deviations from normal operation	IN1.5: Expected frequency of failures and deviations from normal operation. AL1.5: Lower than that in the reference design.
	CR1.6: Occupational dose	IN1.6: Occupational dose values during normal operation and AOOs. AL1.6: Lower than the dose constraints.
UR2: Detection and interception of AOOs: The assessed reprocessing facility has improved capabilities to detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions.	CR2.1: I&C systems and operator procedures	IN2.1: I&C system to monitor, detect, trigger alarms and, together with operator actions, intercept and compensate AOOs. AL2.1: Availability of such systems and operator procedures.
	CR2.2: Grace periods for AOOs	IN2.2: Grace periods until human actions are required after AOOs. AL2.2: Adequate grace periods are defined in design analyses.

TABLE 25. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF SPENT FUEL REPROCESSING FACILITY IN THE AREA OF NFCF SAFETY (cont.)

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR3: Design basis accidents: The frequency of occurrence of DBAs in the assessed reprocessing facility is reduced. If an accident occurs, engineered safety features and/or operator actions are able to restore the assessed facility to a controlled state and subsequently to a safe state, and the consequences are mitigated to ensure the confinement of nuclear and/or toxic chemical material. Reliance on human intervention is minimal, and only required after sufficient grace period.	CR3.1: Frequency of DBAs	IN3.1: Calculated frequency of occurrence of DBAs. AL3.1: Lower than that in the reference design.
	CR3.2: Engineered safety features and operator procedures	IN3.2: Reliability and capability of engineered safety features and/or operator procedures. AL3.2: Superior to those in the reference design.
	CR3.3: Grace periods for DBAs	IN3.3: Grace periods for DBAs until human intervention is necessary. AL3.3: Longer than those in the reference design.
	CR3.4: Barriers	IN3.4: Number of confinement barriers maintained (intact) after DBAs. AL3.4: At least one.
	CR3.5: Robustness of containment design	IN3.5: Containment loads covered by design of the facility assessed. AL3.5: Greater than those in the reference design.
UR4: Severe plant conditions: The frequency of an accidental release of radioactivity into the environment is reduced. The source term of accidental release into the environment remains well within the envelope of the reference facility source term and is so low that calculated consequences would not require public evacuation.	CR4.1: In-facility severe accident management	IN4.1: Natural or engineered processes, equipment, and AM procedures and training to prevent an accidental release to the environment in the case of accident. AL4.1: Sufficient to prevent an accidental release to the environment and regain control of the facility.
	CR4.2: Frequency of accidental release into environment	IN4.2: Calculated frequency of an accidental release of radioactive materials and/or toxic chemicals into the environment. AL4.2: Lower than that in the reference facility.
	CR4.3: Source term of accidental release into environment	IN4.3: Calculated inventory and characteristics (release height, pressure, temperature, liquids/gas/aerosols, etc) of an accidental release. AL4.3: Remains well within the inventory and characteristics envelope of the reference facility source term and is so low that calculated consequences would not require evacuation of population.

TABLE 25. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF SPENT FUEL REPROCESSING FACILITY IN THE AREA OF NFCF SAFETY (cont.)

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR5: Independence of DID levels and inherent safety characteristics: An assessment is performed for the reprocessing facility to demonstrate that the DID levels are more independent from each other than in the reference design. To excel in safety and reliability, the assessed reprocessing facility strives for better elimination or minimization of hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics.	CR5.1: Independence of DID levels	IN5.1: Independence of different levels of DID in the assessed reprocessing facility. AL5.1: More independence of the DID levels is demonstrated compared to that in the reference design, e.g. through deterministic and probabilistic means, hazards analysis, etc.
	CR5.2: Minimization of hazards	IN5.2: Examples of hazards: fire, flooding, release of radioactive material, radiation exposure, etc. AL5.2: Hazards are reduced in relation to those in the reference facility.
UR6: Human factors related to safety: Safe operation of the assessed reprocessing facility is supported by accounting for HF requirements in the design and operation of the facility, and by establishing and maintaining a strong safety culture in all organizations involved in the life cycle of the facility.	CR6.1: Human factors	IN6.1: Human factors addressed systematically over the life cycle of the reprocessing facility. AL6.1: Evidence is available.
	CR6.2: Attitude to safety	IN6.2: Prevailing safety culture. AL6.2: Evidence is provided by periodic safety reviews.
UR7: RD&D for advanced designs: The development of innovative design features of the assessed reprocessing facility includes associated RD&D to bring the knowledge of facility characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for operating facilities.	CR7.1: RD&D	IN7.1: RD&D status. AL7.1: RD&D defined, performed and database developed.
	CR7.2: Safety assessment	IN7.2: Adequate safety assessment. AL7.2: Approved by a responsible regulatory authority.

9.4.2. User requirement UR1: Robustness of design during normal operation

Rationale of UR1 was provided in Section 5.3. UR1 deals with prevention of AOOs. For a reprocessing facility, examples[62] of AOOs are [1, 37, 118–132]:

- Leakage (e.g. due to corrosion) of flammable (explosive) gases such as H_2;
- Leakage (small) of radioactive and/or toxic chemicals;
- Change in process parameters such as flow and temperature that lead to process malfunction;
- Fire in a room with significant amount of fissile material or toxic chemicals;

[62] Assuming that these occurrences have moderate consequences and cannot be qualified as accident conditions.

- Loss of utilities such as electrical power, pressurized air, coolant, ventilation.

Criteria selected for user requirement UR1 are presented in Table 26.

TABLE 26. CRITERIA FOR USER REQUIREMENT UR1

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR1: Robustness of design during normal operation: The assessed reprocessing facility is more robust than the reference design with regard to operation and systems, structures and components failures.	CR1.1: Design of normal operation systems	IN1.1: Robustness of design of normal operation systems. AL1.1: Superior to that in the reference design.
	CR1.2: Subcriticality	IN1.2: Subcriticality margins. AL1.2: Sufficient to cover uncertainties and avoid criticality.
	CR1.3: Facility performance	IN1.3: Facility performance attributes. AL1.3: Superior to those in the reference design.
	CR1.4: Inspection, testing and maintenance	IN1.4: Capability to inspect, test and maintain. AL1.4: Superior to that in the reference design.
	CR1.5: Failures and deviations from normal operation	IN1.5: Expected frequency of failures and deviations from normal operation. AL1.5: Lower than that in the reference design.
	CR1.6: Occupational dose	IN1.6: Occupational dose values during normal operation and AOOs. AL1.6: Lower than the dose constraints.

9.4.2.1. Criterion CR1.1: Design of normal operation systems

Indicator IN1.1: Robustness of design of normal operation systems.

Acceptance limit AL1.1: Superior to that in the reference design.

Normal operation equipment and systems relevant for safety used in a spent nuclear fuel reprocessing facility need to be designed against loads caused by internal and external hazards (see Section 4.2.1). The design of normal operation systems in a fuel production facility can be made more robust, i.e. reducing the likelihood of failures, by increasing the design margins, improving the quality of manufacture and construction, and by use of materials of higher quality. It is acknowledged that increasing the robustness of a facility design is a challenging task for a designer because enhancing one aspect could have a negative influence on other aspects. Thus, an optimised combination of design measures is necessary to increase the overall robustness of a design.

The general approach to the analysis of robustness can be explained as a two steps algorithm:

- Definition of ‘challenge’ for a given system, structure or component;
- Analysis of reaction to the challenge which may be one of the following:
  a) Withstand with full recovery;
  b) Withstand with some loss of functionality;
  c) Loss of function.

Systems, structures and components with increased capacity for ‘a’ option (and ‘b’, when appropriate) have superior robustness compared against alternatives.

An example of robustness of the chemical process in a reprocessing facility is linked to the concentration of Pu and U in the liquid phase. To avoid that any minor variations in the organic/aqueous flows or temperature may result in loss of Pu or U to waste streams or formation of a third phase, the concentrations of Pu and U in organic streams in the flow sheet need to be kept well below the theoretical loading limits. The sensitivity of the flow sheet to variations in flow or temperature needs to be analysed and documented in the safety report. Pu accumulation due to second organic phase formation and polymerisation (with or without precipitation) for Pu bearing systems are the anticipated process upsets in facilities reprocessing Pu rich spent nuclear fuels. Thus, sufficient margin between third phase formation limits and prevailing organic Pu concentration needs to be maintained. For solutions containing high concentrations of Pu, the aqueous acidity needs to remain above 0.2 M in order to prevent hydrolysis and polymerisation of Pu. To prevent a solvent (diluents) flash resulting in a fire and/or explosion**,** the operating temperature of the extractors is limited to 50 ºC.

Inside a hot cell, a likelihood of break or leak is expected to be minimised by design. Improved materials and welding and inspection practices can ensure that there are no breaks during the entire life time of the plant.

The **acceptance limit AL1.1** of **CR1.1** is met if evidence available to the INPRO assessor shows that the design of the facility assessed is superior to a reference design (e.g. has increased design margins, improved quality of manufacture and construction, or uses materials of higher quality), or, in case a reference facility could not be defined, took best international practice into account and is therefore state of the art technology.

9.4.2.2. Criterion CR1.2: Subcriticality

Indicator IN1.2: Subcriticality margins.

Acceptance limit AL1.2: Sufficient to cover uncertainties and avoid criticality.

As it was discussed in section 7.4.2.2. for uranium refining/ conversion or enrichment facility, to avoid criticality accident in a reprocessing facility a criticality analysis needs to be performed demonstrating a design margin of $k_{eff} < 0.90$ for all possible configurations of fissile material. In this analysis, mass concentration, shape, moderation, etc. have to be considered. All process equipment in the material handling area has to be designed for criticality for submerged and water filled conditions.

The **acceptance limit AL1.2** of **CR 1.2** is met if evidence available to the INPRO assessor shows that in the facility assessed no critical configuration can occur taking uncertainties into account.

9.4.2.3. Criterion CR1.3: Facility performance

Indicator IN1.3: Facility performance attributes.

Acceptance limit AL1.3: Superior to those in the reference design.

The strategy of ageing management is expected to cover all relevant stages in the reprocessing facility lifecycle, all normal operation states, AOOs and accidents influencing a given system, and all relevant mechanisms of ageing. The designer of reprocessing facility has to determine the design life of SSC important to safety, to provide appropriate design margins to take due account of age related degradation and to provide methods and tools for assessing ageing during the reprocessing facility operation. The reprocessing facility operating organization has to develop a plan for preparing, coordinating, maintaining and improving activities for ageing management implementation at the different stages of the reprocessing facility lifecycle. Implementation of this plan will involve activities on managing ageing mechanisms, detecting and assessing ageing effects, and managing ageing effects.

Enhancement of the operation quality could be achieved through increased emphasis on automation and on-line monitoring. Similar to the more detailed discussion for fuel fabrication facilities in section 8.5.2.3, the acceptance criteria for quality of operation can be taken to be:

- High(er) degree of remote control;
- Availability of clear operating procedures and manuals, providing comprehensive data on the permissible range of various parameters, and emergency instructions manuals;
- Availability of procedure for the feedback on application of operations manuals including a system of recording and analysing deviations from operating procedures, consequences of the events and methods to avoid recurrences.;
- Availability of surveillance requirements including periodic tests to verify the performance level for safe operation;
- Consideration of ageing management in the design documentation;
- Availability of plan for implementation of ageing management;
- Periodic and intensive training of operators;
- Periodic mock-ups to ensure readiness of operators to handle emergencies.

The **acceptance limit AL1.3** of **CR1.3** is met if evidence available to the INPRO assessor shows that the design of the facility assessed is superior to a reference design, or, in case a reference facility could not be defined, took best international practice into account and is therefore state of the art technology.

9.4.2.4. Criterion CR1.4: Inspection, testing and maintenance

Indicator IN1.4: Capability to inspect, test and maintain.

Acceptance limit AL1.4: Superior to that in the reference design.

Improved capabilities to inspect, to test and to maintain means that the reprocessing facility design assessed is expected to permit efficient and intelligent inspection, testing and maintenance, not just require more inspections and more testing, i.e. the programs of inspection, testing and maintenance need to be driven by a sound understanding of failure mechanisms (corrosion, erosion, fatigue, etc.), so that the right locations are inspected and right systems, structures and components are tested and maintained at the right time intervals. For example, provision for in-service inspection (ISI) of the components and equipment installed inside the hot cells is an essential requirement for fuel reprocessing plants in order that corrosion of equipment is detected at an early stage and actions taken to avoid leakage of radioactive solutions from the equipment. Particular innovations possible in this area include techniques for the measurement of thickness of the dissolver vessels to evaluate their residual life and ISI of welds of dissolver and hold up tanks.

The **acceptance limit AL1.4** of **CR1.4** is met if evidence available to the INPRO assessor shows that the capabilities to inspect, test and maintain systems relevant to safety in the facility assessed are superior to those in the reference design or, in case a reference facility could not be defined, are state of the art and allow easy inspection, testing and maintenance.

9.4.2.5. Criterion CR1.5: Failures and deviations from normal operation

Indicator IN1.5: Expected frequency of failures and deviations from normal operation.

Acceptance limit AL1.5: Lower than that in the reference design.

Examples of AOOs are provided in the beginning of Section 9.4.2. The frequency of AOOs for a reprocessing facility need to be derived from operational experience of comparable facilities and supported by PSA.

The probability of occurrence of various types of failures has been analysed via a probabilistic safety assessment of reprocessing plants in Refs [37, 131, 132]. Failure probabilities for various events in existing facilities such as loss of cooling water to high level waste storage tanks have been determined in Ref [131]. The database for equipment failures is usually derived from data available for equipment in reactors, e.g. see Refs [133, 134].

For the facility assessed, it can be possible to reduce the frequencies of AOOs, by increased robustness of the design, high quality of operation, and efficient and intelligent inspection. The consequences of all AOOs that can take place in the plant (e.g. inadvertent closure of valves, change in flows, mixing of solutions, transfer of fissile materials, etc.) need to be clearly addressed in the design analysis.

The **acceptance limit AL1.5** of **CR1.5** is met if evidence available to the INPRO assessor shows that that in the facility assessed the frequencies of AOOs (see beginning of Section 9.4.2 above) have been reduced in comparison to a reference design or, in case a reference facility could not be defined, that the facility assessed took best international practice into account and is therefore state of the art technology. If quantitative results from operational experience and PSA are not available, alternatively, deterministic analysis can be developed that support a reduction of probability of occurrence for AOOs in the facility assessed.

9.4.2.6. Criterion CR1.6: Occupational dose

Indicator IN1.6: Occupational dose values during normal operation and AOOs.

Acceptance limit AL1.6: Lower than the dose constraints.

The assessment of CR1.6 for a conversion and enrichment facility presented in Section 7.4.2.6 is deemed to be sufficiently similar to the assessment of a reprocessing facility. Therefore, the INPRO assessor is requested to use the assessment approach described for a conversion and enrichment facility also for a reprocessing facility.

9.4.3. User requirement UR2: Detection and interception of AOOs

The rationale of UR2 was provided in Section 5.4. Criteria selected for user requirement UR2 are presented in Table 27.

TABLE 27. CRITERIA FOR USER REQUIREMENT UR2

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR2: Detection and interception of AOOs: The assessed reprocessing facility has improved capabilities to detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions.	CR2.1: I&C systems and operator procedures	IN2.1: I&C system to monitor, detect, trigger alarms and, together with operator actions, intercept and compensate AOOs. AL2.1: Availability of such systems and operator procedures.
	CR2.2: Grace periods for AOOs	IN2.2: Grace periods until human actions are required after AOOs. AL2.2: Adequate grace periods are defined in design analyses.

9.4.3.1. Criterion CR2.1: I&C systems and operator procedures

Indicator IN2.1: I&C system to monitor, detect, trigger alarms and, together with operator actions, intercept and compensate AOOs.

Acceptance limit AL2.1: Availability of such systems and operator procedures.

A reprocessing facility is expected to be designed to cope with AOOs (see beginning of Section 9.4.2) using automatic operational systems, i.e. I&C systems that bring the facility back to normal operating conditions. In case automatic systems are not available, adequate operator procedures are necessary. Passive and automatic active control systems are deemed more reliable than administrative (manual) control, but it is acknowledged that they are difficult to design for such facilities.

The design analysis is expected to specify clearly the regime of safe operating conditions for all equipment and processes. Necessary instrumentation for detecting malfunctions needs to be clearly identified. The availability of redundant monitoring systems based on different principles will ensure that the deviations from the intended conditions are detected efficiently. For example, reliable, continuous air monitoring systems to detect release of radioactivity to operating areas, criticality and temperature monitors need to be provided, with necessary interlocks and alarm annunciation systems.

Precise and reliable liquid flow metering devices need to be provided for the process streams to ensure that changes in flow ratios that may lead to process malfunction can be quickly detected and corrected. Fail-safe process interlocks are expected to be provided to maintain the desired solvent to aqueous ratio in the solvent extractors (pulse columns, mixer-settlers or centrifugal extractors). In the event of a major transient in a particular flow to the extractor, the interlock logic needs to stop all the input flows of the related extractor without fail.

On-line monitoring of Pu in process streams is essential to detect and intercept any process malfunction leading to Pu accumulation in undesired streams and in vessels of critically unsafe geometry. I&C systems play a vital role in avoiding runaway conditions in evaporators and preventing the formation of red oil. The provision of automatically closing fire barriers/ fire dampers in ventilation systems can ensure that fire does not spread to other areas.

Continuous monitoring of the temperatures and pressures in the process tanks can provide timely indications of process malfunctions. Pressures, temperatures and gamma activity levels inside the process enclosures need to be monitored to ensure detection of fire and criticality events. The monitoring of Pu concentrations in process streams is vital for not only detecting process malfunctions but also detecting accumulations of Pu in certain streams due to the phenomenon of third phase formation.

The **acceptance limit AL2.1** of **CR2.1** is met if evidence available to the INPRO assessor shows that I&C systems are available in the facility assessed that can detect failures and deviations from normal operation of systems relevant to safety, provide alarms, initiate automatic and/or manual actions, and bring the facility back to normal operation.

9.4.3.2. Criterion CR2.2: Grace periods for AOOs

Indicator IN2.2: Grace periods until human actions are required after AOOs.

Acceptance limit AL2.2: Adequate grace periods are defined in design analyses.

An explanation of ‘adequate grace period’ is provided in section 6.3.3.2. The grace period available to the operator for each AOO needs to be defined in the safety analysis of the facility design. After detection of an AOO (see beginning of Section 9.4.2) in a reprocessing facility, automatic operational (I&C) systems (presented in Section 9.4.3.1 above) need to control these incidents before the operator intervention.

A minimum period of 30 minutes is envisaged as an adequate grace period with regard to disturbances in the process, due to flow variations, loss of power at site, loss of ventilation, loss of process coolant water, etc. For example, the failure of ventilation systems for hot cells does not lead to leakage of radioactivity to the operating areas beyond permissible limits within 30 minutes. This grace period will be adequate for human intervention to start auxiliary ventilation systems, and complete other safety actions such as the evacuation of operating areas.

The system is expected to have sufficient inertia to withstand transients, i.e. react slowly after AOOs. For example, the flow sheet of the processes needs to be robustly designed such that transients in flows do not lead to large losses of fissile material to waste streams. Sufficient heat transfer area needs to be available for liquid waste storage tanks to dissipate the large inventory of decay heat by natural convection for sufficient long periods in the event of transients in the cooling water flow. Build-up of radiolytic hydrogen in a liquid waste storage tank can take place in the event of a failure of the air sparging system. Availability of enough vapour space in liquid waste storage tanks would ensure that the radiolytic hydrogen level can be kept below explosion limit for a minimum period of eight hours.

To demonstrate the adequacy of the reprocessing facility design, the system behaviour for all AOOs needs to be analysed with validated and verified computer models.

The **acceptance limit AL2.2** of **CR2.2** is met if evidence available to the INPRO assessor shows that an adequate grace period has been determined in the design analysis for the facility assessed for all AOOs.

9.4.4. User requirement UR3: Design basis accidents

Rationale of UR3 was provided in Section 5.5. Examples of DBAs for a reprocessing plant are [130]:

- A criticality accident;
- A large scale leakage of radioactive material;
- A total loss of power leading to ventilation failure;
- Pressurization in the evaporator due to red-oil reactions;
- A large fire and explosion inside a hot cell.

External hazards (defined in Section 4.2.1 and 4.2.6) such as earthquake, flooding, etc. could also lead to a DBA in a reprocessing facility. As stated before, the facilities need to be designed against all external and internal hazards.

The criteria selected for user requirement UR3 are presented in Table 28.

TABLE 28. CRITERIA FOR USER REQUIREMENT UR3

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR3: Design basis accidents: The frequency of occurrence of DBAs in the assessed reprocessing facility is reduced. If an accident occurs, engineered safety features and/or operator actions are able to restore the assessed facility to a controlled state and subsequently to a safe state, and the consequences are mitigated to ensure the confinement of nuclear and/or toxic chemical material. Reliance on human intervention is minimal, and only required after sufficient grace period.	CR3.1: Frequency of DBAs	IN3.1: Calculated frequency of occurrence of DBAs. AL3.1: Lower than that in the reference design.
	CR3.2: Engineered safety features and operator procedures	IN3.2: Reliability and capability of engineered safety features and/or operator procedures. AL3.2: Superior to those in the reference design.
	CR3.3: Grace periods for DBAs	IN3.3: Grace periods for DBAs until human intervention is necessary. AL3.3: Longer than those in the reference design.
	CR3.4: Barriers	IN3.4: Number of confinement barriers maintained (intact) after DBAs. AL3.4: At least one.
	CR3.5: Robustness of containment design	IN3.5: Containment loads covered by design of the facility assessed. AL3.5: Greater than those in the reference design.

9.4.4.1. Criterion CR3.1: Frequency of DBAs

Indicator IN3.1: Calculated frequency of occurrence of DBAs.

Acceptance limit AL3.1: Lower than that in the reference design.

Several examples of potential DBA in spent fuel reprocessing facility are introduced in the beginning of section 9.4.4. The frequency of occurrence of a DBA in the facility assessed is to be determined via a probabilistic risk assessment. Ref [135] estimates the frequency of pressurization due to red-oil reactions in the evaporator to be 10^{-6} per year. Such an assessment is necessary for other events such as loss of control of flow metering systems, total loss of electrical power leading to ventilation failure, etc.

The probability of a fire can be estimated based on a comprehensive knowledge of various fire loads in the plant, using codes such as COMPBRN-III [136].

The calculated frequency of DBA caused by external hazards can be influenced by the designer, e.g. via an increase of robustness of the confinement wall, and by the owner/ operator of the facility by selecting an appropriate site (see also user requirement UR7).

The **acceptance limit AL3.1** of **CR3.1** is met if evidence available to the INPRO assessor shows that in the facility assessed based on probabilistic analyses the frequency for the defined DBA is superior to a reference design. If quantitative results are not available the deterministic analysis can be developed that support a reduction of these frequencies based on an increase of design robustness, high quality of operation, an intelligent inspection and maintenance programs, advanced I&C systems, and increased inertia.

9.4.4.2. Criterion CR3.2: Engineered safety features and operator procedures

Indicator IN3.2: Reliability and capability of engineered safety features and/or operator procedures.

Acceptance limit AL3.2: Superior to those in the reference design.

Engineered safety features need to be available that, after detection of a DBA, can automatically and reliably control the accident, restoring the facility to a controlled state and keeping the consequences within authorized limits. To assure necessary reliability, these features have to be designed with sufficient levels of redundancy, diversity and independence. In case automatic systems are not available, adequate operator procedures are necessary. Redundant, diversified and independent passive and automatic active systems are deemed more reliable than administrative controls (operator interventions) yet are nevertheless acknowledged to be difficult to design for spent fuel reprocessing facilities.

An example of such engineered safety features is seen in the provision of a secondary enclosure with an automatic exhaust system to restore the negative pressure momentarily lost because of a breach of the barrier (e.g. breaking of a glove box panel). The availability and performance of alternate power supply systems based on diesel generators (or batteries) needs to be periodically checked to ensure that they act reliably during an electrical power supply failure. The facility is also expected to have engineered safety features protecting against DBAs caused by external hazards (see Section 4.2.1 and 4.2.6), e.g. shock absorbers and dampers to mitigate an earthquake.

The **acceptance limit AL3.2** of **CR3.2** is met if evidence available to the INPRO assessor shows that the reliability and capability of engineered safety features in the facility assessed is superior to a reference design and assure that after the beginning of a DBA the necessary actions to mitigate the consequences of the accidents will be timely initiated. Alternatively, if a reference facility cannot be found, it could be demonstrated that the design of the facility assessed took available information on best international practice into account and is therefore state of the art.

9.4.4.3. Criterion CR3.3: Grace periods for DBAs

Indicator IN3.3: Grace periods for DBAs until human intervention is necessary.

Acceptance limit AL3.3: Longer than those in the reference design.

An explanation of 'adequate grace period' is provided in section 6.3.3.2 as introduced earlier for control of AOOs (see CR2.2) in Level 2 of DID. The criterion CR3.3 'grace period for DBAs' implies a similar concept. For DBAs (caused by events associated with internal and external hazards), the criterion requires that the system response (inertia) and/or automatic actions of active (and/or passive) safety features provide an adequate grace period for interventions by the operator. Adequate grace periods in new reprocessing facilities are also assumed to be longer than those in the reference design.

There is practically no grace period for a criticality event in a spent fuel reprocessing facility, since the excursion occurs rather rapidly. However, prompt operator action can avert further excursions and consequent release of radioactivity and exposure to personnel. Grace periods for such actions are estimated to be only a few minutes.

Large scale leakages of process vessels leading to releases of radioactivity inside an enclosure such as glove box or hot cell are expected to be among the design basis events for a reprocessing facility. While the time necessary for attending to the large scale leak would depend upon the volume of the process/storage vessel and the leak, a grace period of at least several minutes can be expected.

Since the process solutions in a reprocessing plant include large inventories of combustible organics, a fire incident inside a process enclosure needs to be attended to expeditiously to ensure that it does not lead to a major fire and an explosion. Grace periods available for action in such a case could be 5 to 10 minutes depending upon the location of the fire and the layout of the process vessels.

The available grace periods have to be determined for each DBA in the design analyses. In the analysis, the DBAs need to be clearly specified and those identified that may require human intervention within a given period.

The **acceptance limit AL3.3** of **CR3.3** is met if evidence available to the INPRO assessor shows that in the facility assessed the grace periods are longer than those in the reference design. Alternatively, if a reference facility cannot be found, it needs to be demonstrated that the design of the facility assessed took available information on best international practice into account and is therefore state of the art.

9.4.4.4. Criterion CR3.4: Barriers

Indicator IN3.4: Number of confinement barriers maintained (intact) after DBAs.

Acceptance limit AL3.4: At least one.

The design of engineered safety features is expected to provide deterministically for continued integrity of at least of one barrier containing the radioactive and chemically toxic material following any DBA caused by events associated with internal or external hazards. Alternatively, the probability of losing all barriers could be used as an INPRO methodology indicator with a sufficiently low value of its acceptance limit.

In most NFCFs, the operator is protected by two confinement systems (e.g. a static and a dynamic confinement) that are designed against a release of radioactivity into the working area of the facility. However, due to the large volume and high radiotoxicity (and chemical toxicity) of materials handled in a reprocessing facility, two static safety barriers, e.g. the casing of the equipment and a hot cell or glove box, plus a dynamic confinement system in form of a ventilation system (inside a hot cell and in the working area), are required to protect the operator against releases of radioactive materials and toxic chemicals. Against the release of radioactivity and/or toxic chemicals to the environment, the last physical barrier is provided by the walls of the facility building. Thus, in the design of a reprocessing facility, there are usually three physical barriers, which are combined with ventilation of the hot cell and working area to prevent an accidental release of radioactivity into the environment.

The INPRO task group proposed that, after a DBA in reprocessing facility, at least one barrier will remain intact between the radioactive material and/or toxic chemicals and the operator. A second barrier – the walls of the facility building – will also remain intact between the radioactivity and the environment.

The **acceptance limit AL3.4** of **CR3.4** is met if evidence available to the INPRO assessor shows that, after a DBA in the facility assessed, at least one barrier remains intact between the radioactive material and/or toxic chemicals and the operator.

9.4.4.5. Criterion CR3.5: Robustness of containment design

The INPRO assessment of CR3.5 presented for a uranium conversion and enrichment facility in Sections 7.4.4.5 is deemed to be sufficiently similar to the assessment of a reprocessing facility. Thus, this approach can be used by the INPRO assessor also for reprocessing facilities.

9.4.5. User requirement UR4: Severe plant conditions

The rationale of UR4 was provided in Section 5.6. The INPRO methodology has defined three criteria for UR4: in-facility severe accident management, frequency of accidental release to the environment, source term of accidental release to the environment.

It is noted that a spent fuel reprocessing facility using enriched uranium (> 1 % of ^{235}U) and plutonium has a higher probability of a criticality accident due to the existence of fissile material in liquid form than do other kinds of NFCFs. However, the INPRO assessment of spent fuel reprocessing facilities against user requirement UR4 (severe plant conditions) is deemed to be sufficiently similar to the assessment of an enrichment facility. Therefore, the assessor is requested to use the assessment method of UR4 described in Section 7.4.5 for an enrichment facility (including criteria, indicators and acceptance limits) also for a spent fuel reprocessing facility.

9.4.6. User requirement UR5: Independence of DID levels and inherent safety characteristics

The rationale of UR5 was provided in Section 5.7. The criteria selected for user requirement UR5 are presented in Table 29.

TABLE 29. CRITERIA FOR USER REQUIREMENT UR5

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR5: Independence of DID levels and inherent safety characteristics: An assessment is performed for the reprocessing facility to demonstrate that the DID levels are more independent from each other than in the reference design. To excel in safety and reliability, the assessed reprocessing facility strives for better elimination or minimization of hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics.	CR5.1: Independence of DID levels	IN5.1: Independence of different levels of DID in the assessed reprocessing facility. AL5.1: More independence of the DID levels is demonstrated compared to that in the reference design, e.g. through deterministic and probabilistic means, hazards analysis, etc.
	CR5.2: Minimization of hazards	IN5.2: Examples of hazards: fire, flooding, release of radioactive material, radiation exposure, etc. AL5.2: Hazards are reduced in relation to those in the reference facility.

9.4.6.1. Criterion CR5.1: Independence of DID levels

Indicator IN5.1: Independence of different levels of DID in the assessed reprocessing facility.

Acceptance limit AL5.1: More independence of the DID levels is demonstrated compared to that in the reference design, e.g. through deterministic and probabilistic means, hazards analysis, etc.

The INPRO assessment method of CR5.1 presented in Section 8.5.6.1 for a fuel fabrication facility is deemed to be formulated in sufficiently general terms so that it can be used also to assess a reprocessing facility.

9.4.6.2. Criterion CR5.2: Minimization of hazards

Indicator IN5.2: Examples of hazards: fire, criticality, release of radioactive material, radiation exposure, etc.

Acceptance limit AL5.2: Hazards are reduced in relation to those in the reference facility.

Ref [60] explains the concept of inherent safety as the achievement of safety through fundamental conceptual design choices that eliminate or exclude inherent hazards. An inherent safety characteristic is a fundamental property of a design concept that results from basic choices in the materials used or in other aspects of the design and assures that a particular potential hazard cannot in any way become a safety concern.

Inherent safety can be built into the design of reprocessing plants through careful examination of the major events and by introducing innovations that circumvent such events. For example, the use of borated steels and use of vessels coated with boron compounds can ensure that criticality is not possible in a process vessel at any concentration of fissile material. The use of air operated motors in place of electrically operated motors (to avoid sparks) is an example of an inherently safe characteristic. The use of alternate extractants that are analogous to TBP but have a higher number of carbon atoms (e.g. tri-n-amyl phosphate) can ensure that third phase formation (which can lead to criticality) can be avoided altogether. By limiting the temperatures and concentrations of nitric acid in the evaporators, red oil formation can be avoided. Using fire resistant material and reducing the amount of burnable material in a reprocessing facility would reduce the hazard of a fire.

The **acceptance limit AL5.2** of CR5.2 is met if evidence available to the INPRO assessor shows that hazards in the NFCF assessed have been reduced compared to those in the reference facility. Alternatively, if a reference facility cannot be found, it needs to be demonstrated that the design of the facility assessed took available information on best international practice into account and is therefore state of the art.

9.4.7. User requirements UR6 and UR7

The rationales for UR6 and UR7 are provided in Sections 5.8 and 5.9. The assessment of user requirements UR6 (human factors related to safety) and UR7 (RD&D for advanced designs) for the reprocessing facilities is deemed to be sufficiently similar to the assessment method of UR6 and UR7 described in Sections 6.3.7 and 6.3.8 for mining and milling facilities (including criteria, indicators and acceptance limits).

10. STORAGE OF SPENT NUCLEAR FUEL

In this section, firstly, a short description of the main processes in a storage facility for spent nuclear fuel (SNF) is given and the corresponding specific safety issues are discussed. Secondly, the assessment method will be discussed based on the corresponding criteria of the INPRO methodology in the area of safety, which have been, where necessary, adapted to the specific issues of this kind of NFCF.

The term ‘storage’ refers to the retention of radioactive waste in a facility or a location with the intention of retrieving the waste. Thus, storage is a temporary measure for which some future action is planned such as further conditioning or packaging of the waste and, ultimately, its disposal [135].

After unloading from the reactor core, SNF is usually stored for several years at the reactor site in storage pools. This initial period of storage allows a considerable reduction in the intensity of radiation fields and decay heat generation. Thereafter, SNF is usually transferred into another wet or dry storage facility away from the reactor[63].

Spent nuclear fuel transportation has not been considered in this manual as an independent stage of nuclear fuel cycle. The INPRO methodology implies that safety of spent fuel transportation is to be considered as part of the INPRO assessment either of the spent fuel storage facility or the spent fuel reprocessing facility.

10.1. AT-REACTOR STORAGE POOLS FOR SNF

At-reactor SNF storage pools are either within the reactor building or in an adjacent fuel building, which is linked to the reactor by a transfer tunnel. Access to the SNF in the storage pool is usually by means of immersing a cask in the pool, loading it with SNF and then removing the cask for lid closure, decontamination and transport. A method developed in France involves a cask loading concept with bottom access ports for transferring SNF from the pool into the cask. The advantages of this design are that contamination of the external surface of the cask by immersion in the pool is avoided, and also the requirement to lift the cask (empty and loaded) between the inlet/outlet location and the pool and a heavy duty crane is not needed. There are some cases, e.g. at gas cooled reactors (Magnox), at Sellafield in the UK and at La Hague in France, where SNF can be loaded into (unloaded from) the casks in a dry shielded cave avoiding cask immersion in water.

At-reactor SNF storage pools are constructed of reinforced concrete and usually lined with stainless steel. The pools are filled with de-ionized water with or without additive depending on the type of fuel to be stored and the adopted method of treatment. Water activity levels are maintained as low as reasonably practicable (ALARP) by either in-pool or external ion exchange systems or by limiting activity release to the bulk pool water. The pools are further equipped with leakage monitoring, as well as coolant temperature, transparency and chemical monitoring systems, and systems necessary to maintain monitored parameters within the prescribed ranges. Chemical monitoring normally involves pH measurements, soluble boron concentration measurements for criticality control where necessary, and measurements of levels of aggressive anions such as chloride and sulphate to minimize fuel degradation. Maintenance of good water chemistry provides good water clarity and usually prevents the occurrence of micro-biological organisms. If these do occur, they are treated with specific chemical dosing.

[63] Storage can be located at the reactor site or at a separate site.

Subcriticality in SNF pools was originally maintained by spacing within the storage racks or baskets. However, with the need to store greater quantities of SNF, higher storage density has been achieved by the introduction of neutron absorbing materials in storage racks and baskets such as boronated stainless steel, boral or boraflex.

10.2. AWAY FROM THE REACTOR SITE STORAGE OF SNF

Storage away from the reactor site (AFR) can be wet, in the form of centralized pools in support of reprocessing activities, secondary pools or additional pools, or most often dry, in the form of dry cask storage facilities, which may or may not have capability for off-site transport.

10.2.1. AFR wet storage facilities

A variety of AFR wet storage facilities are in use. A typical AFR wet storage facility has the following features:

- Cask reception, decontamination, unloading, maintenance and dispatch;
- Underwater spent fuel storage (pool);
- Auxiliary services (radiation monitoring, water cooling and purification, solid radioactive waste handling, ventilation, power supply, etc.).

An AFR storage pool is a reinforced concrete structure usually built above ground or at least at ground elevation, however, one wholly underground facility is in operation. Some early pools were open to the atmosphere, but operational experience and the need to control pool water purity has resulted in the pools now being covered. The reinforced concrete structure of the pool, including the covering building, needs to be seismically qualified depending upon national requirements.

Most pools are stainless steel lined; some are coated with epoxy resin based paint. However, there has been experience with degradation of the latter after a number of years. A further option is for the pool to be unlined and untreated. In some situations, the pool may be stainless steel lined or epoxy treated only at the water line or at other locations. Regarding unlined and untreated pools, properly selected and applied concrete can be proved to have negligible corrosive ion leaching and permeability to water. The water is either a fixed quantity or a once through pond purge. Leakage from the pool is monitored, either by means of an integrated leakage collection system or via the inter-space in pools with two walls. In both cases any recovered pool water may be cleaned up and returned to the main pool. In addition to the control of activity by ion exchange or purge, some pools are operated with an imposed chemical regime (see Section 10.1).

10.2.2. AFR dry storage facilities

Dry storage of SNF differs from wet storage by making use of gas or air instead of water as the coolant (often an inert gas such as helium, or an only modestly reactive gas such as nitrogen, to limit oxidation of the fuel while in storage) and metal or concrete instead of water as the radiation barrier. SNF is normally stored in pools for several years before it becomes cool enough for dry storage to be possible.

10.2.2.1. Dry storage vaults.

In a vault, the SNF is stored in a large concrete building, whose exterior structure serves as a radiation barrier, and whose interior has large numbers of cavities suitable for SNF storage units. The SNF is typically stored in sealed metal storage tubes or storage cylinders, which may hold one or several fuel assemblies; these tubes or cylinders provide containment of the radioactive material in the spent fuel. Heat is removed in vault systems by either forced or natural air convection. In some vault systems, SNF is removed from the transport cask and moved in a shielded charge machine to its storage tube, while in others the SNF stays in the container in which it arrives, which is then placed in a transfer cask and moved by crane to its storage cylinder. Thus, vault systems typically also require cranes or fuel-handling machines.

10.2.2.2. Dry storage silos.

In a silo storage system, the SNF is stored in concrete casks, i.e. either vertical or horizontal cylinders fitted with metal inner liners or separate metal canisters. The concrete provides radiation shielding (as the building exterior does in the case of a vault) while the sealed inner metal liner or canister provides containment. Transfer casks are often used for loading of the fuel into the silos. Heat removal is by air convection.

One example of a horizontal concrete silo design is the NUHOMS storage system. The system uses vertically loaded metal canisters and the horizontal concrete storage modules. The metal canisters use a double lid closure, are seal-welded, and tested for leak tightness [137].

10.2.2.3. Dry storage metal casks.

Metal casks are massive containers used in transport, storage and eventual disposal of SNF. A typical metal cask has a capacity of 4 to 26 PWR or 10 to 60 BWR fuel assemblies [31]. The structural materials for metal casks may be forged steel, nodular cast iron, or a steel/lead sandwich structure. They are fitted with an internal basket or sealed metal canister which provides structural strength as well as assures subcriticality. Metal casks usually have a double lid closure system[64] that can be either bolted or seal welded and can be monitored for leak tightness. Metal casks are usually transferred directly from the fuel loading area to the storage site. Some metal casks are licensed for both storage and off-site transportation. Fuel is loaded vertically into the casks which are usually stored in a vertical position.

10.3. OPERATION OF AN SNF DISPOSAL FACILITY BEFORE CLOSURE

Storage of SNF was defined above as the retention of SNF in a facility or a location with the intention of retrieving the SNF [135]. Disposal of SNF is defined as the emplacement of SNF into a facility with no intention of retrieval [79]. Commonly, SNF is envisaged to be disposed of in a deep geological depository [138].

Before SNF can be put into a disposal facility, it needs to be packed in an appropriate way for disposal, for example in special containers with long term durability. This packaging process can be performed at the site of the disposal facility or in a separate plant; again the safety issues related to packaging of SNF are comparable to those for handling SNF in a storage facility. A special safety case is necessary for an SNF disposal facility [91].

However, it is noted that before closure, i.e. during the phase where a disposal facility receives SNF, its safety issues are comparable to those of an SNF storage facility (presented in Section 10.4). Thus, the INPRO assessment methodology described in Section 10.5 for SNF storage facilities can also be applied to SNF disposal facilities during their operational phase, i.e. before closure.

It is to be emphasized that, after closure of a disposal facility, the safety of SNF is assured by passive safety features without the need of human intervention.

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[64] In some designs a single lid is used with two seals.

Safety issues concerning potential releases of radioactive material into the environment after the closure of an SNF disposal facility are to be considered in another area of the INPRO methodology called 'waste management'.

10.4. SAFETY ISSUES IN A STORAGE FACILITY FOR SNF

SNF is usually transferred to storage facilities only after an initial period of storage in the reactor pool. As stated before, this initial period of storage allows a considerable reduction in radiation emissions and decay heat. Hence, the development of conditions that could lead to potential accidents during SNF storage in the storage facilities will theoretically occur comparatively slowly and the safety of SNF storage can thus be maintained with relatively unsophisticated protective systems. However, this statement is not related to the hazards associated with handling of SNF within the storage facility and with potential external events. A comprehensive description of safety aspects in SNF storage facilities is provided in IAEA Safety Standards and other publications [35, 92, 139–141]. Various aspects of SNF management are discussed in Refs [31, 142–145].

For safe operation and maintenance of SNF storage facilities, their design incorporates features to keep the fuel subcritical, remove the spent fuel decay heat, provide radiation protection, and maintain containment over the anticipated lifetime of the facilities as normally stated in the design specifications. These objectives need to be met for all anticipated operational occurrences (AOOs) and design basis accidents (DBAs) in accordance with the design basis approved by the regulatory body.

10.4.1. Criticality

The design of the facility needs to assure subcriticality during loading, transfer, storage and retrieval.

In case of pool storage, subcriticality needs to be guaranteed for all credible water densities including boiling, the reliance on soluble neutron poison such as borated water needs to be avoided and solid neutron absorbers such as borated steel can be used. In case of dry storage, the fuel baskets and containers are normally designed to remain subcritical in all credible situations including the introduction of a moderator due to flooding. Detailed guidance on subcriticality issues in the design of SNF storage facilities is provided in Ref [35].

10.4.2. Radiation exposure

Radiation exposure of workers needs to be minimized. SNF has a high radiotoxicity. For example, the beta-gamma activity in PWR spent fuel six months after unloading from reactors still amounts to about 150 TBq/t_{HM} [31] and the dose rate at 1 m perpendicular to the centre of SNF assembly for the first ca. 10 years after discharge remains above 10 Sv/hr [146].

In case of pool storage, usually the water level above the SNF (typically at least 4 m [31]) is used as radiation shielding for the protection of workers. This level is normally maintained in all credible situations by adequate water supply systems. Another source of radiation is radioactive material released from the SNF into the pool; thus, this material needs to be controlled and removed. The clarity of the pool water and sufficient lighting can help to reduce the time that workers spend exposed to radiation during operations at the pool.

In case of dry storage, SNF loading and unloading actions need to be performed in a way that limits reflection of radiation to the workers. To minimize internal exposure of workers, the concentration of airborne radionuclides (e.g. due to fuel failures) in closed facilities are assumed to be kept within acceptable limits by ventilation with filtering of the air.

Detailed guidance on radiation protection in the design of SNF storage facilities is provided in Ref [35].

Radiation exposure to the public and environment at normal operation conditions is discussed in the INPRO methodology manual on environmental impact of stressors [5].

10.4.3. Heat removal

Heat is generated in SNF due to the decay of fission products and actinides. For example, one year after discharge, the decay heat from a PWR SNF assembly remains higher than 10 kW/tHM and after 10 years it is still higher than 1 kW/tHM [146]. This heat has to be removed safely to avoid overheating of the SNF and subsequent failure, but also to keep the temperature of the equipment and structures in the storage facility below design limits. Pool storage facilities need reliable active heat removal systems. Dry storage facilities are expected to use passive cooling systems to the maximum extent practicable. Detailed guidance on the residual heat removal in the design of SNF storage facilities is provided in Ref [35].

10.4.4. Containment of radioactive material

Apart from potential releases of radionuclides during accident conditions, the SNF assemblies may release radioactive isotopes (e.g. solid and gaseous fission products like ^{85}Kr, ^{134}Cs and ^{137}Cs) due to defects in the fuel cladding. Furthermore, the outer surface of the SNF assembly claddings may be contaminated with radionuclides during the reactor operation or previous storage. To avoid exposure of workers in wet and dry storage facilities, the radionuclides released respectively in the water and in the air have to be controlled and removed.

Ref [35] provides detailed guidance for considering issues related to the containment of radioactive materials at the design and operation stages of SNF storage facilities.

10.4.5. External hazards

External natural phenomena and external human induced phenomena that can influence safety of the SNF storage facilities are discussed in detail in Ref [35]. SNF storage facilities need to be designed against all credible external hazards (see Section 4.2.1 and 4.2.6).

10.5. ADAPTATION OF THE INPRO METHODOLOGY TO A STORAGE FACILITY FOR SPENT NUCLEAR FUEL

Adapting the INPRO methodology for use in assessing SNF storage facilities in the area of NFCF safety required significant modifications and adjustments in relation to the methodology used for other types of NFCFs.

It is noted that general requirements for sustainable management of all types of radioactive waste generated during the operation and decommissioning of all facilities in a nuclear energy system are discussed in the INPRO methodology manual on waste management. The following sections describe how the INPRO methodology in the area of safety is adapted and applied to an SNF storage facility.

10.5.1. INPRO basic principle for sustainability assessment of spent nuclear fuel storage facility in the area of safety

INPRO basic principle for sustainability assessment of spent nuclear fuel storage facility in the area of safety: The planned spent nuclear fuel storage facility is safer than the reference spent nuclear fuel storage facility. In the event of an accident, off-site releases of radionuclides and/or toxic chemicals are prevented or mitigated so that there will be no need for public evacuation[65].

The rationale of the BP was provided in Section 5.2. An explanation on the requirement of superiority in the INPRO methodology area of NFCF safety is provided in section 6.3.1. The INPRO methodology has defined a set of requirements for spent fuel storage facilities as displayed in Table 30.

TABLE 30. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF SPENT NUCLEAR FUEL STORAGE FACILITIES IN THE AREA OF NFCF SAFETY

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR1: Robustness of design during normal operation: The assessed SNF storage facility is more robust than the reference design with regard to operation and systems, structures and components failures.	CR1.1: Design of normal operation systems	IN1.1: Robustness of design of normal operation systems. AL1.1: Superior to that in the reference design.
	CR1.2: Subcriticality	IN1.2: Subcriticality margins. AL1.2: Sufficient to cover uncertainties and avoid criticality.
	CR1.3: Facility performance	IN1.3: Facility performance attributes. AL1.3: Superior to those in the reference design.
	CR1.4: Inspection, testing and maintenance	IN1.4: Capability to inspect, test and maintain. AL1.4: Superior to that in the reference design.
	CR1.5: Failures and deviations from normal operation	IN1.5: Expected frequency of failures and deviations from normal operation. AL1.5: Lower than that in the reference design.
	CR1.6: Occupational dose	IN1.6: Occupational dose values during normal operation and AOOs. AL1.6: Lower than the dose constraints.
UR2: Detection and interception of AOOs: The assessed SNF storage facility has improved capabilities to detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions.	CR2.1: I&C systems and operator procedures	IN2.1: I&C system to monitor, detect, trigger alarms and, together with operator actions, intercept and compensate AOOs. AL2.1: Availability of such systems and operator procedures.
	CR2.2: Grace periods for AOOs	IN2.2: Grace periods until human actions are required after AOOs. AL2.2: Adequate grace periods are defined in design analyses.

[65] Other protective measures still may be needed. Effective emergency planning, preparedness and response capabilities will remain a prudent requirement as discussed in the INPRO methodology area of Infrastructure.

TABLE 30. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF SPENT NUCLEAR FUEL STORAGE FACILITIES IN THE AREA OF NFCF SAFETY (cont.)

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR3: Design basis accidents: The frequency of occurrence of DBAs in the assessed SNF storage facility is reduced. If an accident occurs, engineered safety features and/or operator actions are able to restore the assessed facility to a controlled state and subsequently to a safe state, and the consequences are mitigated to ensure the confinement of nuclear and/or toxic chemical material. Reliance on human intervention is minimal, and only required after sufficient grace period.	CR3.1: Frequency of DBAs	IN3.1: Calculated frequency of occurrence of DBAs. AL3.1: Lower than that in the reference design.
	CR3.2: Engineered safety features and operator procedures	IN3.2: Reliability and capability of engineered safety features and/or operator procedures. AL3.2: Superior to those in the reference design.
	CR3.3: Grace periods for DBAs	IN3.3: Grace periods for DBAs until human intervention is necessary. AL3.3: Longer than those in the reference design.
	CR3.4: Barriers	IN3.4: Number of confinement barriers maintained (intact) after DBAs. AL3.4: At least one.
	CR3.5: Robustness of containment design	IN3.5: Containment loads covered by design of the facility assessed. AL3.5: Greater than those in the reference design.
UR4: Severe plant conditions: The frequency of an accidental release of radioactivity into the environment is reduced. The source term of accidental release into the environment remains well within the envelope of the reference facility source term and is so low that calculated consequences would not require public evacuation.	CR4.1: In-facility severe accident management	IN4.1: Natural or engineered processes, equipment, and AM procedures and training to prevent an accidental release to the environment in the case of accident. AL4.1: Sufficient to prevent an accidental release to the environment and regain control of the facility.
	CR4.2: Frequency of accidental release into environment	IN4.2: Calculated frequency of an accidental release of radioactive materials and/or toxic chemicals into the environment. AL4.2: Lower than that in the reference facility.
	CR4.3: Source term of accidental release into environment	IN4.3: Calculated inventory and characteristics (release height, pressure, temperature, liquids/gas/aerosols, etc) of an accidental release. AL4.3: Remains well within the inventory and characteristics envelope of the reference facility source term and is so low that calculated consequences would not require evacuation of population.

TABLE 30. INPRO USER REQUIREMENTS AND CRITERIA FOR SUSTAINABILITY ASSESSMENT OF SPENT NUCLEAR FUEL STORAGE FACILITIES IN THE AREA OF NFCF SAFETY (cont.)

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR5: Independence of DID levels and inherent safety characteristics: An assessment is performed for the SNF storage facility to demonstrate that the DID levels are more independent from each other than in the reference design. To excel in safety and reliability, the assessed SNF storage facility strives for better elimination or minimization of hazards relative to the reference design by incorporating into its design an increased emphasis on inherently safe characteristics.	CR5.1: Independence of DID levels	IN5.1: Independence of different levels of DID in the assessed SNF storage facility. AL5.1: More independence of the DID levels is demonstrated compared to that in the reference design, e.g. through deterministic and probabilistic means, hazards analysis, etc.
	CR5.2: Minimization of hazards	IN5.2: Examples of hazards: fire, flooding, release of radioactive material, radiation exposure, etc. AL5.2: Hazards minimized according to the state of the art.
UR6: Human factors related to safety: Safe operation of the assessed SNF storage facility is supported by accounting for HF requirements in the design and operation of the facility, and by establishing and maintaining a strong safety culture in all organizations involved in the life cycle of the facility.	CR6.1: Human factors	IN6.1: Human factors addressed systematically over the life cycle of the SNF storage facility. AL6.1: Evidence is available.
	CR6.2: Attitude to safety	IN6.2: Prevailing safety culture. AL6.2: Evidence is provided by periodic safety reviews.
UR7: RD&D for advanced designs: The development of innovative design features of the assessed SNF storage facility includes associated RD&D to bring the knowledge of facility characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for operating facilities.	CR7.1: RD&D	IN7.1: RD&D status. AL7.1: RD&D defined, performed and database developed.
	CR7.2: Safety assessment	IN7.2: Adequate safety assessment. AL7.2: Approved by a responsible regulatory authority.

10.5.2. User requirement UR1: Robustness of design during normal operation

The rationale of UR1 was provided in Section 5.3. UR1 deals with prevention of AOOs. For an SNF storage facility, examples of AOOs include a temporary loss of:

- Ventilation;
- Forced cooling in a dry or wet storage facility;
- Utilities such as supply of electricity and pressurized air.

The criteria selected for user requirement UR1 are presented in Table 31.

TABLE 31. CRITERIA FOR USER REQUIREMENT UR1

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR1: Robustness of design during normal operation: The assessed SNF storage facility is more robust than the reference design with regard to operation and systems, structures and components failures.	CR1.1: Design of normal operation systems	IN1.1: Robustness of design of normal operation systems. AL1.1: Superior to that in the reference design.
	CR1.2: Subcriticality	IN1.2: Subcriticality margins. AL1.2: Sufficient to cover uncertainties and avoid criticality.
	CR1.3: Facility performance	IN1.3: Facility performance attributes. AL1.3: Superior to those in the reference design.
	CR1.4: Inspection, testing and maintenance	IN1.4: Capability to inspect, test and maintain. AL1.4: Superior to that in the reference design.
	CR1.5: Failures and deviations from normal operation	IN1.5: Expected frequency of failures and deviations from normal operation. AL1.5: Lower than that in the reference design.
	CR1.6: Occupational dose	IN1.6: Occupational dose values during normal operation and AOOs. AL1.6: Lower than the dose constraints.

10.5.2.1. Criterion CR1.1: Design of normal operation systems

Indicator IN1.1: Robustness of design of normal operation systems.

Acceptance limit AL1.1: Superior to that in the reference design.

All safety-relevant equipment and systems in an SNF storage facility are normally designed against loads caused by events associated with internal and external hazards (see Section 4.2.1). It is acknowledged that increasing the robustness of a spent fuel storage facility design is a challenging task for a designer because enhancing one aspect could have a negative influence on other aspects. Accordingly, an optimum combination of design measures is necessary to increase the overall robustness of a design. The design of an SNF storage facility can be made more robust, i.e. reducing the likelihood of failures, by increasing the design margins, improving the quality of manufacture and construction, and by using materials of higher quality.

The design of structures and components of an SNF storage facility needs to consider relevant loading conditions (stress, temperature, corrosive environment, radiation levels, etc.) and creep, fatigue, thermal stresses, corrosion and changes in material properties with time (e.g. concrete shrinkage). For example, materials of structures and components of the SNF storage facility that are in direct contact with the spent fuel need to be compatible with the material of the spent fuel to minimize chemical and galvanic reactions that could degrade the integrity of the spent fuel during its storage.

The **acceptance limit AL1.1** of **CR1.1** is met if evidence available to the INPRO assessor shows that the design of the facility assessed is superior in this regard to the reference design, or, in case a reference facility could not be defined, took best international practice into account and is therefore state of the art.

10.5.2.2. Criterion CR1.2: Subcriticality

Indicator IN1.2: Subcriticality margins.

Acceptance limit AL1.2: Sufficient to cover uncertainties and avoid criticality.

As discussed in section 7.4.2.2. for uranium refining/ conversion and enrichment facilities, to avoid a criticality accident in an SNF storage facility that could result in a large release of radiation and radioactive material, a criticality analysis needs to be performed that demonstrates a design margin of $k_{eff} < 0.90$ for all possible configurations of fissile material. In this analysis, mass concentration, shape, moderation, etc. have to be considered.

The **acceptance limit AL1.2** of **CR 1.2** is met if evidence available to the INPRO assessor shows that in the facility assessed no critical configuration can occur, taking uncertainties into account.

10.5.2.3. Criterion CR1.3: Facility performance

Indicator IN1.3: Facility performance attributes.

Acceptance limit AL1.3: Superior to those in the reference design.

The strategy of ageing management needs to cover all relevant stages in the SNF storage facility lifecycle, all normal operation states, all AOOs and accidents influencing a given system, and all relevant mechanisms of ageing. The designer of an SNF storage facility has to determine the design life of safety related equipment, provide appropriate design margins to take due account of age related degradation and provide methods and tools for the assessment of ageing during operation. The operating organization has to develop a plan for ageing management implementation at the different stages of the facility lifecycle.

Superior performance of the facility is aligned with increased robustness of its design. Superior performance implies:

- Increased emphasis on automation and on-line monitoring;
- A system of recording and analysing deviations from operating procedures, consequences of events and methods to avoid recurrences;
- Availability of clear operating procedures and manuals, providing comprehensive data on the permissible ranges of various parameters;
- Consideration of ageing management in the design documentation;
- Availability of a plan for implementation of ageing management;
- Operator training as an important route to ensuring quality of operation.

The **acceptance limit AL1.3** of **CR1.3** is met if evidence available to the INPRO assessor shows that the performance attributes of the facility assessed are superior to those of the reference design, or, in case a reference facility could not be defined, took best international practice into account and are therefore state of the art.

10.5.2.4. Criterion CR1.4: Inspection, testing and maintenance

Indicator IN1.4: Capability to inspect, test and maintain.

Acceptance limit AL1.4: Superior to that in the reference design.

The assessed design of SNF storage facility is expected to permit efficient and intelligent inspection, testing and maintenance and not just require more inspections and more testing. In particular, the programs of inspection, testing and maintenance need to be driven by a sound understanding of failure mechanisms (corrosion, erosion, fatigue, etc.), so that the right locations are inspected and the right systems, structures and components are tested and maintained at the right time intervals.

The **acceptance limit AL1.4** of **CR1.4** is met if evidence available to the INPRO assessor shows that the capability to inspect, to test and to maintain systems relevant for safety in the facility assessed is superior to a reference design or, in case a reference facility could not be defined, is state of the art and allows easy inspection, testing and maintenance.

10.5.2.5. Criterion CR1.5: Failures and deviations from normal operation

Indicator IN1.5: Expected frequency of failures and deviations from normal operation.

Acceptance limit AL1.5: Lower than that in the reference design.

The frequencies of the AOOs selected (see beginning of Section 10.5.2) for an SNF storage facility need to be derived from operational experience of comparable facilities and supported by PSA. For the facility assessed, it can be possible to reduce these frequencies by increased robustness of the design, high quality of operation, and efficient and intelligent inspection.

The **acceptance limit AL1.5** of **CR1.5** is met if evidence available to the INPRO assessor shows that in the facility assessed the frequencies of AOOs have been reduced in comparison to those in the reference design or, in case a reference facility could not be defined, that the facility assessed took best international practice into account and is therefore state of the art. If quantitative results from operational experience and PSA are not available, alternatively, a deterministic analysis can be developed that indicates a reduced probability of occurrence for AOOs in the facility assessed.

10.5.2.6. Criterion CR1.6: Occupational dose

Indicator IN1.6: Occupational dose values during normal operation and AOOs.

Acceptance limit AL1.6: Lower than the dose constraints.

The assessment of CR1.6 presented in Section 7.4.2.6 for a conversion and enrichment facility is deemed to be substantially similar to the assessment of a storage facility for spent nuclear fuel. Therefore, the INPRO assessor is requested to use the assessment approach described for a conversion and enrichment facility also for such a storage facility.

10.5.3. User requirement UR2: Detection and interception of AOOs

The rationale of UR2 was provided in Section 5.4. The criteria selected for user requirement UR2 are presented in Table 32.

TABLE 32. CRITERIA FOR USER REQUIREMENT UR2

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR2: Detection and interception of AOOs: The assessed SNF storage facility has improved capabilities to detect and intercept deviations from normal operational states in order to prevent AOOs from escalating to accident conditions.	CR2.1: I&C systems and operator procedures	IN2.1: I&C system to monitor, detect, trigger alarms and, together with operator actions, intercept and compensate AOOs. AL2.1: Availability of such systems and operator procedures.
	CR2.2: Grace periods for AOOs	IN2.2: Grace periods until human actions are required after AOOs. AL2.2: Adequate grace periods are defined in design analyses.

10.5.3.1. Criterion CR2.1: I&C systems and operator procedures

Indicator IN2.1: I&C system to monitor, detect, trigger alarms and, together with operator actions, intercept and compensate AOOs.

Acceptance limit AL2.1: Availability of such systems and operator procedures.

The design analysis is expected to specify the regime of safe operating conditions for all equipment and processes. Necessary instrumentation for detecting malfunctions needs to be clearly identified. For example, reliable, continuous air monitoring systems to detect release of radioactivity to operating areas, water level in pools, criticality and temperature monitors can be provided, with necessary interlocks and alarm annunciation systems.

Different from other kinds of NFCFs, an automatic compensation of AOOs is deemed not necessary in an SNF storage facility, i.e. timely operator intervention can be sufficient.

The **acceptance limit AL2.1** of **CR2.1** is met if evidence available to the INPRO assessor shows that the I&C systems in the facility assessed can detect failures and deviations from normal operation of systems relevant to safety and provide alarms. The operator is able to perform interventions that bring the facility back to normal operation.

10.5.3.2. Criterion CR2.2: Grace periods for AOOs

Indicator IN2.2: Grace periods until human actions are required after AOOs.

Acceptance limit AL2.2: Adequate grace periods are defined in design analyses.

An explanation of ‘adequate grace period’ is provided in section 6.3.3.2. The grace period available to the operator for each AOO needs to be defined in the safety analysis of the facility design. The appropriate value of this grace period depends on the ease of failure diagnosis and the complexity of the human action to be taken; i.e. simple failures and consecutive straightforward actions allow for shorter grace periods.

Compared to other kinds of NFCFs, the inertia of an SNF storage facility is very high, resulting in a very slow response to deviations from normal operation. For example, analyses typically show that the cooling system of the pool of an SNF storage facility can be stopped for about 10 days without loss of integrity of the SNF [31]. Thus, it is expected that the design analysis of such a facility will clearly demonstrate sufficient inertia to cope with AOOs.

The **acceptance limit AL2.2** of **CR2.2** is met if evidence available to the INPRO assessor shows that adequate grace periods have been determined for all AOOs in the design analysis for the facility assessed.

10.5.4. User requirement UR3: Design basis accidents

The rationale of UR3 was provided in Section 5.5. Ref [35] provides examples of events that may be associated with DBAs[66] in a SNF storage facility:

> “It should be noted that many events would be addressed either as anticipated operational occurrences or as design basis accidents. However, some of these events could also lead to severe accidents, which are beyond the design basis. Whilst the probability of such beyond design basis accidents occurring is extremely low, in the preparation of operating procedures and contingency plans the operating organization should consider events such as the following:
>
> (a) Crane failure with a water filled and loaded cask, suspended outside the pool;
> (b) Loss of safety related facility process systems such as supplies of electricity, process water, compressed air and ventilation;
> (c) Explosions due to the buildup of radiolytic gases;
> (d) Fires leading to the damage of items important to safety (to reduce the risk of fire, the amount of combustible material or waste should be controlled, as should be the amount of other flammable materials (…));

[66] Depending on the consequences some of these events may be associated either with AOO or with design extension conditions.

(e) Extreme weather conditions, which could alter operating characteristics or impair pool or cask heat removal systems;
(f) Other natural events such as earthquake or tornado;
(g) External human induced events (airplane crash, sabotage, etc.);
(h) Failure of the physical protection system.

Consideration should also be given to the possible misuse of chemicals (e.g. unintended introduction into the pool water of acidic or alkaline fluids used for the regeneration of ion exchange resin)."

As stated before, the facilities need to be designed against all external and internal hazards. The criteria selected for user requirement UR3 are presented in Table 33.

TABLE 33. CRITERIA FOR USER REQUIREMENT UR3

User requirement	Criteria	Indicator (IN) and Acceptance Limit (AL)
UR3: Design basis accidents: The frequency of occurrence of DBAs in the assessed SNF storage facility is reduced. If an accident occurs, engineered safety features and/or operator actions are able to restore the assessed facility to a controlled state and subsequently to a safe state, and the consequences are mitigated to ensure the confinement of nuclear and/or toxic chemical material. Reliance on human intervention is minimal, and only required after sufficient grace period.	CR3.1: Frequency of DBAs	IN3.1: Calculated frequency of occurrence of DBAs. AL3.1: Lower than that in the reference design.
	CR3.2: Engineered safety features and operator procedures	IN3.2: Reliability and capability of engineered safety features and/or operator procedures. AL3.2: Superior to those in the reference design.
	CR3.3: Grace periods for DBAs	IN3.3: Grace periods for DBAs until human intervention is necessary. AL3.3: Longer than those in the reference design.
	CR3.4: Barriers	IN3.4: Number of confinement barriers maintained (intact) after DBAs. AL3.4: At least one.
	CR3.5: Robustness of containment design	IN3.5: Containment loads covered by design of the facility assessed. AL3.5: Greater than those in the reference design.

10.5.4.1. Criterion CR3.1: Frequency of DBAs

Indicator IN3.1: Calculated frequency of occurrence of DBAs.

Acceptance limit AL3.1: Lower than that in the reference design.

Examples of DBAs to be considered in a SNF storage facility have been defined above (beginning of Section 10.5.4). The frequency of occurrence of a DBA in the facility assessed needs to be determined via a probabilistic risk assessment.

The calculated frequency of DBAs caused by external hazards can be influenced by the designer via an increased robustness of the confinement (building) walls, and by the future owner/operator of the facility by selecting an appropriate site (see UR7).

The **acceptance limit AL3.1** of **CR3.1** is met if evidence available to the INPRO assessor uses probabilistic analyse to show that the frequency of the defined DBAs in the facility assessed is lower than that in the reference design. If quantitative results are not available, technical arguments can be developed that indicate a reduction of these frequencies based on an increase of design robustness, high quality of operation, intelligent inspection and maintenance programs, advanced I&C systems, and increased inertia.

10.5.4.2. Criterion CR3.2: Engineered safety features and operator procedures

Indicator IN3.2: Reliability and capability of engineered safety features and/or operator procedures.

Acceptance limit AL3.2: Superior to those in the reference design.

For accidents in SNF storage facilities, automatic engineered safety features are deemed to be not necessary due to the slow response of the system to accidents other than criticality accidents. Thus, to meet the acceptance limit AL3.2 of CR3.2, superior operator procedures (in comparison to those for the reference facility) need to be available to control accidents, restore the facility to a controlled state and keep the consequences (e.g. dose) below authorized limits. In case a reference facility cannot be defined, it can be demonstrated that the design took best international practice into account and is therefore state of the art.

10.5.4.3. Criterion CR3.3: Grace periods for DBAs

Indicator IN3.3: Grace periods for DBAs until human intervention is necessary.

Acceptance limit AL3.3: Longer than those in the reference design.

The criterion CR3.3 'grace periods for DBAs' implies a similar concept as introduced earlier for control of AOOs (see CR2.2). However, similar to the situation for AOOs, the system response to DBAs in an SNF storage facility is rather slow due to the high inertia, thus leaving ample time for the operator to intervene. Thus, it is assumed that sufficient grace periods are available for all DBAs in such a facility.

The **acceptance limit AL3.3** of **CR3.3** is met if evidence available to the INPRO assessor shows that in the facility assessed the grace periods are longer than those in the reference design. Alternatively, if a reference facility cannot be found, it can be demonstrated that the design of the facility assessed took available information on best international practice into account and is therefore state of the art.

10.5.4.4. Criterion CR3.4: Barriers

Indicator IN3.4: Number of confinement barriers maintained (intact) after DBAs.

Acceptance limit AL3.4: At least one.

The design of the facility is expected to provide deterministically for continued integrity at least of one barrier containing the radioactive material following any DBA caused by events associated with internal or external hazards. Alternatively, the probability of losing all barriers could be used as an INPRO methodology indicator with a sufficient low value as its acceptance limit.

The **acceptance limit AL3.4** of **CR3.4** is met if evidence available to the INPRO assessor shows that after a DBA in the facility assessed, at least one barrier remains intact, avoiding a release of radioactive material from the facility.

10.5.4.6. Criterion CR3.5: Robustness of containment design

The INPRO assessment of CR3.5 presented for a uranium conversion and enrichment facility in Section 7.4.4.5 is deemed to be substantially similar to that for a spent nuclear fuel storage facility. Thus, that approach can be used by the INPRO assessor also for the storage facility.

10.4.6. User requirements UR4 – UR7

The rationale for UR4 – UR7 is provided in Sections 5.6 – 5.9.

The INPRO assessment of the spent fuel storage facility against user requirement UR4 (severe plant conditions) is deemed to be substantially similar to the assessment method of UR4 described in Section 7.4.5 for an enrichment facility (including criteria, indicators and acceptance limits).

The INPRO assessment of the spent fuel storage facility against user requirement UR5 (independence of DID levels and inherent safety characteristics) is deemed to be substantially similar to the assessment method of UR5 described in Section 8.5.6 for fuel manufacturing facilities and in Section 7.4.6.1 (criterion on minimisation of hazards) for uranium conversion and enrichment facilities.

The INPRO assessment of the spent fuel storage facility against user requirement UR6 (human factors related to safety) is deemed to be substantially similar to the assessment method of UR6 described in Section 6.3.7 for mining and milling facilities (including criteria, indicators and acceptance limits).

The INPRO assessment of the spent fuel storage facility against user requirement UR7 (RD&D for advanced designs) is deemed to be substantially similar to the assessment method of UR7 described in Section 6.3.8 for mining and milling facilities (including criteria, indicators and acceptance limits).

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GLOSSARY

In this publication the safety related terms are used as they defined in the IAEA Safety Glossary [53].

assessment (INPRO assessment of NES sustainability): An assessment using the INPRO methodology is a process of making a judgment about the long term sustainability of a nuclear energy system. In principle, analyses using analytical tools are not part of an INPRO assessment but could provide necessary input for the assessment. The assessment of a nuclear energy system is done at the criterion level of the INPRO methodology. In the case of a numerical criterion, the assessment process consists of comparing the value of an indicator with the value of the acceptance limit of a criterion. In the case of a logical criterion – mostly phrased in the form of a question – the assessment is done by answering the question raised.

assessor: The INPRO assessor is an expert or a team of experts applying the INPRO methodology in a nuclear energy system assessment. The assessor is typically a member of the academic society of the host country (e.g. an academy of science). The assessor may also be from a nuclear research centre, a utility, a supplier, or an organization of the regulator.

basic principle: As defined in the INPRO methodology, an INPRO basic principle is a statement of a general goal that has to be achieved in order to make a nuclear energy system sustainable in the long term. It therefore provides a basic impetus for the development of necessary capabilities and design features.

closed fuel cycle: This is a nuclear fuel cycle that recycles spent fuel. An example of a partly closed fuel cycle is one where spent uranium fuel is reprocessed to (mono) recycle the fuel's bred plutonium for use in producing mixed oxide (MOX) fuel. A completely closed fuel cycle is foreseen in proposed nuclear energy systems where fast breeder reactors would continuously recycle all of their spent fuel.

criterion: As defined in the INPRO methodology, an INPRO criterion enables the assessor to determine whether and how well a user requirement for sustainability assessment is being met by a given nuclear energy system. A criterion consists of an indicator (IN) and an acceptance limit (AL). INs may be based on a single parameter, on an aggregate variable, or on a status statement. ALs may be international or national regulatory limits or limits defined by the INPRO methodology. Two types of criteria are distinguished: numerical and logical. A numerical criterion has an IN and AL that is based on a measured or calculated value that reflects a property of a NES. A logical criterion is associated with some important feature of (or measure for) a NES and is usually presented in the form of a question that has to be answered positively.

event: In the context of the reporting and analysis of events, an event is any occurrence unintended by the operator, including operating error, equipment failure or other mishap, and deliberate action on the part of others, the consequences or potential consequences of which are not negligible from the point of view of protection or safety.

> *internal event:* An event that originates inside the facility and potentially affects the safety of the facility. Typical examples of internal events are failures of equipment.
>
> *external event:* see definition in Ref [53].

evolutionary design: A facility design that achieves improvements over previous designs through small to moderate modifications, with a strong emphasis on maintaining design features that are proven to minimize technological risks.

innovative design: This is an advanced nuclear facility design that incorporates radical conceptual changes in design approaches or system configuration in comparison with existing practice.

nuclear energy system (NES): A NES comprises the complete spectrum of nuclear facilities and associated legal and institutional measures (infrastructure). Nuclear facilities include nuclear reactor facilities as well as facilities for mining and milling, refining, conversion and enrichment of uranium, manufacturing of nuclear fuel, reprocessing of nuclear fuel (if a closed nuclear fuel cycle is used), and facilities for related materials management activities, including transportation and waste management (storage and disposal). Legal measures consist of the national nuclear law and international agreements, treaties, and conventions. Institutional measures include the corresponding national institutions such as regulatory bodies.

open fuel cycle: This is a nuclear fuel cycle that defines spent fuel as waste to be disposed of. It is also called a *once through* fuel cycle.

reference facility: Within the updated INPRO methodology, a reference facility (or design) is an NFCF of most recent design operating in 2013, preferably from the same designer as the assessed facility, and complying with the current safety standards. If a reference design cannot be identified within the same technology lineage, a similar existing design using comparable technology or, when other options are not available, an existing facility design using different technology for the same purpose can be used as the reference. This reference design is to be compared in the INPRO assessment to the assessed facility (assumed to be installed after 2013). Previous experience with INPRO assessments shows that defining the assumed installation date of the selected reference design can help to reduce the potential for misinterpretation of terms. Note that 2013 was the date selected at the beginning of the latest methodology update. This date should be revised periodically along with the rest of methodology.

sustainability: In the INPRO methodology, sustainability is defined as the ability of a nuclear energy system to operate until at least the end of the twenty-first century.

user requirement: A user requirement defines what needs to be done to meet the target/goal of an INPRO methodology basic principle. It is directed at specific institutions (users) involved in nuclear power development, deployment and operation, i.e. the developers/designers, government agencies, facility operators, and support industries.

ABBREVIATIONS

ADU	ammonium diuranate
AL	acceptance limit (INPRO)
AOO	anticipated operational occurrence
BP	basic principle (INPRO)
CR	criterion (INPRO)
DBA	design basis accident
DID	defence in depth
HF	human factor
HFE	human factors engineering
I&C	instrumentation and control
IN	indicator (INPRO)
INPRO	International Project for Innovative Nuclear Reactors and Fuel Cycles
LEU	low enriched uranium
MDU	magnesium di-uranate
MOX	uranium-plutonium mixed oxide
NFCF	nuclear fuel cycle facility
PIRT	phenomena identification and ranking table
PSA	probabilistic safety assessment
RD&D	research, development and demonstration
SNF	spent nuclear fuel
UR	user requirement (INPRO)

ANNEX
IAEA SAFETY STANDARDS AND THE INPRO METHODOLOGY

A-1. DIFFERENCES BETWEEN THE IAEA SAFETY STANDARDS AND THE INPRO METHODODLOGY

The IAEA Safety Standards are internationally endorsed requirements and guides for achieving sufficient levels of safety over the full spectrum of nuclear installations and activities. The standards can be used by Member States as references in developing and applying each nation's regulations for nuclear safety.

The INPRO methodology is a tool for assessing the long term sustainability of a planned nuclear energy system (NES). Developed by IAEA task groups of internal and external experts, the assessment methodology covers all areas relevant to NES sustainability, addressing all planned reactor types from design to decommissioning and all necessary fuel cycle and support activities from mining to waste management. The methodology is structured around NES sustainability considerations in the following areas: economics, infrastructure, proliferation resistance, waste management, environmental impacts of stressors and resource depletion, and safety.

The INPRO sustainability assessment is to be performed by a competent institution within a country that plans to start (or increase) a national NES. The outcome of an assessment is the identification of gaps to be filled by the country with appropriate actions to achieve long-term sustainability of the system.

The application of the INPRO methodology does not replace any aspect of a licensing process. It is emphasized that for a NES to achieve long term sustainability, it must comply with the IAEA Safety Standards while also meeting the INPRO methodology's sustainability criteria in all areas including safety.

A-2. HIGH LEVEL SAFETY STANDARDS CONSIDERING FUEL CYCLE FACILITIES

A-2.1. Fundamental safety objective and fundamental safety principles

The Fundamental Safety Objective defined in the Safety Fundamentals [A-1] is to "*protect people and the environment from harmful effects of ionizing radiation*". This objective applies for all nuclear activities and facilities including reactors, and for all stages during the lifetime of a nuclear facility from planning through normal operation up to decommissioning and closure.

To attain the highest standards of safety that can reasonably be achieved, the Safety Fundamentals state that measures have to be taken to:

- Control the radiation exposure of people and the release of radioactive material to the environment;
- Restrict the likelihood of events that might lead to a loss of control over (the nuclear reactor core), a nuclear chain reaction, radioactive sources, or any other sources of radiation;
- Mitigate the consequences of such events if they were to occur.

Ref [A-1] sets out ten Fundamental Safety Principles[67] that need to be addressed in a nuclear power program to meet the fundamental safety objective without unduly limiting the operation of facilities. The Fundamental Safety Principles provide the basis for the IAEA safety standards and safety related programs. Application of the Fundamental Safety Principles will facilitate the use of the international IAEA safety standards and will lead to greater consistency between the approaches to safety of different States.

A-2.2. IAEA general requirements for assessing the safety of facilities and activities

Ref [A-2] defines twenty-four high level requirements for a comprehensive safety assessment of nuclear facilities and activities based on the ten Fundamental Safety Principles. These high level requirements relate to any human activity that may cause people or the environment to be exposed to radiation risks. Safety assessment is the systematic process that is to be carried out throughout the lifetime of the facility or activity to ensure that all the relevant safety requirements are met by the proposed (or actual) design.

The safety assessment using Ref [A-2] is to be carried out and documented by the licensee, i.e. the organization responsible for operating a nuclear facility or conducting an activity; is to be independently verified in a peer review; and is to be submitted to the regulatory body as part of the licensing process.

A-2.3. IAEA specific safety requirements for the design of nuclear fuel cycle facilities

The IAEA has issued a safety standard on the specific safety requirements for NFCFs [A-3]. In that standard, it is stated that the following measures have to be taken to achieve the Fundamental Safety Objective (Section I.1 above):

- To control the radiation exposure of people and the release of radioactive material to the environment;
- To restrict the likelihood of events that might lead to a loss of control over sources of radiation; and
- To mitigate the consequences of such events, were they to occur.

It is in the responsibility of the licensee to ensure that a comprehensive safety assessment [A-2] of the design and the operation of the NFCF is carried out to demonstrate that the Fundamental Safety Objective is achieved and the Fundamental Safety Principles are met [A-1]. The safety standard [A-3] presents a series of requirements for the legal framework and regulatory supervision, the management system and verification of safety, and the siting, design, construction, commissioning, operation, and decommissioning of the facility.

In addition to safety requirements, there are numerous (general and specific) IAEA safety guides based on international consensus on how to comply with the safety requirements (see for example Refs [A-4 – A-7]).

[67] A distinction is made between the Fundamental Safety Principles, which are defined in Ref [5], and the INPRO sustainability assessment requirements for nuclear safety which have been derived from the Safety Fundamentals but are specific to the INPRO assessment of sustainability in the area of NFCF safety.

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