SAFETY OF NUCLEAR FUEL CYCLE FACILITIES

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SAFETY OF NUCLEAR FUEL CYCLE FACILITIES

1.0. INTRODUCTION

1.1. Background

Nuclear Fuel Cycle activities have been an integral part of the development of nuclear technology in every country. Nuclear Fuel Cycle facilities are facilities in which radioactive material is processed, used, stored or disposed of. Development of processes for fuel cycle facilities - mining, milling, enrichment, fuel fabrication, interim spent fuel storage, transport, reprocessing and waste management has been a core activity, which has significantly contributed to the growth of nuclear energy. The success of the nuclear programme greatly depends on the strengths of the country in nuclear fuel cycle activities. It is however necessary to emphasize that global success in the growth of nuclear energy to a large extent, depends on the safe and economical operation of the fuel cycle facilities as much as it depends on a safe and economical operation of nuclear reactors themselves. Indeed, the public acceptance of the safety of nuclear fuel cycle facilities and waste management options needs to be addressed in a comprehensive fashion for the nuclear energy to register a significant growth in the coming decades.

It is realized that the experience with respect to nuclear fuel cycle facilities in various countries has not been collated and harmonized to the extent that has been done for the reactor systems. For example, the limits for exposures in various levels vary from country to country (see Fig.1.1, from Ref.1.1). Arriving at such limits, falls strictly under the purview of the regulatory body in the respective country, even though it is presumed that the ICRP limits in general form the guidelines for the regulatory process. However, it is desirable to harmonize the approaches adopted by various countries to arrive at an envelope of limits and codes which can be used by all the countries. IAEA has taken several measures to initiate the process of harmonization of the approaches of various countries through a set of systematic documentation (Ref.1.2-1.8). IAEA has published several reports, nuclear safety standards and safety requirements for design (Series on Safety Standards, Safety Guides). A programme to update all the IAEA nuclear safety standards under the guidance of the IAEA Nuclear Safety Standards Committee has been
in progress. The basic safety principles proposed in the International Safety Advisory Group reports (INSAG 1-14) have served as important guidelines for developing methodologies for assessing nuclear systems for their safety.

![Diagram](Fig.1.1 Variation of frequency of occupational exposure with dose (Ref. 1.1))

Documentation and harmonization of safety issues under the auspices of IAEA facilitates a convergence of views on the desirable goals for these processes, and an early advancement of technologies required to achieve these, through efficient use of resources and knowledge from a wide range of international expertise. It is thus appropriate that under the INPRO project, a manual dealing with the safety of nuclear fuel cycle facilities is prepared. The present manual has thus arisen out of the commitment of the IAEA to achieve nuclear energy growth through innovations in key factors related to safety of Nuclear Fuel Cycle Facilities (NFCF).

The overall objective of INPRO is to ensure that nuclear energy provides a substantial contribution in the form of sustainable and environment friendly energy to the growing needs of the society in 21st century. Hence, in the assessment of an Innovative
Nuclear System, various parameters such as safety, economics, waste management, proliferation resistance and environment impact are all intimately connected.

This document discusses the safety issues only, though safety has a large influence on economics, environment and the other aspects. The interfaces between various aspects and interactive approaches would emerge in the INPRO manual for assessment of methodology.

There is ample scope for enhancing safety. This is well illustrated in the case of the annual occupational exposure, one of the important safety parameters (see Fig 1.3, from Ref.1.9). Though the ICRP annual limit is 20 mSv/a, it is seen that majority of the workers of CEA received doses less than 1 mSv/a and only 7% of the workers received dose between 5 to 10 mSv/a.

![Fig.1.3. Results reported by CEA (Ref. 1.9) illustrating the scope for reduction in radiation exposure to workers](image)

It is observed that the radiation awareness of occupational workers and their response to radiation protection measures have improved considerably over the years. The regulations for radiation exposures have been standardized by IAEA and are widely accepted. Better radiation detectors and alarm systems are now available. Mechanisms for implementation of radiation protection regulations are in place. A clear definition of
responsibilities and appropriate training for the operation and maintenance activities must be ensured. These would help in establishing safety as a culture, rather than as a practice. The annual occupational exposure per unit energy output could serve as an index of prevalent safety culture in the fuel cycle facility.

1.2. Objectives

The objectives of the present safety manual are:

a) To provide a framework for assessing the innovations in the safety of the fuel cycle facilities by evolving a comprehensive methodology based on the INPRO approach.

b) To describe the operations carried out in nuclear fuel cycle facilities with focus on the safety aspects.

c) To underline areas where RD&D should catalyze enhanced safety in these facilities.

d) To indicate areas where further deliberation is required to arrive at clear indicators for innovation.

1.3. Scope

This manual mainly deals with the nuclear fuel cycle facilities excluding reactors. The development of standards and safety guides for these facilities is an active pursuit of IAEA. A number of draft safety standards for specific nuclear fuel cycle facilities are available, and these provide a comprehensive overview of the safety issues and practices with respect to fuel cycle facilities. The present manual deals with safety issues related to design and operation of mining, milling, refining, conversion, enrichment, fuel fabrication, fuel storage and fuel reprocessing facilities. The application of INPRO methodology in terms of identifying indicators and acceptance criteria are discussed for mining, milling, enrichment, fuel fabrication and fuel reprocessing facilities. As the safety issues involved in refining and conversion are similar to those of enrichment and fuel fabrication facilities which handle UF₆, the indicators are not discussed separately for these facilities. Safety issues in fuel storage have been discussed. However,
identification of indicators and acceptance are best carried out along with reactors/ waste management facilities, as most of the safety issues are closely related. As the decommissioning activities of each of the fuel cycle facilities are different in nature, application of INPRO methodology to decommissioning is not discussed in this document.

It is clear that the fuel cycle operations are more varied in the processes and approaches, as compared to reactor systems. Most significant of these variations is the fact that some countries are pursuing storage of spent fuels with long term options, while some others have a policy of closing the fuel cycle. Further, diversity is large when 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 (fissile material: low enriched uranium/ natural uranium/ uranium-plutonium; fuel form: metal/ oxide/ carbide/ nitride) and varying burn-up 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 which are specific to certain fuel types. This approach has been taken in order to arrive at a generalized procedure that could enable the users of the manual to apply it with suitable variations as applicable to the specific fuel cycle technologies. In addition, it is recognized that the defense in depth approach and ultimate goal of inherent safety form the fundamental tenets of safety philosophy. However, for many of the specific parameters, international codes, guidelines are not readily available in open literature. The important issues have been highlighted in this manual to provide a framework for further development of the safety codes for NFCF.

1.4. Structure

Chapter 2 of this manual provides an introduction to the safety aspects and discusses the defense in depth approach. Chapter 3 deals with the INPRO assessment methodology, describing various basic principles and user requirements and the approach to the quantification of the assessment procedure.
In Chapter 4, specific fuel cycle operations are discussed from the point of view of safety indicating possible innovations. Chapter 5 provides a summary and also recommends future activities, which can be undertaken by IAEA, which would enable a robust assessment of the safety of nuclear fuel cycle facilities in the framework of INPRO. In the appendix, application of INPRO methodology to a hypothetical fuel fabrication facility using sol-gel process is given as an illustration.
2. SAFETY ASPECTS OF NUCLEAR FUEL CYCLE FACILITIES

2.1 Introduction to Safety

Basic objectives, concepts and principles are defined for ensuring safety of nuclear installations in which the stored energy or the energy developed in certain situations could potentially result in the release of radioactive material from its designated location with the consequent risk of radiation exposure to personnel. These principles are derived from the following fundamental safety objectives.

2.1.1. Nuclear Safety Objective:

To protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defense against radiological hazards.

This general nuclear safety objective is supported by two complementary safety objectives dealing with radiation protection and technical aspects. They are interdependent. The technical aspects in conjunction with administrative and procedural measures ensure defense against hazards due to exposure to radiation.

- **The radiation protection objective** is to ensure that in all operational states, exposures to radiation are kept below prescribed limits and as low as reasonable and practicable, economic and social factors taken into account (ALARP) and to ensure mitigation of the radiological consequences of accidents.

- **The technical safety objective** is to take all reasonably practical measures to prevent accidents and to mitigate their consequences, should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor or below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.
Safety objectives require that nuclear installations are designed and operated so as to keep all sources of radiation exposure under strict technical and administrative control. However, the radiation protection objective does not preclude limited exposure of people or the release of legally authorized quantities of radioactive materials to the environment from installations in operational states. Such exposures and releases, however, must be strictly controlled and must be in compliance with specified operational limits and radiation protection standards.

In order to achieve these objectives in the design of a nuclear installation, comprehensive safety analyses should be carried out to identify all sources of exposure and to evaluate radiation doses that could be received by the public and by workers at the installation, as well as potential effects of radiation on the environment. The safety analysis must examine: (1) all planned normal operational modes of the plant (2) plant performance in anticipated operational occurrences (3) design basis events and (4) selected severe accidents of low probability. The design for safety of a nuclear facility should follow the principle that plant states that could result in high radiation doses or radionuclide releases are of very low probability of occurrence, and plant states with significant probability of occurrence have only minor or no potential radiological consequences. The safety approach should ensure that the need for external intervention measures is limited or even eliminated in technical terms, although authorities, for emergency preparedness, would still require such measures.

2.2 Nuclear Fuel Cycle

Nuclear Fuel cycle comprises a number of activities other than reactor operation, the possible combinations of which provide the various fuel cycle options (see Fig.2.1). These are:

- Uranium/Thorium Mining and Milling
- Uranium Refining and Conversion
- Uranium Enrichment
- Fuel Fabrication
- Transportation
- Spent Fuel Storage
- Spent Fuel Reprocessing including MA partitioning
- Re-fabrication including MA fuels and targets
- Radioactive Waste Management
- Waste disposal
- Decommissioning

![Fig.2.1 Schematic diagram of fuel cycle (Ref.2.1)](image)

Depending upon the requirements and perceptions of the individual country, either open or closed fuel cycle option is followed. In an open fuel cycle, the spent fuel is disposed of directly, without reprocessing. In a closed fuel cycle, spent fuel is reprocessed and the reprocessed fuel is used, thus closing the fuel cycle. A comprehensive review of the activities related to nuclear fuel cycle is given in Refs.2.2, 2.3 and 2.4. The latest trends on the reactors fuels and their technology could be found in
the comprehensive report on the technology roadmap for Gen-IV nuclear energy systems by US-DOE (Ref.2.5). The characteristics of nuclear fuel cycle depend upon the type of nuclear reactors and the fuel. A few examples illustrate this statement. PHWRs use natural uranium as fuel, whereas PWRs and BWRs use low enriched uranium (LEU) as fuel. Fast reactors use mixed U/Pu oxide as fuel. Metallic fuels can also be used in fast reactors. The predominant nuclear fuel cycle is based on U/Pu, where Pu obtained by reprocessing of the spent fuel of U in thermal reactors, is used in fast reactors. The $^{232}\text{Th}/^{233}\text{U}$ fuel cycle would be required for large-scale use of thorium. Fast reactors with U/Pu fuel cycle and thermal reactors with $^{232}\text{Th}/^{233}\text{U}$ fuel cycle provide closed fuel cycle options (Ref.2.4). Typical fuel cycle options are illustrated in Fig.2.2.

![Fig.2.2 Typical fuel cycle options (Ref.2.6)](image)

As stated in section 3 of Ref. [1.2], fuel cycle facilities employ a great diversity of technologies and processes. They differ from reactors in several important aspects. First, fissile material and wastes are handled, processed, treated, and stored throughout the nuclear fuel cycle facilities in dispersible (open) forms. Consequently, the materials of interest to nuclear safety are more distributed throughout the nuclear installations in
contrast to reactors, where the bulk of the nuclear material is located in the reactor core or fuel storage areas. For example, the nuclear materials in reprocessing plants are present for most part of the process in solutions that are transferred between vessels used for different parts of the processes, whereas in reactors the nuclear material is present in concentrated form as solid fuel. Second, these treatment processes use large quantities of hazardous chemicals, which can be toxic, corrosive and/or combustible. Third, the facilities 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. Fourth, in fuel cycle facilities a significantly greater reliance is placed on the operator, not only to run the facility during its normal operation, but also to respond to fault and accident conditions (Ref 2.7). 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, the major steps in the NFCFs consist of chemical processing of fissile materials, which may lead to inadvertent release of hazardous chemical and/or radioactive substances, if not properly managed.

Whereas the reactor core of an NPP presents a very large inventory of radioactive material at high temperature, pressure, and within a relatively small volume, an NFCF operates at near ambient pressure and temperature and with comparatively low inventories at each stage of the overall process. In nuclear waste repositories, the total nuclide inventory will progressively increase to a maximum over the operating period of the facility.

Usually in NFCFs, there are long timescales involved in the development of accidents except in the case of criticality and less stringent process shutdown requirements are required to maintain the facility in a safe state, as compared to reactors. Such facilities also often differ from reactors with respect to the enhanced importance of ventilation systems in maintaining their safety— even under normal operation. This is because materials in these facilities are in direct contact with ventilation or off-gas systems. The robustness of barriers between radioactive inventories and the operators as
well as the environment must be ensured more stringently as compared to reactor systems. Fire protection and mitigation assume greater importance in NFCFs 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 must all 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 reactors. In addition, radioactivity release and criticality issues warrant more attention in NFCFs compared to NPPs. A further comparison of relevant features of an NPP, a chemical process plant and an NFCF is presented below and in Table 2.1 (Ref.1.7).

### Table 2.1. Typical Differences between NPPs, Chemical Process Plants and NFCFs

<table>
<thead>
<tr>
<th>Feature</th>
<th>NPP</th>
<th>Chemical Process Plant Feature</th>
<th>NFCF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Areas of hazardous sources and inventories</td>
<td>Localized at core and spent fuel pool. Standardized containment system. Cooling of residual heat. Criticality management.</td>
<td>Distributed in the process. Present through out the process equipments.</td>
<td>Consisting both of nuclear materials and chemical materials. Co-existence of NPP features and chemical plant features. Present through out the process equipments in the facility.</td>
</tr>
<tr>
<td>Type of hazardous materials</td>
<td>Mainly nuclear materials.</td>
<td>A wide variety of materials dependent on the plant, e.g., poisons, acids, toxins, combustibles and explosives.</td>
<td>Fissile materials, nitric acid, hydrogen fluoride, solvents, process and radiolytic hydrogen, etc.</td>
</tr>
<tr>
<td>Physical forms of hazardous materials</td>
<td>The core in general is in solid form. Liquid, gas and dust (aerosol) of radioactive materials released to the environment in accident phase.</td>
<td>A wide variety of physical forms dependent on the process, e.g., as solid, liquid, gas, slurry, powder.</td>
<td>All physical forms of fissile material and a wide variety of chemical materials. Immobilized radioactive materials.</td>
</tr>
<tr>
<td>Typical causes of accidents</td>
<td>Incidents related to the core and the safety system, initiated by internal or external events.</td>
<td>Operator and equipment failures, e.g., Loading of the wrong amount of or wrong raw material into the vessel or</td>
<td>Incidents related to safety function and barriers, fire, explosion, loss of ventilation, loss of barriers, transport failures.</td>
</tr>
</tbody>
</table>
From 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, gas and solids. Accordingly, it is necessary to provide correspondingly a wide range of specific safety measures as inherent parts of these activities. Radiation protection requirement of the personnel is more demanding especially in view of the many human interventions required for the operation and maintenance of fuel cycle facilities. The safety issues encountered in various fuel cycle facilities have been discussed in Ref.1.2, and these are summarized (partially) in Table 2.2. A comprehensive review of the safety of fuel cycle facilities is given in Ref.2.7.

For existing NFCFs, the emphasis is on the control of operations using administrative and operator controls to ensure safety, as opposed to 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 contact with nuclear material in the process, which may include open handling and transfer of nuclear material in routine processing, special attention is warranted to ensure worker safety. Potential intakes of radioactive material require control to prevent and
minimize contamination and thus ensure adherence to specified operational dose limits. In addition, releases of radioactive material into the facilities and through monitored and unmonitored pathways can result in significant exposures.

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 are different in number and strength for different kinds of nuclear installations. For example, in mining, focus is on preventing contamination of ground or surface water with releases from uranium mining tails. Chemicals and uranium by-products are the potential hazards of the conversion stage. In a fuel fabrication facility, safety is focused on preventing criticality in addition to contamination via low-level radioactive material. It is possible to enhance safety features in an INS by co-location of front end (e.g. mining and enrichment facilities) and back end (reprocessing and waste management) facilities. This would have benefits through minimal transport and avoiding multiple handling of radioactive materials in different plants of the fuel cycle facility.

Compared to safety of operating nuclear power plants, only limited open literature is available on the experience related to safety in the operation of nuclear fuel cycle facilities. IAEA has recognized the need for international efforts towards defining safety concepts and regulations for NFCFs. Safety of and regulations for nuclear fuel cycle facilities have been discussed in IAEA Technical committee meeting and brought out as IAEA TECDOC Series No.1221 (2001) (Ref.1.2). Safety of and regulations for nuclear fuel cycle facilities have also been discussed in the International Conference on Topical Issues in Nuclear Safety, 2001, organized by IAEA. (Ref.2.7). Some aspects of nuclear fuel cycle such as uranium mining have also been reported extensively (Ref 4.1-4.8). A group of experts of the NEA (OECD) committee have also prepared a state of the art report on safety of nuclear installations in 1981 and 1993 (Ref 2.2,2.3). Recently draft safety guides on conversion/enrichment facilities and fuel fabrication facilities and status reports on fuel reprocessing have also been brought out by IAEA. (Refs.4.12, 4.24 and 4.39.)
### TABLE 2.2 SAFETY ASPECTS FOR FUEL CYCLE FACILITIES
(Adopted from Ref. 1.2)

<table>
<thead>
<tr>
<th></th>
<th>Criticality</th>
<th>Radiation</th>
<th>Chemical Toxicity</th>
<th>Fire/Explosion</th>
<th>Product/Residue Storage</th>
<th>Waste Storage</th>
<th>Ageing Facilities</th>
<th>Decommissioning</th>
<th>Effluents</th>
<th>Maintenance</th>
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<tbody>
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<td>Mining/Milling</td>
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<tr>
<td>Conversion</td>
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<tr>
<td>Enrichment</td>
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<td>Fuel Fabrication</td>
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<td>Interim Storage</td>
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<td>Reprocessing</td>
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<td>MOX fuel fabrication</td>
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<tr>
<td>Transportation</td>
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</tbody>
</table>

- may be a concern depending on specific conditions (enrichments, composition etc.,)

@ - concern at most facilities
2.3 Basic safety functions

Fundamental safety functions for nuclear fuel cycle facilities (including spent fuel storage at reactor sites), are to:

- maintain sub-criticality and control chemistry
- remove decay heat and process heat from chemical processes
- confine radioactive materials and shield sources of radiation

To ensure that the fundamental safety functions are adequately fulfilled, an effective defense-in-depth strategy should be implemented, combined with an increased use of inherent safety characteristics and passive systems in fuel cycle installation designs. As manual operations cannot be completely avoided, much emphasis needs to be put on administrative procedures, including a clear definition of responsibilities and appropriate training for the operation and maintenance. At the next level of hierarchy, safety culture is essential if the knowledge and experience gained has to be translated into enhanced safety features and practices.

2.4 Defense-in-depth

Defense-in-depth (DID) provides an overall strategy for safety measures and features of nuclear facilities (see Ref. 2.8 for more details). The strategy is twofold: first, to prevent accidents and second, if prevention fails, to limit their potential consequences and prevent any evolution to more serious conditions.

Accident prevention is the first priority. The rationale for the priority is that provisions to prevent deviations of the plant state from well known operating conditions are more effective and more predictable than measures aimed at mitigation of such departure, because the plant’s performance and safety status may deteriorate when the status of the plant or a component departs from normal conditions. Therefore preventing the degradation of plant safety status and performance will provide an effective protection to the public and the environment as well as the protection of the investment.
Should preventive measures fail, however, control, management and mitigation measures, in particular the use of a well designed confinement features can provide the necessary additional protection to the public and the environment. An increased use of inherent safety characteristics will strengthen accident prevention in nuclear installations. A plant has an inherently safe characteristic against a potential hazard if the hazard is rendered physically impossible, without human intervention. The term inherent safety is normally used with respect to a particular characteristic, not to the plant as a whole. For example, a fuel cycle facility is inherently safe against criticality if it cannot attain a critical configuration of material under any circumstance. This can be achieved for example, through the use of ever safe geometries for the process tanks in a reprocessing plant. Another such example is the choice of suitable reactants to prevent red-oil explosion. Suitable choice of materials and fabrication techniques would provide safety against leaks caused by corrosion.

The concept of DID, as applied to organizational, operational or design related safety activities, ensures that they are subject to functionally redundant provisions, so that if a failure were to occur, it would be detected and compensated for or corrected by appropriate measures. Application of DID in the design of a plant provides a series of levels of defense (inherent features, equipment and procedures) aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails. This strategy has been proven to be effective in compensating for human and equipment failures, both potential and actual.

There is no unique way to implement DID (i.e. no unique technical solution to meet the safety objectives). For instance, several successive physical barriers are put in place for the confinement of radioactive material. Their specific design may vary depending on the radioactivity of the material and on the possible deviations from normal operation that could result in the failure of some barriers. So, the number and type of barriers confining the radioactive/chemically hazardous material is dependent on the adopted technology. Defense-in-depth has a structure of five levels. Should one level fail, the subsequent level comes into play. Table 2.3, summarizes the objectives of each one of
the five levels and the corresponding means of achieving effectiveness of each level (Ref.2.8).

The general objective of DID is to ensure that a failure, whether equipment failure or human failure, at one level of defense, and even combinations of failures at more than one level of defense, would not propagate to defeat DID at subsequent levels. The independence of different levels of defense, i.e. the independence of the features implemented to fulfill the requested functions at different levels, is a key element in meeting this objective. The logic flow of DID at different levels is shown in Fig.2.1. At each level, the DID approach needs to be carefully studied, evaluated and strengthened. It is possible to take up analogous assessments based on well established approaches that have evolved for nuclear reactor systems and make the approach suitable to specific nuclear fuel cycle facility.

<table>
<thead>
<tr>
<th>Levels of defense</th>
<th>Objective</th>
<th>Essential means</th>
</tr>
</thead>
<tbody>
<tr>
<td>Level 1</td>
<td>Prevention of abnormal operation and failures</td>
<td>Conservative design and high quality in construction and operation</td>
</tr>
<tr>
<td>Level 2</td>
<td>Control of abnormal operation and detection of failures</td>
<td>Control, limiting and protection systems and other surveillance features</td>
</tr>
<tr>
<td>Level 3</td>
<td>Control of accidents within the design basis</td>
<td>Engineered safety features and accident procedures</td>
</tr>
<tr>
<td>Level 4</td>
<td>Control of severe plant conditions including prevention of accident progression and mitigation of the consequences of severe accidents</td>
<td>Complementary measures and accident management</td>
</tr>
<tr>
<td>Level 5</td>
<td>Mitigation of radiological consequences of significant releases of radioactive materials</td>
<td>Off-site emergency response</td>
</tr>
</tbody>
</table>

Defense-in-depth should be applied to NFCFs taking into account the following features of the fuel cycle facilities:
The energy potentially released in a criticality accident in a fuel cycle facility is relatively small. However generalization is difficult as there is several fuel fabrication or reprocessing options for the same or different type of fuels.

Fig 2.1. Logic Flow Diagram of Defense-in-depth
• The power density in a fuel cycle facility is typically two to three orders of magnitudes less in comparison to a reactor core.

• In the reprocessing facility, irradiated fuel pins are mechanically cut (chopped) into small lengths, suitable for dissolution and the resultant solution is further subjected to chemical processes. This makes it possible for larger releases of radioactivity to environment on a routine basis as compared to reactors.

• The likelihood of 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 the spent nuclear fuel.

A few examples directed to enhance DID are as follows:
  • Balanced design options and configurations
  • Increased emphasis on inherent safety characteristics
  • Multiple and redundant control and instrumentation systems

2.5 Postulated initiating events for use in Safety Analysis

It is recognized that for evolving indicators and acceptance criteria for INPRO basic principles and user requirements, the first step is to identify the initiating events that may lead to an unsafe situation. These events could be categorized into two classes- external postulated events and the internal postulated events.

The external events could be further listed into two sub-categories- natural events and man-made events.

Natural events:

These include natural calamities such as earthquakes, inundation in the flooding, tsunamis, extreme weather conditions such as excessive snowfall, avalanches, tornadoes/storms/cyclones, lightning and extreme temperatures (high or low). Precaution against these is required at the selection of site as well as in the design and construction of civil works. The plant and machinery should also be protected as per the recommended practices as required by statutory and risk-managing organizations.
**Man-made events**

These events include the damages/risks caused by man-made factors like potential loss of power 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 neighborhood/sky/space, accidental or willful (terrorist) aircraft crash and public unrest including violent strikes and sabotage.

**Internal events**

The internal events proving to be potential threats include electricity related malfunctioning, inadequate original design, modifications of equipment, processes or procedures and I&C, fires originating from the process malfunctions and operational mistakes and finally due to failures such as flooding inside the cell due to overflows, plugging or blocking and loss of ventilation.

Typical initiating events for various nuclear fuel cycle facilities are discussed in Ref.1.7 and 2.9.

It is necessary to emphasize that the safety requirements adopted for a particular nuclear fuel cycle facility should take into account the hazard potential and the probability of occurrence of a particular event, and thus should result in a “graded” approach (Ref.1.3) to ensure that the design and operation philosophies are commensurate with the hazards. The most significant hazards in NFCFs are discussed in the following sections.

2.6 **Criticality**

Criticality safety is one of the dominant safety issues for the fuel cycle facilities. These facilities employ a great diversity of technologies and processes, thus the materials of interest to nuclear safety are more distributed throughout these nuclear fuel cycle facilities. They may be used not only in a bulk form (fuel pellets, fuel elements, fuel rods, fuel assemblies, and so on), but in the distributed and mobile forms as well (different kinds of solutions, slurries, gases, powders, and so on). As a result the fissile materials may accumulate in some parts of the equipment and may also escape from the facility as
a result of equipment leakage. The distribution and transfer of potentially critical nuclear material requires operator attention to account for this material throughout the installation and thus ensure that nuclear criticality safety is maintained and to prevent the potentially lethal effects of gamma and neutron radiation doses to workers and the subsequent release of fission products from an inadvertent nuclear criticality.

Fuel cycle facilities 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 (2) those where the 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 critical parameters and controls are followed. Designs of such facilities need to ensure sub-criticality in all areas, first by engineering design, utilizing where possible ‘criticality safe designed equipment’. Similarly for the operation of these facilities, critical parameters and controls have to be maintained.

A review of some criticality accidents that occurred during nuclear fuel process operation is provided in (Ref. 2.10). The criticality accident at Tokai Mura was the highest level event in the International Nuclear Event Scale reported since 1991. Of the nearly 60 criticality accidents which have occurred since 1945, about a third occurred at nuclear fuel cycle facilities. 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 calculations. The main cause of criticality accidents appears to be the failure to identify the range of possible accident scenarios, particularly those involving potential human error.

2.7 Chemical hazards

Fuel cycle facilities may also pose hazards to workers and members of the public from releases of chemically toxic and corrosive materials. Major steps in the nuclear fuel
cycle consist of chemical processing of fissile materials, which, if not properly managed, may lead to the inadvertent release of radioactive substances.

Chemical hazards differ considerably from facility to facility. The production of uranium hexafluoride (UF6) involves the use of significant quantities of hydrogen fluoride, which is both a powerful reducing agent and is chemo-toxic. This poses a significant hazard to workers, although the hydrogen fluoride is not in itself a radioactive material. Other examples include the use of strong chemical acids to dissolve uranium and other materials and to remove, in some cases, the fuel cladding. These acids are also used to chemically dissolve the spent fuel during reactor fuel element reprocessing, enabling the separation of the plutonium and uranium from the residual fission products. In addition, the residual fission products, which comprise approximately 99% of the total radioactivity and toxicity in the spent fuel, pose a significant radiological hazard in what is typically complex chemical slurry. During the solvent extraction processes, strong acids and organic solvents are used to remove the plutonium and uranium from the slurries. These processes can generate toxic chemical by-products that must be sampled, monitored and controlled. Other chemicals encountered at fuel cycle facilities in significant quantities include chemicals such as ammonia, nitric acid, sulphuric acid, phosphoric acid and hydrazine. It is important to recognize that unplanned releases of the 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 facilities worldwide. The chemical toxicity hazards associated with UF6 processing were evident in two accidents in 1986 in the USA and Germany (Ref.1.2).

2.8 Fire and explosion hazards

Many fuel cycle facilities use flammable, combustible and explosive material in their process operations, such as a tributyl 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 fault
operation when unexpected chemical reactions take place.

Fire and explosion hazards have been recorded at fuel cycle facilities. In 1990, for
example, there was an ammonium-nitrate reaction in an off-gas scrubber at a low
enriched uranium (LEU) scrap recovery plant in Germany which injured two workers and
destroyed the scrubber (Ref.1.2). Fire is an especially significant accident scenario
because it can be both an initiating event for the accident sequence and can also disrupt
safety systems. 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 is the fire and explosion at the Tokaimura
reprocessing plant in Japan in March 1997, which contaminated 37 workers with
radioactive material.

The design of the facilities should provide for minimum inventories of
combustible materials and should ensure adequate control of thermal processes and
ignition sources to reduce the potential for fire and explosions. For example, extreme care
should be taken to prevent the accumulation of radiolytic hydrogen, which is generated in
high activity waste tanks in fuel reprocessing plants. In addition, fire can become a
motive force for significant releases of radioactive and toxic material from the facilities.
Consequently, fire detection, suppression, and mitigation controls are usually required.

A fuel cycle facility design and operation should consider the radiological and
other consequences from fires and explosions. Suitable safety controls should be
instituted to protect against the potential consequences of fire and explosive hazards.
These safety controls should be designed to provide the requisite protection during
normal operations, anticipated operational occurrences and credible accidents at a
facility. Similar to the chemical hazards, fires and explosions which might adversely
affect any nuclear safety measures should be given adequate consideration.
2.9 Radiation hazards

Radiation safety is an important consideration at nuclear fuel cycle facilities. Special attention is warranted, when developing and using standards and establishing operational practices, to ensure worker safety in the operational process, which may include the open handling and transfer of nuclear material in routine processing. Although external exposures 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 into the facilities and through monitored and unmonitored pathways can result in significant exposures to workers, particularly from long lived radiotoxic isotopes. Some facilities, such as MOX fuel fabrication and reprocessing facilities require shielding design, containment, ventilation and maintenance measures to reduce potential exposures to workers.

General principles, whose effective application will ensure appropriate protection and safety in any situation which involves or might involve exposure to radiation, are defined in the IAEA Safety Fundamentals on ‘Radiation Protection and the Safety of Radiation Sources’ (Ref.2.11). Based on these principles and objectives, requirements with respect to radiation safety are established in the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (Ref.2.12), which is applicable to all type of nuclear installations. However, fuel cycle facilities pose a higher risk to facility operators than to the public and this fact should be given proper consideration when establishing relevant safety criteria and developing safety guides.

2.10 Decommissioning of Nuclear Fuel Cycle Facilities:

The safety aspects of decommissioning of nuclear fuel cycle facilities deserve as much attention as the safety aspects of operation of the facilities. The decommissioning of the fuel cycle facility has to be factored in to the design of the facility, and a clear plan for decommissioning should be available even at the time of commissioning of the
facility. The safety aspects of decommissioning have been dealt with by the IAEA safety guide WS-G-2.4, 2001 (Ref.2.13). An IAEA draft standard DS 333 covering the safety requirements of decommissioning is now available (Ref.2.14). The safety aspects related to release of the fuel cycle facilities from regulatory control upon termination of practices is covered by draft standard DS 332 (Ref.2.15).

In all phases of decommissioning, workers, the public and the environment shall be properly protected from both radiological and non-radiological hazards resulting from the decommissioning activities. Special safety issues that should be considered in the decommissioning of nuclear fuel cycle facilities may include:
(a) The presence and nature of all types of radioactive contamination and, in particular, alpha contamination;
(b) The significantly higher radiation levels in some facilities, necessitating the consideration of remote handling;
(c) The increased hazards associated with the possible in-growth of radionuclides (such as americium produced due to decay of plutonium), during storage of the spent fuel;
(d) The potential in some facilities for criticality hazards associated with the possible accumulation of fissile material during activities for decontamination or dismantling;
(e) The complexity of strategies for waste management owing to the diversity of waste streams;
(f) The hazards, such as fire or explosion, associated with the original chemical processing activities.

The specific characteristics of each type of nuclear fuel cycle facility will strongly influence the selection of the decommissioning option. A safety assessment should form an integral part of the decommissioning plan. Non-radiological as well as radiological hazards associated with the decommissioning activities should be identified and evaluated in the safety assessment and should be factored in to the design of the facility. The extent and detail of the safety assessment shall be commensurate with the complexity and the hazard associated with the facility or operation.
3.0 INPRO METHODOLOGY FOR ASSESSMENT OF FUEL CYCLE FACILITIES

3.1 Basic Principles and User Requirements

Towards assessment of innovative fuel cycle facilities, a set of Basic Principles and User Requirements are defined. Initially five basic principles were defined in IAEA TECDOC-1362. Based on discussions, these have been modified and only four basic principles are defined in IAEA TECDOC-1434. The holistic life-cycle analysis encompassing the effect on people and on the environment of the entire integrated fuel cycle is also an integral part of TECDOC-1434. Basic Principles and User Requirements defined in this chapter and discussed in the subsequent chapters with regard to nuclear fuel cycle facilities have been taken from IAEA TECDOC-1434. Criteria consisting of indicators and acceptance limits have also been defined for each of the User Requirements (Ref.3.1 and 3.2).

The basic principles shall be met by all INS. However it is recognized that the user requirements may not be applied in their entirety, because the range of innovative fuel cycle installations is large and their safety characteristics varied; so it is not practical that all user requirements and criteria should apply to all types. It is also assumed that requirements and practices set out in IAEA Safety Standards and Guides will be followed where applicable.

3.1.1 Basic principles

Installations of an innovative nuclear fuel cycle facility should:

1. Incorporate enhanced defense-in-depth as a part of their fundamental safety approach and ensure that the levels of protection in defense-in-depth shall be more independent from each other than in existing installations.
2. Excel in safety and reliability by incorporating into their designs, when appropriate, increased emphasis on inherently safe characteristics and passive systems as a part of their fundamental safety approach.
3. Ensure that the risk from radiation exposures to workers, the public and the environment during construction/commissioning, operation, and decommissioning, shall be comparable to that of other industrial facilities used for similar purposes. 

Further, the development of innovative nuclear fuel cycle facility should:

4. Include associated RD&D work to bring the knowledge of plant characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for existing plants.

This set of basic principles will apply to any type of innovative design. It should foster an appropriate level of safety that can be communicated to and be accepted by users.

3.1.2 User requirements and criteria for all basic principles

In the following, for each basic principle defined above, the corresponding user requirements are set out. To confirm that a user requirement is met, it is necessary to define the criteria for each aspect of the fuel cycle facilities. Corresponding indicators and acceptance limits are described for these criteria, to facilitate assessment. In chapter 4, user requirements, INPRO criteria, corresponding indicators and acceptance limits are described for the fuel cycle activities mining and milling, enrichment, fuel fabrication and fuel reprocessing.

The user requirements for Basic Principle 1 are directed towards strengthening of the defense-in-depth strategy so that for future nuclear installations – even in the case of severe accidents – evacuation measures outside the plant site are not needed. The concept of defense in depth and typical examples of how it can be achieved has been discussed in Chapter 2. The user requirements related to Basic Principle 2 are focused on reducing and eliminating hazards by various means such as inherent safety features in future designs. These user requirements are complementary to those that are aimed at strengthening defense-in-depth. One example of such an approach is the adoption of designs of process vessels in a reprocessing plant with ever safe geometry with respect to criticality instead of criticality controls which are based more on administrative measures. The user requirements related to Basic Principle 3 are focused on achieving a very low risk as a
result of radiation exposures. The User Requirements related to Basic Principle 4 focus on the RD&D that needs to be performed prior to the deployment of INS. Safety analyses should cover all modes of operation of the installation to obtain a complete assessment of compliance with DID. In the case of simple installations related to the fuel cycle (e.g. mining facilities), a deterministic analysis is recommended, if DID is demonstrated. But for the other installations, probabilistic analyses should be included, for example, with respect to release of radioactivity in the event of ventilation failure.

![Diagram of safety characteristics comparison between Reactor and Reprocessing plants](image)

**Fig.3.1. Conceptual Comparison of Safety Characteristics between Reprocessing Plants and Reactor Plants**

It is obvious that in well designed facilities, the safety related events that have a high hazard potential will also have low frequency of occurrence and vice versa. For example, a comparison of the relationship between the exposure and the frequency for safety related events in a reactor and a reprocessing facility is given in Fig.3.1, adopted from Ref.3.3. For the purpose of safety analysis, the events can be broadly classified based on the frequency of occurrence into three categories: very unlikely, unlikely and not unlikely. In general, these categories are taken to correspond to frequencies of $<10^{-6}$/year (Category A), $10^{-6}$-year to $10^{-4}$/year (Category B) and $10^{-4}$-$10^{-2}$ / year (Category C). In Ref.3.4, for example, a similar event categorization has been adopted for probabilistic safety assessment of fuel fabrication facilities. The most serious events in
fuel cycle facilities such as large scale leakage of radioactivity into environment and criticality can be categorized as Category A events and in this manual, the frequency of occurrence of such events is taken to be $< 10^{-6}$ / year. Events such as fire, loss of cooling water, loss of ventilation, which have higher probability but will have less serious implications on safety, are taken as Category B events and a frequency of $10^{-4}$ is taken as the limit for such events. Events that have minimal consequences such as process malfunctions, temporary loss of power etc., are taken as Category C events and a frequency of $10^{-2}$ is taken as the limit for such events.

Probabilistic safety criteria for a large radioactive release have been discussed in Ref.3.5 and the objective for future plants is given as $10^{-6}$/ reactor-year for nuclear power plants. Same value is adopted for the fuel cycle facilities. A good discussion on the risk informed decision making, approaches to reduction of risks at NFCFs and a good discussion of the probability of various types of accidents averaged over the last 45 years are given in Ref.2.7.

Grace period available for human intervention in the case of a design basis event before it escalates into an accident involving large scale release of radioactivity depends upon the nature of fuel cycle facility, the type of incident, the system parameters at the time of the incident etc. However, based on the available international experience, a grace period of 10-30 minutes is given as the typical decision interval for the operator in the event of a design basis event. (Ref.3.5). A similar approach has been adapted for the fuel cycle facilities, other than mining and milling activities.

It is recognized that for innovative fuel cycle facilities, integration of development in the fuel cycle is emphasized, to ensure that releases of radioactive material from all components of the system are considered and optimized for a given concept. Ideally, the impact of the whole fuel cycle (including the associated waste disposal facilities and decommissioning activities) should be evaluated. As a consequence, the overall risk associated with INS should be less than that of existing plants.
4. NUCLEAR FUEL CYCLE FACILITIES

4.1. Mining/milling Facilities:

4.1.1. Introduction:

The safety issues related to the entire cycle of ore processing extend from the mining activity to the decommissioning of the mines. Though the stages are different, the hazard potential remains more or less similar. The siting of uranium mining and the related components should consider the disposal of waste and the final decommissioning without unduly affecting the environment (displacement of population, change in the land use, flora and fauna etc). Having selected a site through a detailed survey, an extensive base line survey of the area is undertaken prior to actual operation. The study of the 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 the mining and milling activities and can be used for safety and environmental assessment.

Operation of the uranium/ thorium processing industry generates by-products and wastes that are chemically and radiologically hazardous and needs to be safely stored and ultimately disposed off into the environment. The limit of radiation exposures to general public from the normal operations of the ore processing industry and also due to anticipated events should be less than 1 mSv in a year. The disposal of radioactive waste and the consequent dose from all the pathways of exposure to the general public residing near the industry shall not exceed the derived quantities that are based on the limits prescribed. The safety measures for the processing operations, transport, the storage and disposal of the tailings and the ultimate decommissioning of the mines shall result in exposures to the public well below the regulatory limits.
4.1.2. Mining and Milling of uranium

Uranium processing technology, right from ore exploration to the ultimate recovery of the product, has developed rapidly over past several decades. Significant innovations in mining, milling and leaching processes as well as in the use of equipment have also been achieved. Open pit mining, underground mining and in-situ leaching are the processes that are adopted world over for extraction of uranium. Open pit mining is preferable since the productivity is higher, ore recovery is better due to minimum dilution, dewatering of the open pit is easier and mining conditions are safer. The disadvantages include environmental contamination, large-scale excavation resulting in huge overburden of soil, land degradation etc. Underground mining is preferable when the ore is available at greater depths (> 200m). The in-situ leaching process does not introduce surface disturbance, eliminates crushing, milling, grinding and leaching steps in the process, provides safer working conditions, does not generate large solid radioactive waste and requires less manpower. However this process is recommended only for certain type of ores and detailed precautions are required to be taken to prevent the contamination reaching water sources.

4.1.3. Operations carried out in underground Uranium mines: Very often “cut and fill method” and “open stope” methods of underground uranium extractions are used. Following identification of the area with rich uranium, a shaft is sunk in the vicinity of the ore veins and cross cuts are driven horizontally to the veins at various levels at an interval of 100-150 meters. Tunnels known as “drifts” are driven along the ore veins 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. 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 are required for mining.

Mining consists of the following steps: blasting by remote control process, identification and delineation of ore body using radiation detectors, drilling, loose
dressing and support, mucking, tramming and stowing. Entry of miners after blasting should be restricted till the dust and fumes disperse with ventilation air to the permissible level.

Milling consists of the following steps: Transferring the ore by conveyor belt, crushing, wet grinding, leaching 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. Finally, precipitation is carried out to purify the uranium from iron and other metals and to recover uranium as magnesium diuranate (MDU).

4.1.4. Uranium Tailings neutralisation & impounding: The barren liquor from ion exchange generates acidic liquid waste 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 solid containing waste is secondary filter cake slurry. It contains the undissolved uranium, radium and other radionuclides. These are mixed with the lime stone for neutralisation and sent to a hydro cyclone, where sand and slime get separated. The sand goes to the mine for backfilling and slime to tailings pond.

4.1.5. The Uranium effluent treatment plant: The decanted liquid from the tailings pond is sent to the Effluent Treatment Plant for chemical treatment and activity removal so as to discharge to the public domain as per the norms specified by the regulatory authorities.

4.1.6. In- situ leaching U mines:

_In situ leach_ (ISL) mining is defined as the leaching of uranium from a host sandstone by chemical solutions and the recovery of uranium at the surface. 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.
4.1.7. Safety Issues in Uranium Mining/Milling:

A large number of reports discuss the safety and environmental issues associated with uranium mining and milling activities (Ref. 4.1-4.8). In contrast to underground and open pit mining, in the case of ISL, 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 (Ref.4.8).

While the occupational and environmental hazards of mining and milling of uranium are not very different from the other mineral extraction processes, the additional risks involved in uranium mining include exposure to radioactive materials. The major safety issues in the entire process – mining, milling, leaching, product recovery, storage and disposal of tailings- are dust, noise, 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.

Mining and milling operations involving uranium have a potential for generation of dust, which has radioactivity in varying quantities. The hazard potential is higher if the operations are dry and dusty rather than wet operations. Radiation exposure to occupational workers is through both external and internal modes of exposure. The daughter products of natural uranium are in equilibrium with uranium and some of the daughter products such as $^{214}$Bi and $^{214}$Pb are strong gamma emitters, which pose external exposure hazard. (83% of the gamma energy is from $^{214}$Bi and 12% is from $^{214}$Pb). A dose rate of 5 $\mu$Gy/h can be measured from a 0.1% uranium ore body and the annual exposures could be about 50 mSv with an ore grade of about 0.5% (Ref. 4.9). Similarly, the gaseous daughter product Radon ($^{222}$Rn) is another major source of radiation exposure to the occupational workers. It would be similar in the milling and product extraction areas too. Hence safety in the design and operation of the process is of paramount importance. Monitoring of workers as per national regulatory requirement is also essential. The limit for occupational workers shall be as per national regulatory
requirement (ICRP limits are 100 mSv for a defined calendar period of 5 years, with an average dose of 20 mSv in any single year). Innovative and proven techniques such as increased automation, improved O&M techniques and effective engineered safety features are required to keep the exposures ALARA.

4.1.8 Thorium Mining and Milling

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

4.1.9. Beach Sands Mining and Separation of Thorium

The beach sand minerals 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. Mining, separation and processing of these minerals involve operation of floating dredge, application of high voltage and high magnetic fields, operation of dryers, operation of material handling equipment like belt conveyors, bucket elevators, mixer-settlers involving flammable materials etc.

Monazite and zircon are subjected to further processing to obtain thorium oxalate/thorium nitrate, Ammonium diuranate (ADU) and zircon frit powder respectively. There are standard chemical processes involving digestion, solvent extraction, precipitation, filtration etc. For example, processing of monazite is carried out by digestion of finely ground monazite with caustic soda which results in three components namely byproduct trisodium phosphate, mixed hydroxides of rare earths, thorium and uranium as well as unreacted monazite. After the majority of rare earths is first separated from the mixed hydroxide, the mixed hydroxides of Th, U & residual rare earth are extracted through acid leaching followed by solvent extraction to ultimately produce thorium oxalate and crude uranyl chloride solution besides recycling the residual rare earths. The crude uranium chloride solution is subsequently refined to produce nuclear grade U₃O₈. While processing of every ton of monazite, approximately 0.2 ton of thorium oxalate, 0.08 ton of insolubles and 0.06 ton of Pb-Ba cakes are produced.
(Ref.4.10). The solid wastes are buried in underground RCC trenches. Liquid effluent is treated in the effluent treatment plant and then discharged after monitoring.

4.1.10. 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 underground uranium mining. Inhalation hazards arise mainly from dust produced during the physical separation of the mineral constituents of placers or from thoron 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 should include an assessment of thorium and its long-lived daughter products in the working atmosphere, in addition to thoron. The dose delivered to the lung 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 the 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, 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).

4.1.11. Surface Contamination with thorium

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 in 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 very small amount of uranium, and although the typical ratio of thorium to uranium is 25:1, $^{222}\text{Rn}$ and radon daughters may occur in significant air concentrations along with $^{220}\text{Rn}$ and thorium in the initial chemical treatment areas of the plant.

Since the hazard from thoron is predominantly attributable to $^{212}\text{Pb}$, which occurs with thoron in all practical situations, it is permissible to apply the value for $^{212}\text{Pb}$ as the standard of control for both radionuclides. Because of the very short half-lives of thoron $^{220}\text{Rn}(55.3\text{s})$ and $^{216}\text{Po}(0.15\text{s})$ compared to $^{212}\text{Pb}(10.6\text{h})$, dilution ventilation is relatively ineffective for these radionuclides but it can reduce the concentration of $^{212}\text{Pb}$ by a large factor. Thus, in some atmospheres, the concentration of thoron may exceed that of $^{212}\text{Pb}$ 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 $^{212}\text{Pb}$ 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 $^{232}\text{Th}$ and $^{228}\text{Th}$ from the thorium series, and $^{234}\text{Th}$, $^{230}\text{Th}$, $^{231}\text{Th}$ and $^{227}\text{Th}$ from the uranium series) and the non-thorium fraction (consisting of $^{228}\text{Ra}$, $^{224}\text{Ra}$ and other daughters from the thorium series, and $^{226}\text{Ra}$ with daughters from the uranium series). The processing of monazite to extract thorium gives rise to generation of solid, liquid and gaseous wastes. 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.

4.1.12. Application of INPRO Methodology to uranium and thorium mining/milling

**Basic Principle 1:** Installations of an INS shall incorporate enhanced defense-in-depth as a part of their fundamental safety approach and the levels of protection in defense-in-depth shall be more independent from each other than in current installations.

**User Requirement (UR) 1.1:** *Installations of an INS should be more robust relative to existing designs regarding system and component failures as well as operation.*

**Indicator (IN) 1.1.1:** Robustness of design

Flooding of the mines and consequent release of radioactive material to environment is one of the design basis events for the mines. Thus, if there are no dams upstream and no catchment areas, the INS would be superior from safety point of view. In the case of underground mining, prior testing and analysis of the rock beds, analysis of rock mechanics and incorporation of these data in design would result in a more robust design. The site must be checked for design basis floods. The site, tailing pond and dam must be checked for design basis seismic loads. The tailing pond should be lined appropriately to prevent leakage of radioactive material to environment. The ventilation system and its design should be more robust in terms of reliability and should ensure to keep the radon levels within safety margin. The design analysis should demonstrate compliance of the limits of radon levels and ensure that radon concentration results in exposures in a single year, which is less than 4 Working Level Month- WLM (Ref.4.11).

**IN 1.1.2:** High Quality of Operation:

In the context of mining and milling facilities, operational methodologies that ensure that the dust generated and the consequent spread of contamination is minimised constitute an example of high quality of operation. This can be achieved using higher levels of automation and more modern milling techniques. Automation and tele-operation using an integrated high speed network would result in minimum human interference and more robust and safe operation.
**IN 1.1.3: Grace Period**

As the migration of radionuclides takes a long time (a few months to several years) to reach the public domain, a grace period of two months, is recommended as the acceptance limit for the INS. However, it is recognized that this period is quite arbitrary, and no international standards or codes are available to arrive at a realistic period. Further, it is recognized that the period would depend upon the type of soil and ground conditions (water table etc) at the given site of the pond, and may vary from site to site. Hence the target value for each site must be arrived at, after a careful consideration of site conditions.

**IN 1.1.4: Capability to inspect**

Provisions should exist for inspection of bore wells and nearby public water sources to ensure that there is no migration of radioactive materials into public domain. Instrumentation systems of the mines should incorporate monitoring of oxygen and toxic gases and healthiness of ventilation systems.

**IN 1.1.5: Expected frequency of failures and disturbances**

Frequency of failure of electric power and consequently ventilation system should be assessed and it should be ensured that the frequency of failure of ventilation system is less than $10^{-2}$/year and should be substantiated by PSA analyses.

**IN 1.1.6: Inertia to cope with transients**

The ventilation system of the mining and milling facilities should be able to operate safely for brief periods of electrical power failure. Adequate secondary power sources should be available to address brief power failures.

**UR1.2- Provision for detection of deviation from normal operation:** *Installations of an INS should detect and intercept deviations from normal operational states in order to prevent anticipated operational occurrences from escalating to accident conditions.*
**IN 1.2.1:** Capability of control & instrumentation system/ inherent characteristics to detect/intercept/compensate deviations:

As indicated in UR 1.1, prevention of the leakage of the radionuclide from the tailing ponds to the environment constitutes an important element of safety. In the event that the barriers are breached and radionuclides find their way into ground water, it is important that the leakage is detected at an early stage and actions initiated to arrest further leakage. This necessitates a regular system of monitoring of the radioactivity in nearby water bodies and bore wells. Availability of such a monitoring system is thus an acceptance criterion.

**IN 1.2.2:** Another important deviation from normal state is the maloperation of ventilation system leading to build-up of radon in the atmosphere inside the mine. To take timely corrective action, it is necessary to have continuous monitoring of radon levels in the atmosphere, and associated alarm systems. Availability of such a monitoring system is thus an acceptance criterion.

**UR1.3:** Frequency of occurrence of accidents and safety features: The frequency of occurrence of accidents should be reduced, consistent with the overall safety objectives. If an accident occurs, engineered safety features should be able to restore an installation of an INS to a controlled state, and subsequently to a safe shutdown state and ensure the confinement of radioactive material. Reliance on human intervention should be minimal and should only be required after some grace period.

**IN 1.3.1:** Calculated frequency of occurrence of design basis events:

The design basis events in a mining and milling facility with regard to a radiological hazard are a) a large leak of the tailing bond and b) a failure of the ventilation system. A large leak from the tailing pond either due to flooding or breach in the dam or migration through soil should have low probability of $10^{-6}$. The frequency of ventilation system failure should be arrived at by a probabilistic assessment, and should also be low.

**IN 1.3.2:** Grace period until human intervention is necessary:

The grace time required for human intervention would vary depending upon site characteristics, as discussed in an earlier section. However, the design analysis of the
specific site should establish a conservative estimate of the time period, so that all safety actions are initiated well within the time period. Availability of a clear estimate of the grace period is thus an acceptance criterion.

**IN 1.3.3:** Reliability of engineered safety features:

For ensuring that the radioactive material does not leak from the tailing ponds to public domain, the design of the tailing pond should incorporate adequate number of redundant barriers against migration. Safety analysis should show that the number of barriers provided is such that even in the event of failure of one of the barriers, a grace period of two months should be available, as discussed earlier.

**UR1.4:** The frequency of a major release of radioactivity into the containment/confine of an INS due to internal events should be reduced. Should a release occur, the consequences should be mitigated.

A major release of radioactivity into containment/confine is not conceivable in a mining and milling facility.

**UR1.5:** A major release of radioactivity from an installation of an INS should be prevented for all practical purposes, so that INS installations would not need relocation or evacuation measures outside the plant site, apart from those generic emergency measures developed for any industrial facility used for similar purpose.

A major release of radioactivity into environment is not conceivable in a mining and milling facility.

**UR1.6:** An assessment should be performed for an INS to demonstrate that the different levels of defense-in-depth are met and are more independent from each other than for existing systems.

**IN 1.6.1:** Independence of different levels of DID

In the case of mining and milling facilities, the major defense in depth parameters are the provision of multiple barriers for radioactive leakage and diverse ventilation systems to ensure low radon levels in the working atmosphere.
**UR1.7:** Safe operation of installations of an INS should be supported by an improved Human Machine Interface resulting from systematic application of human factors requirements to the design, construction, operation and decommissioning.

**IN 1.7.1:** Human Factors:
Frequent training of the staff and safety awareness, ensures a good safety culture (discussed in more detail in the manual on “infrastructure”). A multi-tiered system for design safety assessment and operational safety aspects, which constantly supervises the facility and recommends implementation of safety procedures, takes into account the human factors.

**Basic principle 2:** Installations of an INS shall excel in safety and reliability by incorporating into their designs, when appropriate increased emphasis on inherently safe characteristics and passive systems as a part of their fundamental safety approach

**UR 2.1:** INS should strive for elimination of minimization of some hazards relative to existing plants by incorporating inherently safe characteristics and/or passive systems, when appropriate.

Inherently safe systems cannot be visualised for mining/milling facilities

**Basic Principle 3:** Installations of an Innovative INS shall ensure that the risk from radiation exposures to workers, the public and the environment during construction/commissioning, operation, and decommissioning, are comparable to the risk from other industrial facilities used for similar purposes.

**UR 3.1:** INS installations should ensure an efficient implementation of the concept of optimization of radiation protection through the use of automation, remote maintenance and operational experience from existing designs.

**IN 3.1.1 & IN 3.1.2:** Occupational dose values should be minimized ALARP and shown to be meeting the regulatory requirements and also be lower than the values for existing plants.
UR 3.2: Dose to an individual member of the public from an individual INS installation during normal operation should reflect an efficient implementation of the concept of optimization and for increased flexibility in siting may be reduced below levels from existing facilities.

IN 3.2.1: Public dose values should be minimised ALARP and shown to be meeting the regulatory requirements and also be lower than the values for existing plants.

The radiation exposure resulting from mining and milling operations has been discussed in section 4.1.7 and 4.1.10 for uranium and thorium respectively. It is necessary that dose received by occupational workers is under 100 mSv for a defined period of 5 years (20 mSv per year) and for public, under 1 mSv per year through all routes (air, water and land), in line with ICRP recommendations.

Transportation of ore from underground mines through conveyors and remote handling ensure an efficient implementation of the ALARA principle. Proper planning of O&M schedule may reduce occupational exposure to individuals. Inventory of activity released to the environment should be such that the estimated dose from the released activity is significantly less than the regulatory limit of the national body. Use of better technologies for drilling operations, such as IT enabled drill jumbos, would also result in less occupational exposures. Backfilling of worked out areas would also reduce the radiation exposure.

Basic Principle 4: The development of INS shall include associated Research, Development and Demonstration work to bring the knowledge of plant characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for existing plants.

UR4.1: The safety basis of INS installations should be confidently established prior to commercial deployment.

IN 4.1.1: Safety concept defined

The safety concept of the mining and milling facilities, including safety of residues, control of effluent releases, long term monitoring etc., should be described comprehensively in the safety report of the facility.
**IN 4.1.2:** Design related safety requirement specified

The design related safety requirements should be clearly specified in the safety report and it should be demonstrated that all necessary facilities and technologies are available to handle design basis events in addition to minor failures and disturbances.

**IN 4.1.3:** Clear process for addressing safety issues

Use of validated models and codes for safety analysis would increase the confidence level in the safety analysis methods used in analysing the accident scenarios and postulated hazards. Such codes are not presently available and need to be developed. However, it is imperative that based on the available knowledge, safety manuals and design safety analysis reports should be prepared and reviewed by experts groups and regulatory authorities. The existence of a clear process of preparation of documents and their systematic and comprehensive review is thus essential for an innovative INS.

**UR 4.2:** Research, Development and Demonstration on the reliability of components and systems including passive systems and inherent safety characteristics should be performed to achieve a thorough understanding of all relevant physical and engineering phenomena required to support the safety assessment

**IN 4.2.1:** RD&D defined and performed and database developed.

The facility safety report should identify areas for RD & D which would further enhance the safety status of the facility. A database of RD & D information should be available to assist in safety assessment. RD & D should be performed on methods such as in-situ leaching which would avoid generation of large quantities of waste.

**IN 4.2.2:** Computer codes or analytical methods developed and validated.

Codes for probabilistic safety assessment of the mining and milling facilities have not been reported. There is considerable scope for development in this area. Innovative facilities should incorporate safety features based on a systematic analysis using necessary codes.
**IN 4.2.3:** Scaling understood and/or full scale tests performed

A well defined RD & D programme is essential to realize continuous improvements in the safety status of the fuel cycle facility. Indicators of a mature RD & D would include reports and other publications; evidence of development of computer codes or analytical methods is another indication of an innovative INS.

The operational philosophy of the mining and milling facilities should take into account the scale of operation. Thus a “graded approach” can be adopted which can strike a balance between safety and economy.

**UR4.3:** *A reduced-scale pilot plant or large-scale demonstration facility should be built for reactors and/or fuel cycle processes, which represent a major departure from existing operating experience.*

**IN 4.3.1:** Degree of novelty of the process:

Novel processes which differ from the existing mining and milling processes should be examined in depth to ensure that the prevailing safety concepts can be applied.

**IN 4.3.2:** Level of adequacy of the pilot facility:

A demonstration facility is not conceivable for mining operations. However, for milling operations, wherever a high degree of novelty is envisaged, it is desirable to have operated a pilot plant at an adequate scale and for sufficient period of time so as to generate confidence in the new process. Where the degree of novelty is low, it may be adequate to have a rationale for the new process based on experience of other existing plants.

**UR4.4:** *For the safety analysis, both deterministic and probabilistic methods should be used, where feasible to ensure that a thorough and sufficient safety assessment is made. As the technology matures, “Best Estimate (plus Uncertainty Analysis)” approaches are useful to determine the real hazard, especially for limiting severe accidents.*

**IN 4.4.1:** Use of a risk informed approach:

A risk informed approach should be adopted in design, construction and operation of mining and milling facilities. In line with the risks involved, the emphasis should be more on long term effects on environment and public.
IN 4.4.2: Uncertainties and sensitivities identified and dealt with

The safety report of the facility should include analysis of safety issues based on a risk informed approach. The uncertainties in all important operational parameters should have been addressed in the design analysis of the facility.

4.2. REFINING/CONVERSION FACILITIES

4.2.1. Uranium Refining and Conversion to Hexafluoride

The end product from the mining and milling stage of the fuel cycle is “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. Various stages in the purification process are dissolution, solvent extraction, concentration and thermal denitration to uranium trioxide. The processes are those of a chemical industry handling a chemi-toxic rather than radioactive material, the toxicity of natural uranium being about the same as that of lead. There are several methods for the production of uranium hexafluoride; a batch process using chlorine trifluoride; direct fluorination; and the modern method using pure uranium trioxide, UO$_3$, which is converted to UF$_4$, uranium tetrafluoride, then to UF$_6$, uranium hexafluoride, prior to enrichment. The reactions:

\[
\begin{align*}
\text{UO}_3 + \text{H}_2 & \rightarrow \text{UO}_2 + \text{HF} \\
\text{UF}_4 + \text{F}_2 & \rightarrow \text{UF}_6 
\end{align*}
\]

are generally carried out using fluidized bed technology. A typical flowchart of uranium refining and conversion is shown in Fig.4.2.1.

4.2.2. Safety issues:

The safety issues related to conversion facilities are dealt with in IAEA draft safety guide DS 344 (Ref.4.12). Generally in a refining / conversion facility, only natural or slightly enriched uranium is processed. The radio toxicity of this is low, and thus the expected off-site radiological consequences following potential accidents are limited. However, it is noted that the radiological consequences of an accidental release of any
reprocessed uranium, when licensed, are likely to be greater and should be taken into account in the safety assessment of refining and conversion facilities.

The existing processes for uranium refining and conversion to hexafluoride give rise to no significant radio-active hazards, and the safety problems associated with the handling of this material are essentially those of a conventional chemical industry dealing with toxic materials. Uranium hexafluoride handling is common to several stages of the fuel cycle, such as enrichment and fuel fabrication.

The conversion to UF₆ of uranium recovered from the reprocessing of spent fuel from power reactors could give rise to an increase in radioactive hazards associated with UF₆ handling. The uranium product will contain small quantities of plutonium, other actinides and fission products. The latter may accumulate in some parts of the process, particularly in the hexafluoride production stage. Assessment and control of these hazards should 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.

4.2.3. Causes of UF₆ Release

Failure of vessels, gaskets, valves, instruments or lines can give either liquid or gaseous UF₆ release. These failures could result from corrosion, mechanical damage, mal-operation of the system, or overheating of all or part of the system.

Four types of cylinder with capacities ranging from 2 to 12 t are in common use for storage and transport of UF₆ which is then in solid form. These cylinders need to be heated up for transfer of UF₆. Though UF₆ is not in itself inflammable, 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 impurities in storage cylinder or process equipment could lead to an exothermic reaction, which might give rise to a UF₆ release. (Ref. 4.13).
Fig. 4.2.1. Flowchart of Uranium Refining and Conversion

- Dissolution
  - U extraction
    - U loaded solvent
  - HNO₃
  - Solvent
  - Pure Uranium nitrate solution
    - Denitration
    - Hydrogen
    - Hydrofluoric acid
    - Fluorine
    - Hydrofluorination
    - Fluorination
    - UF₆
4.2.4. UF₆ Release Hazards

The chemical toxicity of Uranium in soluble form such as UF₆ is more significant than its radio toxicity. At room temperature, at which it is handled and stored, UF₆ 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 UF₆ release arises from the reaction between UF₆ and the moisture which is normally present in the atmosphere producing two toxic substances hydrofluoric acid (HF) and uranyl fluoride (UO₂F₂) according to the equation:

\[
UF₆ + 2H₂O → UO₂F₂ + 4HF.
\]

With gaseous UF₆, this reaction proceeds rapidly liberating some heat and is accompanied by a substantial volume increase at atmospheric pressure. Both UO₂F₂ and HF are toxic. Deposition of UO₂F₂ from the cloud formed following a release 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 UF₆ release is complicated by virtue of the high density levels of some of the products and the chemical reactions which occur. UF₆ leakage should be restricted to less than 0.2 mg/m³ (chemical toxicity limit for Natural ‘U’ and up to 2.5% enrichment).

4.2.5. Accidents with UF₆ handling:

There have been several accidents involving uranium hexafluoride. As early as 1944, in the United States, a weld ruptured on an 8-ft long cylinder containing gaseous
natural UF6 that was being heated by steam. An estimated 400 lb of UF6 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 UF6 accident involving a cylinder rupture occurred at a commercial uranium conversion facility (Sequoyah Fuels Corp., USA) in 1986. The accident occurred when an over-loaded shipping cylinder was reheated to remove an excess of UF6. The cylinder ruptured, releasing a dense cloud of UF6 and its reaction products. This accident resulted in the death of one individual from HF inhalation.

As mentioned above, the main safety issues in the conversion steps relate to hazards in UF6 handling. These are discussed from INPRO viewpoint in subsequent sections on fuel enrichment and fuel fabrication.

4.2.6. Internal exposure: Inhalation of uranium compounds would lead to internal exposure. Depending on solubility, inhalations of different uranium compounds give different doses. Activity and uranium mass (which gives 20 mSv on inhalation) are given in Table 4.2.1.

Table 4.2.1. Activity and Uranium mass (which gives 20 mSv on inhalation)

<table>
<thead>
<tr>
<th>Class (depending on solubility)</th>
<th>Activity (Bq) to get 20mSv from inhalation (subjected to small variations with respect to enrichment but for practical purpose can be considered as independent of enrichment)</th>
<th>Corresponding mass of uranium (mg) to get 20msv from inhalation (different for different enrichment)</th>
</tr>
</thead>
<tbody>
<tr>
<td>S Class – UO2, U3O8, U</td>
<td>3120</td>
<td>For Natural 125 For 3.5% enriched 29</td>
</tr>
<tr>
<td>M Class – UO3, UF4, MDU, ADU</td>
<td>10800</td>
<td>For Natural 435 For 3.5% enriched 99</td>
</tr>
<tr>
<td>F Class – UF6, UO2F2, UO2(NO3)2</td>
<td>31200</td>
<td>For Natural 1248 For 3.5% enriched 286</td>
</tr>
</tbody>
</table>
4.3 URANIUM ENRICHMENT FACILITIES

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

4.3.1 Uranium Enrichment Processes

The uranium enrichment can be performed in several ways: (i) Electromagnetic isotope separation (EMIS), (ii) Thermal diffusion, (iii) Aerodynamic uranium enrichment process, (iv) Chemical exchange isotope separation, (v) Ion-exchange process, (vi) Plasma separation process, (vii) Gaseous diffusion process, (viii) Gas centrifuge process, and (ix) Laser isotope separation. Of these, Gaseous diffusion process and Gas centrifuge process are used in industries. For details on various processes see Ref.4.14-4.19.

4.3.1.1 Gaseous diffusion process: Uranium arrives at the plant in the form of solid UF₆. It is vaporized and advantage is taken of the difference in the molar masses of the three isotopes to separate them selectively by passage of UF₆ through a porous wall, a “barrier.” The lightest isotopes, ²³⁵U and ²³⁴U pass more easily than ²³⁸U. 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 UF₆ 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 joined into a whole of ten to twenty units that constitute a group. Several groups constitute the cascade. UF₆ is introduced into the center of the cascade. The UF₆ that has been enriched in uranium 235 is withdrawn at one end and the depleted UF₆ at the other. One of the disadvantages in gaseous diffusion process is the heavy use of electricity.
4.3.1.2. Gas centrifuge process:

In the gas centrifuge uranium-enrichment process, gaseous UF₆ 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 UF₆ molecules to tend to move closer to the wall than the lighter 235 UF₆ molecules, thus partially separating the uranium isotopes. This separation is increased by a relatively slow axial countercurrent flow of gas within the centrifuge that concentrates enriched gas at one end and 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.4.3.1.

The main subsystems of the centrifuge are (1) rotor and end caps; (2) top and bottom bearing/suspension system; (3) electric motor and power supply (frequency changer); (4) center post, scoops and baffles; (5) vacuum system; and (6) casing. Because of the corrosive nature of UF₆, all components that come in direct contact with UF₆ must 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 must be capable of damping some amount of vibration.

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 must 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 must 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 required enrichment. Once started, a modern centrifuge runs for more than 10 years with no maintenance.

Fig.4.3.1. Gas Centrifuge Process
4.3.1.3 Enrichment of UF₆ resulting from uranium recovered after reprocessing (RepU):
The enrichment of ²³⁵U in repU fuel has to be higher than in standard fuel, in order to compensate for the decrease in reactivity due to the presence of ²³⁴U and ²³⁶U. Use of repU has a major impact on the choice of an enrichment process. If fuel is to be fabricated using only reprocessed uranium, gas centrifuge process is better suited than 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 ²³⁴U and ²³⁶U than those of a gaseous diffusion plant.

4.3.1.4. Atomic vapor laser isotope separation process: This is based on the difference in the ionization energies of the isotopes of a given element. A laser beam illuminates vapor of uranium metal or uranium metal alloy and selectively ionizes the atoms of ²³⁵U, removing an electron from each and leaving them with a positive charge. ²³⁵U is then collected on negatively charged plates. Neutral ²³⁸U, condenses on collectors on the roof of the separator.

4.3.2 Safety issues in Enrichment Facility

As mentioned elsewhere in the manual, the fuel cycle operations present a great variety which is an important factor to be considered in evolving assessment methodologies. This is especially true for enrichment facilities. The methodology for assessment and innovations that can be introduced in the enrichment process depend on the choice of the process itself. In the present manual, the application of INPRO methodology to gas centrifuge based enrichment process has been explored. While an attempt has been made to obtain generic assessment parameters, it is quite possible that a different set of parameters has to be evolved for other enrichment processes.

The safety issues related to uranium enrichment facility have been covered by IAEA draft safety guide DS 344 (Ref.4.12). The safety issues relevant in uranium enrichment facility are (i) Chemical hazard, (ii) Radiation exposure, (iii) Criticality, and (iv) Storage and maintenance of depleted uranium.
The chemical hazards due to release of uranium hexafluoride have been discussed in section 4.2.4. UF₆ leakage should be restricted to less than 0.2 mg/m³ (chemical toxicity limit for Natural ‘U’ and up to 2.5% enrichment). (see Table 4.3.1)

4.3.2.1. Radiation exposure: The enriched uranium and depleted uranium in UF₆ cylinder emit neutron and gamma radiation. The neutron radiation results from (α, n) reaction of the uranium with fluorine. In the proximity of cylinders carrying enriched uranium, up to 70% of the radiation exposure can be due to neutron radiation. In the proximity of cylinders carrying depleted uranium, up to 20% of the radiation exposure can be due to the neutron radiation.
Radiation limit: 13Bq/m³, which implies for U with 5% enrichment - 80 µg/m³ and for U with 10% enrichment- 32 µg/m³

4.3.3. Criticality

Depending upon the concentration, the fissile material can attain criticality in some geometries. Hence safe geometries must be ensured. There is no criticality possible with gaseous UF₆. At low enrichments of less than 1%, even liquids with moderation do not go critical. For higher enrichments, moderation is important. Typically at 7% enrichment, H/U atom ratio must be kept below 0.38.

4.3.4. Storage and Maintenance of Depleted Uranium

In addition to the radiological and chemical health hazards associated with depleted UF₆, there are also risks of industrial accidents and transportation-related accidents during handling, storage or transport of depleted UF₆.

4.3.5. User requirements, INPRO Indicators and Acceptance Criteria for an enrichment facility based on centrifuges

UR 1.1: Robustness

IN 1.1.1: Robustness of design (simplicity, margins):
The separating element should be designed with lesser number of probable leakage points. 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 hermetically sealed design would ensure robustness. Vessels should be designed for preventing criticality, considering the maximum enrichment targeted. Isolation of cascade hall and handling area, provision of secondary seals in centrifuges, clear operation limits for critical parameters and adequate factors of safety in containment are other measures towards robustness.

**IN1.1.2:** High quality of operation.

A stable power supply is considered as an important requirement of enrichment processes based on centrifuge. The power supply should be of a high standard. Frequency of loss of power supply at the facility can be assessed from available data and it should be demonstrated that the frequency would be less than \(10^{-2} / \text{plant year}\). Sufficient margin in the design should be provided so that any small deviation of system parameters from normal operation will not lead to accident. The deviation should be corrected in a feedback loop. Training of personnel in handling of UF\(_6\) gas cylinders, action to be taken in the event of leakage of UF\(_6\) gas, etc., will ensure that the plant would operate in a safe regime.

**IN 1.1.3:** Capability to inspect

On line monitoring systems, with capability to inspect and more than one-way to measure the same parameter, are important options for ensuring safe operation of the centrifuges. The system should be designed with adequate condition monitoring systems and trending to predict incipient failures.

**IN 1.1.4:** Expected frequency of failures and disturbances

This aspect has to be controlled and continuously improved. The frequency of events such as leakage of UF\(_6\) gas, criticality and explosion have to be arrived at based on probabilistic as well as deterministic methods.
IN 1.1.5: Grace period until human actions are required
30 minutes in case of leak of UF₆ gas during normal operation. Low dependence on human action during regular operation facilitates longer grace periods.

IN 1.1.6: Inertia to cope with transients
The system should be robust to withstand transients. Surge suppression limiters, provision of fly wheel in the driving system of centrifuge machine in case of electricity fault, thermal inertia of heating furnace, and multi-stage control for reducing transients are required.

UR1.2- Detection and interception of deviations from normal operational states

IN1.2.1 Capability of instrumentation to detect / intercept / compensate deviations
Safe operating conditions must be clearly defined in the safety analysis report and different limits for alarm and shutdown conditions (pressure and overloading) should be indicated. Provision of Digital Control System with intelligent controller and hot stand by would ensure that the enrichment facility could be safely operated. Redundancy in devices for detecting overloading of separation system should be provided. Measurement of the parameter based on different principles wherever applicable, would provide enhanced safety. For example, use of two independent parameters to indicate maloperation of centrifuges (e.g. current drawn by motor and vibration) would ensure prompt correct action. As a feedback system to bring back to normalcy, strategy to isolate and limit damage to separation system must be available.

UR 1.3. Frequency of occurrence of accidents
IN 1.3.1: Calculated frequency of occurrence of design basis accidents
Protection against seismic activities should be provided as part of the design. Large scale leakage of centrifuges leading to loss of enriched material and resulting in criticality is an extreme event that can form the design basis. Such events should be shown by safety analysis to have a frequency of less than 10⁻⁶/plant year.
**IN 1.3.2:** Grace period until human intervention is necessary

A grace period of a few minutes for criticality accident should be achieved by provision of shielded enclosures wherever concentration of uranium is expected to be high, providing criticality monitors and negative pressure in the process handling area. Risk to humans should be limited to material handling area only. Since large scale gas leak has a risk of propagation to the public domain, a grace time of 15 minutes should be provided, for the gas leak to be managed by scrubber/ventilation system capability.

**IN 1.3.3:** Reliability of engineered safety features

(a) Reliability of secondary back-up seals in the centrifuges should be excellent, with failure rate better than 10^{-4} per operation year. This could be achieved by accelerated tests under simulated conditions.

(b) Provision should be available in the form of suitable brakes, to absorb the momentum of the centrifuge. This would localize the damage caused in the event of failure and prevent the centrifuge from becoming a missile.

**IN 1.3.4:** Number of confinement barriers maintained

Machinery casing, back up seals in the rotating parts and area isolation are examples of barriers. Provision of more than one barrier of each type can ensure defense in depth.

**IN 1.3.5:** Capability of the engineered safety features to restore to a controlled state

Safety interlocks must be provided for addressing the instability and vibration in motors for the centrifuges. The detection of gas leakage into operating area should result in the shutting down of gas supplies.

**IN 1.3.6:** Sub-criticality margins

To ensure that any accident resulting in large scale release of enriched uranium does not lead to criticality, the system should be analysed and shown to have $k_{eff} < 0.90$ for all possible configurations. In this process, mass concentration, shape, moderation etc. have
to be considered. All process equipments in material handling area have to be designed for criticality for submerged and water filled conditions.

**UR 1.4- Major release of radioactive materials into the containment / confinement**

**IN 1.4.1:** Calculated frequency of major release of radioactive materials into the containment/confinement:
In uranium enrichment facilities, the release of UF₆ gas due to catastrophic failure of centrifuges would lead to major release of radioactivity into the containment. The frequency of such release should be less than 10⁻⁶ per operation year. The UF₆ release in the working area has to be contained within the process area itself.
Process specific, sub-atmospheric pressure operation is likely to ensure that this can be achieved.

**IN 1.4.2:** Natural or engineered processes sufficient for controlling relevant system parameters and activity levels in containment / confinement.
Cascade segment isolation and cascade isolation based on pressure rise are such processes.
Emergency exhaust scrubber with alkali washing should be provided to bring down concentration of UF₆ to less than 0.2 mg/m³ within 30 minutes.

**IN 1.4.3:** In-plant severe accident management
(a) Cut off source, area isolation, emergency evacuation- for example, the cascade room should be isolated.
(b) Activation of on-site emergency plan to prevent the spread into uncontrolled area.
For instance, process isolation and area isolation, followed by evacuation/scrubbing.
The safety manual for the facility should include a carefully prepared comprehensive emergency action plan, and there should be periodic mock-up drills and training programmes to ensure that the operators are in readiness to handle emergencies.
UR 1.5: Major release of radioactive materials to the environment.

**IN 1.5.1:** Calculated frequency of a major release of radioactive materials to the environment.

Though it is process specific, it should be less than $10^{-6}$ per plant year.

**IN 1.5.2:** Calculated consequences of releases (e.g. dose).

A facility specific realistic, enveloping and robust (i.e. conservative) estimation of internal and external doses to workers and the public should be performed. Source term calculations should use: (i) material with the highest specific activity, (ii) licensed inventory, (iii) maximum process throughput. When considering the efficiency of barriers, their lowest performances in normal operation should be used. Public dose calculations should be based on maximum estimated releases to the air, water and deposition to ground. Conservative models and parameters should be used to estimate doses to the public. (Ref.4.12)

**IN 1.5.3:** Calculated individual and collective risk.

Objective should be as low as reasonably practicable.

UR 1.6. Defense in depth

**IN 1.6.1:** Independence of different levels of DID

(a) Cascade segment isolation, (b) cascade isolation, (c) area isolation and (d) evacuation, ventilation and scrubbing of affected segment/area, provide various levels of independence. Failure of a system should not lead to the failure of other systems by preventing transmission of shock or vibration, disturbance to other cascades, redundancy in material handling system. Each cascade and handling system should be made as independent modules.

UR 1.7: Systematic application of human factors

**IN 1.7.1:** Evidence that human factors (HF) are addressed systematically in the plant life cycle
(a) Emphasis on Human Resources Development
(b) Qualification scheme
(c) Mimicking operation panel and DCS operation with self-diagnostics, enhance human factor involvement.

Workers training, qualification and monitored working hours lead to continual improvement. Periodic up-gradation training, qualification incentive and minimization of operator fatigue through automation are good examples. The simulator training and health check up should be made mandatory.

**UR 2.1: Minimization of some hazards by incorporating inherently safe characteristics and/or passive systems**

**IN 2.1.1:** Sample indicators: stored energy, flammability, criticality, inventory of radioactive materials

Control of the inventory of radioactive materials is the first step towards prevention of criticality accidents. This should 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.

Other indicators which need to be considered are the fire hazard and a clear indication through safety analysis about steps to decrease the fire hazard.

**IN 2.1.2:** Expected frequency of abnormal operation and accidents.

The accidents that should be considered as design basis events for enrichment plants include (Ref.4.12) breach of an overfilled cylinder during heating, breach of a cylinder or pipe containing liquid UF₆, large fire and criticality events. Among these, the first three can result in adverse off-site consequences in addition to on-site consequences. It is necessary that the safety report of the facility establishes procedures to ensure that the frequency of such accidents are minimized and in any case <10⁻⁴ / plant year.
**IN 2.1.3: Consequences of abnormal operation and accidents.**

The design of the facility should incorporate features that will ensure that the consequences of the abnormal operations and accidents are minimized. For example, with respect to fire accidents, compartmentalizing buildings and ventilation ducts should be followed as far as possible in order to prevent spreading of fires. Buildings should be divided into fire areas. Should a fire break out within a given fire area, it must not be able to spread beyond the sector. The higher the fire risk, the greater the number of areas a building should have. Similarly, damage to the separation system and process handling system should be confined within the given area and should not spread to other areas.

**UR 3.1 & 3.2- Radiation Protection**

**Radiation hazard:** Both internal and external exposure can occur to a worker. Radiotoxicity is significant for U with enrichment above 2.5%. The radiological toxicity limit for UF$_6$ in air is Uranium$< 13$ Bq/m$^3$ subject to small variations with respect to enrichment but for practical purpose can be considered as independent of enrichment. (Derived from ALI values of ICRP International Commission on Radiological Protection-60). But 200 $\mu$g/m$^3$ corresponds to different values of Bq/m$^3$ depending on enrichment (specific activity). Typical values are given in Table 4.3.1.

<table>
<thead>
<tr>
<th>Enrichment</th>
<th>Chemical toxicity limit for UF$_6$ as U in air</th>
<th>Radiological toxicity limit for UF$_6$ as U in air (2400 m$^3$ air inhalation per year)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$\mu$g/ m$^3$</td>
<td>Bq/ m$^3$</td>
</tr>
<tr>
<td>0.7% (Natural)</td>
<td>200</td>
<td>5</td>
</tr>
<tr>
<td>1%</td>
<td>200</td>
<td>6</td>
</tr>
<tr>
<td>2%</td>
<td>200</td>
<td>12</td>
</tr>
<tr>
<td>2.3%</td>
<td>200</td>
<td>13</td>
</tr>
<tr>
<td>3%</td>
<td>200</td>
<td>18</td>
</tr>
<tr>
<td>3.5%</td>
<td>200</td>
<td>22</td>
</tr>
<tr>
<td>5%</td>
<td>200</td>
<td>33</td>
</tr>
</tbody>
</table>
From the above table it can be seen that for UF₆ chemical toxicity is important up to around 2.3% enrichment (limit U – 200μg /m³). However, for enrichment higher than 2.3%, radiological toxicity is important and in terms of mass of uranium the limit depends on the level of enrichment. For example the radiological toxicity limit for uranium is 78 μg/m³ for 5% enriched UF₆, as compared to its chemical toxicity limit of 200 μg /m³. Dose limits for occupational workers are 100 mSv for a defined period of 5 years (20 mSv per year) and for public, 1 mSv per year through all routes (air, water and land), in line with ICRP recommendations.

**UR 4.1-** Safety basis to be established with confidence

**IN 4.1.1:** Safety concept defined

Based on the available data and the specific RD & D carried out in the country, the safety concept for the facility should be established in a comprehensive manner and detailed in the safety report. All aspects of design, construction, operations and decommissioning should be covered in this effort.

**IN 4.1.2:** Design-related safety requirements specified

Based on the safety concept, the safety requirements have to be spelt out clearly in the safety report, with regard to each aspect of safety such as fire safety, criticality and other design basis accidents. The provisions in the plant which address these aspects have to be brought out. For example, requirement of monitoring for criticality, fire hazard evaluation and fire fighting provisions, handling of leaks etc.

**IN 4.1.3** Clear process for addressing safety issues

A safety review process has to be identified.

**UR 4.2- RD & D to achieve understanding of various phenomena to support safety assessment**

**IN 4.2.1:** RD&D defined and performed and database developed.

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 w.r.t 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

**IN 4.2.2:** Computer codes or analytical methods developed and validated.

Use of validated models and codes for safety analysis would increase the confidence level in the safety analysis methods used in analysing the accident scenarios and postulated hazards. However, such codes are not presently available and need to be developed

**IN 4.2.3:** Scaling understood and/or full scale tests performed

The scaling up of existing experience on separation technologies demands knowledge of all factors that need to be taken into account to ensure that the safety of the new plant would be on par with existing plants or better.

**UR 4.3.** Pilot plant or large scale demonstration facility to be built

**IN 4.3.1:** Degree of novelty of the process

The novelty will be measured on the basis of the safety features as well as production capacities. Intrinsic safety, proliferation resistance and simple, robust and energy efficient separation system are some of the novelty features. Maintaining minimum inventory, less criticality risk and low energy consumption – kWh/SWU can be some of the other parameters indicating novelty.

**IN 4.3.2:** Level adequacy of the pilot facility

In modular concept, minimum 3-5 stages must be present in the pilot facility. The actual number to be based on detailed studies and testing of individual sub- systems.

**UR 4.4- Risk informed approach**

Detailed studies have to be carried out as part of a risk informed approach, based on deterministic as well as probabilistic approaches, as appropriate (Ref 3.2). Uncertainties
in various probabilities and their sensitivities with respect to overall risk have to be evaluated, in order to arrive at a graded approach which optimizes the level of safety measures with respect to the probability and consequences of various events.

4.4 FUEL FABRICATION FACILITIES

As there are various types of fuel cycles, different kinds of fuel are fabricated in different forms. Light water and heavy water reactors most often use uranium oxide fuel. For light water reactors, the uranium is enriched to about 1.5 to 5 %. In heavy water reactors, natural uranium is used. In PWRs and FBRs, mixed oxide (MOX) fuel is being used. The majority of the research reactors use metallic/alloy fuel in plate form. Many countries are also planning to use metal fuels in fast reactors. In the present manual, the discussion is restricted to the fabrication of the fuels most commonly used in power reactors, UO₂ fuel and mixed UO₂-PuO₂ Fuel.

4.4.1 Natural UO₂ Fuel

Starting with uranium concentrate in the form of impure ammonium diuranate (ADU)/ magnesium diuranate (MDU)/uranium peroxide or UO₃, high-density UO₂ pellets are prepared by ‘powder-pellet’ route (Ref.2.4). The major process steps for fabricating the ‘fuel pellets’ are as follows:

- Dissolution in nitric acid followed by solvent extraction purification of impure uranium nitrate solution by using tributyl phosphate (TBP) in kerosene as solvent.
- Addition of ammonium hydroxide to pure uranium nitrate solution to precipitate pure ADU or addition of NH₃ and CO₂ 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 UO₂ powder.
- Cold-pelletisation of powder followed by high temperature sintering (1700-1725°C) in hydrogen atmosphere and centreless grinding to desired diameter.
In most countries, sinterable grade UO$_2$ powder for PHWRs is obtained by adapting the ADU route. The 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 the granulation step for making free-flowing press-feed granules. UO$_2$ granules (1-2 mm) are obtained by either ‘roll-compaction-granulation’ or ‘precompaction-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. The AUC route is followed in South Korea and Argentina. In the AUC route, calcination, reduction and stabilization are simultaneously carried out in vertical fluidized bed reactor. In the beginning 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 high-speed ‘rotary compaction’ press has been selected. For densification of green pellets, high temperature sintering is carried out at ~1700 °C in continuous sintering furnace.

For instance in INDIA, natural UO$_2$ powder is produced from magnesium diuranate (MDU). The UO$_2$ powder is subjected to either ‘roll compaction-granulation’ or ‘precompaction-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 °C in cracked ammonia in pusher type continuous sintering furnace. The sintered pellets are finally subjected to wet centreless-grinding to obtain UO$_2$ pellets. The pellets are then loaded into the clad tube which is subsequently welded.

### 4.4.2 Enriched Uranium Oxide Fuels for LWRs

The light water reactors all over the world use zircaloy clad low enriched uranium (LEU) oxide fuel assemblies with $^{235}$U content in the range of 1.5 to 5%. The enriched UO$_2$ pellets are universally manufactured by ‘powder-pellet’ route involving preparation
of enriched UO₂ powder using UF₆ as starting material. The integrated dry route (IDR) is followed in most countries for preparation of fine UO₂ powder by reacting UF₆ vapour with a mixture of super heated dry steam and hydrogen at ~600 – 700 °C. The chemical reaction is as follows → UF₆ + 2H₂O + H₂ = UO₂ + 6HF

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 UO₂ powder is low, about 2 m²/g compared with the powder produced by the wet chemical route. The IDP powder is extremely fine (~0.2 μ) and requires granulation. The powder is usually transferred into orbital screw blenders for homogenization. Pore formers such as polyvinyl alcohol, methyl cellulose or U₃O₈ are added at the blending stage. The UO₂ powder is subjected to cold-pelletisation and high temperature sintering in hydrogen atmosphere.

4.4.3 MOX Fuel Fabrication

Fuel containing Pu can be in the form of MOX, 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 need to be very reliable. Fabrication of Th-Pu MOX fuel can be done in a similar manner. Th-Pu MOX can be sintered in air, which adds to economy and convenience. (Th-²³³U) MOX fuel fabrication calls for development of automated and remote fabrication technology due to the presence of ²³²U. For more details see Ref. 4.20.

4.4.4 Safety Issues

Safety issues in fuel fabrication facilities (FFFs) have been discussed in Ref. 4.21-4.30. The draft safety guides IAEA DS-317 and IAEA DS-318 deal extensively with safety aspects of fuel fabrication facilities for uranium and MOX fuels respectively. Typical initiating events for incidents in fuel fabrication facilities are given in Table 4.4.1. Criticality accidents and the accidental release of hazardous materials are the major safety issues. Some of the incidents to be considered are

- Nuclear criticality accidents, e.g. in a wet process area
- Release of uranium, e.g. explosion of a reaction vessel in the conversion process
- Release of UF₆ from breach of a hot cylinder
- Release of HF from breach of a storage tank
- Spread of fire

The first two events would primarily result in radiological consequences to workers on site but may also result in some adverse off site and environmental consequences. The last three events would have both on-site and off-site chemical consequences.

### Table 4.4.1. Typical Initiating Events for Fuel Fabrication Facilities

<table>
<thead>
<tr>
<th>INCIDENT</th>
<th>INITIATING EVENT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flow transient</td>
<td>Degradation of control function&lt;br&gt;Single active component failure/Operator error</td>
</tr>
<tr>
<td>Fissile material density transient</td>
<td>Degradation of control function&lt;br&gt;Single active component failure/Operator error</td>
</tr>
<tr>
<td>Solvent density transient</td>
<td>Degradation of control function&lt;br&gt;Single active component failure/Operator error</td>
</tr>
<tr>
<td>Inventory transient</td>
<td>Single active component failure/Operator error&lt;br&gt;Leakage from pipes/vessels</td>
</tr>
<tr>
<td>Uncontrolled changes in heating capability</td>
<td>Degradation of control function&lt;br&gt;Single active component failure/Operator error</td>
</tr>
<tr>
<td>Uncontrolled process change</td>
<td>Leakage from pipes/vessels&lt;br&gt;Interventional operator actions leading to deviation&lt;br&gt;Loss of power supply&lt;br&gt;Fire/explosion in plant</td>
</tr>
<tr>
<td>Disturbance of temperature regime</td>
<td>Degradation of caution function</td>
</tr>
<tr>
<td>Hydrogen burner extinguished in UF₆ conversion unit</td>
<td>Operator error</td>
</tr>
<tr>
<td>Self-supported chain reaction-criticality</td>
<td>Increase of fissile material concentration&lt;br&gt;Violation of loading norms&lt;br&gt;Increase of the product moisture&lt;br&gt;Clog of gas-line&lt;br&gt;Decrease of the unit temperature&lt;br&gt;Extinguished hydrogen burner&lt;br&gt;Depressurization of the filter&lt;br&gt;Leakage of the cooling system</td>
</tr>
</tbody>
</table>

The facility should be designed to restrict exposure from normal operations to as low as reasonably achievable level (ALARA). Some of the design principles are prevention of criticality by design (the double contingency principle is the preferred
approach) and confinement of chemical hazards (includes the control of any route into the workplace and the environment).

In case of enriched uranium/MOX fuel fabrication, special care is taken to minimize contamination. Shielding may be needed for protection of the workers due to higher gamma dose rates.

4.4.4.1 Prevention of 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 (for example 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. safe diameter of vessels, distances in storages)
- Concentration of fissile material in solutions (e.g. wet process for recycling uranium)
- Presence of appropriate neutron absorbers (e.g. construction of storages, drums for powder, fuel shipment containers)
- Moderation limitation (e.g. moisture and amount of additives in powder)

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

- The use of a conservative approach (uncertainties on physical parameters, optimum moderation conditions physically possible, etc)
- The use of appropriate and qualified computer codes within their applicable range and use of appropriate cross section libraries

4.4.4.2 Confinement against internal exposure and chemical hazards: Consideration is given to protecting the workers, public and environment from releases of hazardous material in both operational states and design basis events conditions.

- Containment is the primary method for protection against the spreading of dust contamination, e.g. areas where significant quantities of either uranium or hazardous
substances are held in a gaseous form. A cascade of reducing absolute pressures can be established between the environment outside the building and the hazardous material inside.

- The design takes into account the performance criteria for the ventilation and containment systems including (i) the pressure difference between premises (ii) the air renewal ratio (iii) the type of filters to be used (iv) the maximum differential pressure across filters and (v) the appropriate flow velocity at the openings in these systems (e.g. acceptable range of air speed at the opening of a hood).

### 4.4.4.3 Protection of the workers:

- Ventilation systems are used as one of the means of minimizing the workers exposure to hazardous material that may become airborne in the facility.

- Careful design of containment and ventilation systems is done to minimize the need for use of respiratory protective equipment.

- Primary filters are located as close to the source of contamination as practical. This is to minimize the buildup of uranium powder in the ventilation ducts. Having multiple filters in series is preferable since it avoids reliance on a single barrier.

- Monitoring equipment such as differential pressure gauges (on filters between rooms or between a glove box and its operating area), air monitors, fire alarms, area radiation monitors are installed as necessary.

- Personnel radiation monitoring instruments are provided for radiation protection.

- In order to prevent the propagation of a fire through the ventilation ducts and to maintain the integrity of firewalls, ventilation systems are equipped with fireproof shutters.

- To facilitate decontamination, the walls, floors and ceilings in areas of the FFF where contamination may exist are nonporous and easy to clean.

### 4.4.4.4 Environment protection:

The design of the fuel fabrication facility should provide for the monitoring of the facility environment and identification of barrier
breaches. Uncontrolled dispersion of radioactive substances to the environment from
accidents can occur if the containment barrier(s) are impaired. In addition, ventilation of
these containment systems, with discharge of exhaust gases through stack via a gas
cleaning process such as a filter, reduces normal environmental discharges of radioactive
materials to very low levels. Efficiency and resistance of the filters to chemicals (e.g.
HF), temperatures in the exhaust gases and fire conditions need to be taken into
consideration.

4.4.4.5 Fire and explosion: Design must account for fire safety on the basis of a fire
safety analysis and implementation of defense 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.

Special fire hazard analysis should be carried out for:

- Processes involving H₂ such as conversion, sintering and reduction of uranium oxide
- Processes involving Zr powder form or mechanical treatment of Zr metal
- Workshops using flammable liquids like recycling shop and laboratories
- Storages of reactive chemicals (NH₃, H₂SO₄, HNO₃, H₂O₂…)
- Areas with high fire loads like waste storages or waste treatment areas, especially if
  incineration is used and
- Control rooms.

Explosions can occur due to gases (H₂ used in conversion process and sintering
furnaces etc.) and chemical compounds (ammonium nitrate in recycling facilities).
Explosion can be prevented by using inert gas atmosphere or dilution systems. Recycling
systems should be regularly monitored to prevent ammonium nitrate deposits. In areas
with potentially explosive atmospheres, the electrical network and equipment are
protected according the industrial safety regulations.
4.4.4.6 Flooding: Flooding can lead to the dispersion of radioactive materials and changes in moderation conditions. In installations where there are vessels and/or pipes with water the criticality analyses should consider the presence of the largest volume of water within the considered room as well as in connected ones. Walls of rooms with potential flooding should withstand the water load in order to avoid any domino effect.

4.4.4.7 Leaks and spills: Leaking of components like pumps, valves and pipes can lead to dispersion of radioactive material (UO₂, U₃O₈ powder and UF₆) and toxic chemicals (e.g. HF). Leaks of hydrogenous fluids (water, oil, etc) can change the moderation in fissile materials and reduce criticality safety. Leaks of flammable gases (H₂, natural gas, propane) or liquids can lead to explosions and/or fire. Leak detection systems are used in such cases. Vessels containing significant quantities of nuclear materials in liquid form are equipped with alarms to prevent overfilling.

UF₆ leakage should be restricted to less than 0.2 mg/m³ (chemical toxicity limit for natural ‘U’ and upto 2.5% enrichment). (See Table 4.3.1.) Radiation limit: 13Bq/m³; This implies 80 µg/m³ for U with 5% enrichment and 32 µg/m³ for U with 10% enrichment.

4.4.4.8 FFF instrumentation and control (I&C):

Instrumentation: Instrumentation should be provided to monitor facility variables and systems over the respective ranges for (i) normal operation (ii) anticipated operational occurrences and (iii) design basis accidents in order to ensure that adequate information can be obtained on the status of the facility and proper actions from operating procedures or automatic systems can be undertaken.

Control System: Passive and active engineering controls are more reliable than administrative control and should be preferred for normal operational states and DBE conditions. When used, automatic systems are designed to keep process parameters within the operational limit and conditions or to bring the process to its safe state.

Safety related I&C during normal operations:

During normal operations, the FFFs safety related I&C includes
Instrumentation and process control: (e.g. temperatures, pressures, flow rates, concentration of chemicals and/or radioactive materials or tank level). A key process control parameter related to safety is the concentration of hydrogen in the gas being fed to sintering furnace.

Control and monitoring of ventilation: Mainly on differential pressures across HEPA filters and airflows

Radiation dosimetry: Sensitive dosimeters with real time display and/or alarm (for external exposure); Continuous air monitors (for internal exposure)

Gaseous and liquid effluents: Real time measurements are necessary if there is a risk of exceeding regulatory limits

All rooms with fissile and/or toxic chemical materials are equipped with fire alarm

Gas detectors are used where leakage of gases (H₂, heating gas) could lead to an explosive atmosphere.

Criticality: Radiation detectors (gamma and/or neutrons), with audible and where necessary visible alarms for initiating immediate evacuation from the affected areas

Chemical release: Detectors and alarm systems are installed in areas with a significant chemical hazard (e.g. UF₆, HF)

Release of effluents: The devices used to measure releases of gaseous and liquid effluents under operational states should also be able to measure the releases in case of DBEs.

4.4.5 Radiation Protection in MOX FFF

Typical design principles for safety are:

- Minimization of inventory
- Shielding inside the glove boxes as well as double skin glove boxes
- Use of combined shielding- polythene + lead + neutron absorber
- Minimization of human intervention by means of automated processes
- Barriers to prevent contamination
- Air, surface and personnel monitoring
4.4.6 Criticality Safety in MOX FFF

- Priority of technical measures for criticality safety: ever safe geometry and mass control, neutron poisoning and moderation control
- Administrative control in area with low inventory only
- Ability to cope with double failure

4.4.7 $^{233}$U Fuel Fabrication Plants

- These plants have the additional problem of handling $^{232}$U. As already indicated, this leads to a high external gamma dose and all the facilities in the fabrication plant have to be well shielded. Remotisation and automation are necessary to reduce occupational dose to the workers.
- The noble gas Radon-220 of $^{232}$U decay chain can pass through HEPA filers and then decay to solid $^{208}$Tl. To prevent escape of the decay products, off-gas system requires two HEPA filters with a charcoal-bed delay system between them.
- Reliable static and dynamic containment systems with single entry/exit should be in place for effective management of contamination.

4.4.8. User requirements, INPRO Indicators and Acceptance Criteria for Fuel Fabrication Facilities

UR 1.1. Robustness

**IN 1.1.1:** Robustness of design (simplicity, margins):

Fuel fabrication facilities (specially those handling plutonium or U-233) should be designed for withstanding earthquakes. For fire prevention, use of fire resistant materials for primary confinement system and use of passive cooling systems for high temperature operation are envisaged. For prevention of explosion, avoidance of combustible gases or use within fire safety limits (e.g., Ar-H$_2$ mixtures or cracked ammonia); where H$_2$ is used, use of flame curtain is recommended.

**IN 1.1.2:** High quality of operation.
The distinctive feature of the fuel fabrication facilities are the presence of large inventories of powders of uranium oxide/plutonium oxide/mixed oxide. These are usually in finely divided form, and unless a high quality of operation is ensured, spillage of the fuel materials inside the enclosures would lead to long term accumulation in various difficult-to-access areas and the glass panels of glove boxes, which would 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 accumulation 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). A high degree of automation/remotisation/robotics would lead to reduction of dose received by the operators. An inappropriate response to an emergency is also a result of inadequate operator training.

Thus, acceptance criteria for high quality of operation can be taken to be:

a) High degree of remotisation
b) Periodic and intensive training of operators
d) Availability of operations manuals and emergency instructions manuals
e) Periodic mock-ups to ensure readiness of operators to handle emergencies

IN 1.1.3: Capability to inspect

On line 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. Continuous monitoring of pressure drops across HEPA filters in the ventilation systems would ensure 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 criteria for this indicator is thus the availability of continuous monitoring of radiation levels as well as all the safety critical systems such as ventilation, cooling water, etc.,
**IN 1.1.4:** Expected frequency of failures and disturbances:
Examples of failures and disturbances include minor leakages of UF₆ gas to the operating area, temporary loss of power leading to ventilation failure, and disturbance in cooling water flow to furnace. Since these events would not lead to catastrophic failures especially in well designed facilities, a value of $<10^{-4}$ per year is recommended as the acceptance criterion.

**IN 1.1.5:** Grace period until human actions are required.
The design of glove boxes in MOX fabrication facilities should ensure that in the event of ventilation failure, the 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. The operation manual must list the anticipated incidents action plan and time before which the actions have to be completed.

**IN 1.1.6:** Inertia to cope with transients.
The system should be robust to withstand transients. Redundant cooling systems and design of furnaces for furnaces should ensure that in the event of temporary loss of cooling water supply, the furnace casing temperature will not exceed safe limits. It should be possible to bring the furnaces to a safe shut down state if necessary or continue to operate at a lower power level. The leak tightness of the glove box units should be such that in the event of temporary loss of the glove box ventilation, the radioactivity level in the operating area would not cross unsafe levels.

**UR 1.2. Detection and interception of deviations from normal operational states**
**IN 1.2.1:** Capability of control and instrumentation system and/or inherent characteristics to detect and intercept and/or compensate such deviations.

The 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 the healthiness and safety of various systems and ensuring that they operate in safe regimes of parameters. Thus, the utility must define safe operating
conditions for every system, and different limits for alarm and shutdown conditions should be indicated. Furnaces should be equipped with power supply shut down systems to prevent escalation of temperature. Pressure control systems in glove boxes must be able to intervene in the event of 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 negative w.r.t operating area.

Measurement of the parameter based on different principles wherever applicable and more than one devise for measurement, would provide enhanced safety.

**UR 1.3. Frequency of occurrence of accidents**

**IN 1.3.1:** Calculated frequency of occurrence of design basis events.

While all the postulated initiating events described earlier may not be analysed quantitatively, it is important to ensure that the safety analysis includes probabilistic safety assessment of important initiating events such as a) Earthquake b) Fire c) criticality d) large scale release of UF₆ e) major release of radioactivity due to an explosion or damage to the glove box panel due to a missile such as press tooling or grinder bowl f) Flooding of glove boxes due to rupture of cooling water lines. A frequency of less than 10⁻⁶ per plant year is recommended

**IN 1.3.2:** Grace period until human intervention is necessary:

Criticality accident can be caused by human errors such as double batching or due to flooding of glove boxes containing large inventories of fissile material. Provision of criticality monitor is essential. In the event of criticality, a grace time of a few minutes only may be available to avoid further excursions. In the event of flooding of glove boxes due to coolant pipe rupture, the grace time available to avoid criticality or release of radioactive material would depend on the design of the box and the flow rate of water. The safety report should take into account these factors and define the time limits for human action.
**IN 1.3.3:** Reliability of engineered safety features:
Examples of engineered safety features in a fuel fabrication facility include temperature control systems to shut down furnaces in the event of loss of cooling water, and secondary ventilation systems which would take over in the event of loss of a glove box barrier (e.g. puncture or tear of a glove). The systems deployed and their reliability should be demonstrated through design analysis as well as actual trial experiments before the facility goes into active operation.

**IN 1.3.4:** Number of confinement barriers maintained:
The most important safety feature of the fuel fabrication facility is the barrier for release of radioactive material into the environment. Pu based materials are handled in glove boxes, whose panels and gloves constitute a barrier. However, it is important to ensure that the glove box is treated as the secondary barrier and larger inventories of fuel materials are always maintained in another suitable enclosure which would constitute the primary barrier. For example, in boxes housing equipment with moving parts such as press or grinder, the equipment should 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. A secondary ventilation enclosure should be provided for glove boxes, so that in the event of damage to the glove box panel, resulting in loss of negative pressure inside the box, the secondary exhaust system would ensure containment of active material. 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 defense in depth. Release of UF₆ due to inadvertent breaking of cylinders is one of probable incidents. Hence sufficient barriers should be provided, for access of the cylinders.

**IN 1.3.5:** Capability of engineered safety features to restore INS to controlled state:
This is dealt with also under 1.3.3. The safety review procedures should include a demonstration of the reliability of the engineered safety features and their capability to ensure safety of the system without any operator intervention.
IN 1.3.6: Sub-criticality margins.
The $k_{\text{eff}}$ value has to be calculated for all possible configurations and should be $< 0.90$.
(a) Mass concentration, shape, moderation, isotopic concentration of fissile element etc. have to be considered. For uranium fuel fabrication, typical limits of isotopic concentrations that can be considered safe are 5 wt% U-235 for uranium metal and compounds under dry/unmoderated conditions, 0.9% U-235 for homogenous aqueous U solution and 1.8% U-235 for uranyl nitrate solution ($\text{N/ U} = 4$).
(b) All process equipments in material handling area have to be designed for criticality for sub-merged and water filled conditions.

UR 1.4. Frequency of release of major amounts of radioactivity to the confinement

IN 1.4.1: Calculated frequency of major release of radioactive materials into the containment / confinement
The frequency of occurrence of major release of radioactivity into the confinement should be arrived at by probabilistic safety assessment, considering various accidents and failures discussed above, and taking into account human factors. A value of $10^{-4}$ per operation year should be taken as the indicator.

IN 1.4.2: Natural or engineered processes sufficient for controlling relevant system parameters and active levels in containment / confinement.
The number of air changes provided in the operating area should be such that the air activity in the operating area can be brought down quickly below the safety limits. Secondary enclosures and exhaust systems should be able to provide necessary air changes around the glove box in the event of a failure

IN 1.4.3: In-plant severe accident management.
A carefully prepared comprehensive emergency plan should be available for managing severe accidents (e.g., criticality, explosion) that may occur in the fuel fabrication facility. The main components of this plan would be:
(a) Cutting off the source, area isolation, emergency evacuation
(b) Activation of on-site emergency plan to prevent the spread into uncontrolled area.

**UR 1.5. Major release of radioactivity to environment**

**IN 1.5.1:** Calculated frequency of major release of radioactive materials to the environment
Major release of radioactivity into environment from a fuel fabrication plant can take place through explosions and criticality incidents. The frequency of such incidents should be analyzed and shown to be less than $10^{-6}$ / plant year.

**IN 1.5.2:** Calculated consequences of releases (dose).
The safety analysis must demonstrate that in the event of large scale radioactivity release to environment, the dose to public should be minimized and is ALARP.

**IN 1.5.3:** Calculated individual and collective risk.
Objective should be as low as reasonably achievable.

**UR 1.6. Defense in depth**

**IN 1.6.1:** Independence of different levels of DID
Maintenance of differential pressure in process enclosures and operating areas, easy access of the equipments in operating areas, automation / robotics for handling radioactive materials, zoning in layout of the plant for hazardous operations, single port entry and exit for personnel and equipment and multiple levels of filtration are some of the independent strategies for control of contamination. The safety analysis report of the facility must clearly demonstrate the independence of the levels of defense.
Mass control of fissile material, on-line NUMAC, use of safe geometry (with respect to criticality) in equipment layout to provide safe separation between equipment as well as storage systems, minimization of hydrogenous materials in process and use of neutron absorbing materials are necessary for criticality control.
The use of defense in depth approach for fuel fabrication plants is discussed in Ref. 3.4.

**UR 1.7. Systematic application of human factors**

**IN 1.7.1.** Evidence that human factors (HF) are addressed systematically in the plant life cycle.

(a) Emphasis on Human Resource Development (b) qualification scheme (c) rotation of shifts and breaks enhance human factor involvement. Workers training, qualification and monitored working hours lead to continual improvement. Periodic up-gradation training, qualification incentive and minimization of operator fatigue through automation are good examples. The simulator training and health check up should be made mandatory.

**BP 2: Increased emphasis on inherent safety characteristics:**

**UR 2.1. Incorporation of inherent / passive safety**

**IN 2.1.1:** Sample indicators: stored energy, flammability, criticality, inventory of radioactive materials

Fuel material inventory in each stage should be as per limits arrived at in safety analysis. It must be ensured that double batching would not lead to accumulation of fissile material beyond the critical limits. Deposition of powders containing Pu inside the glove box surfaces constitutes a long term radiation hazard in MOX fuel fabrication facilities. The use of sol-gel microsphere based fabrication processes provides inherent safety in this regard. Use of Ar/H₂ gas mixtures or cracked ammonia in place of hydrogen provides inherent safety against fire

**IN 2.1.2:** Expected frequency of abnormal operations and accidents:

Abnormal operations in FFFs include runaway conditions in furnaces, overpressurisation of glove boxes, etc. Accidents in FFFs mainly involve fire, chemical explosion and criticality. These have to be analysed using both probabilistic and deterministic approaches to ensure that the accidental conditions or abnormal operations would not have a frequency > 10⁻⁶.
IN 2.1.3: Consequences of abnormal operations and accidents:
The design safety report for the facility should comprehensively indicate the abnormal
operations (e.g., runaway conditions in furnaces, overpressurisation of glove boxes) and
accidents (explosions, fire, criticality, large scale release of radioactivity) and the impact
of these operations and accidents, such as release of chemicals or radioactivity into public
domain.

IN 2.1.4: Confidence in innovative components and approaches:
Wherever innovative changes are introduced in the plant, it is necessary that the
reliability of the innovative system is demonstrated to be higher than existing system

BP 3: Risk from radiation exposures
UR 3.1. Optimisation of radiation protection to workers
IN 3.1.1 and 3.2.1: Radiation Protection
Dose limits for occupational workers are 100 mSv for a defined period of 5 years (20
mSv per year) and for public, 1 mSv per year through all routes (air, water and land), in
line with ICRP recommendations.

BP 4: Improving confidence in design and safety assessment
UR 4.1. Safety basis to be established with confidence
IN 4.1.1: Safety concept defined- Each concept should be defined individually and the
safety features demonstrated to be better than the existing ones. For this, a suitable
combination of deterministic and probabilistic approaches should be used.
IN 4.1.2: Design-related safety requirements specified- The specific design safety
requirements for fuel fabrication plants have been described in Ref. 4.21 and 4.24. The
design report of the facility should address all the issues indicated in these reports.
IN 4.1.3: Clear process for addressing safety issue
The facility should conform to a well established review process by independent
regulation, to ensure that the safety issues are all addressed.
**UR4.2. RD & D to achieve understanding of various phenomena to support safety assessment**

**IN 4.2.1:** RD&D to be defined and performed and the database have to be developed and checked periodically.

RD&D is essential to introduce innovative features in the fabrication process which would ensure enhanced safety. For instance, the powder-pellet route has several steps involving generation and handling of fine powder and leads to radiotoxic dust. Processes such as sol-gel microsphere pelletisation could address some of these issues. Similarly, development of remote fabrication techniques would aid in reduction in manrem exposure.

**IN 4.2.2:** Computer codes or analytical methods have to be developed and validated on benchmark experiments.

**IN 4.2.3:** Scaling should be understood and/or full scale tests performed:

The tested model should not violate the basic safety principles while enlarging the size. It is thus important that any novel fuel fabrication route proposed should be tested on a pilot plant scale and the experience obtained fully integrated into the design of the full scale plant.

**UR4.3. Pilot plant or large scale demonstration facility to be built**

**IN 4.3.1:** Degree of novelty of the process

Intrinsic safety, proliferation resistance and simple, robust and energy efficient system are some of the novelty features. The novelty will be measured on the basis of its safety features and production.

**IN 4.3.2:** Level of adequacy of the pilot facility: To ensure that the pilot plant provides all necessary safety related inputs to the design of the full scale facility, it is important that the pilot plant is planned at an adequate scale, so that the extrapolation of results from its operation would be meaningful.

**IN 4.4.1 & 4.4.2:** use of a risk informed approach

Detailed studies have to be carried out as part of a risk informed approach. Uncertainties and sensitivities have to be identified and appropriately dealt with. As pointed out in
IAEA DS316, the implementation of safety requirements for the nuclear fuel fabrication facilities should be commensurate with the potential hazards (“graded approach”). The utilization of risk information for fuel fabrication facilities is well illustrated, for example, by the studies reported by JAERI, Japan (Ref.3.4)

4.5. SPENT FUEL STORAGE FACILITIES

Interim spent fuel storage facilities are required for the safe, stable and secure storage of spent nuclear fuel after it has been removed from the reactor pool and before it is reprocessed or disposed of as radioactive waste. Spent fuel is usually transferred to interim spent fuel storage facilities only after an initial period of storage at the reactor. Fig.4.5.1 gives a schematic of the spent fuel movement. The initial period of storage allows a considerable reduction in the quantity of volatile radionuclides, the radiation fields and the production of decay heat. Hence, the development of conditions which could lead to accidents in interim spent fuel storage facilities will generally occur comparatively slowly, allowing ample time for corrective action before limiting conditions may be approached. The safety of spent fuel handling and storage operations can thus be maintained without reliance on complex, automatically initiated protective systems. Various aspects of spent fuel management are discussed in Ref.4.31 and 4.32.

For the safe operation and maintenance of spent fuel storage facilities, their design must incorporate features to maintain fuel sub-critical, to remove spent fuel decay heat, to provide for radiation protection, and to maintain containment over the anticipated lifetime of the facilities as specified in the design specifications. These objectives must be met in all anticipated operational occurrences and design basis events in accordance with the design basis as approved by the Regulatory Body.

AR (At-Reactor) storage facilities are essentially storage pools in which spent fuel is kept under water following discharge from the reactor. Two technologies have been developed for AFR (Away-From-Reactor) storage: (1) AFR (Reactor Site) storage is independent of the reactor and its AR pool. This can be wet in the form of secondary or
additional pools or most often in the form of dry storage facilities which may or may not have capability for off-site transport. (2) AFR storage off the reactor site (OS) at an independent location, for example, at reprocessing plants. This AFR (OS) interim storage can also be centrally located at a selected power plant complex and receive fuel from other power plants.

![Fig.4.5.1. Interim storage in spent fuel reprocessing/disposal routes](image)

**4.5.2. Wet storage**

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.).
4.5.2.1. At-reactor (AR) storage pools

At-reactor spent fuel storage pools are either within the reactor building or in an adjacent spent fuel building which is linked to the reactor by a transfer tunnel. Access to the fuel in the storage pool is usually by means of immersing a cask in the pool, loading it with fuel and then removing the cask for lid closure, decontamination and transport. A recent development, unique to France, is a cask loading concept with bottom access ports from the pool for fuel transfer into the cask. The advantages of this design are that contamination of the external surface of the flask by immersion in the pool is avoided, and also the requirement to lift the flask (empty and loaded) between the inlet/outlet location and the pool and a heavy duty crane is no longer needed. There are some cases, for example at gas cooled reactors and at Sellafield in the UK, where spent fuel is loaded into casks in a dry shielded cave and the cask is never immersed in water.

The 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 all 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 pond 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 should be proved to have negligible corrosive ion leaching and permeability to water. The pools are filled with deionized water with or without additive depending on the type of fuel to be stored and the adopted method of treatment. The water is either a fixed quantity or a once through pond purge. Water activity levels are maintained ALARA (as low as reasonably achievable) by either in-pool or external ion exchange systems or by limiting activity release to the bulk pool water. Leakage from the pool is monitored, either by
means of an integrated leakage collection system or via the interspace 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. This is for pH control, maintaining boron levels for criticality control where necessary, and the maintenance of acceptably low 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.

Sub-criticality was originally maintained by spacing within the storage racks or baskets. However with the need to store greater quantities of fuel, 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.

### 4.5.3. Dry Storage of Spent Fuel

Wet storage requires a large amount of space. With increasing amounts of spent fuel being accumulated, it has become necessary to consider the dry storage option, which is more compact. Fig 4.5.2 gives a comparison of the amount of spent fuel in storage and the fuel which is reprocessed. This clearly illustrates the increasing need for large spent fuel storage facilities.
Fig. 4.5.2 Global statistics on spent fuel management

Dry storage of spent fuel 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. Fuel must be stored in pools for several years before it becomes cool enough for dry storage to be possible. For those seeking economies of scale in storing large quantities of spent fuel for a prolonged period, vaults and silos are attractive, while for those seeking the flexibility of a modular, piece-by-piece storage system, dry casks are preferred.

4.5.3.1. Dry Storage Vaults

In a vault, the spent fuel is stored in a large concrete building, whose exterior structure serves as the radiation barrier, and whose interior has large numbers of cavities suitable for spent fuel storage units. The fuel is typically stored in sealed metal storage tubes or storage cylinders, which may hold one or several fuel assemblies; these 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, fuel is removed from the transport cask and moved without any container to its storage tube, while in
others the fuel 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.

4.5.3.2. Dry Storage Silos

In a silo storage system, the fuel is stored in concrete cylinders, either vertical or horizontal, fitted with metal inner liners or separate metal canisters. The concrete provides the 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.

4.5.3.3. Dry Storage Casks

In a cask system, a flat concrete pad is provided (either outdoors or within a building), and large casks (metal or concrete with metal liner), that contain the spent fuel can be added as needed to store the amount of fuel required. The casks provide both shielding and containment. Originally, like vaults and silos, casks were designed only for storage (so-called “single purpose” casks). More recently, some cask designs have been licensed for both storage and transport (“dual purpose”), and design work continues on casks intended to serve for storage, transport, and permanent disposal (“multi-purpose”).

4.5.3.4. Metal casks

Metal casks are massive containers used in transport, storage and eventual disposal of spent fuel. 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 sub-criticality. Metal casks usually have a double lid closure system that may be bolted or seal welded and may 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.
4.5.4. Safety Issues in Storage Facilities:

The safety assessment of spent fuel storage facilities and the safety aspects related to design of spent fuel storage facilities are discussed in Ref.4.33 and Ref.4.34. The safety analysis of the fuel storage facility should address the following aspects:

— Design life of the storage facility;
— Selection of component materials;
— Fuel cladding temperature;
— Material temperatures;
— Radiation fields;
— Pool water chemistry and radioactivity;
— Gaseous and liquid releases inside the facility;
— Gaseous and liquid releases outside the facility.

All choices of this kind should be justified with appropriate data, analyses and well-justified arguments.

For the safety systems and safety related systems and components to perform properly, the components of the facility should maintain their structural integrity in operational states and accident conditions. Therefore, the integrity of the components and systems for these conditions should be demonstrated by structural analysis. This should take account of relevant loading conditions (stress, temperature, corrosive environment, etc.), and should consider creep, fatigue, thermal stresses, corrosion and material property changes with time (e.g. concrete shrinkage). Because the fuel produces heat, all thermal loads and processes should be given appropriate consideration in the design.

Among the faults to be analysed during safety assessment (Ref.4.34) are: receipt of defective fuel, fuel handling faults (dropped loads, impacts), loss of external electrical supplies, faults in heat removal system, faults in coolant or pool water circulation systems, excessive coolant or pool water leakage, faults in containment (e.g. container leakage), seismic events, fire and explosion. Care should be taken to consider all situations where mechanisms might jam, leaving a fuel element or a basket less than
adequately shielded. Consideration should also be given to the possibility of a basket jamming within the storage facility. In addition to the shielding issue, it should be considered whether the handling equipment and methods are such that recovery from such situations could be endangered by excessive stresses having been applied.

The SAR should show by an appropriate analysis that for the geometry defined by the design and for the construction materials, sub-criticality will be maintained under all circumstances. The methodology used for performing the analysis should be described and evidence provided to show that the methodology has been validated for the specific application against criticality experiments. Account should be taken of possible variations in the parameters used in the analysis.

The response of the fuel containment to a seismic event should be evaluated. A seismic analysis of the integrity of the overall containment should be carried out at a level of detail commensurate with the hazard to the storage facility.

Handling equipment shall be designed to minimize the potential for damage to fuel, fuel assemblies, and to storage or transport containers. Equipment should be provided with positive latching mechanisms to prevent accidental release.
(c) Moving equipment should have defined speed limitations.
(d) Systems should be designed so that fuel cannot be dropped as a result of loss of power.

If the safe functioning of the facility requires electric power (e.g. forced convection, ventilation, pool water circulation), then the assessment should demonstrate adequate reliability of the electric power supply or, failing that, demonstrate that an adequate level of safety is maintained in the event of loss of electric power.

4.5.4.1 Wet Storage Safety

The following safety requirements need to be addressed:
(1) Fuel cladding integrity should be maintained during handling and exposure to corrosion effects of the storage environment
(2) Fuel degradation during storage should be prevented through providing adequate cooling in order not to exceed fuel temperature limits, under all situations.
(3) Sub-criticality of the spent fuel is to be maintained under normal and accidental conditions.
(4) Radiological shielding of the spent fuel should protect plant operators, the public and the environment from receiving radiation doses in excess of regulatory limits.
(5) Environmental protection should be assured by minimizing the release of radioisotopes.
(6) Fuel retrievability must always be possible.

In wet storage, shielding is provided mainly by the depth of water above the fuel. The SAR should describe the means of maintaining the design level. This might include a description of the engineered system to supply make-up water, an analysis of the ability of the pool to withstand external events (specified for the site of the facility) and a description of any engineered means to prevent raising the fuel too high.

4.5.4.2 Dry Storage Safety

Dry cask storage of spent fuel is among the safest of all the phases of the nuclear fuel cycle. The basic safety goals that must be met are to ensure that (a) sufficient shielding is provided so that workers at the facility are not exposed to hazardous levels of radiation, and (b) the fuel is contained so that any release of radioactive material from the casks to the surrounding environment is reliably prevented. To ensure that dry cask storage systems provide adequate shielding and containment, such systems are designed to meet the following requirements: (1) fuel cladding must maintain its integrity while in storage; (2) high temperatures that could cause fuel degradation must be avoided; (3) accidental chain reactions (“criticality”) must be prevented; (4) effective radiation shielding must be provided; (5) radiation releases must be avoided; and (6) fuel retrievability must be ensured in case any problem arises. For some fuel types such as those of Magnox reactors, fuel being a metal, not an oxide, corrodes relatively rapidly in
water (if the chemistry is not controlled extremely carefully). Effective regulation and monitoring is essential to ensure high-quality construction of the casks and supporting facilities. The system must be designed to be safe not only during normal operations, but in the event of plausible accidents as well, including earthquakes, tornadoes, or plane crashes.

The primary safety related systems for radiological protection during dry storage are shielding, containment and contamination control. Shielding is provided by the storage structures which are designed to attenuate radiation levels to values acceptable to the Regulatory Body. The SAR should describe the shielding system with regard to its ability to limit both gamma and neutron radiation to these acceptable levels. The description should show that there will be sufficient shielding when moving the fuel from the storage pool to the dry storage facility and that there will be no unreasonable limitations on access time by operating personnel. The SAR should describe how the storage system will contain the radionuclides present in the fuel so that the dose limits are not exceeded.

Dropping a fuel container is of particular significance when fuel is transferred from a pool to dry storage. A complete analysis of a dropped fuel container should be done for each of the several possible scenarios. Examples include:

— Drop of a basket or an empty transfer cask within the pool;
— Drop of a basket into a silo or pool;
— Drop of a basket from a transfer cask onto the ground;
— Drop of a loaded transfer cask (with a basket inside) into the pool or onto the ground;
— Drop of a storage cask (loaded or unloaded) within the pool;
— Drop of a loaded storage cask onto the ground;
— Vehicle accident during transfer.

The design should anticipate the possibility of an accidental drop anywhere between the loading facility and the storage site. The design of dry fuel storage facilities should allow for any consequences likely to result from the redistribution or the introduction of a moderator as a consequence of an internal or external event.
One important concern related to the long-term behavior of the fuel is degradation of the fuel cladding as it is exposed to the high temperatures generated by the spent fuel in a dry storage environment, resulting in a potential contamination risk. For this reason, a strict limit on the maximum temperature for dry storage is specified (effectively a limit of 380°C in the U.S.). A Standard Guide for Evaluation of Materials Used in Extended Service of Interim Spent Nuclear Fuel Dry Storage Systems has been issued by the American Society for Testing of Materials (ASTM-C 1562).

4.5.5. Accidents in Storage Facilities:

A few situations have highlighted the need for effective regulatory oversight. In 1996, for example, after the fuel had been loaded into a cask at the U.S. nuclear plant at Point Beach, Wisconsin, and the cask lid was being welded into place, hydrogen inside the cask ignited, lifting the three-ton lid approximately 3 inches and tilting it at a slight angle. The spent fuel was not damaged, and no measurable radioactivity was released. Nor were there any unanticipated exposures of workers. It appears that the hydrogen resulted from an electrochemical reaction between the zinc coating used in the storage canister and the borated water in the spent fuel pool. Similarly, cracks were found in the welds of the inner lids of some casks, and if there had been cracks in both the inner and outer lids, helium could have leaked out and moist air could have leaked in, increasing temperatures and causing additional fuel corrosion.

4.5.6. Safety in Transport of Fuel Materials:

Spent fuel transportation is a vital link to fuel reprocessing. The majority of the spent fuel transportation activities accomplished until now are associated with reprocessing. Several companies have been developed to provide the transportation service, operating a number of spent fuel transportation casks that have been licensed and employed in compliance with national and international regulations. The extensive industrial experience in spent fuel transportation accumulated in the past several decades, including an excellent safety record, will be entirely applicable to spent fuel
transportation operations for other activities in the fuel cycle backend, such as interim storage and disposal. In fact, concerns are raised on the future transportation of spent fuel from reactor sites to repository is regarded as a big challenge in countries like the US, for example, because of the sheer quantity and long distance involved [38].

Based on the 1971-1990 accident data, DOE calculated accident and incident rates for commercial spent fuel shipments to a repository. Based on this analysis, DOE recommended use of a truck accident rate of 0.7 - 3.0 accidents per million shipment miles and a rail accident rate of 11.9 accidents per million shipment miles. (Transportation of Spent Nuclear Fuel and High-Level Radioactive Waste to a Repository: Fact sheet issued by State of Nevada Nuclear Waste Project Office).

Safety regulations for fuel transport have been well documented in detail and are strictly enforced, as most of the transport takes place in public domain. Co-location of two or more of the fuel cycle facilities, such as reactor and the reprocessing facilities would be an added safety feature in reducing the public radiation exposures and aid proliferation resistance.

4.6 FUEL REPROCESSING FACILITIES

4.6.1. Introduction:

Fuel reprocessing is the technology for recovery and recycle of unused nuclear fuel as well as bred fissile material from the fuel subassemblies discharged from the nuclear reactor after irradiation. For a complete review of fuel reprocessing, its techniques and applications to different type of fuels, see Ref.4.35 and 4.36. The present status of reprocessing and trends in reprocessing technologies and strategies are discussed in detail in Ref. 4.37. It is well appreciated that only with the closed fuel cycle long term energy sustainability can be ensured. Proliferation resistance built into the process, in addition to physical protection and administrative control, is a highly desirable innovative feature. Use of reprocessing will also reduce the volume for waste storage
requirement. If partitioning and transmutation option is exercised, this will reduce long term radiotoxic inventory.

Current reprocessing design and development have the following main objectives:

- Ensuring the safety of operators as well as general public
- Releasing minimum quantity of effluents
- Generating minimum inventory of wastes.
- Conditioning of the process waste to meet safe interim storage and transport requirements, as well as long term final disposal.

With the state-of-art technology, it is now possible to design and operate fuel reprocessing plants with as low an environmental release as any other conventional chemical industry. For instance, it has now been demonstrated that the volume of waste generated in the reprocessing operations have considerably reduced in recent years. The dose to the public due to waste discharged from reprocessing has also shown a steady decrease. (Ref.4.36). A combination of learning from experience and continuous improvements, modifying both plant and practice, such as introduction of automated operations has reduced the average employee radiation exposures at reprocessing facilities from over 10mSv to 1.5mSv per person pa over the past two decades. (By comparison, the average annual exposure for airline crew is about 2mSv). Radiation exposure of the public has also reduced, largely in line with the reductions in radioactive discharges. As the quantity of radioactivity being discharged has declined each year, the proportion of radiation exposure that is attributable to current discharges has declined. In the UK today, the average annual exposure of individuals due to radioactive discharges is less than 0.1mSv. By comparison, the average annual exposure to individuals in the UK from natural background radiation is about 2.2mSv.

In the aqueous route of reprocessing, extremely high separation factors (also called decontamination factors) of $10^7$ and high recovery of U and Pu ($\geq 99.8\%$) are routinely achieved. The current state of reprocessing of the spent nuclear fuel is primarily based on solvent extraction process. The contemporary design of reprocessing plants has
evolved significantly but still there is ample scope of improvement in the process steps as well as equipment, in the light of stricter norms for performance, economic viability and environmental liabilities. The newer process steps are based on extensive modeling and validation at the pilot plant testing level. Instead of several cycles needed for required purification levels as well as massive cell volumes needed to house equipment, there is emphasis on a single cycle flow sheet with lesser number of process steps and small continuous equipment with matching decontamination factors. The long term goal of reducing the radio toxicity of the waste has necessitated separation of actinides and long lived fission product isotopes from the high level waste produced in Purex process. Several new processes employing novel reagents have been designed for effective partitioning of long lived isotopes like $^{99}$Tc, $^{129}$I as well as actinides.

4.6.2. PUREX Process

Several solvent extraction systems were explored for reprocessing of nuclear fuel, before an efficient extraction system was identified. The combination known generically as Purex, which utilizes the extractant tributyl phosphate (TBP) mixed in a largely inert hydrocarbon solvent, has now replaced all earlier solvent extraction media. The Purex process has a number of advantages 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 is still being used in all commercial reprocessing plants currently operating.

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 viable up to a burn-up of at least 100 GWD/t, radiation and chemical degradation of the extractant being the limitations for processing fuels of higher burn-up. A typical schematic of conventional PUREX process (Ref. 4.39) is shown in Fig.4.4.1.
The major steps in the Purex process are:
(1) Head end process- the fuel is received, chopped and prepared for dissolution. The U, Pu, minor actinides and fission products go into solution in nitric acid. The hull is monitored for residual fissile material content and treated as solid waste.
(2) 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.
(3) Purification of U and Pu,
(4) Concentration of U and Pu and its storage and
(5) Concentration of high active liquid waste
(6) Recovery of minor actinides and long lived fission products from high level waste
Variations of the flow sheet include recovery of uranium and plutonium without their mutual separation (a choice focusing on proliferation resistance and economics) followed by co-precipitation, partial separation of Pu from U, crystallization for selective removal of uranium, etc. However, these variations do not significantly affect the safety issues in reprocessing.
The recent studies are focused on improvement in the process and solvent extraction equipment performance enabling reduced operating losses, maximized recoveries and possible lowest environmental burden. Late partitioning without intercycle evaporation and higher acid flow sheets are adopted for high Pu bearing fuels especially of fast reactors.

4.6.3. Pyrochemical Processing:

Pyrochemical processes use molten salts and molten alloys which are highly resistant to radiation (Ref.4.40). Hence short cooled fuels can be reprocessed using these processes. The process volumes for a given amount of fuel are very low compared to aqueous processes and hence the plant will be more compact which also enables co-location of the reprocessing plant with the reactor and refabrication plant. The waste produced by these processes is in the form of solid and the volume of waste is less compared to PUREX process and hence waste management problems are less. It is estimated that the volume of the TRU waste generated from a pyrochemical reprocessing plant will be around 12 times less than that from an aqueous reprocessing plant (Ref. 4.35). In the absence of aqueous reagents, criticality related problems are also less, which could enable the processing of plutonium rich fuels.

Another specific advantage of using the pyrochemical processing techniques is that the minor actinides are co-deposited along with the fuel materials and can be incorporated in the fuel for burning them and producing energy.
Fig. 4.6.2 Pyrochemical Reprocessing Flow sheet for Metallic Fuels
Fig. 4.6.3 Pyrochemical Reprocessing Flow sheet for MOX fuel
Pyrochemical processes have been developed for metallic fuels as well as oxide fuels. The molten salt electrorefining process, for reprocessing the U-Pu-Zr alloy fuels of the Integral Fast Reactor IFR, was developed by ANL, USA. The pyrochemical method for reprocessing of oxide fuel (in combination with vibro-pac method of fuel fabrication) was developed by RIAR, Russia. Recently, USA, Japan and Korea are developing a process to convert oxide fuel of LWR into metal form for fast reactor using pyroprocess. The pyrochemical process flow sheet based on the molten salt electrorefining process for metallic fuels developed by ANL, USA is shown in Fig. 4.6.2. The flow sheet shown in Fig. 4.6.3. is based on the oxide electrowinning process for oxide fuels developed by RIAR, Russia. There has been a renewed interest worldwide in pyrochemical methods not only for processing but also for partitioning and transmutation of transuranium elements.

To the extent that the pyrochemical processing promises smaller waste volumes in solid form and have much less criticality implications, they represent innovative processes for reprocessing metallic as well as oxide fuels with high burn-up. However, these processes operate at high temperatures and involve handling of molten salts. The safety aspects of these processes therefore need to be studied in greater depth. This manual focuses on aqueous reprocessing techniques based on Purex process, which have reached a high level of industrial maturity and represent default options for reprocessing.

### 4.6.4. Safety Issues in Reprocessing

Among the nuclear fuel cycle facilities other than reactors, reprocessing facilities are the most complex with respect to safety analysis. The potential safety hazards to which fuel reprocessing facilities are prone include criticality excursions, radiation exposure, chemical reactions, fire and explosion.

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 part of the plant. Highly irradiated fuel is processed and the radioactive materials in process are in dispersible forms, such as solutions, powders and are subjected to vigorous
chemical and physical reactions. Hence control of radiation exposure to operators and possible release of major inventories of radioactivity is one of the main issues with regard to safety of reprocessing plants. In addition, flammable, combustible and explosive materials, such as tributyl phosphate-organic solvent mixtures are used in the reprocessing process. Red oil formation leading to runaway reaction and explosion is known to be caused by violent TBP-nitric acid reaction and subsequent rapid pressurization (typical place of occurrence: HLW evaporator or Pu evaporator) The use of high concentrations of nitric acid and oxidising / reducing agents in the process introduces problems of corrosion in the process equipment and piping, which can translate into leakage incidents.

4.6.5. Accidents in Reprocessing Plants:

4.6.5.1. Leakage of Radioactive Liquid:

One of the most recent major incidents in a nuclear facility was the substantial leakage of radioactive liquid on 21 April 2005 in the feed clarification cell of the THORP plant at Sellafield in Great Britain. The THORP plant – operated by the British Nuclear Group Sellafield Limited (BNGSL) – is designed to reprocess irradiated fuels produced by advanced gas-cooled reactors (AGR) and light water reactors. Such uranium oxide fuels are enriched in uranium-235 by up to 5%. The plant had already reprocessed around 5,700 tonnes since its commissioning in 1994. During the incident in April 2005, about 83 m³ 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 said to come from a fractured pipe running between two accountancy tanks. The origin of this fracture is not yet known (welding error, mechanical stress or corrosion, etc.).
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.

4.6.5.2. Red Oil Explosion:

Red oil is the name of a substance of nonspecific composition formed when an organic phase consisting of TBP and diluent in contact with concentrated nitric acid is heated above 120°C under reflux. At temperatures above 130°C, the degradation of TBP, diluent, and nitric acid proceeds at rates fast enough to generate heat and voluminous amounts of detonable vapor. 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 Ref.4.41, 4.42, 4.43).

Three red oil events have occurred in the Department of Energy’s (DOE) defense nuclear facilities complex (complex): at the Hanford Site in 1953, and at the Savannah River Site (SRS) in 1953 and 1975 (Ref. 4.44, 4.45). A red oil explosion also occurred in 1993 at the Tomsk-7 facility in Seversk, Russia. A recent report by Defense Nuclear Facilities Safety Board, USA has highlighted the conditions under which red oil formation and explosion can take place, means to avoid such incidents (Ref. 4.46).

4.6.6. User requirements, INPRO Assessment parameters and Acceptance Criteria

UR 1.1- Robustness of design (simplicity, margins):

IN 1.1.1: Robustness of design (simplicity, margins)
Technology employed in the plant/facility should be robust and reliable. If the flow sheet involves concentrations of Pu and U in organic streams at values well below the theoretical loading limits, any minor variations in the organic/aqueous flows may not result in loss of Pu to waste streams or formation of third phase. The sensitivity of the
flow sheet to variations in flow ratios must be analysed and documented in the safety report. Similarly, the process flow sheet should be relatively unaffected by small changes in temperature.

**IN 1.1.2: High quality of operation**

High quality of operation implies:

a) availability of clear operating procedures and manuals, providing comprehensive data on the permissible range of various parameters;

b) system of recording and analyzing deviations from operating procedures, consequences of the events and methods to avoid recurrences

Thus, operator training is an important route to ensuring quality of operation. Increased emphasis on automation and on-line monitoring would enhance the quality of operation

**IN 1.1.3: Capability to inspect**

Provision for in-service inspection 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.

**IN 1.1.4: Expected frequency of failures and disturbances**

The probability of occurrence of various type of failures have been analysed in the context of probabilistic safety assessment of reprocessing plants. (Ref 4.47-4.49). The equipment failure database is usually derived from the data available for reactors. (e.g. Ref. 4.50-4.51)

Failure probabilities for various events such as loss of cooling water to high level waste storage tanks have been worked out (Ref. 4.48). The failure probabilities for various events in the innovative plant should be demonstrated to be less than the reported probabilities.
IN 1.1.5: Grace period until human actions are required

A minimum period of 30 minutes is envisaged as grace time until human actions are required 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 the hot cells or glove boxes should 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, complete the evacuation of operating areas and complete other such safety actions.

Build up of radiolytic hydrogen in waste storage tank can take place in the event of air sparging system failure. Availability of enough vapour space in liquid waste storage tanks can ensure that radiolytic hydrogen level can be kept below explosion limit for a minimum period of eight hours.

IN 1.1.6: Inertia to cope with transients.

The flow sheet must be robustly designed such that transients in flows do not lead to large losses of fissile material to waste streams. Sufficient heat transfer area should be available for liquid waste storage tanks to dissipate the large inventory of decay heat by natural convection in the event of transients in the cooling water flow.

UR 1.2: Detection and interception of deviations from normal operational states in order to prevent anticipated operational occurrences from escalating to accident conditions.

The safety analysis report must clearly specify the regime of safe operating conditions for various equipment and processes. Provisions for detecting malfunctions should be clearly described. Reliable, continuous air monitoring systems to detect release of radioactivity to operating areas, criticality and temperature monitors should be provided, with necessary interlocks and alarm annunciation systems.
**IN 1.2.1:** Capability of control and instrumentation system and/or inherent characteristics to detect and intercept and/or compensate such deviations.

Precise and reliable liquid flow metering devices should be provided for the process streams to ensure that the change in flow ratios that may lead to process malfunction can be quickly detected and corrected. Fail-safe process-interlocks should be provided to maintain the desired solvent to aqueous ratio in the solvent extractors (pulse columns, mixer-settlers or centrifugal extractors) constant. In the event of major transient in a particular flow to the extractor, the interlock logic should 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. Instrumentation systems play a vital role in avoiding run away conditions in evaporators and preventing the formation of red oil. Provision of fire barriers / fire dampers in ventilation systems can ensure that fire does not spread to other areas. The availability of redundant monitoring systems based on different principles will ensure that the deviations from the intended conditions are detected efficiently.

The safety report of the plant should clearly bring out the availability of these features.

**UR 1.3- Probability of occurrence of upsets.**

**IN 1.3.1:** Calculated frequency of occurrence of design basis events

Criticality, fire and large scale leakage of plutonium containing solutions are examples of design basis events for reprocessing plants. Inside the hot cell, no break or leak is allowed by design. Improved materials and welding and inspection practices have to ensure that there are no breaks during the entire life time of the plant. The possibility of fire should be predicted from a comprehensive knowledge of various fire loads in the plant, using codes such as COMPBRN-III (Ref. 4.52). Mitsubishi, Japan has developed the code FEVER that can be used for analyzing solvent fires.
IN 1.3.2: Grace period until human intervention is necessary

“Grace period” is defined as the period available for the operator to take action to prevent the escalation of an event into a major catastrophe. Thus, there is practically no grace period for a criticality event, since the excursion occurs rather rapidly. However, prompt operator action can avert further excursions and consequent release of radioactivity and exposure to personnel. Grace period for such action is estimated to be only a few minutes.

Large scale leakage of process vessels leading to release of radioactivity inside an enclosure such as glove box or hot cell is visualized as one of the design basis events for a reprocessing facility. While the grace period for attending to the large scale leak would depend upon the volume of the process/storage vessel and the leak, a grace period of xx minutes is recommended as the grace period. Since the process solutions in a reprocessing plant include large inventories of combustible organics, a fire incident inside a process enclosure should be attended to expeditiously to ensure that it does not lead to a major fire and an explosion. Grace period available for action in such a case could be 5-10 minutes depending upon the location of the fire and the lay out of the process vessels. The safety manual must clearly specify the design basis events and identify those that may require human intervention within a given period.

IN 1.3.3: Reliability of engineered safety features.

The reliability of the interlocks for ventilation systems needs to be established by standard analytical techniques. The availability and performance of alternate power supply systems based on diesel generators or batteries should be periodically checked to ensure that they act reliably during an electrical power supply failure.

IN 1.3.4: Number of confinement barriers maintained.

The reprocessing plants are characterised by a large inventory of radioactive materials in dispersed forms. Thus the maintenance of confinement barriers is of paramount importance in the operation of the reprocessing plants. In the event that the primary containment (e.g. storage tank) fails, there should be secondary containment systems
(either storage tanks or trays) to hold the leaked liquid. In any case even in the event of such a leakage, the number of barriers to be maintained between the operator and the radioactive material should be two. In the case of glove boxes, this can be achieved by erecting secondary enclosures around the glove boxes where large quantities of fissile material is handled or where the mechanical operations carried out inside the box introduce a higher risk of breach of the glove box panel.

**IN 1.3.5:** Capability of the engineered safety features to restore the INS to a controlled state (without operator actions)

One example of engineered safety feature is a secondary enclosure with exhaust system to restore the negative pressure momentarily lost because of breach of the barrier (e.g., breaking of glove box panel). The automation of operation of these systems is necessary to ensure that the engineered safety feature would intervene to maintain the safety status of the plant.

**IN 1.3.6:** Sub-criticality margins

The $k_{\text{eff}}$ of the system under normal operating conditions should be demonstrated to be less than 0.9. A margin of 0.05 will thus be available to accommodate any process malfunctions, taking into account the uncertainties in the calculation of the $k_{\text{eff}}$. All process equipments in material handling area should be designed for criticality for submerged and water filled conditions.

**UR 1.4: Major release of radioactivity**

**IN 1.4.1:** Calculated frequency of major release of radioactive materials into the containment/confineement.

The calculation of frequency of various events in reprocessing plants is illustrated by Ref. 4.48 which discusses the frequency of events such as boiling of solution in a storage tank. The frequency of occurrence of each event may be quantified by the event tree and fault tree analysis approaches, with due consideration to uncertainty related to equipment
failure rate and human error data. It must be demonstrated by such analysis that the calculated frequency of major release of radioactive materials into the containment/confinement is less than $10^{-4}$/year.

**IN 1.4.2:** Natural or engineered processes sufficient for controlling relevant system parameters and activity levels in containment/confinement.

As an example, the design of the cooling systems of the storage tanks should ensure that in the absence of flow of cooling water, natural convection can ensure heat dissipation to ensure that there is no boiling of the solutions.

**IN 1.4.3:** In-plant severe accident management.

The plant safety procedures should ensure the availability of comprehensive emergency manual clearly outlining duties of various individuals, and the actions to be taken in the event of an emergency. Periodic drills should be conducted to ensure that the operators are well prepared to handle emergencies.

**UR 1.5: Major release of radioactivity:**

**IN 1.5.1:** Calculated frequency of a major release of radioactive materials to the environment.

The calculations based on safety analysis (see IN 1.4.1) should demonstrate that the frequency of release of large quantities of radioactive materials into the environment would be less than $10^{-6}$/year.

**IN 1.5.2:** Calculated consequences of releases (e.g. dose)

The dose to public due to any release of radioactivity to public domain should not exceed 1 mSv.

**IN 1.5.3:** Calculated individual and collective risk.

The calculated values of individual and collective risks should be less than those for existing facilities.
UR 1.6: Assessment to be performed regarding the levels of defense-in-depth.

IN 1.6.1: Independence of different levels of DID

Defense in depth is an important element of safety for all fuel cycle facilities and especially for the reprocessing facilities. For every event scenario, it is important that the defense levels provided are able to act independent of each other. For instance, the prevention of criticality should be achieved by a variety of independent steps such as control of flows and concentrations, safe geometries, instrumentation and administrative procedures. The escalation of major fires may be achieved through fire barriers and dousing/extinguishing systems.

UR 1.7: Safe operation of installations of an INS should be supported by an improved Human Machine Interface resulting from systematic application of human factors requirements to the design, construction, operation and decommissioning.

IN 1.7.1: Evidence that human factors (HF) are addressed systematically in the plant life cycle.

A criticality incident inside a shielded facility (such as the hot cells of a reprocessing plant) does not result in a significant exposure to personnel. However, when a criticality incident occurs in a non-shielded facility such as a fuel fabrication facility or in unshielded areas of the reprocessing plant such as the reconversion laboratories, the exposure to operators can be significant. A review of criticality incidents in nuclear fuel cycle facilities clearly indicates that most of the accidents have human factors as one of the root causes. Thus operator training and periodic certification should be an essential feature of the management system of the fuel cycle facilities. It has also been noted that the criticality excursions could be terminated with prompt operator attention.

The above discussion equally applies to other emergencies such as large leaks. The plant manual must clearly bring out a scheme which would ensure that the operators are fully trained not only to carry out routine operations but also to meet emergencies.
UR 2.1. Use of inherent safety characteristics /passive systems to eliminate/minimize hazards

Inherent safety can be built in the design of reprocessing plants through a careful examination of the major events and introducing innovations that circumvent these events. For instance, the use of borated steels and use of vessels coated with boron compounds can ensure that the criticality is not possible in a process vessel at any concentration. By limiting the temperatures and concentration of nitric acid in the evaporators, red oil formation can be avoided. Use of alternate extractants that are analogous to TBP but have a higher no. of carbon atoms (e.g. tri-n-amyl phosphate) can ensure that third phase formation (which can indeed lead to criticality) can be avoided altogether. Use of air operated motors in place of electrically operated motors is another example of passive safety feature.

IN 2.1.1: System variables (e.g. temperature, pressure).

Pu accumulation due to second organic phase formation and polymerisation (with or without precipitation) for Pu bearing systems are the anticipated process upsets in plants reprocessing Pu rich spent fuels. Thus sufficient margin between third phase formation limits and prevailing organic Pu concentration should be maintained. For solutions containing high concentrations of Pu, the aqueous acidity should not be less than 0.2 M in order to prevent hydrolysis and polymerisation of Pu. To prevent solvent (diluent) flash resulting in fire and/or explosion, the operating temperature of the extractors should be limited to 50 °C.

Continuous monitoring of the temperatures and pressures in the process tanks would provide timely indication of process malfunction. Pressures temperatures and gamma activity levels inside the process enclosures should be monitored to ensure detection of fire and criticality events. The monitoring of concentration of Pu in process streams is vital for not only detecting process malfunction but also detecting accumulation of Pu in certain streams due to the phenomenon of third phase formation.
IN 2.1.2: Expected frequency of abnormal operation and accidents.

The frequency of pressurization due to red-oil reactions in the evaporator has been estimated to be $10^{-6}$ per year (Ref.4.38). Such an assessment is necessary for other abnormal events such as loss of control of flow metering systems, total loss of electrical power leading to ventilation failure etc. Probabilistic safety assessment should be carried out based on modeling of human errors in process operations to arrive at the frequency of criticality events.

IN 2.1.3: Consequences of abnormal operation and accidents.

The consequences of all the abnormal operations that can take place in the plant (e.g. inadvertent closure of valves, change in flows, mixing of solutions, transfer of fissile materials, etc.) should be clearly addressed in the safety manual. Similarly the consequences of accidents such as criticality should be described and it must be demonstrated that in any event the dose to public would remain below the regulatory limits.

IN 2.1.4: Confidence in innovative components and approaches

Wherever innovative equipment or processes are used, the safety analysis should establish that the safety is not compromised. For instance, use of new, corrosion resistant materials for process tanks should be based on extensive long term corrosion tests. Fire resistant materials should be tested as per standard codes to establish confidence levels in their deployment.

UR 3.1 & 3.2. Radiation protection to public and operators:
Remotisation and automation significantly contribute to reduction in radiation exposure; Simulators for training in operation and maintenance and periodic preventive maintenance would result in reduced occupational exposures and better safety. For instance, introduction of preventive maintenance and optimization of inspection have brought down the exposure of personnel for UP2 and UP3 plants in France (Ref. 4.36).
Dose limits for occupational workers are 100 mSv for a defined period of 5 years (20 mSv per year) and for public, 1 mSv per year through all routes (air, water and land), in line with ICRP recommendations.

**UR 4.1: Safety basis of installations to be confidently established.**

**IN 4.1.1:** Safety concept defined.

The safety concepts of the plant must be clearly defined in the safety manual. The manual must typically address a) limits for radiation exposure to workers b) limits for release of radioactivity in routine operations and c) losses of fissile material to waste streams and indicate strategies to achieve these objectives

**IN 4.1.2:** Design-related safety requirements specified

The control of the nature, quantity and concentration of fissile materials along the process line, the control of the geometry (dimension and shape) of the equipment used, under all conditions, and the presence of appropriate neutron absorbers are among the preventative measures which, in combination, can be used to avoid criticality excursions. These design related safety requirements must be described in detail in the safety report.
**IN 4.1.3:** Clear process for addressing safety issues.

The reprocessing plant management must provide a mechanism such as plant safety review committee with adequate independence, for periodic review of the operational status of the plant as well as for ensuring that the regulatory practices are strictly adhered to.

**UR 4.2. RD & D to be performed to achieve understanding of the related phenomena**

**IN 4.2.1:** RD&D defined and performed and database developed.

A strong base in RD&D is a catalyst for the safe operation of reprocessing plants. A number of areas of RD & D are being pursued in various countries towards enhancing the safety of operations of reprocessing plants. These include development of alternate reductants in place of hydrazine, development of corrosion resistant materials for dissolver and other process equipment; on-line monitoring systems for the fissile nuclides; constant volume feeders that would help to avoid drastic changes in flow leading to process malfunctions; alternate solvents for extracting minor actinides and useful long lived fission products from waste solutions.

**IN 4.2.2:** Computer codes or analytical methods developed and validated.

Presently, the versatile Monte Carlo Neutron Transport code KENO, developed at ORNL, USA, is used for criticality studies. This code has been validated internationally and is also continually improved. With more accurate cross-section libraries, the uncertainty level in $k_{eff}$ prediction could be reduced significantly. Codes for probabilistic safety assessment of reprocessing plants need to be developed comprehensively. Ref.4.41, for example, highlights the utilization of risk information for reprocessing facilities in Japan. The application of probabilistic safety assessment to Rokkasho reprocessing plant (Ref. 4.42) and Tokai reprocessing plant (Ref. 4.43) are other examples.
IN 4.2.3: Scaling understood and/or full scale tests performed.
Reprocessing being a complex operation involving large quantities of radioactivity, any new process must be demonstrated at a reasonable scale. The consequence of the scaling up from the pilot plant to the commercial scale must be understood clearly.

4.6.7 Radioactive Waste Management

After reprocessing, more than 99% of the non-volatile radionuclides are retained in the high level liquid waste (HLW), High level solid waste and low level solid/liquid wastes. The HLW is stored, vitrified and finally disposed as solid wastes. Solid wastes are mostly disposed in geological repositories.

Twenty years of continuous operations in industrial reprocessing plants have led to process and waste treatment optimizations. Improved sorting procedures and increased package concentrations have allowed operational waste quantities arising from process and plant maintenance to be reduced. An example is the recent improvement resulting from fuel hulls and end fittings compaction implemented in La Hague where some three to five-fold volume reduction has been achieved (Ref. 4.36) (Fig. 4.6.4).

Fig.4.6.4. UP2 and UP3 plant release into sea (Ref.4.36)

Some of the safety issues related to waste management in reprocessing plants are discussed briefly below.
High temperatures are essential for HLW solidification process. The process should ensure minimum release of the radionuclides to the environment during the vitrification process. Suitable insulation of furnaces, availability of cooling systems, comprehensive analysis of failure modes etc need to be considered with respect to waste management facilities.

The off-gas cleaning system has to be designed to take care of possible fire in HEPA filters.

Wastes need to be monitored for the concentration of Pu, as well as other radioactive nuclides. The monitoring techniques must be reliable and should be periodically qualified.

4.6.8. Decommissioning

With the first generation of reprocessing plants being retired from operation, experience is now being gained in decommissioning of reprocessing facilities. Decommissioning begins immediately following the final closure of a facility and continues to the point of leaving a clear site where the facility once stood. According to generally accepted principles, decommissioning operations comprise three major stages:
- Initial clean-up and preliminary decontamination, where necessary, of plant and related facilities;
- Dismantling and removal of the systems, equipment and pipe work within the facilities, with decontamination as appropriate;
- Demolition or reuse (restricted or unrestricted) of buildings and structures.

There are a number of challenges associated with the decommissioning of these facilities, including the high radioactivity levels inside certain parts (due to fission products) and the presence of various types of contamination (alpha, beta and gamma emitting radionuclides). Consideration must be given to how the radiological hazard will change with time, for example because of the decay of Pu-241 to Am-241. In addition the criticality hazard potential should be taken into account while decommissioning and dismantling plant areas which may contain residual plutonium and/or other fissile material. Another major consideration is the hidden presence of alpha emitters in
confined areas, such as small diameter pipes, where contamination assessment is difficult (for example due to the difficulties involved in alpha measurements). IAEA has published a safety guide (WS-G-2.4 (2001)) for decommissioning of nuclear fuel cycle facilities.

5. CONCLUSIONS

In this manual, the safety issues related to a typical set of nuclear fuel cycle facilities has been discussed. The application of the basic principles and user requirements described in IAEA TECDOC 1434 to the nuclear fuel cycle facilities has been explored. The indicators and acceptance criteria for assessing innovations in fuel cycle facilities have been described.

The approach to safety is based on the application of a defense-in-depth strategy, supported by increased emphasis on inherent safety characteristics and passive features. The end point of the enhanced defense-in-depth strategy is that even in case of severe accidents, the radiation releases would be so low as to ensure no need for evacuation of people living nearby the plant, apart from those generic emergency measures developed for any industrial facility. The innovations should finally result in a robust, sustainable and radiologically and environmentally safe facility.

Compared to safety of operating nuclear power plants, only limited experience and literature is available on the safety in the operation of nuclear fuel cycle facilities. It is obviously due to less number of fuel cycle facilities-years operating experience available and that too in limited number of countries. IAEA has recognized the need for international efforts towards defining safety concepts and regulations for NFCFs. The publications of IAEA on the subject of safety of fuel cycle installations provide a good starting point for the design, construction, commissioning, operation and decommissioning of fuel cycle facilities.
A perusal of this safety manual would provide ample illustration of the wide diversity prevalent in fuel cycle facilities with respect to technologies deployed, safety issues to be addressed and RD & D to be performed. This manual at best provides a broad approach to safety of various fuel cycle facilities, at the same time providing some examples of fuel cycle operations as illustrative guidelines towards further discussion and debate, and to highlight the areas where there are gaps in applications of INPRO methodologies thus defining areas for RD & D. In addition, INPRO methodology has been applied to a hypothetical fuel fabrication plant, using SOL-gel process (see Appendix). This provides a starting point for a member state embarking upon a novel/new fuel cycle facility, to assess the innovations in the facility.

It is apparent that intensive consultations between experts related to each specific area of fuel cycle operation are necessary to ensure that the safety issues are crystallized and the indicators and acceptance limits fine tuned. As the reported literature does not provide quantitative data, there is a need to have an international assessment of the acceptance criteria for the safety parameters, especially in the MOX fuel fabrication and fuel reprocessing areas. It is actually possible to envisage a dedicated safety manual for each stage of fuel cycle operation, such as fuel fabrication or fuel reprocessing, in the same way as a separate manual for waste management is being addressed.

In addition to the development of safety manuals in a phased manner with involvement of experts in various aspects of safety as applicable to fuel cycle facilities, it is also necessary to evolve safety codes and guides addressing the specific parameters indicated in the manual, such as probabilistic safety analysis as applied to fuel cycle facilities, exposure and radioactivity discharge limits, response times, etc. Hence, indicators and acceptance limits for the nuclear fuel cycle facilities can be expected to undergo revisions with evolution in technologies, development of new codes and accumulation of more experience.
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GLOSSARY

The definitions presented in this publication are intended principally for use in the IAEA safety related documents for fuel cycle facilities and may not necessarily conform to definitions adopted elsewhere for other use. In all cases these definitions are identical with or, at least, consistent with those used in the IAEA Nuclear Safety Standards (NUSS) for nuclear power reactors.

accident conditions. Deviations from normal operation more severe than anticipated operational occurrences, including design basis events and severe accidents.

anticipated operational occurrence. An operational process deviating from normal operation which is expected to occur at least once during the operating lifetime of a Facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety nor lead to accident conditions.

authorization. The granting, by a regulatory body or other governmental body, of written permission for an operator to perform specified activities. Authorization could include, for example, licensing, certification, registration, etc.

beyond design basis event. Accident conditions more severe than a design basis event (see facility state).

commissioning. The process during which fuel cycle facility components and systems, having been constructed, are made operational and verified to be in accordance with design and to have met the performance criteria. Commissioning includes both non-nuclear and nuclear tests.

confined / contain. A barrier which surrounds the main parts of a Facility or equipment carrying radioactive materials and which is designed to prevent or mitigate the uncontrolled release of radioactive material to the environment in operational states or design basis events.

decommissioning. Administrative and technical actions taken to allow the removal of some or all of the regulatory controls from the Facility.

defense in depth. To compensate for potential failures, a defense in depth concept is implemented, centred on several levels of protection including successive barriers preventing the release of radioactive material to the environment. The concept includes protection of the barriers by averting damage to the Facility and to the barriers themselves. It includes further measures to protect the public and the environment from harm in case these barriers are not fully effective.
**design basis.** The range of conditions and events taken explicitly into account in the design of a Facility, according to established criteria, such that the Facility can withstand them without exceeding authorized limits by the planned operation of safety systems.

**design basis events.** A postulated event against which measures are taken when designing the Facility. The measures may be preventive (e.g. to prevent criticality, fire etc) or mitigative (e.g. to mitigate consequences of a spill).

Design Basis Events (DBEs) are reference design events of very low probability of which the consequences to the public or likelihood have be acceptable. (See Acceptability Diagram after final design of the FCF). By definition, no PIEs of higher frequency of occurrence can result in higher consequences (worst postulated accident). These reference design events can be: (i) Events the occurrence of which have to be designed out (e.g. criticality event); (ii) Events the consequences of which have to be acceptable (e.g. earthquake) For safety analysis, DBEs are postulated to occur and the most serious consequences to the public should be evaluated according to best estimate scenarios. And the Emergency Planning and Preparedness plan shall define the mitigation measures to make such off-site consequences acceptable. (see also: “safety methodology for design”)

**disposal.** Emplacement of waste in an appropriate Facility without the intention of retrieval.

**diversity.** The presence of two or more redundant components or systems to perform an identified function, where the different components or systems have different attributes so as to reduce the possibility of common cause failure.

**engineered safety features (see safety systems)**

**facility states.**

Operational states: Normal operation, Anticipated operational occurrences

Accident conditions: Design Basis Events, Beyond Design Basis Events

**grace Period:** Period available to the operator in which the event does not escalate into a major failure or catastrophe. Safety actions initiated within this period could bring back the system to a safe condition.

**inherent safety:** The term inherent safety is normally used with respect to a particular characteristic, not to the plant as a whole. A plant has an inherently safe characteristic against a potential hazard if the hazard is rendered physically impossible, without human intervention. For example, a fuel cycle facility is inherently safe against criticality if it cannot attain a critical configuration of material under any circumstance.

**maintenance.** The organized activity both administrative and technical of keeping SSCs in good operating condition, including both preventive and corrective (or repair) aspects.

**modification.** A deliberate change in or an addition to the existing Facility configuration,
with potential safety implications, intended for continuation of the Facility operation. It may involve safety systems, or safety related items or systems, procedures, documentation or operating conditions.

**monitoring.** Continuous or periodic measurement of parameters or determination of the status of a system. Sampling may be involved as a preliminary step to measurement.

**normal operation.** Operation within specified operational limits and conditions (OLCs).

**Operating Limits and Conditions (OLCs):** The OLCs are the set of rules setting forth parameter limits, the functional capability and the performance levels of equipment and personnel for safe operation of a Facility.

**operation.** All activities performed to achieve the purpose for which fuel cycle facility was constructed, including maintenance, and other associated activities.

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

**operator.** Individual workers engaged in the operation of an authorized Facility.

**periodic safety review.** A systematic reassessment of the safety of an operational facility or activity carried out at regular intervals to deal with the cumulative effects of ageing, modifications, operating experience and technical developments, and aimed at ensuring a high level of safety throughout the operating lifetime of the Facility or activity.

**postulated initiating event.** An event identified during design as capable of leading to anticipated operational occurrences or accident conditions. PIEs could lead to a release of significant quantities of radiation and/or radiological materials and associated chemicals in line with the hazards

**protection (or radiation protection).** The protection of people from the effects of exposure to ionising radiation, and the means for achieving this.

**protection system.** A system which monitors the operation of a fuel cycle facility and which, on sending an abnormal condition, automatically initiates actions to prevent an unsafe or potentially unsafe condition.

**quality assurance.** All those planned and systematic actions necessary to provide adequate confidence that an item, process or service will satisfy given requirements for quality, for example, those specified in the licence.

**management.** The members of the operating organization who have been delegated responsibility and authority for directing the operation of a fuel cycle facility.
**redundancy.** Provision of alternative (identical or diverse) structures, systems and components, so that any one can perform the required function regardless of the state of operation or failure of any other.

**safety.** The achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of site personnel, the public and the environment from undue radiation hazards.

**safety analysis report (or SAR).** A document provided by the applicant to the regulatory body containing information concerning the fuel cycle facility, its design, safety analysis and provisions to minimize the risk to the public, the operating personnel and the environment.

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

**safety function.** Function of which loss may lead to radiological or chemical consequences to the workforce, the public or the environment:
- Containment against dispersion of radioactive material and chemical hazards;
  Associated Secondary Safety Functions: (1) Structure; (2) Cooling (evacuation of decay heat) and (3) Prevention of radiolysis;
- Protection against external irradiation;
  - Prevention of criticality.

**safety margin.** The difference between safety limits and operational limits. It is also sometimes expressed as the ratio of the two values.

**safety methodology.** The safety methodology for design, commonly consists of 5 steps:

1. Definition of the Facility design basis data (products, processes, capacities etc...) and definition of the Facility Safety Functions to be fulfilled.
2. Hazard identification: Identification of all (i) external hazards from pre-established list and (ii) nuclear and chemical internal hazards (Facility specific and/or from established list for non nuclear internal hazards). Chemical hazards are taken into account only when leading to nuclear consequences
3. Hazard evaluation
4. Development of event scenarios and identification of Postulated Initiating Events: The hazards identified during the hazard identification step, are linked with the PIEs to produce event scenarios. These event scenarios may be grouped by event and hazard type (e.g., loss of confinement, criticality, fire, etc.)
(3.B) Evaluation of consequences of event scenarios: For each of the event scenarios, consequences to the public, the worker, and the environment should be estimated.

(3.C) Identification of Structures, systems and components important to safety and their safety requirements. For those scenarios of unacceptable consequences (Design Basis Events) identify SSCs to fulfil the safety functions

(3.D) Evaluation of mitigated consequences and likelihood: If they are not acceptable, iterate the evaluation (3.B) and modify the SSCs (3.C) until the results (likelihood or consequences) are acceptable.

(4). Establishment of Operating Limits and Conditions

(5). Safety Justification: Preparation of the Facility Safety case

**safety system.** A system important to safety provided to ensure the safe shutdown of a fuel cycle facility or to limit the consequences of anticipated operational occurrences and design basis events.

**single failure.** A failure, which results in the loss of capability of a component or system to perform its intended safety function(s) and any consequential failure(s) which result from it.

**site.** The area containing the fuel cycle facility defined by a boundary and under effective control of the Facility management.

**surveillance (testing).** Periodic testing to verify that the SSCs continue to function or are in a state of readiness to perform their functions.
APPENDIX

Application of INPRO Methodology to a MOX fuel fabrication plant based on sol-gel technology

1.0 Introduction

Sol-gel technique is an important route for the preparation of MOX fuel in the form of microspheres which can then be used to fabricate fuel pins either by vibro-compacting into a fuel pin or by pelletising the microspheres and loading the pellets in the fuel pin. Of the two options, the vibrocompaction option is especially attractive due to its simplicity. While sol-gel / vibrocompaction technique has been studied by many countries up to pilot plant scale [1-3], an industrial scale plant for fabrication of fuel through sol-gel process has not been set up so far.

Sol-gel based fuel fabrication methods offer several advantages over the conventional powder-pellet route: elimination of problems associated with radioactive dust such as radiation exposure to personnel, amenability for automation and remote handling in view of the fluid like behaviour of the sol-gel derived microspheres, much better microhomogenity in case of mixed oxides, carbides etc., suitability for dove tailing to the reprocessing plant, significantly higher throughputs and lower scrap recycle requirements. Further, sol-gel based fabrication methods are also proliferation resistant, since the uranium, plutonium mixed nitrate solutions from the reprocessing plants can be used as the feed, obviating the need for partitioning of uranium and plutonium.

In view of these advantages, an exercise to apply the INPRO methodology to a conceptual sol-gel fuel fabrication plant (SFFP) is attempted here. The capacity of this plant will be in the range of 5 tonnes of MOX fuel per year (25 kg per day), which is roughly the amount of fuel required per year for a typical 500 MWe fast breeder reactor. There are three sol-gel based processes for the production of microspheres namely, internal gelation process, external gelation process and sol dehydration process. The SFFP is based on the internal gelation process.
2.0 Features of the Process

The process flow sheet for the internal gelation process of MOX microspheres is shown in Fig.A.1. The major advantage of sol-gel process over the powder pellet route is, as mentioned earlier, that the nitrate solutions from the reprocessing plants can serve as the feed for the fabrication plant and thus powder handling can be avoided. However, nitrate solutions of 2.5 to 3.0 M are needed for the sol-gel process whereas the reprocessing plant output will be of the order of 10-15 g/l. Hence, the first step is the adjustment of concentration. The optimum process for this step has not yet been developed since there are no operating plants based on sol-gel process. After this concentration step, the uranyl and plutonium nitrate solutions of 2.5 to 3.0 M (pH approx.2) are cooled to –2°C and mixed with 3 M solutions of urea and ammonium nitrate mixture also cooled to –2°C and the feed solution thus prepared is dispersed in the form of droplets by forcing it through a nozzle which is being vibrated at a known frequency. The droplets are contacted with hot silicone oil at 363 K in a glass column. The hot silicone oil is pumped upwards through the glass column to control the temperature as well as the residence time of the gel inside the gelation column. As the droplets travel down the silicone oil column the gelation occurs to form the gel particles. The gel particles are washed with CCl₄ to remove the silicone oil and then with 2 M ammonia solution to remove ammonium nitrate formed during gelation as well as excess urea. The washing is continued until the conductivity of the solution becomes equal to that of ammonia solution. Then the microspheres are dried at 373 K in air followed by calcination at 773 K and then reduced and sintered in flowing N₂+8% H₂ atmosphere at 873 K and 1523 K respectively, to get (U,Pu)O₂ microspheres of ~98% T.D. The microspheres are vibrocompacted into the fuel pin which is welded and subjected to inspection and qualification steps for irradiation in reactors.

Recycling of the reagents used in the process is essential for reducing the volume of the waste. The CCl₄ used is recovered by distillation which is recycled. Recently, ion exchange based processes have been reported for the recovery and recycle of ammonia [4].
3.0 Safety issues

Criticality accidents, accidental release of hazardous materials, fire and explosion are the major safety issues.

The SFFP is designed to restrict exposure from normal operations to as low as reasonably achievable level (ALARA). The main source term for exposure is neutrons and gamma radiation from plutonium. Some of the design principles are prevention of criticality by design (the double contingency principle is the preferred approach) and confinement of chemical hazards (includes the control of any route into the workplace and the environment). Shielding would be needed for protection of the workers due to higher gamma dose rates.

Typical initiating events for the SFFP are given in the table:

<table>
<thead>
<tr>
<th>INCIDENT</th>
<th>INITIATING EVENT</th>
</tr>
</thead>
<tbody>
<tr>
<td>Leakage of plutonium nitrate solutions</td>
<td>Faulty valve, corrosion of vessels or pipes</td>
</tr>
<tr>
<td>Self sustained chain reaction-criticality</td>
<td>Increase of fissile material concentration</td>
</tr>
<tr>
<td></td>
<td>Overloading of tanks</td>
</tr>
<tr>
<td>Inventory transients</td>
<td>Malfunctioning of level probes</td>
</tr>
<tr>
<td></td>
<td>Faulty valves and pumps</td>
</tr>
<tr>
<td></td>
<td>Operator error</td>
</tr>
<tr>
<td></td>
<td>Leakage from pipes/vessels</td>
</tr>
<tr>
<td>Premature gelation in transfer line of feed</td>
<td>Failure of cooling water leading to increase of</td>
</tr>
<tr>
<td>solution, inside feed tank etc., column</td>
<td>temperature</td>
</tr>
<tr>
<td>Uncontrolled process change</td>
<td>Problems in mixing ratios due to operator error or</td>
</tr>
<tr>
<td></td>
<td>error in control systems such as</td>
</tr>
<tr>
<td></td>
<td>level monitors, valves etc.,</td>
</tr>
<tr>
<td>Spillage of nitric acid used for cleaning</td>
<td>Leakage in pipe lines/vessels</td>
</tr>
<tr>
<td>of vessels between batch to batch</td>
<td></td>
</tr>
</tbody>
</table>

3.1. Prevention of criticality

Nuclear criticality safety is achieved by maintaining the amount of fissile materials in sub-critical limits during anticipated events and design basis accident conditions:

i) Ever safe geometry will be adopted for the design of the vessel
(ii) The transfer of solutions from one vessel to another is always metered and level monitors are in place. Visual/ audio alarms are actuated by the level probes when the set levels are exceeded. Level monitoring is planned wherever possible.

(iii) Presence of appropriate neutron absorbers (e.g. construction of storage vessels using borated steel or external thermalisation of neutrons and absorption in cadmium).

3.2 Confinement against internal exposure and chemical hazards

Protection of workers, public and environment from releases of hazardous material in operational states as well as design basis events/conditions is given the highest level of importance.

- Containment is the primary method for protection against the spreading contamination. This is achieved by maintaining the different zones of radioactive area under a pressure gradient and following the administrative procedures strictly during the movement of personnel and materials between different zones.

- The design takes into account the performance criteria for the ventilation and containment systems including (i) the pressure difference between different zones (ii) the air renewal ratio (iii) the type of filters used (iv) the maximum differential pressure across filters and (v) the appropriate flow velocity at the openings in these systems (e.g. acceptable range of air speed at the opening of a hood).

3.3 Protection of the workers

- Ventilation systems are used as one of the means of minimizing the exposure of workers to hazardous material that may become airborne in the facility.

- Careful design of containment and ventilation systems is done to minimize the need for use of respiratory protective equipment.

- Primary filters are located as close to the source of contamination as practical. This is to minimize the buildup of activity in the ventilation ducts. Having multiple filters in series is adopted since it avoids reliance on a single barrier.
• Monitoring equipment such as differential pressure gauges (on filters between rooms or between a glove box and its operating area), criticality monitors, air monitors, fire alarms, area radiation monitors are installed as necessary.

• Personnel radiation monitoring instruments are provided for radiation protection.

• In order to prevent the propagation of fire through the ventilation ducts and to maintain the integrity of firewalls, ventilation systems are equipped with fireproof shutters.

• To facilitate decontamination, the walls, floors and ceilings in areas of the FFF where contamination may exist are made nonporous and easy to clean.

• Class III power supply is provided for ventilation back-up and Class I and II for health physics instruments

3.4 Environment protection

Uncontrolled dispersion of radioactive substances to the environment from accidents can occur if the containment barrier(s) are impaired. The design of the SFFP, therefore, provides for monitoring the facility environment and identification of barrier breaches. In addition, ventilation of these containment systems, with discharge of exhaust gases through stack via a filter, reduces radioactive materials to very low levels. Efficiency and resistance of the filters to chemicals (e.g. CCl₄, ammonia), temperature of the exhaust gases and fire conditions are taken into consideration. Scrubbers are provided for the gaseous effluent from the drying and calcination vessel to remove ammonia and carbon tetrachloride.

3.5 Fire and explosion

Fire in SFFP can lead to dispersion of radioactive or toxic materials by destroying the containment barriers or cause a criticality accident by modifying the safe conditions. Therefore, the design of the SFFP takes into account fire safety on the basis of a safety analysis and implementation of defence in depth (prevention, detection, control and mitigation).

Special fire hazard analysis is carried out for:
- Processes involving H₂ (reduction and sintering of MOX)
- Storages of reactive chemicals (NH₃, HMTA, urea, nitric acid)

Explosions can occur due to gases (H₂ used in reduction and sintering furnaces etc.) and chemical compounds (ammonium nitrate). Explosion is prevented by using gas mixtures containing nitrogen or argon mixed with hydrogen instead of pure hydrogen. It is ensured that the pH is always above unity and the temperature is below ambient, when nitrate solutions are mixed with HMTA and urea to prevent formation of explosive mixtures. Similarly, reduction of the volume of aqueous waste containing ammonium nitrate, HMTA, urea and ammonium hydroxide will be carried out only after removing the nitrate ion. Care is taken (a) to store HMTA, urea in the area and acids separately (b) to allow only solutions of HMTA/urea and not solids in the active plant area (c) not to mix acidic and alkaline waste and to carry out the distillation of carbon tetrachloride under low temperature and pressure conditions to prevent any possible decomposition. Electrical short circuit is prevented in calcination and sintering furnaces. Suction transfer rather than pressure transfer is used for transfer of microspheres and solutions inside glove boxes to avoid pressurization of the glove boxes. Accidental ingress of ammonia into oil tank which can lead to explosions is completely eliminated by appropriately sequencing the wash operations. Runaway increase of oil temperature to flash point is avoided by suitable temperature controller.

3.6 Flooding

Flooding can lead to the dispersion of the radioactive materials and changes in neutron moderation conditions. Prevention of leakage of cooling water of the process vessels and drying, calcinations and sintering furnaces is ensured by using metal hoses and quick release couplings for the end connections for water lines. Wherever possible, furnaces are attached to the bottom of the glove boxes such that the cooling water lines to the casing do not enter the glove box thereby limiting the possibility of flooding the glove box. (Some parts of the furnace which are inside the glove box may however be cooled by process water to prevent glove box surface from heating up during furnace operation). Water lines carrying below ambient temperature water are insulated to prevent condensation of water which can lead to formation of water pools inside the glove boxes.
The criticality analyses takes into account the presence of the largest volume of water within the considered room as well as in connected ones. Walls of the laboratories housing the glove boxes are designed to withstand the water load in order to avoid any domino effect. Sensors are provided to detect water leak in various laboratory areas which shall be interlocked with the power supply for the pumps to prevent major disasters.

3.7 Leaks and spills

Leaking of components such as pumps, valves and pipes can lead to dispersion of radioactive material (MOX, uranyl nitrate and plutonium nitrate) and toxic chemicals (e.g. ammonium hydroxide). Leaks of hydrogenous fluids (water, oil, etc) can change the moderation in fissile materials and reduce criticality safety. Leaks of flammable gases (H₂) can lead to explosions and/or fire. Multiple leak detection systems are used in such cases. Vessels containing significant quantities of plutonium nitrate solution are equipped with level probes and alarm features to prevent overfilling. The cooling water system incorporates monitoring equipment for the temperature and flow at the outlet which can provide information on the overall health of the system such as any possible choking of the lines etc. Breakage of the silicone oil column will lead to spilling of oil as well as MOX gel. A collection tray is provided on the glove box floor with sufficient capacity to contain one column charge.

4.0 Instrumentation and control (I&C)

4.1 Instrumentation

Instrumentation is provided to monitor facility variables and systems over the respective ranges for (i) normal operation (ii) anticipated operational occurrences and (iii) design basis accidents in order to ensure that adequate information can be obtained on the status of the facility and proper actions from operating procedures or automatic systems can be undertaken.

4.2 Control System

Passive and active engineering controls are more reliable than administrative control and hence are preferred for normal operational states and DBE conditions.
Automatic systems such as metering pumps and level probes are used to control the ratios of uranyl and plutonium nitrate contents as well as the ratio of HMTA + urea mixture to the metal nitrate solutions. Similarly, temperature of gelation, temperature of feed solution etc. are controlled within the operational limits and conditions.

### 4.3 Safety related I&C during normal operations

During normal operations, the safety related I&C includes:

- **Instrumentation and process control:** (e.g. temperatures, pressures, flow rates, concentration of chemicals and/or radioactive materials or tank level). One of the key process control parameters related to safety is the concentration of hydrogen in the gas being fed to sintering furnace.
- **Control and monitoring of ventilation:** Mainly on differential pressures across HEPA filters and airflows.
- **Radiation dosimetry:** Sensitive dosimeters with real time display and/or alarm (for external exposure); Continuous air monitors (for internal exposure).
- **Gaseous and liquid effluents:** Real time measurements are done to prevent mixing of acid and alkaline waste as well as to limit the level of nitrates to regulatory limits.
- **All laboratory areas with fissile and/or toxic chemical materials are equipped with fire alarm**
- **H₂ detectors are used near the calcination and sintering furnaces where leakage of H₂ can occur**
- **Criticality:** Radiation detectors (gamma and/or neutrons), with audio-visual alarms for initiating immediate evacuation from the affected areas.
- **Chemical release:** Detectors and alarm systems are installed in areas with a significant chemical hazard (e.g. ammonia, Carbon tetrachloride).
- **Release of effluents:** The devices used to measure releases of gaseous and liquid effluents under operational states are also capable of measuring the releases in case of DBEs.
- **Emergency power supply is provided for critical equipment**
• Emergency exhaust system is provided to maintain the glove boxes at negative pressure in the event of failure of ventilation of the glove box due to failure of power, blower etc.

• Emergency cooling water system for furnaces using gravity flow to take care of cooling water stoppage due to power or pumping system failure etc.

• Single button emergency shut down system for high temperature sintering furnaces to be used in case of failure/malfunctioning of control circuits

• Thermocouple break protection to switch off power when temperature control thermocouple fails.

• Routine checks will be carried out for the functioning of all the control systems which include built-in checks as well as checking of auxiliary systems such as manual overrides. All sensors and operating/control hardware will be checked.

5.0 Radiation Protection in MOX FFF

Typical design principles for safety are:

• Minimization of inventory

• Shielding inside the glove boxes

• Use of combined shielding – consisting of polythene, lead and cadmium for neutron absorption

• Minimization of human intervention by means of automated processes

• Barriers to prevent contamination

• Air, surface and personnel monitoring

6.0 Criticality Safety

• Priority of technical measures for criticality safety: ever-safe geometry and mass control, neutron poisoning and moderation control

• Administrative control in area with low inventory only

• Provision to cope with double failure
User requirements, INPRO Indicators and Acceptance Criteria for Fuel Fabrication Facilities

At the outset, it must be emphasised that safety analysis of a sol-gel based fuel fabrication plant has not been discussed in detail in open literature. Also, there is no report on operation experience on a plant based on this process. Hence, the purpose of this exercise is two fold: to provide an illustration of the elements of innovation in fuel fabrication and a methodology for their assessment based on INPRO, and to indicate typical areas where further work is necessary to evolve a robust and safe fuel fabrication technology based on sol-gel process

UR 1.1 Robustness

IN 1.1.1 Robustness of design (simplicity, margins):

The sol-gel fuel fabrication facility is designed for withstanding earthquakes. For fire prevention, use of fire resistant materials for primary confinement system and use of passive cooling systems for high temperature operation is envisaged. Minimum use of flammable materials is ensured.

For prevention of explosion, Ar-H₂ or N₂-H₂ gas mixtures are used for calcinations and sintering instead of pure hydrogen. Mixing of nitric acid or nitrate solutions with HMTA/ Urea under low pH and high temperatures is avoided.

Maintenance of differential pressure in glove boxes and operating areas, easy access of the equipment in operating areas, automation for process operations, zoning in layout of the plant for hazardous operations, single port entry and exit for personnel and equipment and multiple levels of filtration are used for control of contamination.

Mass control of fissile material, on-line nuclear material accounting (NUMAC), use of ever-safe geometry, layout with sufficient separation between equipment as well as fissile
material storage tanks, minimization of hydrogenous materials in process and use of neutron absorbing materials are ensured for criticality control.

**IN 1.1.2** High quality of operation.

The distinctive feature of the fuel fabrication facility is the presence of large inventories of mixed oxide. These are usually in the form of microspheres which are free from loose powder and can therefore be transferred by vacuum suction, exploiting the fluid like behaviour of the microspheres. However, microspheres of poor quality can give rise to powders by erosion. Alternate procedures would therefore be available for handling powders during such eventualities to ensure that the main advantage of the sol-gel process – namely the absence of powder generation- is not lost. A high quality of operation is ensured, by avoiding spillage of the fuel materials inside the enclosures which would lead to long term accumulation leading to increased dosage to the operator. Intensive training is given to operators to ensure that human factors do not lead to unexpected accumulation of fissile material in any part of the plant and leading to criticality. Adherence to administrative procedures is strictly implemented.

Automation/ Remotisation of all the process steps is ensured to minimize exposure of personnel to radiation. It must be emphasised here that sol-gel process is much more amenable to remote operation than powder-pellet processes. Every operator is made to undergo training in the regular operations and emergency procedures every six months.

**IN 1.1.3** Capability to inspect

Monitoring systems are installed to provide information on the levels of liquids in process vessels, temperature, flow of cooling water and gas flow /vacuum in furnaces, radiation levels in glove boxes and operating areas and pressure drops across filters. Fire detectors are installed in all glove boxes.

**IN 1.1.4** Expected frequency of failures and disturbances:

Temporary loss of power leading to premature gelation which in turn can lead to choking of feed lines, leakage of solutions, ventilation failure, and loss of cooling water flow to
furnace are the possible events of failures. Based on operating experience and available literature, it can be concluded that the frequency of occurrence of these failures and disturbances will not exceed $<10^{-2}$ per year.

**IN 1.1.5** Grace period until human actions are required.

In the event of ventilation failure (e.g. due to loss of electrical power), a grace period of one hour would be available to restart the ventilation systems. This is due to the leak tightness of the glove boxes systems (typically less than 0.5 % box volume per hour at a pressure differential of 4 inch water column) and also because the handling of radioactive materials in containments.

In the event of loss of flow of cooling water to the furnace, the gravity fed cooling water supply would be able to supply cooling water for a period of at least two hours, within which the furnace can be brought to a safe shutdown in case the pumped cooling water does not get initiated. This can be ensured by designing a water storage system with adequate capacity.

In the event of chocking of feed line due to premature gelling, only process upset is expected and there is no safety implication.

The operation manual lists the anticipated incidents, action plan and time before which the actions have to be completed so that the incident does not escalate into an emergency.

**IN 1.1.6** Inertia to cope with transients.

Adequate flow of cooling water to the furnace is ensured, combined with a flow monitoring system, to ensure that in the event of transient reduction in coolant flow rate, the temperature of the furnace shell will not rise above safe limits. The leak tightness of the glove box (usually $<0.5\%$ box volume/hr when inside of glove box is at a pressure of $+100\text{mm water column}$) is such that in the event of temporary loss of the glove box ventilation or under slight positive pressure, the radioactivity level in the operating area would not cross regulatory limits.

**UR 1.2 Detection and interception of deviations from normal operational states**

**IN 1.2.1** Capability of control and instrumentation system and/or inherent characteristics to detect and intercept and/or compensate such deviations.
The fuel fabrication facilities incorporate several critical systems such as glove boxes, furnaces, vacuum systems etc. Thus, instrumentation and control play an important role in ensuring the healthiness and safety, ensuring that they operate in a safe manner. The safe operating conditions for every system indicating different limits for alarm and shutdown conditions are clearly defined. Furnaces are equipped with power supply shut down feature to prevent escalation of temperature.

**UR 1.3 Frequency of occurrence of accidents**

**IN 1.3.1 Calculated frequency of occurrence of design basis accidents**

While all the postulated initiating events described earlier may not be analysed quantitatively, it is ensured that the safety analysis includes probabilistic safety assessment of important initiating events such as a) Earthquake, b) Fire, c) criticality and d) Major release of radioactivity.

The description of the sol-gel process is given in Section 2.0. In a sol-gel facility plutonium and uranium nitrate solutions are mixed and a broth is formed by the addition of urea and HMTA. This broth is fed to a vibrating nozzle and the droplets formed fall through a hot silicone oil column to gel into solid microspheres. The vessels that hold the plutonium solution as well the mixed uranium-plutonium solution are made of ever safe geometry with respect to criticality. However, in the event of simultaneous multiple failures of a) uranium addition and b) addition of HMTA/Urea, the plutonium nitrate solution will not undergo gelation and will end up in the geometrically unsafe holding the silicone oil leading to criticality. A gamma ray detection system can be incorporated to detect the presence of plutonium in the silicone oil tank and stop the accumulation of Pu in the silicone oil tank, by stopping the flow of feed Pu solution. In the event that this instrumentation also fails, criticality can occur since batch size is expected to be of the order of 5 kg.

Fig.A.2. analyses the sequence of events leading to criticality. The individual events leading to the failure are estimated to have a probability of $10^{-3}$, $10^{-3}$ and $10^{-2}$ per year respectively. Hence the overall probability will be $10^{-8}$/year.
A chemical explosion can occur if the microspheres, from which HMTA, urea and ammonium nitrate are not completely washed off, are heated in a furnace. By adequate experimentation and optimisation of process conditions, it can be ensured that the chemicals are completely washed off from the microspheres, and the frequency of such an explosion can be demonstrated to be less than $10^{-6}$/year.

**IN 1.3.2 Grace period until human intervention is necessary:**

As mentioned in 1.3.1, the design basis criticality event can occur in silicone oil tank. Action to terminate criticality can be taken by injecting a solution of neutron poison into this tank. Since this can be carried out by instrumented actions, the criticality event can be handled in a matter of a few minutes. A large release of radioactivity into the operating area can take place due to an explosion resulting in breakage of glove box panel and spillage of solution into working area. However, while this would increase air activity in the environment, it can be safely handled by trained staff with adequate precautions including use of respirators. The most important safety related action would be the evacuation of the laboratory area, which can be completed in about 15 minutes. No escalation of the incident is expected in this time period.
IN 1.3.3 Reliability of engineered safety features:
Examples of engineered safety features in the sol-gel fuel fabrication facility would include:

a) Automatic switch over of water supply from pumped cooling water to gravity based cooling water in the event of failure of cooling water pump
b) temperature control systems to shut down calcination and sintering furnaces in the event of total loss of cooling water
c) secondary ventilation systems which would take over in the event of loss of a glove box barrier (e.g. puncture or tear of a glove).

The systems deployed and their reliability is demonstrated through design analysis as well as actual trial experiments before the facility goes into active operation.

IN 1.3.4 Number of confinement barriers maintained:
The most important safety feature of the fuel fabrication facility is the barrier for release of radioactive material into the environment. Pu based materials are handled in glove boxes, whose panels and gloves constitute a barrier. Further, additional barriers such as trays to collect the leaks from glass column of silicone oil are also kept inside the glove box to increase safety. It is ensured that inside the glove boxes the active solutions are always inside closed vessels so that presence of minimum two barriers is always maintained.

IN 1.3.5 Capability of engineered safety features to restore INS to controlled state:
This is dealt with also under 1.3.3. The safety review procedures include a demonstration of the reliability of the engineered safety features and their capability to ensure safety of the system without any operator intervention.

IN 1.3.6 Sub-criticality margins.
The choice of plutonium amount and concentration, design of the process vessels and the lay out of the vessels in the glove boxes will be such that the $k_{eff}$ will be less than 0.90. It
will be ensured by design that even under moderated conditions the value will not exceed 0.95, thus providing a sub-criticality margin of 0.05.

All process equipment in material handling area are designed to eliminate criticality for submerged conditions and fully reflected conditions.

**UR 1.4 Frequency of release of major amounts of radioactivity to the confinement**

**IN 1.4.1** Calculated frequency of major release of radioactive materials into the containment / confinement

The only routes through which a major release of radioactivity into the operating area can take place are: a) damage to glove box panel by an explosion or criticality. frequency of occurrence of major release of radioactivity into the confinement will be arrived at by probabilistic safety assessment. It will be ensured that it is below $10^{-6}$ per operation year.

**IN 1.4.2** Natural or engineered processes sufficient for controlling relevant system parameters and active levels in containment / confinement.

Glove boxes incorporate additional exhaust systems which provide the required face velocity (equivalent to those achieved in fume hoods (40 meter/min)) in the openings created by tearing, loss of gauntlets etc.

**IN 1.4.3** In-plant severe accident management.

A carefully prepared comprehensive emergency plan is available for managing severe accidents (e.g., criticality, explosion, fire, large scale leakage of plutonium solution etc..) that may occur in the fuel fabrication facility. The main components of this plan are:

(a) Cutting off the source, area isolation, emergency evacuation

(b) Activation of on-site emergency plan to prevent the spread into uncontrolled area.

**UR 1.5 Major release of radioactivity to environment**

**IN 1.5.1** Calculated frequency of major release of radioactive materials to the environment – In plants handling only unirradiated fuel materials, a large release of
radioactivity into the environment is not foreseen, and in any case, the frequency is expected to be less than $10^{-6}$ / plant year.

**IN 1.5.2** Calculated consequences of releases (dose).
Dose to Public under normal operating conditions should be $<$1 mSv/annum.

**IN 1.5.3** Calculated individual and collective risk.
Objective would be as low as reasonably achievable.

**UR 1.6. Defence in depth**

**IN 1.6.1** Independence of different levels of DID
Maintenance of negative pressure, ventilation, multiple barriers, system isolation, area isolation and evacuation provide various levels of independence. The safety analysis report of the facility would clearly describe the independence of the levels of defence. Emergency cooling water system based on gravity is provided so that equipment dependent factors such as power failure, non-functioning of pumps etc. are eliminated. Amounts of solutions transferred are controlled using more than one device to provide enhanced safety.

**UR 1.7 Systematic application of human factors**

**IN 1.7.1** Evidence that human factors (HF) are addressed systematically in the plant life cycle.
Human Resource Development is given importance. Incentives are given for upgradation of the qualification of workers. Rotation of shifts and breaks ensures that the operator fatigue is minimized. Workers are periodically trained. Periodic health check up and follow up is made mandatory.

**BP 2 Increased emphasis on inherent safety characteristics**

**UR 2.1 Incorporation of inherent / passive safety**

**IN 2.1.1** Sample indicators: stored energy, flammability, criticality, inventory of radioactive materials:
Fuel material inventory in each stage is strictly maintained as per limits arrived at in safety analysis. Release of radioactivity and release of hazardous chemicals will be confined to the facility by maintaining the radioactive zoning and ensuring proper ventilation barriers between the zones.

**IN 2.1.2 Expected frequency of abnormal operations and accidents:**
This will be arrived at through a probabilistic and deterministic approach.

**IN 2.1.3 Consequences of abnormal operations and accidents:**
The design safety report for the facility comprehensively indicates the abnormal operations (e.g., runaway conditions in furnaces, over pressurisation of glove boxes) and accidents (explosions, fire, criticality, large scale release of radioactivity) and the impact of these operations and accidents.

**IN 2.1.4: Confidence in innovative components and approaches:**
Certain innovative features such as supercritical extraction of the silicone oil from the gelled microspheres and microwave assisted gelation will be tested on a pilot plant scale to ensure higher reliability than those in the existing process.

**BP 3 Risk from radiation exposure**
**UR 3.1 Optimisation of radiation protection to workers**
**IN 3.1.1 and 3.2.1:** Radiation Protection:
Dose limits for occupational workers are 100 mSv for a defined period of 5 years (20 mSv per year) and 1 mSv per year for public through all routes (air, water and land), in line with ICRP recommendations.

**BP 4 Improving confidence in design and safety assessment**
**UR 4.1 Safety basis to be established with confidence**
**IN 4.1.1 Safety concept defined-** Each concept will be tested individually and the safety features demonstrated to be better than the existing techniques. For this, a suitable combination of deterministic and probabilistic approaches will be used. Gaseous waste
release from stack and radioactive effluent generation will be kept as low as reasonable and practical.

**IN 4.1.2** Design-related safety requirements specified- The specific design safety requirements for fuel fabrication plants have been described in Ref. 4.21. The design report of the facility will address all the issues indicated in these reports.

**IN 4.1.3** Clear process for addressing safety issue
The facility will conform to a well established review process by independent regulatory authority, to ensure that all the safety issues are addressed.

**UR4.2 RD&D to achieve understanding of various phenomena to support safety assessment**

**IN 4.2.1** RD&D to be defined and performed and the database has to be developed and checked periodically.
RD&D on the sol-gel process is well documented; however, there are issues such as waste generation and explosive reactions between Urea/HMTA/ammonium nitrate. These will be addressed by RD & D.

**IN 4.2.2** Computer codes or analytical methods have to be developed and validated on benchmark experiments.
Data on heats of combustion of materials such as silicone oil, urea, HMTA and ammonium nitrate as well as data on energy released in the explosion involving these materials need to be generated. Codes will have to be developed to model the explosive reactions involving these materials.

**IN 4.2.3** Scaling should be understood and/or full scale tests performed:
Since the fabrication plant would produce 5 tonnes of MOX fuel per year, a pilot plant of capacity 500 kg/year (2.5 kg per day) would provide adequate safety related inputs to the design of the full scale plant. All the safety related issues such as criticality can be
addressed in this facility. The design of the instrumentation and control systems can be easily extended to the full scale facility without any issues.

**UR4.3  Pilot plant or large scale demonstration facility to be built**

**IN 4.3.1 Degree of novelty of the process**

The process is not novel in the sense that it has been studied by a number of groups. Pilot plant scale facilities have already been built and operated in several countries and reports are available. However, the country developing this technology for the first time needs to set up and operate a pilot plant on an adequate scale, since the operations are significantly different from existing powder-pellet technology. Possibilities of introducing alternate process steps such as microwave induced gelation that result in elimination of silicone oil/carbon tetrachloride handling exist. These features if introduced will be demonstrated in pilot plant scale prior to implementation for production process.

**IN 4.3.2 Level of adequacy of the pilot facility**

A facility fabricating fuel at a capacity of 2.5kg/d (500kg/y) can be considered as adequate for obtaining realistic operating experience fuel needed for FBR.

**IN 4.4.1 & 4.4.2 Use of a risk informed approach**

Detailed studies will be carried out as part of a risk informed approach. Uncertainties and sensitivities will be identified and appropriately dealt with.

**References**

Figure A.1. Flow sheet for the preparation of MOX microspheres