PROBABILISTIC SAFETY ANALYSIS FOR NUCLEAR FUEL CYCLE FACILITIES, AN EXEMPLARY APPLICATION FOR A FUEL FABRICATION PLANT

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Abstract - In order to assess the risk of complex technical systems, the application of the Probabilistic Safety Assessment (PSA) in addition to the Deterministic Safety Analysis becomes of increasing interest. Besides nuclear installations this applies to e. g. chemical plants. A PSA is capable of expanding the basis for the risk assessment and of complementing the conventional deterministic analysis, by which means the existing safety standards of that facility can be improved if necessary. In the available paper, the differences between a PSA for a nuclear power plant and a nuclear fuel cycle facility (NFCF) are discussed in shortness and a basic concept for a PSA for a nuclear fuel cycle facility is described. Furthermore, an exemplary PSA for a partial process in a fuel assembly fabrication facility is described.

The underlying data are partially taken from an older German facility, other parts are generic. Moreover, a selected set of reported events corresponding to this partial process is taken as auxiliary data. The investigation of this partial process from the fuel fabrication as an example application shows that PSA methods are in principle applicable to nuclear fuel cycle facilities. Here, the focus is on preventing an initiating event, so that the system analysis is directed to the modeling of fault trees for initiating events. The quantitative results of this exemplary study are given as point values for the average occurrence frequencies. They include large uncertainties because of the limited documentation and data basis available, and thus have only methodological character. While quantitative results are given, further detailed information on process components and process flow is strongly required for robust conclusions with respect to the real process.

Introduction

Apart from the accepted deterministic safety analysis, whereby usually conservative approaches are applied, the probabilistic safety analysis (PSA) becomes increasingly important. Probabilistic methods are applied in particular to analyze facilities, from which a certain potential hazard for the environment exists. These are for instance plants of the chemical industry and nuclear installations. In the past decades the application of probabilistic methods on nuclear power plants has positively affected the development of PSA methodologies.

In Germany the PSA is seen as a supplement of the deterministic safety analysis, whereby the PSA results should give a realistic picture of the plant concept as far as possible. The results can be used for the optimisation in terms of safety of the plant or also for the evaluation of operating rules they can also extend the basis for regulatory decisions. If one considers that all plants of the nuclear fuel cycle are characterized by handling of radioactive and/or fissile material, whereby frequently the contribution of chemical-reactive and/or toxic substances may not be neglected from the potential of hazard, it becomes rapidly evident that the application of the PSA methodology to such plants is meaningful in every case. The German nuclear regulations do not explicitly require the compilation of probabilistic safety analyses for nuclear fuel cycle facilities. Nevertheless a certain interest on PSA exists on side of the plant operators, as well as at the regulatory body, since a PSA can contribute also to analyze flow charts and plant design in detail and to identify possible undiscovered weak points in order to improve the plant safety.

The available paper is based on a study performed at GRS on the applicability of the PSA methodology on plants of the nuclear fuel supply [1] financed by the German Federal Ministry for Environment, Nature Conservation and Reactor Safety (BMU). Therein first a fundamental concept for the approach is described and second applied this in a generic example from a fuel fabrication plant. The data used thereby are to a large extent generic and do not originate from a certain existing plant. Therefore also the received results are to be regarded as generic and may not be transferred to an existing plant. The goal of this work was rather to demonstrate the applicability of methodology and as well to point out its possibilities and limitations.

Differences between a PSA compiled for nuclear power station and for nuclear fuel cycle facilities (NRN-Facility)

PSA application to nuclear power plants as well as to facilities of the nuclear fuel cycle (NFCF) is based on common principles. Nevertheless some partial substantial changes - mainly due to the different plant design as well as to the kind of the process cycles – are evident. These deviations can briefly be summarized as follows [2]:

Compared with nuclear power plants the nuclear fuel cycle facilities are usually characterized by a larger technological and/or process-justified diversity.

Apart from processing radioactive and/or fissile materials also larger quantities of chemical materials are handled frequently. These materials can act toxically and/or corrosively or can be easily inflammable. Consequently these materials must be included into appropriate PSA evaluations for NFCFs.

The essential potential hazard sources of nuclear reactors - the core and the spent fuel pool - are spatially central arranged. In the comparison to this is the material in NFCFs, which has to be included into the analysis, can be according to the process conditions relatively broadly distributed; it must be e.g. supplied to the production process, processed and stored. This means that under such conditions a spatially relatively expanded plant area must be considered.

NFCFs are more frequently than nuclear reactors influenced by operational- and/or process-conditioned changes. In the intention to improve the production flow continuously and/or to convert new production developments, amongst others also technical equipment can be subject to more frequent changes.

The aspect of human errors must be considered more strongly in case of the NFCFs. Usually a relatively high confidence is brought to the actions of the personnel and/or the operator here. That concerns not only normal operations but also measures for error correction and/or for incident control.

In contrast to PSA for nuclear power plants which have different levels regarding the consequences of end states of the assessed initiating events (from core damage to environmental consequences), PSA analyses for NFCFs have actually only one stage with defined end states which are typical for NFCF-sequences (e.g. release of hazardous material, criticality, violation of regulations).

Basic conception of a PSA for nuclear fuel cycle facilities

Performing a PSA for NFCFs should be proceeded according to the following steps [2]:

Step 1 creation of the bases in management and organisation

The first step covers essentially all activities necessary to create the organisational conditions in order to allow the successful execution of the appropriate PSA.

Step 2 Identification and selection of initiating events

The principal objective of the second step is the provision of a list of so-called initiating events. In principle initiating events are (assumed) disturbances or failures of systems or components which are causing challenges on safety systems [7]; i.e. within a PSA the reliability of the control of these assumed disturbances with the existing relevant safety devices is assessed. Therefore the selection of initiating events within the PSA is very important; an appropriate list must be provided with high accuracy. When generating the list of initiating events the following specified points may be helpful as a guideline:

- Study of the plant characteristic and information composition;
- Incident identification on basis of the plant characteristic;
- Selection of initiating events;
- Provisional identification of undesirable final conditions;
- Identification of safety measures and – functions;
- Compilation of information concerning safety measures;
- Grouping of the initiating events in order to perform the analysis.

The selection of the initiating events, which are to be regarded for the PSA, is in practice usually realized simultaneously to the incident identification. It should be mentioned that - under consideration of the variety of possibilities - it will be difficult to provide a complete list of all possible initiating events. Under the basic conditions given in each case this list should be arranged in such a way that it meets the objectives of the PSA and is in this context as complete as possible. For the selection process all actual available sources of information should find consideration. If necessary further measures, which are suitable to support this process should be seized (e.g. audits etc.). The non-consideration or neglecting of certain initiating events should be justified.

Step 3 Modeling of the incident scenarios

The objective is to develop a comprehensive model which links together the initiating events, the relevant (safety-)system response and the spectrum of the hence resulting final conditions. Such a model describes the progression of events as consequence of an initiating event, with consideration of appropriate actions of the personnel, which are initiated, in order to control the incident. The most common methods for the modeling of such complex systems are the Event Tree Analysis (ETA) as well as Fault Tree Analysis (FTA) (respectively bottom-up and top-down approach). Depending on the complexity of the plant as well as the extent and the objective of the PSA meanwhile also different methods for the modeling of events respectively for the system modeling can be consulted amendatory.

Step 4 Evaluation of data and parameters

The fourth step consists of compiling of all necessary information, relevant for the quantification of the frequencies of the regarded operational sequences and of the consequence evaluation models. During the data acquisition the priority should be granted to the best-estimate approach before appropriate conservative evaluation collars, since the best-estimate approach supplies a more realistic picture of the incident frequencies as well as of the effects to be expected. In each case however the uncertainties of the data, which are used for the appropriate computation models, should be considered. It is to be noted that simplifications with the modeling of complex processes or phenomena have frequently additional uncertainties to the consequence.

Step 5 Quantification of scenarios

The scenario quantification comprises also the necessary analyses of sensitivity as well as appropriate measures to the evaluation of the determined results. The process of the scenario quantification supports thereby considerably the interpretation of the PSA results. It can be divided essentially into two sections:

Quantification of the incident scenarios and risk assessment: The intention of a PSA is to provide qualitative and quantitative results with regard to the safety evaluation of the respective plant. In doing so, it is important to specify the type of required results according to the PSA objective and the evaluation depth resulting from it. The analyses should be aligned in particular in such a way that appropriate regulations are addressed directly by the obtained results (such as prescribed dose limit values etc.). In this connection it should always be considered that the performance of a PSA is corresponding in principle to an iterative process; i.e. computer models are developed, with which first results are obtained, which are submitted to an evaluation, in order to improve - if necessary - subsequently the computer models etc. Quantifying the plant risk is generally - however not exclusive - the intention of a PSA. Depending on the PSA intention, according to respective official regulations or according to the demands, derived from the plant operation, different result categories concerning the risk evaluation can be deviated.

Sensitivity and Uncertainty Analyses: In order to facilitate the interpretation of PSA results, all contributing effects, affecting the final condition of the plant, the frequency of error sequences, the unavailability of systems as well as the general effects etc., should be evaluated according to their respective importance. With a sensitivity analysis generally an evaluation takes place both of the sensitive dependence of the final condition of the plant of appropriate component errors respectively human errors and of the sensitivity of model assumptions.

An uncertainty analysis should be carried out to determine the uncertainty that arises from the data that have been used to quantify the PSA and to provide an indication of the level of confidence on the PSA-results. Therefore uncertainty distributions should be specified for the data used in the PSA. These uncertainties should be propagated through the whole quantification process.

Step 6 Documentation

The documentation of the PSA represents an important part of the quality assurance of the total process. One of the principal goals of the documentation to be accomplished consists in a clear and reproducible compilation of all information with regard to the analysis bases (e.g. assumptions, data and methods etc.) and of the obtained results (e.g. results of the detailed analyses, interpretation of the results etc.), in order to make a later investigation of the entire accomplished analyses details possible.

Example of application

Type and comprehensiveness

In the frame of a generic study the PSA methodology was applied exemplary to a selected sub-process of a UO_2 fuel fabrication facility for light water reactor fuel. Therefore a part of the fuel fabrication process starting with powder processing and ending with pressing of pellets has been selected. This section has been selected for reasons of manageability and availability of data and information. As far as possible information from different, partially from former facilities was used. Missing information was replaced by reasonable assumptions and generic data, in particular with regard to reliability of operational components. For this reasons the PSA described in the following is of pure methodological type and the results may not be assigned to a currently operating facility.

Description of the sub-process and its main components

Before fabrication of the fuel pellets the UO_2 powder must be pretreated in order to get a homogeneous capable of flowing granulate of a defined grain size. The systems for powder treatment are placed in rooms located elevation levels above each other. They comprise of the following processing steps and main components (see also figure 4-1):

- Powder batch mixing station with drum station, two collection container and pneumatic conveying equipment for the powder to a cone mixer.
- Powder reprocessing with collection container, hammer mill , rolling compressor, granulator and drum station.
- Pellet pressing with auxiliary station, vibration mixer, powder filling equipment, collection container, pellet press and filling equipment for sintering boats.

The different components have the following functions:

- Drum station: receiving of delivered Uranium-oxide powder in 170 liter drums or powder from return flow in 20 liter drums, on demand delivery to the suction box; the 170 liter drum are equipped with neutron absorber material.
- Suction box: dispatching the powder into the collection container by pneumatic transfer.
- Collection container: Checking the humidity of the powder and feeding it by gravitation into the cone mixer, which is located at the level below.
- Cone mixer: Powder mixing by means of a homogenizing screw, then pneumatic convey of the powder either into a collection container or into a 170 liter drum.
- Collection container for powder reprocessing: Transport of the powder into the hammer mill by screw conveyer, then by gravitation into the rolling compressor and the granulator. By the sequence of milling, compacting and granulating possible lumps in the powder will be cracked and a required defined grain size is achieved. Subsequently the powder is filled into 170 liter drums and shipped to auxiliary station.
- Auxiliary station: add-on of additives or of milled return flow material into the170 liter drums and carriage of the drums to the vibration mixer; after mixing by vibration back carriage to the powder filling station.

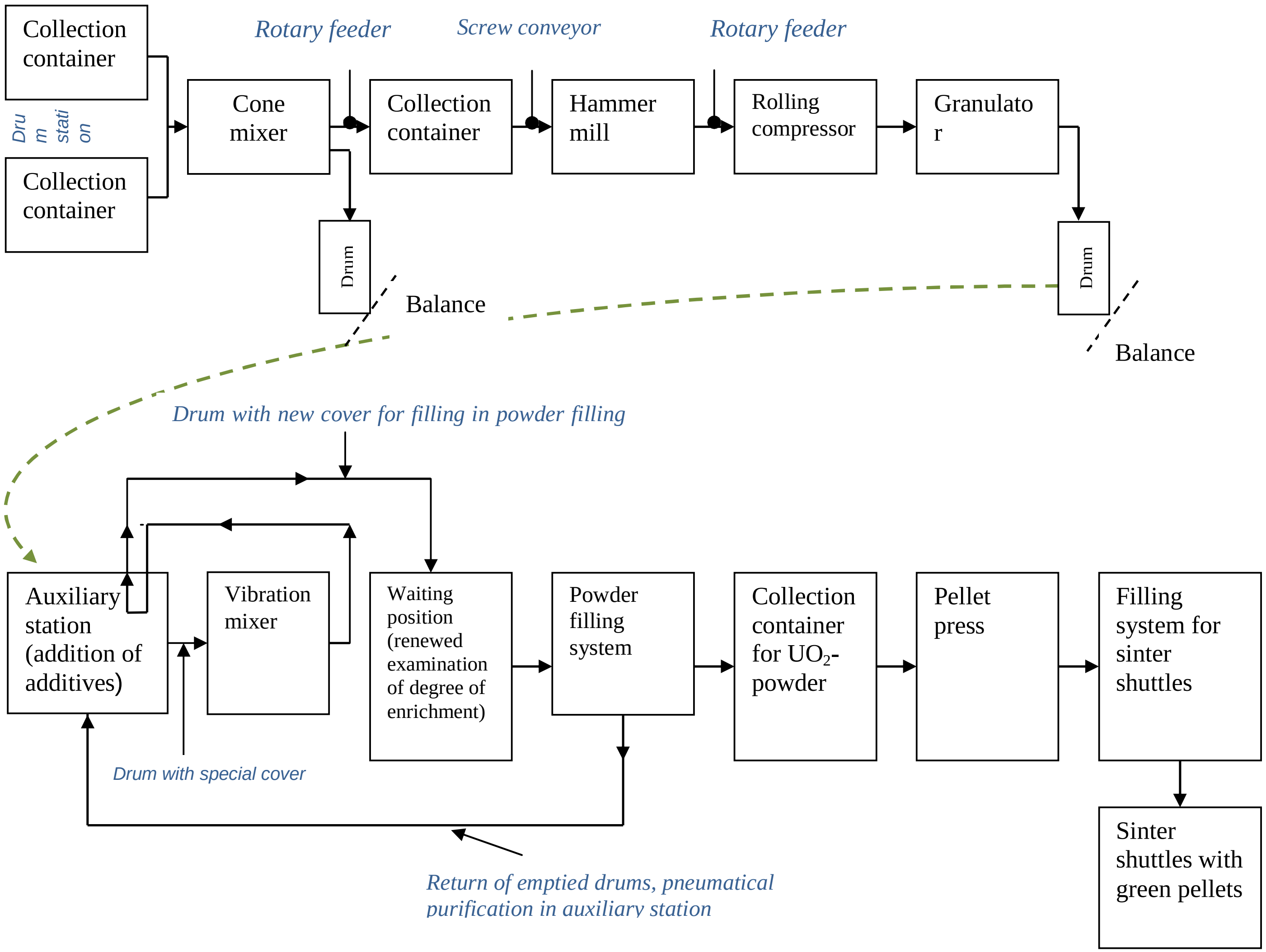
- Powder filling station: Feeding the powder via a collection container to the powder press, pressing of the green pellets, transfer of the green pellets to the filling equipment for sintering boats and carriage of the loaded sintering boats to the sintering furnace.

Figur 4-1 shows a simplified flow chart of the analyzed sub-process.

Undesirable final conditions

In contrast to the reliability analysis, where a system is evaluated with regard to its reliability, the risk analysis determines the probability, at which a system can reach an undesirable state. In this context undesirable states are such states of the system, where administrative requirements or regulatory limits are exceeded. In a fuel fabrication facility these are for instance dose limits for radiation protection (local dose rates, air contamination) or requirements to ensure sub-criticality considering the double contingency principle. According to this principle "process designs must incorporate sufficient factors of safety to require at least two unlikely independent and concurrent changes in process conditions, before a criticality accident is possible (ANSI/ANS-8.1).

Figure 4-1. **Schematic view of the analyzed sub-process**



The relevant requirements are also written down in the German criteria for notification of events in nuclear facilities other than nuclear fission reactors [4]. Consequently for example at hand the following undesirable final states have been identified:

- Contamination of the air respectively surfaces in working areas due to release of radioactive substances, frequently in connection with exceeding the dose rate limit;
- Violation of a safety requirement for ensuring sub-criticality;
- Criticality accident;
- Evaluation of operation experience.

In German fuel fabrication facilities 128 notifiable events occurred between 1979 and 2009, which have been recorded on a data bank and are available to GRS [7]. These notifications have been evaluated with regard to information, which is relevant or applicable to the investigated sub-process. It has been evaluated, whether possible initiating events, possible failures of systems and components as well as performance of the plant after occurrence of an initiating event are relevant for the analysis at hand. From the evaluation twenty events were identified as relevant for this analysis.

Systems analysis

At first the initiating events will be identified i.e. disturbances respectively incidents, which are to be analyzed. For this purpose a linkage diagram (Master Logic) is used.

The identified initiating events are then analyzed with consideration of intended counter measures with the help of the event- and fault-tree analysis, to determine the expected frequencies per year of the unrequested final conditions specified above. In the analyzed sub-process however practically no counter measures are intended respectively these measures are not well-known. For this reason the systems analysis will be limited on the determination of initiating events, which are leading either directly or by a linkage of faults to undesirable final conditions.

Identification of initiating events

For the determination of the relevant initiating events the Master Logic for occurrence of this event will be compiled, i.e. there will be a deductive assessment for all causes, leading separately or in combination with others to „the undesirable final conditions“ in the analyzed sub-process of fuel element production. For the release of radioactive substances the following reasons are possible:

- Faults when filling drums or taking a samples (spillage, overfills)
- Faults when carrying UO_2 powder (toppling over, falling down of drums);
- Release by leakages (leakage in the pneumatic conveyor system or in a drum);
- Faults when handling drums (leakage when changing the different lids);

The event fire was not considered in this analysis for methodical reasons.

An aircraft crash was not analyzed because of the small occurrence probability of a coincidental crash. Violations of the safety requirement for maintaining sub-criticality may have the following causes:

- Wrong degree of enrichment of UO_2;
- Error when controlling humidity;

- Overfilling of one of the two collection containers;
- Overfilling of the cone mixer;
- Jam of material in the powder preparation station (leads to overfilling of the receiver tank);
- Inadequate presence of neutron absorber in 170 liter drum;
- Faults when feeding additives.

Water intrusion into an area of control moderation (burst of a cooling-water pipe, fire extinction with water in contrary to regulations).

The only possible reason for a criticality excursion in the considered system range is overfilling of the cone mixer with faultily enriched UO_2-powder respectively too high humidity.

Fault tree analysis

In this case the fault tree analysis is the systematic implementation of the master Logic discussed above, where the sub-processes are divided into individual components. The undesirable final conditions represent the TOP gates and the individual components are modeled with different logical functions in fault trees. In order to identify general weak points in the system, the undesirable final state „deviation from the target process“ was modeled additionally.

Thus the fault tree analysis covers four TOP gates (possible undesirable final conditions):

- Deviation from the target process;
- Contamination and/or increased activity in the room air;
- Violation of moderation control and/or the safety criteria;
- Criticality accident.

Characteristic reliability data

Compiling a PSA for NFCF implies analog to the PSA guideline for nuclear power stations [6] and Annexes [7] and [8] the application of plant specific reliability data (e.g. failure rates) if possible. For the present analysis plant specific reliability data are - with exception of a few derived from notified events - not available. For this reason reliability data were for the most part determined from generic sources or estimated on the basis of assumptions.

Quantitative evaluation of the fault trees

The fault trees compiled on the basis of the four TOP-Gates have been analyzed with the computer code RiskSpectrum® under consideration of the determined reliability data. The results are summarized in the table 4-1.

Table 4-1**. Results of the fault-tree-analysis for the four examined final conditions**

Nr	Analyzed final conditions	Mean frequency of occurrence per year
1	Deviation from normal operation	6,0
2	Contamination and/or increased activity in the room air	$1{,}4 \cdot 10^{-1}$
3	Violation of moderation control and/or the safety criteria	1,1
4	Criticality accident	$4{,}1 \cdot 10^{-8}$

The first case is essentially a single-train process, thus predominantly single failures are causing the undesirable event. The loss of the power supply of the following components is leading in each case with 7.3 % to the result: Rotary feeder, granulator, pellet press, filling mechanism, hammer mill, screw conveyor, belt conveyor, cone mixer, rolling compressor and compressor. The mechanical failure of the components specified above follows with in each case 1.5 %. The further contributions are in each case below 1 %. The group of the human errors is contributing approx. 1.3 % to the result.

Also in the second analyzed final condition the result is determined by individual faults. The greatest contribution to the result is contributing the leakage of a drum with approx. 35%, followed by a leakage at the discharge side of the pneumatic conveyor (approx. 20%). A leakage on the suction side, which occurs with the same frequency, is leading in case of simultaneous breakdown of the compressor with approx. 8% to a contamination. The further contributions are in each case below 5%. The group of the human errors has approx. 34% share in the result.

In the third analyzed final condition those failures, who are leading to an overfilling of the cone mixer are substantially accounting for the result: Loss of the electrical power supply of the rotary feeder respectively of the compressor for the pneumatic conveyor with each approx. 58 %. Mechanical failure of the compressor of the pneumatic conveyor respectively of the rotary feeder contribute each with approximately 12 %. The further contributions are in each case below 5 %. The group of human errors accounts for approximately 1% to the result.

The occurrence of the final condition criticality accident is negligible. In this context it should be noted that the most frequent failure combinations contain in each case three human errors. Since manual actions in a course of events are usually not completely independent from each other, this result would usually have to be scrutinized. A detailed analysis of the manual actions would lay open a common part, which will increase the result significantly.

The main contributions are: Delivery batch with accidentally increased enrichment of Uranium (100% contribution), loss of the electrical power supply of the rotary feeder respectively of the compressor of the pneumatic conveying with each approx. 40%, leakage at the suction face of the pneumatic conveying with approx. 44%, a series of manual action, which describe the loss of moderation control respectively isotope control for the two collection container (in each case between approx. 29% and approx. 21%), mechanical failure of the rotary feeder respectively the compressor of the pneumatic conveyor with each approx. 8%. The further contributions are in every case below 5%. The group of the manual actions contributes with approx. 100% to the result. This means, that a human error is involved in every combination of failures.

Uncertainty analyses

All parameters, which are affecting the frequency for undesirable final conditions, are statistic quantities, which are afflicted with different types of uncertainties (knowledge uncertainty and/or stochastic uncertainty). The extent of propagation of these uncertainties of the initial parameters into the results can be quantified by means of uncertainty analysis. This means that statistic expected values and confidence intervals should be accounted for each of the calculated results. Since the uncertainties of knowledge concerning the sub-process and the individual components exceed the statistic uncertainty considerably, an uncertainty analysis based on statistic uncertainties is not meaningful in this case and was thus not accomplished.

Summary of the results

The investigations of the example have shown that PSA methods are in principle applicable to a production process of a fuel manufacturing facility. In contrast to a PSA of a nuclear power station, where the plant behavior following an assumed initiating event is modeled up to the control or to the nuclear core damage, active safety systems to control the initiating event are not or much less present in this case. Here the attention is lying on the prevention of an initiating event, so that the systems analysis is concentrating on the modeling of „trigger fault trees “. Furthermore the results of this investigation can be used only qualitatively and have an only methodical character due to the subject and data situation and the large uncertainties in knowledge resulting from this. Quantitative results were presented, however for more reliable conclusions additional information concerning the real process as described above is absolutely necessary. The results presented here cannot be transferred thus under any circumstances directly to a real existing plant of nuclear fuel manufacturing.

Conclusion

A PSA analysis is useful to extending the basis for the risk evaluation of complex plants and to complement the conventional deterministic analyses, whereby the present safety standards of the plant can be improved if necessary. The PSA process requires a systematic procedure of a skilled interdisciplinary team of specialists.

The methodical base and procedures for PSA performance have been permanently refined in the past and have reached a development status, which permits the practice-oriented application of PSA methodology. However it should be always considered that the correct and meaningful application of the existing basic knowledge to a concrete plant results in a partial substantial effort regarding the collection and evaluation of complex and/or specific connections (process and parameter evaluation, component and system dependencies, influence of human errors etc.). Also great attention must be paid to e.g. the determination and composition of appropriate initiating events, the development and the quality assurance of the analysis models, the supply of qualitatively high-quality model input data, which should be determined on a realistic basis, as well as the interpretation of the results of computation and the consequence evaluation in each case.

Finally it should be emphasized again that a PSA - with consideration of the objective target - can lead only then to usable and profitable insights, if the entire PSA process is conscientiously accomplished and documented sufficiently.

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Probabilistic Safety Analysis for Nuclear Fuel Cycle Facilities, an Exemplary Application for a Fuel Fabrication Plant

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- Basic concept of a PSA for nuclear fuel cycle facilities
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Introduction

- Probabilistic methods for risk assessment in industries dealing with hazardous materials are generally accepted
- In Germany probabilistic safety analysis (PSA) is required for NPPs in the frame of safety review, in addition to deterministic analyses
- PSA-methods and -tools have been developed in particular for application to NPPs
- For NFCFs in Germany PSA is not required by law
- PSA is recommended in order to supplement the deterministic approach:
  - Review and in-depth analysis of systems and processes
  - Detection of possible sources of failures, weak points of the design etc.

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Introduction (cont.)

The analysis presented:

- Refers to a sub-process of a fuel fabrication facility
- Based on partially generic data and previous design of a plant
- Work done so far is mainly of methodological character
- Main objectives:
  - Development of the method,
  - Exemplary application, to gain experience
- Results not referring to an existing facility

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Head end fuel cycle facilities in Germany

Uranium Enrichment Plant Gronau

- Operator: URENCO Deutschland GmbH
- Gas ultra centrifuge technique, 5 % U-235 (product)
- In operation since 1985 UTA1: 1800 t SWU/a
- New section UTA2 under construction
- Capacity (end of 2010): 3200 t SWU/a, production 2009: 2300 t UTA/a

Nuclear Fuel Fabrication Plant Lingen

- Operator: Advanced Nuclear Fuel (ANF) GmbH (100% subsidiary of AREVA NP)
- Production of UO_2 fuel assemblies for Light Water Reactors (LWR)
- In operation since 1979
- Capacity: 650 t U/a, increase to 800 t U/a licensed (end of 2009)
- Dry conversion process for UF_6 implemented

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Main Differences between a PSA for nuclear power plants and for nuclear fuel cycle facilities

PSA applications to NPPs and NFCFs is based on common principles. Differences are:

- Larger technological and/or process-justified diversity in case of NFCFs
- Amounts of chemical materials handled in NFCFs are of higher importance with regard to the total risk of the facility :
  - ➢ Toxic, corrosive, easily inflammable or explosive materials may be considered besides nuclear material
- Greater plant areas with hazardous materials stored and handled in case of NFCFs
- More operational- and/or process-related changes in NFCFs
- Greater influence of human errors in NFCFs
- Usually no different levels regarding consequences of PSA (as for NPPs) are performed for NFCFs

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Basic concept of a PSA for nuclear fuel cycle facilities

According to the 'Procedures for conducting probabilistic safety assessment for non-reactor facilities' (IAEA-TECDOC-1267, January 2002), performing a PSA for NFCFs should be performed according to the following six steps:

(1) Creation of the bases in management and organization

(2) Identification and selection of initiating events

- Study of the plant characteristic and information composition
- Incident identification on basis of the plant characteristic
- Provisional identification of undesirable final conditions
- Identification of safety measures and - functions
- Compilation of information concerning safety measures
- Grouping of the initiating events in order to perform the analysis

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Basic conception of a PSA for nuclear fuel cycle facilities (2)

(3) Modeling of the incident scenarios

(4) Evaluation of data and parameters:

- Best-estimate approach to be preferred
- Uncertainties of data to be considered

(5) Quantification of scenarios:

- Sensitivity/uncertainty analyses, evaluation of results
- Results of the PSA should refer to safety requirements, e. g. dose limits

(6) Documentation

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Example of application

- Exemplary application of the PSA methodology in a generic study to a selected sub-process of a UO_2 fuel fabrication facility for light water reactor fuel.
- Selected sub-process: Powder processing (includes handling of drums) ⇒ pellet pressing
- As far as possible information from different facilities, partially from earlier design was used.
- Collected information on notified events has been evaluated with regard to initiating events, failures of systems and components
- Missing information was replaced by generic data and reasonable assumptions, in particular with regard to reliability of operational components.
- For this reasons the PSA described in the following is of pure methodological type and the results may not be assigned to a currently operating facility.

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Description of the sub-process and its main components

- **Powder batch mixing station** with drum station, two collection containers and pneumatic conveying equipment for the powder to a cone mixer
- **Powder reprocessing** with collection container, hammer mill, rolling compressor, granulator and drum station
- **Pellet pressing** with auxiliary station, vibration mixer, powder filling equipment, collection container, pellet press and filling equipment for sintering shuttles.

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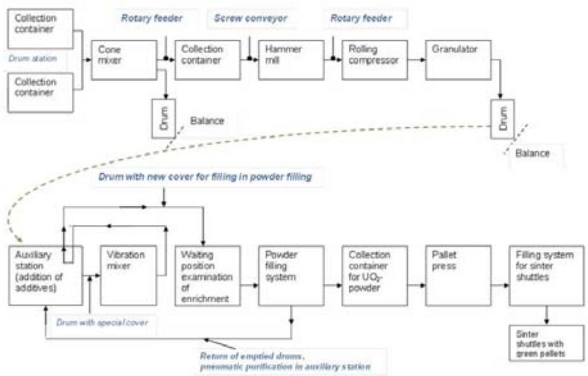
Example of application: Schematic view of the analyzed sub-process



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UO_2 fuel fabrication sub-processes

UO_2-powder filling

Pellet pressing

Pellet sintering

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Identification of undesirable final states

- Risk analysis determines in general the frequencies of undesirable states.
- Undesirable states of the analyzed process are states, where safety requirements are violated or regulatory limits will be exceeded.
- For this example the following possible undesirable final states have been identified:
  - Contamination of the air respectively surfaces in working areas due to release of radioactive or toxic substances, frequently in connection with exceeding the dose rate limit
  - Violation of a safety requirement for ensuring sub-criticality
  - Criticality accident.

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Identification of initiating events (1)

- Creation of a diagram (Master Logic) to analyze the system
- Deductive assessment for all causes, leading separately or in combination with others to „the undesirable final states".
- For the release of radioactive substances the following relevant causes were identified
  - Failures when filling drums or taking samples (spillage, overfills)
  - Failures during transport of UO_2 powder (toppling over, falling down of drums)
  - Release by leakages (leakage in the pneumatic conveyor system or in a drum)
  - Faults when handling drums (leakage when changing lids)

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Identification of initiating events (2)

- Violation of the safety requirement for maintaining sub-criticality may have the following causes:
  - Wrong degree of enrichment of UO_2
  - Error when controlling humidity
  - Overfilling of one of the two collection containers
  - Overfilling of the cone mixer
  - Jam of material in the powder preparation station (leads to overfilling of the receiver tank)
  - Inadequate presence of neutron absorber in 170 liter drum
  - Faults when feeding additives
  - Water intrusion into an area of controlled moderation (e. g. leakage of a cooling-water pipe etc.).

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Identification of initiating events (3)

- Occurrence of a criticality
  - Overfilling of the cone mixer with faultily higher enriched UO_2-powder
  - Significant excess of the moderation limit

The events fire and aircraft crash have not been considered in this analysis for methodical reasons.

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Fault tree analysis

- Fault tree analysis: Systematic implementation of the Master Logic. The sub-processes are modeled on the level of individual components.
- Undesirable final states: Represented by TOP gates; failures of individual components are modeled and linked by means of different logical functions in fault trees (and-gate, or-gate etc.).
- Undesirable final states analyzed:
  - ➢ Deviation from the target process (additionally modeled to identify possible weak points in the system)
  - ➢ Contamination and/or increased activity in the room air
  - ➢ Violation of moderation control and/or safety criteria
  - ➢ Occurrence of a criticality accident.

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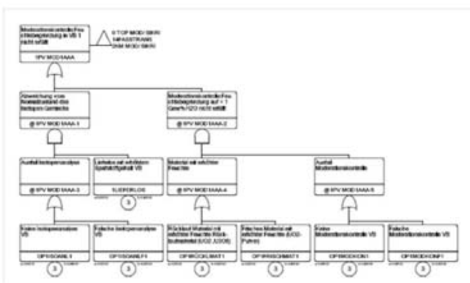
Example of a fault tree



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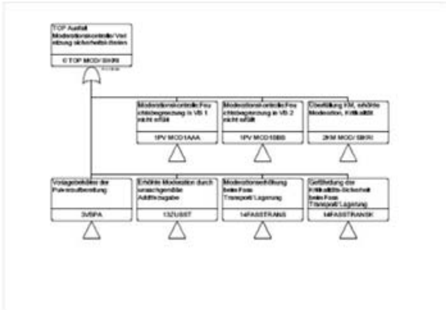
Example of a fault tree (cont.)



Quantitative evaluation of the fault trees

- The fault trees were analyzed with the computer code RiskSpectrum® (the used reliability data mainly derived from generic sources or estimated on the basis of assumptions).

Results of the fault tree analysis

No	Analyzed final state	Occurrence frequency per year (point value)
1	Deviation from normal operation	6,0
2	Contamination and/or increased activity in the room air	$1,4 \cdot 10^{-1}$
3	Violation of moderation control and/or the safety criteria	1,1
4	Criticality accident	$4,1 \cdot 10^{-8}$

Evaluation of the results

- Most of these cases are essentially single-train processes, thus predominant single failures are causing the undesirable states
- Contributions of different failures to the result of analyzed final states are quantified
- Human errors in same failure sequence were assumed as independent (detailed dependency analysis would be desirable)
- The determined occurrence of the final state criticality accident is negligible
- Uncertainty analysis based on statistic uncertainties was not meaningful in this case, since uncertainties of knowledge are considerably larger
- PSA in NFCF: The attention is lying more on the prevention of an initiating event than on the availability of safety systems, so that the system analysis is concentrating on the modeling of „trigger fault trees ".
- The results presented here may be used only qualitatively and have only methodical character due to the subject and data situation and the large knowledge uncertainties.

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Summary and conclusion

- A PSA is useful to extend the basis for the risk evaluation of complex plants and to complement the conventional deterministic analyses.
- The PSA process requires a systematic procedure of a skilled interdisciplinary team of specialists.
- The methodical base and procedures for PSA performance have been permanently refined in the past and have reached a development status, which allows the practice-oriented application of PSA methodology.
- Correct and reasonable application of the existing basic knowledge to a real plant requires great effort regarding the evaluation of complex systems and processes
- A PSA can lead only to useful and beneficial insights, if the entire PSA process is carefully accomplished and documented adequately.

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Thank you