USE OF PROBABILISTIC RISK ASSESSMENT IN FUEL CYCLE FACILITIES

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Abstract – As expressed in its Policy Statement on the Use of Probabilistic Risk Assessment (PRA) Methods in Nuclear Regulatory Activities, the U.S Nuclear Regulatory Commission has been working for decades to increase the use of PRA technology in its regulatory activities. Since the policy statement was issued in 1995, PRA has become a core component of the nuclear power plant (NPP) licensing and oversight processes. In the last several years, interest has increased in PRA technologies and their possible application to other areas including, but not limited to, spent fuel handling, fuel cycle facilities, reprocessing facilities, and advanced reactors.

This paper describes the application of PRA technology currently used in NPPs and its application in other areas such as fuel cycle facilities and advanced reactors. It describes major challenges that are being faced in the application of PRA into new technical areas and possible ways to resolve them.

1. Introduction

The Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities expressed the Commission’s position that the use of PRA technology should be increased in all regulatory activities with the commitment to a risk-informed regulation. A risk informed approach as defined by the NRC is: “An approach to regulation taken by the NRC, which incorporates an assessment of safety significance or relative risk. This approach ensures that the regulatory burden imposed by an individual regulation or process is appropriate to its importance in protecting the health and safety of the public and the environment”.

Risk information is now used in many aspects of NRC’s NPP work such as regulation and guidance, licensing and certification, oversight, and operational experience. With the increased necessity to risk-inform comes the need to have consistent processes for implementing risk-informed regulation of nuclear activities (reactors and non reactors).

As a result of the experience gained through the use of PRA for existing NPPs, the need for the use of PRA in other facilities has increased. PRA use has expanded to other facilities and processes like new reactors, storage and transportation of nuclear waste materials, and fuel cycle facilities. Although fuel cycle facilities (FCFs) rely mostly on the use of Integrated Safety Analysis (ISA) for their safety assessments, an interest in adapting PRA to these facilities has arisen. Work is currently being performed to study the feasibility of adapting previous NPP PRA approaches to FCFs, but several challenges have surfaced during this work. Some of the more significant challenges will be discussed in more detail in the following sections.

2. Probabilistic Risk Assessment

The NRC defines PRA as: "A systematic method for assessing three questions used to define risk. These questions consider (1) what can go wrong, (2) how likely it is, and (3) what its consequences might be. These questions allow the understanding of likely outcomes, sensitivities, areas of importance, system interactions, and areas of uncertainty, which can identify risk-significant scenarios. The PRA determines a numeric estimate of risk to provide insights into the strengths and weaknesses of the design and operation of a nuclear power plant."

For nuclear power plants there are three levels of PRA: Level 1 PRA, Level 2 PRA and Level 3 PRA. Each level considers a sequential step in risk assessment processes as described below:

- "A Level 1 PRA models the various plant responses to an event that challenges plant operation." To measure the Level 1 PRA, the analysts construct a set of event trees to represent the different accident sequences that either can lead to successful recovery or to core damage. The frequency of each core damage accident sequence is estimated, and the frequencies for all core damage sequences are summed to calculate the total core damage frequency. The results of the Level 1 PRA are used as input to the Level 2 PRA.

- A level 2 PRA takes the results from Level 1 PRA accident sequences that resulted in core damage and calculates frequencies of radioactivity releases as the output. This PRA analyzes the progression of the accidents that result in reactor core damage (severe accidents). It considers how the reactor coolant and other relevant systems respond, as well as how the containment responds to the accident. The results of the Level 2 PRA are used as input to the Level 3 PRA.

- A Level 3 PRA takes the results from the Level 2 PRA as input and produces offsite consequences as output. It models the release and transport of radioactive material in a severe accident, and estimates the health and economic impact in terms of early fatalities and latent cancer fatalities, and the economic costs associated with evacuation, relocation, property loss, and decontamination. By combining the results of the Level 1 and Level 2 PRAs with the results of this consequence analysis, only the Level 3 PRA estimates the integrated risk (likelihood x consequences) to the public for the analyzed NPP.

3. Reactor oversight process

The reactor oversight process (ROP) is a risk informed approach used by the NRC to monitor reactor safety performance to ensure that the nuclear power plants meet the NRC regulations in order to ensure and protect public health and safety. The reactor oversight activities include the ROP inspection programme, the significance determination process (SDP), and other assessment activities. Figure 1 shows the connection between the different processes that are part of the ROP.
The main objectives of the ROP are:

- To obtain information on operating facilities and identify safety concerns;
- To evaluate the risk significance of issues to ensure the appropriate regulatory measure;
- To assess licensee performance;
- To take enforcement actions that encourage the resolution of risk-significant issues;
- To verify that licensees effectively identify problems and resolve issues;
- To provide the appropriate regulatory response to operational events on the basis of their safety significance;
- To monitor licensees and encourage them to maintain a safety-conscious work environment.

The regulatory framework for reactor oversight consists of the three key strategic performance areas: reactor safety, radiation safety, and safeguards. Seven cornerstones that reflect the essential safety aspects of facility operation originate from these three performance areas. These cornerstones include:

(accessed on July 13, 2011)
• Initiating events – focuses on operations and events at a nuclear plant that could lead to a possible accident, if plant safety systems did not intervene. These events could include equipment failures leading to a plant shutdown, shutdowns with unexpected complications, or large changes in the plant’s power output.

• Mitigating systems – measures the function of safety systems designed to prevent an accident or reduce the consequences of a possible accident.

• Barrier integrity – refers to the assessment of the physical barriers (cladding, reactor coolant system boundary, and containment) that protect the public from radionuclide releases caused by reactor core damage. The integrity of these barriers is continuously checked for leakage, and the ability of the containment to prevent leakage is measured on a regular basis.

• Emergency preparedness – refers to the validation of the emergency plan actions that provide adequate protection of public health and safety during a radiological emergency.

• Public radiation safety – refers to measures used to ensure adequate protection of public health and safety from exposure to radioactive material released into the public as a result of routine nuclear reactor operations. This cornerstone measures the procedures and systems designed to minimize radioactive releases from a nuclear plant during normal operations and to keep those releases within federal limits.

• Occupational radiation safety – refers to the limit, set by the NRC, on radiation doses received by plant workers. This cornerstone measures the effectiveness of the programme to control and minimize those doses. The main objective is to ensure adequate protection of worker health and safety from exposure to radiation from radioactive material during nuclear reactor operation.

• Security – refers to the NRC’s objective of providing assurance that the physical protection system can protect against the threat of a radiological sabotage, either internal or external.

Each cornerstone contains inspection procedures and performance indicators to ensure that their objectives are being met. A similar approach could be applied to other facilities such as fuel cycle facilities and reprocessing facilities. The possible approaches will be discussed in the following sections.

One of the main reactor oversight activities is the significance determination process (SDP). The SDP is a process that uses risk insights to help NRC inspectors and staff determine the safety or security significance of inspection findings. The safety significance of findings, combined with the results of the performance indicator (PI) programme, are used to define a licensee’s level of safety performance. Each SDP supports a cornerstone associated with the strategic performance areas as defined in the NRC’s Inspection Manual\textsuperscript{28} Chapter 2515.

The ROP SDP process consists of the analysis of inspection findings resulting from NRC’s inspection programme and performance indicators reported by the licensee. Both inspection findings and performance indicators are evaluated, and depending on the results of the evaluations, are assigned a color. Inspection findings are either characterized as a green finding, white finding, yellow finding, or red finding. Green inspection findings indicate a deficiency in licensee performance that has very low risk significance and has little or no adverse impact on safety. Green performance indicators represent acceptable performance in which cornerstone objectives are fully met. A fundamental concept of the ROP is to provide timely feedback on license performance in order to allow for licensee initiatives to correct performance issues before increased regulatory involvement is warranted. White, Yellow, and Red inspection findings or

performance indicators respectively represent successively greater degrees of safety significance and therefore trigger increased regulatory attention.

The SDP uses a three phase process to characterize inspection findings. All inspection findings are initially screened through the SDP Phase 1. If the Phase 1 determination’s result is a green finding, no further screening is necessary. On the other hand, if the determination’s result is greater than green (white, yellow or red), a Phase 2 SDP is necessary. Phase 2 SDPs use plant specific risk-informed information to determine the importance of the finding. Two plant-specific risk tools are available to support the Phase 2 evaluation: pre-solved tables/worksheets and a risk-informed notebook. If the finding cannot be adequately categorized using the pre-solved tables/worksheets, a more detailed evaluation using the risk-informed notebook is performed. If the evaluation of the finding requires departure from the guidance for a Phase 1 or 2 SDP or consideration of factors not adequately addressed by the Phase 2 SDP simplified or risk tools, then a Phase 3 analysis should be performed to characterize the significance of the finding. Phase 3 SDP analyses rely mostly on PRA techniques and NRC risk analysts.

Risk informed SDP tools are intended to estimate the actual incremental risk increase above the nominal baseline level of probabilistic risk in terms of core damage frequency (CDF) — the likelihood that, given the way a reactor is designed and operated, an accident could cause the fuel in the reactor to be damaged; or in terms of large early release frequencies (LERF) — the likelihood that a rapid, unmitigated release of airborne fission products from the containment to the environment occurs before the effective implementation of off-site emergency response and protective actions such that there is a potential for early health effects.

Some of the tools used for the SDP are inspection manual guidance, the SAPHIRE PRA code, and the Standardized Plant Analysis Risk (SPAR) Models. The SDP appendices are tools that were developed to risk inform and characterize the safety significance of findings associated with the seven cornerstones. These appendices are used to guide the user through a process of identifying the significance of the finding. SDP tools either use a combination of quantitative and qualitative risk methods or a risk-informed process developed by an expert panel consisting of staff and industry representatives. Examples of processes developed by an expert panel include emergency preparedness, occupational and public radiation safety, and security. The plant-specific reactor safety SDP tools that use quantitative risk methods include at-power operations, fire protection, shutdown operations, containment integrity, operator requalification, steam generator tube integrity, and maintenance effectiveness.

One of the tools used to quantify the results of an SDP in reactors is the SPAR models. SPAR models are independent plant-specific PRA models developed by the NRC that reflect the as-built, as-operated plant to the extent needed to support the PRA analyses. The SPAR models use a standard set of event trees for each plant design class and standardized input data for initiating event frequencies, equipment performance, and human performance, although these input data may be modified to be more plant- and event-specific when needed. The system fault trees contained in the SPAR models are generally not as detailed as those contained in licensees’ PRA models. These models are used in SDP phase 3 analyses to determine the risk significance of inspection findings or of events to decide the allocation and characterisation of inspection resources, the initiation of an inspection team, or the need for further analysis or action by other agency organisations.

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The SPAR model gives risk analysts the capability to quantify the expected risk of a nuclear power plant in terms of core damage frequency and the change in that risk given an event or an anomalous condition or a change in the design of the plant. More importantly, the model provides the analyst with the ability to identify and understand the attributes that significantly contribute to the risk, and insights on how to manage that risk. Obtaining the results in terms of delta CDF allows the analyst to identify in which range of the SDP thresholds the finding lies (white, green, yellow or red).

4. Introduction to Non-Reactor Nuclear PRA

In the past decade, Probabilistic Risk Analysis (PRA) has found its way to other Non-Reactor related nuclear operations such as storage and transportation, fuel cycle facilities, advanced reactors, and waste applications. Several non-reactor related studies have been completed by NRC, DOE and Industry (EPRI) [See section 8]. Some of these studies have quantified frequency and probability of specific scenarios of concern and estimated individual probability of early and latent cancer fatalities in dry cask storage or spent fuel pools in NPPs. Limited scope studies and methodologies have been developed and proposed to analyze fuel cycles which include uranium conversion, uranium enrichment, fuel fabrication and uranium de-conversion facilities. These facilities have hazards that might or might not be found in other nuclear facilities.

One of the most prominent differences between reactor and non-reactor operations is the lack or an analogue to the reactor core as found in a nuclear power reactor. In a NPP PRA, a radiological release is generally assumed to originate as a result of damage to the nuclear fuel. Therefore, core damage can be used as a surrogate for release. Conversely, non-reactor facilities have many potential radiological (and chemical) sources distributed throughout the facility and no single surrogate can be used. Other notable challenges of PRA analysis of non-reactor operations include different types of hazards, variable source terms throughout a process, event sequences, vulnerability duration, lack of standby systems and reliance on human actions due to process differences. Fuel cycle facilities also have differences in processing technologies which provide challenges during an analysis.

The methodology that has been followed in the past to risk-inform FCFs follows the guidance developed for NPPs such as NUREG/CR-2300 “PRA Procedures Guide: A Guide to Performance of Probabilistic Risk Assessments for Nuclear Power Plants”30, supplemented by staff with PRA and FCF experience and other guidance such as NUREG/CR-6410 “Nuclear Fuel Cycle Facility Accident Analysis Handbook”31 and the NRC report “Risk-Informed Decision-making for Nuclear Material and Waste Applications (RIDM)”32. The process can be divided into three major sections: review of the system; qualitative analysis; and quantitative analysis of processes and hazards that were of most concern in the qualitative analysis. The review of the systems consists of review of historical incidents, review of licensee analysis, and process chemistry and physics; the qualitative analysis reviews possible areas of concerns such as failures and event sequences from an engineering perspective; and the quantitative analysis quantifies the occurrence probability of a release and/or hazard of interest. Most of the studies performed by NRC have been limited scope or preliminary studies due to lack of available resources and have been


limited to a specific hazard (e.g. chemical explosions, criticalities, etc). A full PRA for a FCF has not been performed by the NRC.

5. Decision to Risk-Inform and use of PRA

Performing a PRA on a facility with thousands of processes would likely prove to be resource intensive and time consuming. Therefore it is important that an analyst prioritizes where he will spend his time and that he uses the tools available. In order to accomplish this, methodologies and tools need to be developed that address differences in facility hazards. The NRC is currently exploring which PRA techniques and tools would be most beneficial to a FCF. These can be used to study and analyze different stages of the life of a facility including design and licensing, operation, and oversight and decommissioning.

U.S. facilities licensees and regulators use a process called Integrated Safety Analysis (ISA) to design a facility and identify process safety controls to meet licensing regulations. Oversight of a FCF relies on inspections and uses the ISA, Safety Analysis Report and the License Application to identify risk significant systems. PRA techniques could improve model realism and employ tools such as event trees and fault trees to quantify probability of failure of systems to identify risk significant system, and in case of inspection findings, estimate an increase in risk due to the system vulnerability. PRA techniques and tools have been used in FCF licensing applications and are currently being explored and developed for oversight applications for the Revised Fuel Cycle Oversight Process.

PRA tools can be used for prioritisation of resources (such as increased inspections for higher risk findings), modeling and identification of initiating events and event sequences, realistic quantitative measures for the likelihood of risk contributors, and realistic evaluation of potential consequences with hypothetical accident sequences. All this can potentially help an analyst or inspector to better assess the safety of a system during a safety review or analysis.

Risk analyses for FCF have several challenges that differ from those found in nuclear power plants. Hazards and risks are facility dependent and could vary greatly depending on the processes and technology implemented by the facility; this limits the extent to which tools and methodologies can be generalized. Developing tools for the prioritisation of resources spent in event/accident sequence analysis would prove useful early in a study and would help to focus available resources on higher risk items. Limited resources would not permit the time needed to perform a full probabilistic analysis on a FCF including release mechanisms, quantification of environmental release and consequence analysis. Current NRC regulations (10 Code of Federal Regulations 70.61) require that FCF license holders implement controls to reduce the likelihood of high consequence events to “highly unlikely” and intermediate consequence events to “unlikely.”

Lower than intermediate consequence event sequences (or inspection findings that affect this event sequence) would receive little benefit from a probabilistic analysis. Current US regulations do not require FCF license holders to report and maintain reliable safety controls to lower than intermediate consequence events as required in intermediate and high consequence events. A PRA analysis to lower than intermediate consequence event sequences or findings would be resource intensive as process control information might not be readily available. Event sequences or findings with intermediate to high consequence or with a increased likelihood could potentially be characterized for further evaluation and analysis.
6. Available Guidance

Guidance such as the one found in the RIDM document\(^5\) and in NUREG-1520, “Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility NUREG-1520\(^{33}\)”, discusses possible methodologies (including PRA) that can be applied to FCF and other non-reactor related areas. In particular, the RIDM document suggests possible processes to prioritize safety reviews, tools to assess risk, and quantitative health guidelines to use as the risk metric for accident risk to individuals (workers and public), among other risk relevant discussion topics.

The RIDM document provides a description of 6 quantitative health guidelines (QHGs) that were developed equivalent to the quantitative health objectives (QHOs) for the public contained in the US NRC Safety Goal for the Operations of NPPs\(^{34}\). The QHGs represent a level of individual risk of a health effect (serious injury, and worker and public acute fatality and latent cancer fatality). Accident risk is generally dealt with qualitatively in the regulations and the QHGs provide a benchmark to assess the level of risk.

The four region risk diagram (shown below) divides the risk space for any of the applicable risk metrics into four regions; unacceptable (UR), safety margin (SM), tolerable risk (TR), and negligible risk (NR). The lower line that divides the TR and the NR implies anything that is below the QHGs. The upper line dividing the UR/SM and the TR corresponds to the risk implication of the regulatory limit that constitutes what is implied by adequate protection. The difference between the UR and the regulatory limit is the SM which is a factor to assure reliability of a system under different conditions. An increase in a risk metrics would move it towards the unacceptable risk area and would mean a higher level of scrutiny when analyzing a safety system or deficiency.

![Four region risk diagram](image)

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7. Applications of PRA: Oversight Significance Determination Process:

As explained in Section 3, SDP provides a characterisation of an increase in risk of an inspection finding or a deficiency in a safety system. The ROP SDP assigns colors (Red, Yellow, White and Green) for these risk increases, risk thresholds, which are then used for enforcement of regulations.

The SDP process for NPPs utilizes simple screening tools and more detailed risk models to estimate the increase in risk due to a deficiency. Similarly to the ROP SDP, an SDP in fuel cycle can be composed of two phases:

Phase 1: Initial screening and characterisation of inspection findings

Phase 2: Evaluation of Risk

Given the differences in FCFs design, technology, and risk, developing facility specific SDP Risk tools would prove to be very challenging and resource intensive. Phase 2 and Phase 3 of the ROP SDP would condense into Phase 2 where a finding or deficiency would be evaluated using generalized examples and methodologies in a specific cornerstone. The U.S. NRC has started work to develop guidance and the technical basis for establishing quantitative risk-informed thresholds that could be used for the different phases of SDP of the FCF oversight process.

The initial screening would consider the nature of a finding, associated degradation, and its duration. This step would require analysis of available information related to the process similar to what would have been performed in a deterministic approach. Items not screened would potentially be of greater concern and would be evaluated during a Phase 2 analysis. During Phase 2, a risk analysis would assess the change in likelihood of an event due to the deficiency or the identified degradation. This would use information and approaches used in the ISA, event tree/fault tree modeling, human reliability analysis, and/or other available tools/techniques to appropriately characterize the risk associated with a deficiency.

A methodology for Phase 2 analysis needs to be developed taking into account processes, technologies, and fuel cycle facility differences and risk thresholds. If a high level of detail is needed for an analysis, it is foreseen that data could be need to be developed or gathered. Databases such as the Department of Energy Savannah River Site Generic Database Development have been used in the past for equipment failure probabilities.

8. FCF PRA analysis challenges and other non-reactor related studies

Notable differences and challenges that would need guidance and tools developed to perform a PRA risk-inform analysis of non-reactor operations include:

1. Risk from facility to facility can vary due to variable source terms, the type of facility and technologies and processes employed. These factors complicate the development of a generic methodology that can be used for all facilities. Control systems and human actions can vary greatly due to process differences such as batch or continuous processes, processing technologies (e.g. laser enrichment and gaseous diffusion), hazard physical state (e.g. solid, liquid or gaseous), and radiological and/or chemical hazards.

2. Different types of hazards such as criticalities or chemical explosions can have health consequence to a facility worker but little to no consequence to the public, or could have health consequence to both (worker and public).

3. Information regarding fuel cycle facilities can be considered classified, proprietary, or sensitive, which causes difficulty in peer review and in improving the quality of PRA analyses.

4. Limited publicly available databases exist that can be utilized to acquire failure rate data such as the “Savannah River Site Generic Database Development”36.

Several non-reactors related studies have been performed in the past decade that could potentially serve as a learning experience on how to address some of the non-reactor related risk analyses. These include risk-studies that estimate radiological risk to the public in the areas of dry storage37, wet storage38, and transportation of nuclear waste39. Other work includes limited preliminary work that has been performed in FCF oversight40, 41 and 9.

9. Conclusion

Probabilistic Risk Assessment is a risk analysis methodology that has been used by the nuclear power industry for many decades. Maturing in operating power reactors, it has expanded to new reactors, storage and transportation of nuclear waste materials, and now to fuel cycle facilities. As discussed above, several challenges need to be addressed and resolved for PRA to be successfully applied to FCF. Limited guidance exists on how to conduct a risk analysis on a fuel cycle facility, but further studies could make it possible to adapt PRA processes used in NPPs to FCFs. The US Nuclear Regulatory Commission is currently evaluating cornerstones and developing tools to risk-inform its review and oversight processes for FCFs and other non-reactor related activities.

36 Blanchard, A.; “Savannah River Site Generic Data Base Development”; WSRC-TR-93-262; Revision 1; May 1998.


Use of Probabilistic Risk Assessment in Fuel Cycle Facilities

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Outline
- Introduction
- Probabilistic Risk Assessment
- Reactor oversight process
- Non-reactor nuclear PRA
- Oversight process application
- Challenges
- Available NRC guidance

Introduction
- Risk informed approach
  - "An approach to regulation taken by the NRC, which incorporates an assessment of safety significance or relative risk. This approach ensures that the regulatory burden imposed by an individual regulation or process is appropriate to its importance in protecting the health and safety of the public and the environment".
Probabilistic Risk Assessment

• NRC definition:
  – A systematic method for assessing three questions:
    • What can go wrong?
    • How likely is it?
    • What are its consequences might be?
  – Levels of PRA
    • Level 1 PRA
    • Level 2 PRA
    • Level 3 PRA

Probabilistic Risk Assessment

Reactors Oversight Process

• Risk informed approach used by the NRC to monitor reactor safety performance.
• ROP activities:
  – ROP inspection program
  – Significance determination process (SDP)
  – Other assessment activities
Reactor Oversight Process

- ROP Objectives
  - Obtain information on operating facilities and identify safety concerns
  - Evaluate the risk significance of issues to ensure the appropriate regulatory measure
  - Assess licensee performance
  - Take enforcement actions that encourage the resolution of risk-significant issues

Reactor Oversight Process (cont’d)

- ROP Objectives (cont’d)
  - Verify that licensees effectively identify problems and resolve issues
  - Provide the appropriate regulatory response to operational events on the basis of their safety significance
  - Monitor licensees and encourage them to maintain a safety-conscious work environment
Reactor Oversight Process

- Performance areas:
  - Reactor safety, radiation safety and safeguards
- Cornerstones
  - Initiating events
  - Mitigating systems
  - Barrier integrity
  - Emergency preparedness
  - Public radiation safety
  - Occupational radiation safety
  - Security

Significance Determination Process (SDP)

Non-Reactor Nuclear PRA

- Storage and transportation
- Fuel cycle facilities (FCFs)
- Advanced reactors
- Waste applications
Non-Reactor Nuclear PRA

- Differences between Reactor and Non-Reactor PRA
  - Lack of an analogue to the reactor core as found in a nuclear power reactor.
  - Different types of hazards, variable source terms throughout a process, event sequences, vulnerability duration, lack of standby systems and reliance on human actions due to process differences.
  - Differences in processing technologies which provide challenges during an analysis.

Non-Reactor Nuclear PRA

- Fuel Cycle PRA Pros and Cons
  - Resource intensive and time consuming
    - NRC is exploring which PRA technique would be more beneficial to FCFs
  - PRA techniques could improve model realism.
    - PRA has been used in FCF licensing applications and is being explored and developed for oversight applications for the Revised Fuel Cycle Oversight Process.

Non-Reactor Nuclear PRA

Four-region risk diagram:

- UNACCEPTABLE RISK (UR)
- SAFETY MARGIN (SM)
- TOLERABLE RISK (TR)
- NEGLECTIBLE RISK (NR)

Regulatory Limit

Negligible Risk
Reference Level
Oversight Process Application

- Differences between ROP and FCOP
  - FCOP would be composed of two phases
  - Thresholds
  - Cornerstones

Challenges

- Diversity of technologies and processes
- Different types of hazards
- Available information
- Limitation of databases (i.e. failure rate data)

Available NRC Guidance

- Risk informed Decision Making for Nuclear Material and Waste Applications (RIDM)

*Available at www.nrc.gov
Acronyms:
- CD - Core damage
- FCF - Fuel cycle facility
- LOCA - Loss of coolant accident
- LGSP - Loss of offsite power
- PDS - Plant damage state
- ROP - Reactor oversight process
- SDP - Significance determination process
- SGTR - Steam generator tube rupture