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Safety Assessment of Fuel Cycle Facilities – Regulatory Approaches and Industry Perspectives

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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

OECD/NEA WGFCS Workshop

Safety Assessment of Fuel Cycle Facilities – Regulatory Approaches and Industry Perspectives

Toronto, Canada, 27 – 29 September 2011
The Fairmont Royal York, 100 Front Street West, Toronto, Ontario, Canada

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 34 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

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NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 30 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, the Republic of Korea, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission also takes part in the work of the Agency.

The mission of the NEA is:

– to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as

– to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include the safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information.

The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

Within the OECD framework, the NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made of senior scientists and engineers, with broad responsibilities for safety technology and research programmes, as well as representatives from regulatory authorities. It was set up in 1973 to develop and co-ordinate the activities of the NEA concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations.

The committee's purpose is to foster international co-operation in nuclear safety amongst the NEA member countries. The CSNI's main tasks are to exchange technical information and to promote collaboration between research, development, engineering and regulatory organisations; to review operating experience and the state of knowledge on selected topics of nuclear safety technology and safety assessment; to initiate and conduct programmes to overcome discrepancies, develop improvements and research consensus on technical issues; and to promote the co-ordination of work that serves to maintain competence in nuclear safety matters, including the establishment of joint undertakings.

The clear priority of the committee is on the safety of nuclear installations and the design and construction of new reactors and installations. For advanced reactor designs the committee provides a forum for improving safety related knowledge and a vehicle for joint research.

In implementing its programme, the CSNI establishes co-operate mechanisms with the NEA's Committee on Nuclear Regulatory Activities (CNRA) which is responsible for the programme of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with the other NEA's Standing Committees as well as with key international organisations (e.g., the IAEA) on matters of common interest.

FOREWORD

Nuclear fuel is produced, processed, and stored mainly in industrial-scale facilities. Uranium ores are processed and refined to produce a pure uranium salt stream, Uranium is converted and enriched, nuclear fuel is fabricated (U fuel and U/Pu fuel for the closed cycle option); and spent fuel is stored and reprocessed in some countries (close cycle option). Facilities dedicated to the research and development of new fuel or new processes are also considered as Fuel Cycle Facilties.

The safety assessment of nuclear facilities has often been led by the methodology and techniques initially developed for Nuclear Power Plants. As FCFs cover a wide diversity of installations the various approaches of national regulators, and their technical support organizations, for the Safety Assessment of Fuel Cycle Facilities are also diverse, as are the approaches by their industries in providing safety justifications for their facilities.

The objective of the Working Group on Fuel Cycle Safety is to advance the understanding for both regulators and operators of relevant aspects of nuclear fuel cycle safety in member countries.

A large amount of experience is available in safety assessment of FCFs, which should be shared to develop ideas in this field. To contribute to this task, the Workshop on “Safety Assessment of Fuel Cycle Facilities – Regulatory Approaches and Industry Perspectives” was held in Toronto, on 27 – 29 September 2011. The workshop was hosted by Canadian Nuclear Safety Commission.

The current proceedings provide summary of the results of the workshop with the text of the papers given and presentations made.

ACKNOWLEDGMENTS

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Thanks are also expressed to the Workshop Organising Committee members, and the Session Chairs for their professional attitude, which helped to bring the workshop to a successful end.

B.R. Ravishankar, CNSC, Canada, Workshop Chair
Pierre Nocture, Areva, France, WGFCS Chair
Radomir Rehacek, OECD/NEA, Workshop Secretary
Neil Blundell, ONR, United Kingdom
Jean-Paul Daubard, IRSN, France
Kunio Fujita, JNFL, Japan
Bernhard Gmal, GRS, Germany
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Yoshinori Ueda, JNES, Japan

and the workshop attendance for its active participation and cooperation.

EXECUTIVE SUMMARY

1. Introduction

This report documents the proceedings of the Workshop on “Safety Assessment of Fuel Cycle Facilities – Regulatory Approaches and Industry Perspectives” held in Toronto, Canada, on 27 – 29 September 2011. The workshop was organised by the Committee for the Safety on Nuclear Installations (CSNI) of the OECD/Nuclear Energy Agency (NEA) and hosted by the Canadian Nuclear Safety Commission (CNSC).

More than 60 specialists representing 11 countries and international organisations attended the workshop. A total of 31 papers were presented and discussed in open forum. In addition, an opportunity was taken to consider the impacts of the events at Fukushima on Fuel Cycle Facilities (FCFs) in general.

2. Background

Nuclear fuel is produced, processed, and stored mainly in industrial-scale facilities. Uranium ores are processed and refined to produce a pure uranium salt stream, Uranium is converted and enriched, nuclear fuel is fabricated (U fuel and U/Pu fuel for the close cycle option); and spent fuel is stored and in some countries also reprocessed (close cycle option). Facilities dedicated to the research and development of new fuel or new processes are also considered as Fuel Cycle Facilities.

The safety assessment of nuclear facilities has often been led by the methodology and techniques initially developed for Nuclear Power Plants (NPPs). As FCFs cover a wide diversity of installations, the various approaches of national regulators, and their technical support organizations (TSO), for the Safety Assessment of Fuel Cycle Facilities are also diverse as are the approaches by their industries in providing safety justifications for their facilities.

3. Objectives and structure of the workshop

The objective of this workshop was to review the various approaches of national regulators, and their technical support organizations, in the Safety Assessment of Fuel Cycle Facilities and the experience of their industry in providing safety justification for their facilities. It addressed the present situation in various NEA member countries (MC) and concerned both initial safety assessment of new facilities and reassessment of existing ones (periodic safety review). It also considered trends of future improvement of safety assessment techniques.

The workshop was organised in an opening session, four technical sessions, one special session and a conclusion session. The technical sessions were focussed on:

- General approach including human aspects;
- Front end facilities;
- Chemical hazards – release limits; and
- Back end facilities.

In addition, a special session was held to discuss the lessons learnt for FCFs from the Fukushima accident in Japan. The workshop ended with an organized site visit to Cameco Corporation’s Port Hope Conversion Facility in Port Hope, Ontario on the last day of the workshop.

4. Summary of the technical and special sessions

Each session consisted of a number of presentations followed by a panel discussion moderated by the session Chairs. A summary of each session and subsequent discussion that ensued are provided below.

Session 1: General approach including human aspects

This session was split in two sub-sessions which were chaired by Bernhard Gmal (GRS, Germany) and Yoshinori Ueda (JNES, Japan), respectively. Nine papers were presented in those two sub-sessions.

In the first sub-session, four papers were presented by the safety regulatory body of Canada and France (CNSC and ASN), and the technical support organizations of France (IRSN), where different safety items relevant for licensing and/or operation were addressed, covering:

- Regulatory approach for ensuring competence of personnel in FCFs;
- Periodic safety review for FCFs;
- Development of a guidance document for performing safety assessments for different types of FCFs; and
- Evaluation of human and organizational factors for ensuring safe operation in FCFs.

The presentations gave an informative insight into different activities and measures to ensure safe operation of facilities from the regulator's side and the technical advisor's side. Two papers were presented by Nuclear Safety Authorities and two presentations were given by TSOs.

Regarding competence of personnel at nuclear facilities that has direct impact on safety, a regulatory approach for personnel training and ensuring competence in fuel cycle facilities in Canada was presented. Previously, certification requirement for personnel was applied to reactor operators, shift supervisors and health physicists in nuclear power plants and research reactors as well as exposure device operators who use nuclear substances for the purposes of industrial radiography. As a result of a regulatory review process and progress made in implementing risk-informed regulatory oversight activities as well as a formal suggestion from the International Atomic Energy Agency – International Regulatory Review Service (IRRS) conducted on the CNSC in 2009, a regulatory approach to confirming the competence of operators at Fuel Cycle Facilities has been initiated by CNSC staff. In the first step the new approach is being applied at the Port Hope Facility. The presentation explained the concept of education and training as well as certification. Questions from the audience addressed experience from implementation, in particular efforts in manpower and time for implementation.

Another important regime for ensuring safety of nuclear installations is performing a periodic safety review (PSR). This item was addressed in the presentation given by representatives of the French authority ASN. In France basic nuclear installations are subject to a PSR every ten years. In addition, an integrated approach for the PSR is required, asking the licensees to present an evaluation of all types of risks, radiological and/or chemical and to consider also human and organizational factors. Furthermore a reassessment of bounding accidents may be part of the PSR. The primarily deterministic assessment of the bounding accident can be supplemented either by the "operating conditions" method, which contains probabilistic elements, or by a probabilistic safety analyses (PSA). The application and role of PSA to FCFs was addressed during the discussion. Furthermore interconnections between requirements and the authority as well as feedback of experience were discussed. Comments also addressed the basis for development and variation of the categories of events to be considered in the safety evaluation.

A safety guide as a tool for performing assessments for a variety of different types of fuel cycle facilities was presented by IRSN experts. In France a large variety of FCFs, including front and back-end FCFs, laboratories, facilities for storage and/or waste treatment and finally facilities under decommissioning and dismantling have to be assessed by the TSO. A safety evaluation guide for being used by TSO experts in their often multidisciplinary works was introduced in the presentation.

The guide is also used for training of young experts. It is partially posted on the IRSN website for being used by external users and feed-back to the developers. Experiences from the application of the guide and availability to other than IRSN users were addressed in the discussion.

Another important aspect of operational safety of FCFs, human and organizational factors (HOF) was addressed in the next presentation. This issue has to be considered also in the frame of PSR for FCFs in France. A methodology for analysis was presented and illustrated by examples.

The importance of HOF also in other countries was emphasized in the discussion. Besides operation, also dismantling work of nuclear facilities was addressed as an example, were HOF may be a challenge, due to use of external personnel to a major extent.

The five presentations in the second sub-session were given by safety regulatory body of UK (ONR) and USA (NRC), which covered and addressed different methodologies used in accident and risk analysis in FCFs and other industries, as follows:

- Bow-tie methodology used in petrochemical industry, but unique in FCFs;
- Expectation on Design basic accident analysis in UK;
- Severe accident analysis and management in UK's FCFs;
- Comparison of Integrated safety analysis and probabilistic risk assessment;
- Use of Probabilistic risk assessment in Us FCFs.

There are same requirements in place on installations with major hazard in UK, whether nuclear or conventional, to understand and identify the hazards of their operations, the initiating events, the consequences, the prevention and mitigation barriers. However nuclear and "Seveso" type facilities seem to adopt a different approach to the presentation of their safety cases. While safety cases developed for nuclear fuel cycle facilities are rigorous, detailed and complex, this can have the effect of reducing the visibility. Operators in the oil and gas industries are choosing to use "bow tie methodology", one of in which very simple overview diagrams are produced to illustrate key hazards and corresponding protective measure. Presentation concluded that if used in the nuclear industry the diagrams would be readily accessible in the control room; the operators of nuclear facilities could further improve their understanding of the safety significance of their role in preventing major accidents and mitigating consequences.

The UK Health and Safety Executive's Nuclear Installations Inspectorate has published its most recent regulatory expectations in the 2006 version of its safety assessment principles (SAPs). The presentation given and paper submitted described the published design basis accident analysis (DBAA) logic in context with other technical aspects of the regulatory expectation for safety cases. It will further illustrate the BDAA methodology with practical examples from actual experience on reprocessing plant gained over the last 15 years or so. Among the examples the relevance of conventional safety fault initiators to nuclear safety assessment was demonstrated.

Following paper set out the UK nuclear regulatory expectation on what constitutes a severe accident, irrespective of the type of facility, and describes characteristics of severe accidents focusing on nuclear fuel cycle facilities. Key rules in assessment of severe accidents as well as the relationship to other fault analysis techniques were discussed. The role of severe accident analysis in informing accident management strategies and off-site emergency plans were covered. The paper also presented generic examples of scenarios that could lead to severe accidents in a range of nuclear fuel cycle facilities.

The U.S. Nuclear Regulatory Commission conducted a comparison of two standard tools for risk informing the regulatory process, namely, the Probabilistic Risk Assessment (PRA) and the Integrated Safety Analysis (ISA). The paper provided the background, features, and methodology associated with the PRA and ISA. The application of both methodologies to various inspection and assessment tools were discussed. The paper concluded that, while the ISA method is sufficient to establish an adequate safety basis, PRA is able to provide additional insights such as risk significance, uncertainty assessment, and prioritization of safety features.

As expressed in the Policy Statement on the Use of Probabilistic Risk Assessment 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. The last paper in session 1 described the application of PRA technology currently used in NPPs and its application in other areas such as fuel cycle facilities and advanced reactors. It described major challenges that were being faced in the application of PRA into new technical areas and possible ways to resolve them.

Session 2: Front end facilities

This session was chaired by John Kinneman (USNRC, USA) and Veronique Lhomme (IRSN, France).

Five papers were presented during this session by the safety regulatory body of Canada (CNSC), the technical support organizations of Germany (GRS) and Japan (JNES) and two operators from Canada (Cameco) and United States of America (URENCO).

The session began with a review of two recent safety-related incidents at a Canadian uranium hexafluoride (UF6) conversion facility:

- June 20, 2010: release of UF6 during filling of a 48X cylinder;
- March12, 2010: release of uranium tetrafluoride (UF4) slurry due to failure of a diaphragm valve in a slurry line.

Lessons learned from the events include assuring that procedures are written clearly, in an operator friendly manner, and that steps are completed in the proper order and that it is important to have a program for responding to small “upsets” such as small leaks. This program should include adequate maintenance procedures. In the question and discussion session, the point was made that when controls are thorough and effective, operators may become less aware of what is important in the procedures they follow; this may reflect a poor safety culture. An effective practice of posting diagrams communicating important procedural steps was described.

Then, the four following presenters focused on methodologies and practices regarding safety analysis of nuclear fuel cycle facilities, as:

- Developing a safety report for an existing conversion facility (the above mentioned Canadian UF6 conversion facility);
- Using Probabilistic safety assessment (PSA) methodology to assess the risks of complex technical systems such as a fuel fabrication plant in Germany;
- Developing Integrated safety analysis (ISA) procedures for uranium fuel fabrication and enrichment facilities in Japan;

- Performing an ISA to allow a uranium enrichment facility to operate while constructing the remainder of the facility in the USA.

The presenters discussed a number of successes and challenges with these reports and analyses:

- One significant challenge is that of preparing a new safety report for an old, existing facility: in such a facility the analysis is difficult because complete information is not available and the facility does not use modern design features or equipment;
- There were several discussions of the application of ISA methodology to facilities. These discussions included the use of risk matrices and the grading of Items Relied on for Safety (IROFS). It was underlined that, to proceed to this type of analysis, having actual operational data provides the best result, but that such data is often not available (e.g. data on human error rates). As part of this discussion, there were a number of comments on the availability of data and whether mean values were appropriate for matrix calculations. In general, the participants felt that in nearly all cases the assessments could only be semi quantitative at best. One area that received attention was human performance and error rate. There was a suggestion was that data on human error rates was available in some sources, including the US document NUREG 1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility. Others felt that the available data on human performance was not well collected and was difficult to apply. This discussion suggests that human performance data availability and application in ISAs might be a useful topic at a future meeting. It was also mentioned that when needed data for other areas of an ISA or PSA is not available, information and experience at Nuclear Power Plants (NPPs) may provide useful approximations.

There was a short discussion of the importance of encouraging conservative criteria selection and decision making for whichever analytical method (ISA or PSA) was being employed. In addition, one participant expressed the importance of avoiding analysis in isolation and avoiding "group think" when working with a team. This thought leads to a comment on the importance of some sort of independent review of the choices and conclusions made in preparing an ISA or PSA. While peer review is used in evaluating some ISA or PSA determinations, it is more difficult in many others because of proprietary concerns.

One more interesting point was the possible harmonization of the methods for conducting safety assessments between NPPs and FCFs. It seems that even though this would be desirable, it is not likely to be realized soon. In fact several participants expressed their opinion that it would never be possible due to the diversity of processes within and among the facilities. Of course, the experiences at NPPs should be looked at to see how those experiences might apply to FCFs and what benefit could be obtained from the comparison.

There was the observation from one industry representative that the approach to analyzing and accepting criticality controls varies widely among different countries and that this makes it difficult for a company to work efficiently when they have facilities in different countries. Another participant expressed the same concern about changes in transportation requirements from one country to another.

This led to some comments that in addition to harmonizing methodology between NPPs and FCFs, it would be helpful at a number of levels if the various regulators could have more uniform requirements and methods of review and approval. There was extensive discussion of the approaches used by the various countries to evaluate the current status of fuel facilities with respect to the lessons learned from the events at Fukushima.

In summary, Session 2 identified several important points, including that there is a lack of data which makes it hard to do fully quantitative analyses at FCFs; that there is a need for more work to evaluate human error frequencies; that iinformation from analyses at NPP can sometimes provide insights at FCFs; and that independent reviews of the analyses at FCF in addition to those performed by the regulators might provide some benefit.

Session 3: Chemical hazards – release limits

This session was chaired by Neil Blundell (ONR, UK) and Dave Ingalls (Cameco, Canada). Six papers were presented in this session across a diverse range of subjects.

Six papers were presented during this session by an operator from Canada (GE Hitachi), the safety regulatory body of France (ASN) and USA (NRC), the technical support organization of France (IRSN) and USA (BNL).

The papers covered setting emission limits within Canada and France, the assessment of chemical safety and fire safety within nuclear plant in France, and two papers from the United States covering the general regulation of non-reactor facilities and chemical safety within them. Most of the papers included real examples to demonstrate the practical aspects of their methodologies.

The first paper summarized the system of environmental and public protection constraints currently in place at GEH-C including derived emission limits, operational limits and action levels. It discussed the basis of any values chosen and described the methods used to calculate their magnitude. Perspectives on how to ensure operational effectiveness of these constraints were also addressed.

The French regulation in force concerning chemical hazards inside basic nuclear installations (BNI) or nuclear sites was described as a mix between previous regulation settled down in the 1990s, and the new regulation based on the Nuclear transparency and safety (TSN) act at the moment. This TSN act clearly requires taking into account all the risks generated by a BNI, which is a progress compared to the previous regulation that was not that precise. As a result, the regulation in force concerning chemical hazards inside nuclear sites can look quite complex, because it refers to several texts not necessarily linked to each other, but the regulation under development is expected to help to take all kind of risks into account more accurately and more simply.

An authorization process of released limits was described in the following presentation. The determination of release authorized limits for a French nuclear site is initiated by the request of the operator, based on the maximum nuclear and chemical inventory that could be released during normal operating conditions, accompanied with justifications. Request and justifications are analyzed and discussed by the ASN and the IRSN, taking into account nuclear and chemical inventories expected inside BNI, different regulations, operating feedback and best available technologies that can be used to treat liquid or gaseous waste before release. The release authorized limits are set up in specific ASN prescriptions based on the results of the discussion together with potential public suggestions. These prescriptions have to be ratified by the State secretaries in charge of nuclear safety.

A fire safety analysis (FSA) is requested to justify the adequacy of fire protection measures set by the operator in France. Another presentation described a recent document written by IRSN, which outlines a global process for such a comprehensive fire safety analysis. One of the key points of the fire analysis is the assessment of possible fire scenarios in the facility. Given the large number of possible fire scenarios, it is then necessary to evaluate "reference fires" which are the worst case scenarios of all possible fire scenarios and which are used by the operator for the design of fire protection measures.

Another paper of NRC and its support organisations described a methodology used to model potential accidents in fuel cycle facilities that employ chemical processes to separate and purify nuclear materials. The methodology used probabilistic risk assessment related tools as well as information about the chemical reaction characteristics, information on plant design and operational features, and generic data about component failure rates and human error rates. The accident frequency estimates for the specific reaction can be useful to help to risk-inform a safety review process and assess compliance with regulatory requirements.

Final presentation in this session described historical background of NRC in regulation of chemical safety which has been limited and that is why NRC established memoranda of understanding (MOUs) with other regulatory agencies to encourage exchange of information between the agencies regarding occupational hazards. Over the years NRC's regulation of chemical safety at fuel cycle facilities has improved, but coordination and cooperation with other regulatory agencies is essential to maintaining effective oversight of the industry.

Session 4: Back end facilities

This session was chaired by Pierre Nocture (Areva, France) and Jean-Paul Daubard (IRSN, France).

There were six papers presented during this session by the safety regulatory body of Germany (BfS) and USA (NRC), two operators from France (AREVA) and Japan (JNFL), and a research institute from Hungary (NUBIKI), when three presentations were dedicated to dry spent fuel storage facilities and other three were on the reprocessing/MOX facilities.

After introducing the licensing process in Germany, the first presentation described the assessment of the spent fuel storage in extreme events scenarios (BDA). The scenarios are based upon the imaginable impacts onto the facility and a conjecture to the possible results is drawn. By comparing the results with the regulatory framework (limits for Dose rate etc.) the identification of possible cliff edge effects is possible. Additional attention was given to the impact of the given scenarios onto the surrounding area excluding the impacts coming from the storage facility. The talk was summarized with a table were each extreme events are associated with qualitative attributes (radiological consequences, realism, and specific comments on the impacted area) that confirm the robustness of dry storage facility in respect to the given scenarios.

Further in the next presentation an overview of the PSA developed by the designer of the Paks Modular Vault Dry Storage was presented. For the PSR, the licensee decided to review the PMVDS PSA to include a complete list of initiating events, use the feed-back experience, extend the analysis the internal and external hazards, improve the modeling of power supply.

The authors of the third paper on SF storage focused on the benefit of the risk-informed decision making for long term dry storage. The paper dressed two examples, one on marine stress corrosion cracking of stainless steel canisters and a second on Hydrogen effect on Zirconium cladding integrity.

The presentation on the safety assessment implemented by AREVA on its new back-end facilities insisted on the specificities on these facilities when compared with PWR, the importance of implementing a risk by risk approach essential deterministic. The design looks for the practical elimination of accidents that needs emergency countermeasures to protect the public and not anymore a reference to a pre-determined probability/radiological consequences graph.

The first JNFL presentation introduced the simplified PSA called QSA (Quantitative Safety Assessment) thanks to the comparison of these methods for the evaluation of High Active Level Waste (HALW) storage. The supply, a leakage of an external loop, a loss of active component function at internal loop and then external are similar but the time requested for performing these assessment are quite different: QSA is one fiftieth less demanding than PSA in this example

The second presentation summarized a study performed by the Group on the Hydrogen Consumption Reaction catalyzed by Palladium ions in a simulated HALW. With the assumption of a negligible effect of alpha emitters on the radiolysis of nitrate solutions of HALW, experiments showed the important catalytic effect of PD ions for recombining H_2 with HNO_3 forming H_2O and NOx.

Special session: Lessons learnt for FCFs from the Fukushima accident

This special session was included in the workshop in response to the Fukushima accident in Japan and WGFCS' attempt to exchange information on the principal lessons learnt so far. The session was chaired by Pierre Nocture (Areva, France), the WGFCS Chair.

This special session comprised presentations of Japan, the USA, the UK and France

In the first part of his presentation Yoshinori Ueda (JNES, Japan) gave an overview of the Fukushima accident and an outline of the emergency safety measures and response at the NPP site. The second part was focused on the regulatory issues for FCFs after the accident. The first issue was the emergency safety measures in case of total loss of AC power (loss capabilities of decay heat removal and hydrogen accumulation prevention) and tsunami in the reprocessing facilities and associated spent fuel storages at Tokai and Rokkasho plants. The second issue was the directions to the licensees of these facilities to secure the work environment in the main control rooms in case of complete loss of AC power, to secure communication within the facility in case of such emergency, and to secure material and equipment for radiation protection, and to deploy heavy tools for rubble removal

In the UK, the Secretary of State requested the safety authority to "identify any lessons to be learnt by the UK nuclear industry". Neil Blundell (ONR) presented the findings stated in the interim ONR report. He more specifically highlighted the recommendations relevant to the FCF safety assessment that deals with the need to take into account "cliff-edge effects, the design basis and margins for flooding, the seismic resilience, the analysis of accident sequences for long term severe accidents, the off-site infrastructure resilience, and the human behavior in severe accident conditions. Mr. Blundell provided also 2 examples of extreme weather conditions that occurred in UK FCFs.

On the US side, John D. Kinneman (US NRC) made an introduction of the NRC near term and longer term actions for domestic operating reactors and spent fuel pools. For fuel facilities, the NRC verifies that the licensees' mitigation strategies for the licensing basis events (seismic hazards, flooding hazards, wind and tornado loading, extended loss of AC power and emergency power, fire impacts) are properly implemented with prevention and/or mitigation strategies appropriate for the consequence.

In France, the ASN asked the French licensees to undertake stress test called "complementary safety assessment (CSA). This request covers all the nuclear facilities in France, including the 15 FCFs, according to six issues (flooding, earthquake, loss of electrical power, loss of cooling systems, accidental situations management and management of subcontractors) to determine whether improvements are necessary for each installation. Ms. Dorothee Conte presented also the list of FCF facilities subjects to CSA and the provisional calendar for 2011. She provided the ASN conclusion related to the CSA methodology proposed by the licensees and gave an example of one specific and hypothetical scenario of loss of protections after a dam breach on AREVA Tricastin site. P. NOCTURE then made a presentation on the CSA methodology implemented by AREVA for its FCFs in France. The objectives of this methodology are (a) to determine the extent of facility resistance to external natural phenomena beyond their design basis and to the postulated loss of some safety functions, (b) to verify that resources that can be called-up are sufficient to limit the consequences of severe accidents that may happen simultaneously on several facilities and to limit the impacts of releases into the environment, and (c) if needed, to identify the arrangements to reinforce the organization and the equipment

Although other countries, such as Canada, did not make a formal presentation on their post-Fukushima activities, they reported that their FCFs were also examining their preparedness for beyond design basis events.

5. General Conclusions and Recommendation

WGFCS has been actively pursuing an agenda of knowledge sharing and interaction between the FCF community among its member countries. This Workshop brought together many specialists involved in the FCFs, to discuss regulatory and operational aspects. The meeting was attended by almost twice the number of participants compared to the previous technical workshop held two years ago.

The above sections have summarized the topic specific ideas that were presented and discussed by the participants. In addition, there were several broad conclusions that were drawn during the 3 day workshop:

- Participants recognized that the March 2011 events at Fukushima have lessons for the FCF communities around the world. They confirmed that the FCF community was actively seeking to learn from these events and were at various stages of review.

- Participants recognized the importance of the impact of chemical hazards on safety assessment of FCFs in addition to radiation hazards. Sometimes, public perception of low risk low radiological consequence accidents are more damaging to the reputation and safety record of the industry.

- Participants felt that there is a need for improved coordination between various regulatory bodies within each country.

- It was also noted that it may be beneficial to benchmark with other industries within the nuclear industries (such as NPPs) as well as outside (such as petrochemical industry) to identify cross-learning opportunities.

- It was recognized that the risk-informed decision making process is an established and useful approach not only at a macro level, but also useful for complex, technical tasks in FCFs.

- Both operators and regulators in member countries are engaged in continuous improvement initiatives, including improved evaluation methods and approaches, and recognize the importance of employing ALARA, in order to achieve better safety analysis and better protection of the workers, environment and the public.

- Ageing FCFs face a challenge in that they don’t necessarily have modern design features and equipment, making fully quantitative analysis challenging.

It is clear that the workshop on Safety Assessment of Fuel Cycle Facilities made the exchange of information between technical specialists from different jurisdictions possible, and highlighted some of the common aspects faced by the FCF community. It is recommended that WGFCS continue to support similar technical workshops, and continue to foster similar information exchange.

Future Directions:

WGFCS members discussed the future activities stemming from the above-noted conclusions from the Workshop. Members overwhelmingly agreed that there was benefit for the fuel cycle community in identifying and pooling together best practices within the nuclear industry and other industrial sectors, such as the chemical industry. The group also expressed a strong desire to see the result in a format that is practical and usable by the regulators as well as the FCF operators.

There was recognition that IAEA databases (such as IRS to learn from NPP experience and practices) satisfy some of this need. However the members recognized their limitation as well, since the IAEA mandate does not include chemical hazards, unless it leads to radiation hazards.

WGFCS, therefore, identified the “Development of a Reference Library of Good Practices for FCFs for Safety Assessment” as a worthwhile initiative to pursue. This reference library would likely include topics such as Remediation techniques and standards used in different countries, hazard assessments and approaches used in the chemical industry, regulatory approaches towards chemical hazards in chemical industries versus approaches to chemical hazards in the nuclear industry, etc. The purpose of this Reference Library is not to develop these reports, but to be a repository of available reports and resources on these topics.

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Information Regarding Recent Activities within OECD/NEA Working Group on Integrity and Aging of Components (WGIAGE)
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IAEA Safety Requirements for Safety Assessment of Fuel Cycle Facilities and Activities
G. JONES (IAEA, Austria)

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NEA overview and Workshop on Safety Assessment of Fuel Cycle Facilities some organisational remarks

Radomir REHACEK
NEA, Nuclear Safety Division

W S on Safety Assessment of FCFs, Toronto, Canada, 27 - 29 September 2011

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NEA overview and more

- ❑ NEA history, Mission, Committees
- ❑ Working Group on Fuel Cycle Safety
- ❑ Set up of the Workshop
- ❑ Workshop organisation and outputs

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NEA history, Mission, Committees

- ❑ Nuclear Energy Agency was set up as European Nuclear Energy Agency (ENEA) in February 1958
  - ➢ Renamed in 1972
- ❑ The NEA's current membership consists of 30 countries, in Europe, North America and the Asia-Pacific region
- ❑ http://www.oecd-nea.org

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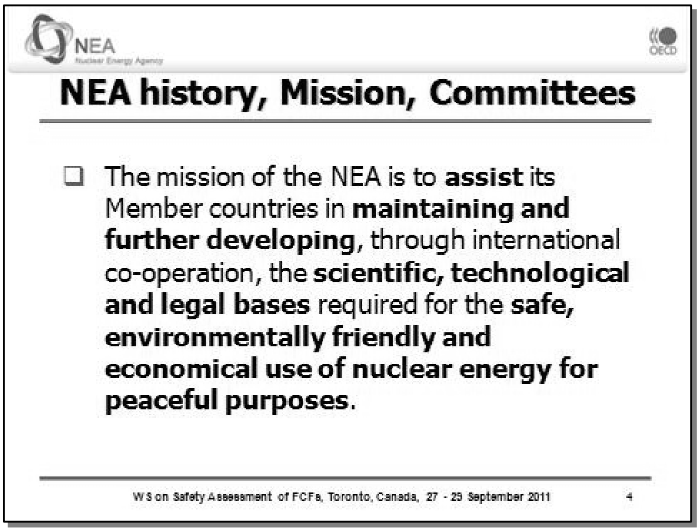
NEA
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OECD
NEA history, Mission, Committees
The mission of the NEA is to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for the safe, environmentally friendly and economical use of nuclear energy for peaceful purposes.
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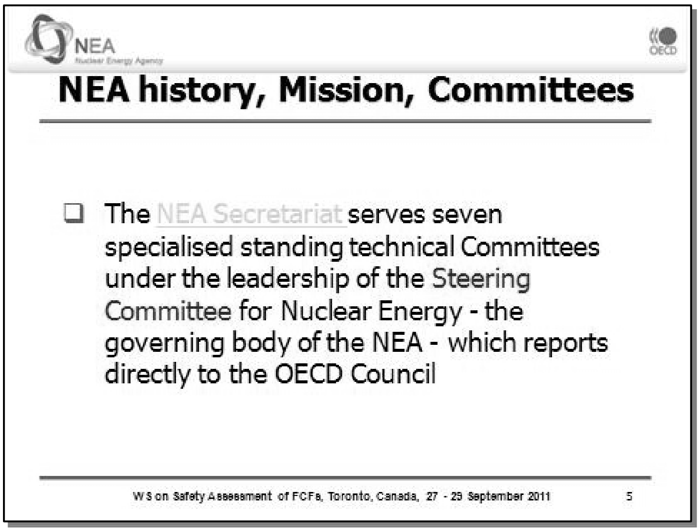
NEA
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NEA history, Mission, Committees
The NEA Secretariat serves seven specialised standing technical Committees under the leadership of the Steering Committee for Nuclear Energy - the governing body of the NEA - which reports directly to the OECD Council
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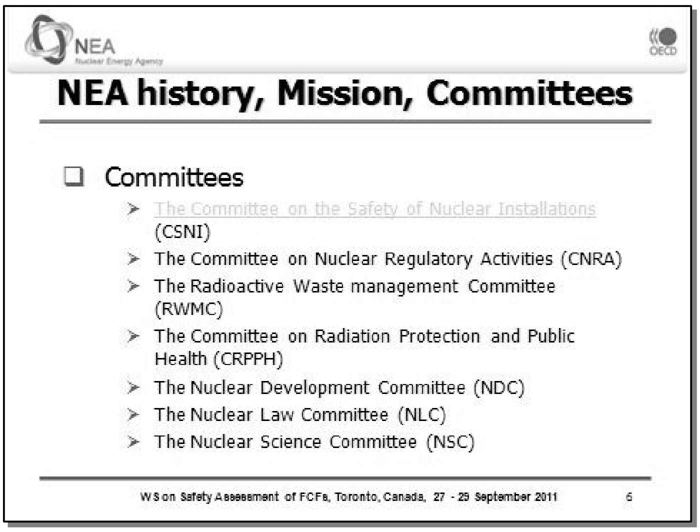
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NEA history, Mission, Committees
Committees
The Committee on the Safety of Nuclear Installations (CSNI)
The Committee on Nuclear Regulatory Activities (CNRA)
The Radioactive Waste management Committee (RWMC)
The Committee on Radiation Protection and Public Health (CRPPH)
The Nuclear Development Committee (NDC)
The Nuclear Law Committee (NLC)
The Nuclear Science Committee (NSC)
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NEA Nuclear Energy Agency — OECD

Working Group on Fuel Cycle Safety

- Mission
  - The mission of the Working Group on Fuel Cycle Safety (WGFCS) is to **advance the understanding for both regulators and operators of relevant aspects of nuclear fuel cycle safety in NEA member countries.**

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Working Group on Fuel Cycle Safety

- Scope
  - uranium mining and milling; uranium refining and conversion to uranium hexafluoride; uranium enrichment; fuel fabrication and storage (including MOX fuel); spent fuel storage; spent fuel reprocessing; decommissioning of nuclear facilities; radioactive waste management and disposal options (including for spent fuel) and the research and demonstration facilities that support these activities.
- Group's activities based on Integrated plan

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Working Group on Fuel Cycle Safety

Item	2008	2009	2010	2011	2012	2013	2014	Duration planned/real	Status
Fuel Incident Notification and Analysis System (FINAS)								Baseload	
Management of ageing of fuel cycle facilities								3,5 / 4 years	
Questionnaire									Some responses received
Workshop									Proceeded in October 2009
Proceedings									Proceedings approved by CSNI in June 2010 and published.
TOP									CSNI accepted 12/2009; TOP Contents accepted by the Group
Safety Assessment of FCFs								2,5 years	
Workshop									
Proceedings									

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Set up of the Workshop

- Background
  - Approved as WGFCS activity in December 2009
  - Host Canadian Nuclear Safety Commission
    - Cameco
- Objectives
  - to review the **various approaches** of national regulators, and their technical support organizations (TSO), in **the Safety Assessment of Fuel Cycle Facilities (FCFs)** and the experience of their industry in providing safety justification for their facilities.

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Set up of the Workshop

- WS Organising Committee

B.R. RAVISHANKAR	CNSC		Chair
Pierre NOCTURE	Areva		Co-Chair and WGFCS Chair
Neil BLUNDELL	NII		
Bernhard GMAL	GRS	Germany	
Dave INGALLS	Cameco		
Jafir JAFERI	CNSC		
Radomir REHACEK	OECD-NEA	International	Secretariat
Michael TSCHILTZ	USNRC	USA	
Yoshinori UEDA	JNES		

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Workshop organisation and outputs

- Workshop is organised in one opening session, four technical sessions, one special session and conclusion session
- **Opening and Summary Sessions**
  - Chair: ***B.R. RAVISHANKAR*** – *Workshop Chair (CNSC, Canada)*
  - Co-Chair: ***Pierre NOCTURE*** – *CSNI/WGFCS Chair (Areva, France)*

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NEA
Workshop organisation and outputs
Session 1 GENERAL APPROACH INCLUDING HUMAN ASPECTS
Sub-session 1 chaired by Bernhard GMAL (GRS, Germany)
Sub-session 2 chaired by Yoshinori UEDA (JNES, Japan)
Session 2 FRONT END FACILITIES
Session chaired by John KINNEMAN (NRC, USA) and Veronique LHOMME (IRSN, France)
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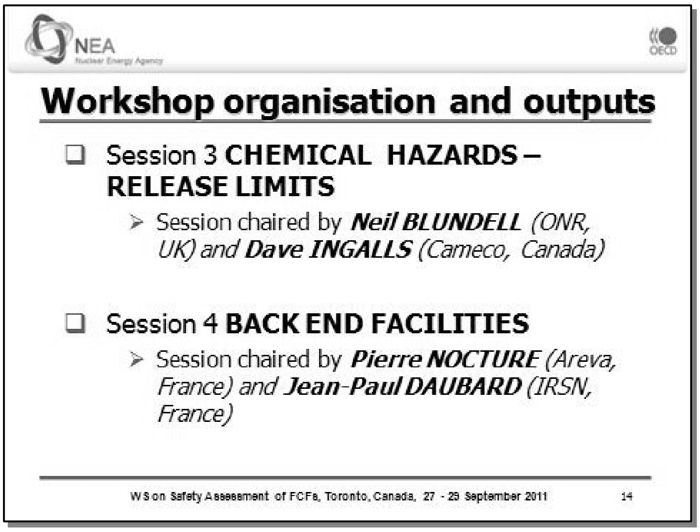
NEA
Workshop organisation and outputs
Session 3 CHEMICAL HAZARDS – RELEASE LIMITS
Session chaired by Neil BLUNDELL (ONR, UK) and Dave INGALLS (Cameco, Canada)
Session 4 BACK END FACILITIES
Session chaired by Pierre NOCTURE (Areva, France) and Jean-Paul DAUBARD (IRSN, France)
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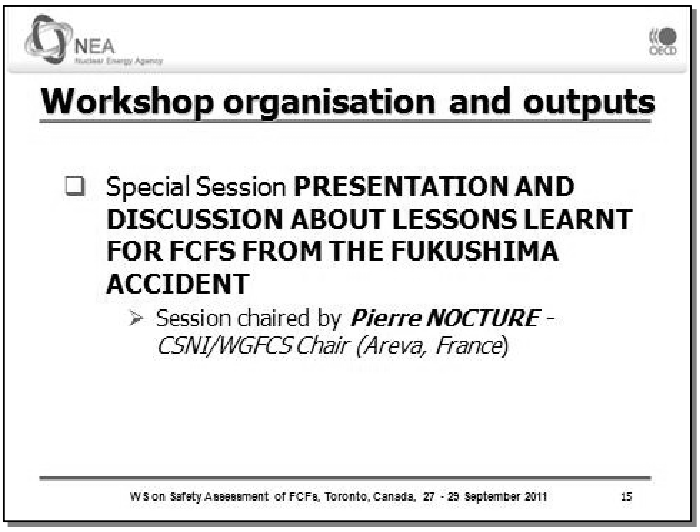
NEA
Workshop organisation and outputs
Special Session PRESENTATION AND DISCUSSION ABOUT LESSONS LEARNT FOR FCFS FROM THE FUKUSHIMA ACCIDENT
Session chaired by Pierre NOCTURE - CSNI/WGFCS Chair (Areva, France)
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Workshop organisation and outputs

- 26 papers will be presented in four technical sessions
- **20' allocated for each presentation including discussion!!!!**
  - **Green Card after 15'**
  - Yellow Card after 18'
  - **Red Card after 20' – end**
- Each technical session will be followed by 30' panel discussion
- Changes will be announced

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Workshop organisation and outputs

- Site visit of Cameco plant in Port Hope
  - Thursday, 29 September, afternoon
  - Confirmation of participation in circulating WS list of participants (V)
  - **Don't forget your ID (passport,...)**
  - Bus departure time and place will be specified on Thursday's morning

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Workshop organisation and outputs

- CNSC ...... with Handouts
- Documents on the NEA web shortly after the WS:
  - http://www.oecd-nea.org/download/fcs/papers2011/
  - Program, list of participants
  - Full Papers, Presentations
  - Conclusions of topic discussions
  - Workshop preliminary summary
- Proceedings will be published one year later

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Thanks for you attention…
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NEA
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NEA member countries
Australia
Austria
Belgium
Canada
Czech Republic
Denmark
Finland
France
Germany
Greece
Hungary
Iceland
Ireland
Italy
Japan
Luxembourg
Mexico
Netherlands
Norway
Poland
Portugal
Republic of Korea
Slovak Republic
Slovenia
Spain
Sweden
Switzerland
Turkey
United Kingdom
United States
30 Member countries
New comers in 2010: Poland, Slovenia
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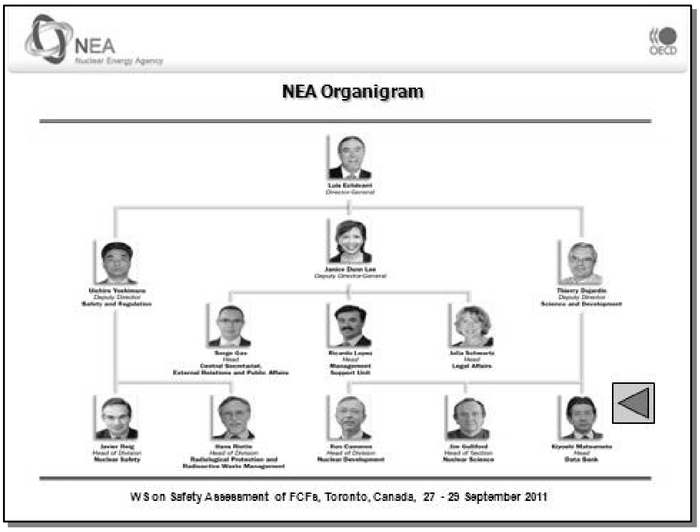
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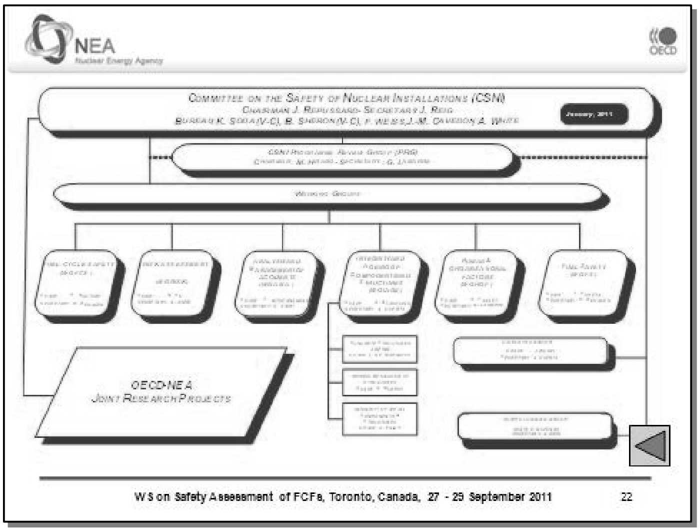
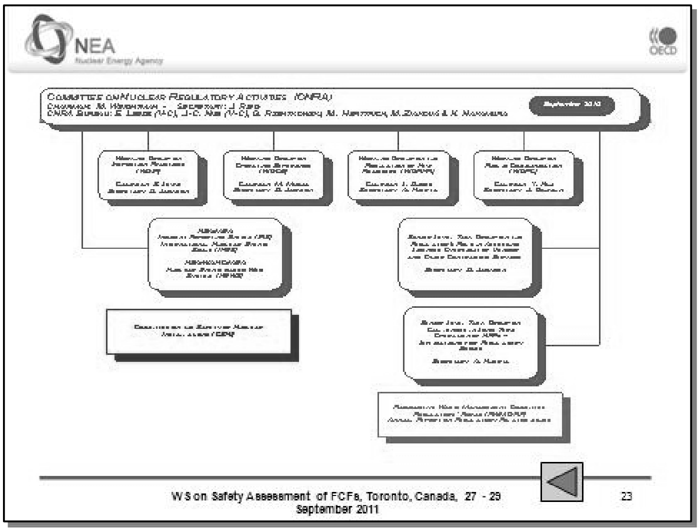
NEA Organigram
WS on Safety Assessment of FCFs, Toronto, Canada, 27 - 29 September 2011





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The 34 OECD Members

- Membership as at the end of 2009
  - 19 European Union countries
    - Austria, Belgium, Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Ireland, Italy, Luxembourg, Netherlands, Poland, Portugal, Slovak Republic, Spain, Sweden, United Kingdom
  - Switzerland, Norway, Iceland and Turkey
  - United States, Canada and Mexico
  - Japan, Korea, Australia and New Zealand

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The 34 OECD Members
New comers in 2010: Chile, Slovenia, Israel and Estonia
Accession candidate countries – Russia
Enhanced engagement countries – Brazil, China, India, Indonesia, South Africa
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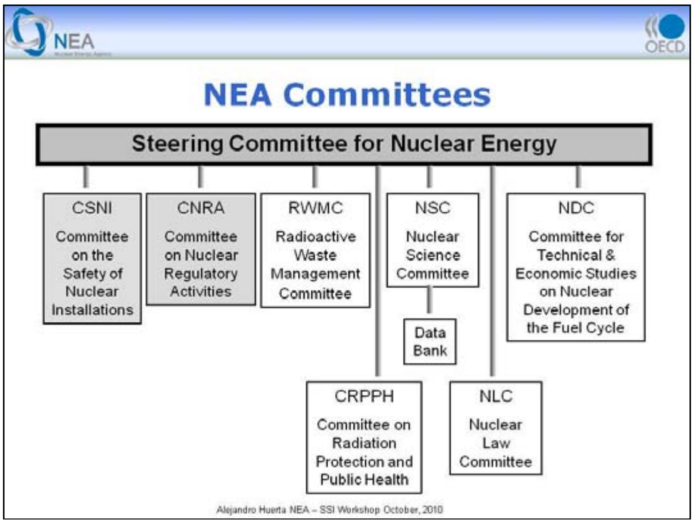
CSNI WGIAGE Activities
Information regarding recent activities within OECD-NEA Working Group on Integrity and Aging of components (WGIAGE)
Andrei Blahoianu, CNSC
WGIAGE Chairman
Presentation for the CSNI Workshop on Safety Assessment on Fuel Cycle Facilities Regulatory Approaches and Industry Perspectives
Toronto, Canada, 27-29 September, 2011


NEA
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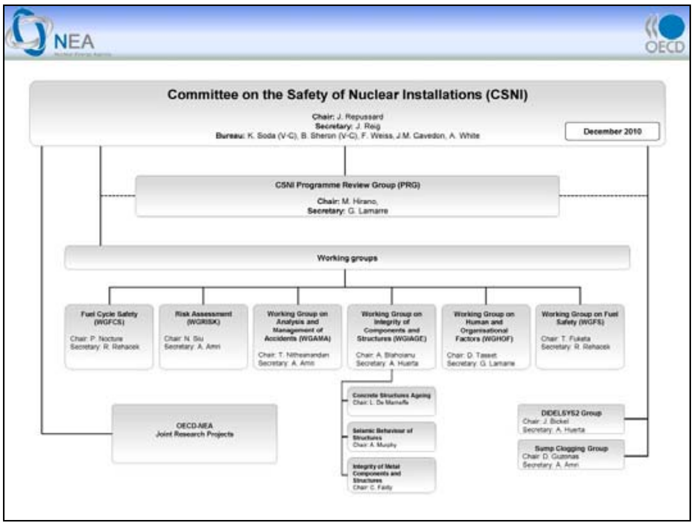
NEA Committees
Steering Committee for Nuclear Energy
CSNI
Committee on the Safety of Nuclear Installations
CNRA
Committee on Nuclear Regulatory Activities
RWMC
Radioactive Waste Management Committee
NSC
Nuclear Science Committee
Data Bank
NDC
Committee for Technical & Economic Studies on Nuclear Development of the Fuel Cycle
CRPPH
Committee on Radiation Protection and Public Health
NLC
Nuclear Law Committee
Alejandro Huerta NEA – SSI Workshop October, 2010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Committee on the Safety of Nuclear Installations (CSNI)
Chair: J. Repussard
Secretary: J. Reig
Bureau: K. Soda (V-C), B. Sheron (V-C), F. Weiss, J.M. Cavedon, A. White
December 2010
CSNI Programme Review Group (PRG)
Chair: M. Hirano,
Secretary: G. Lamarre
Working groups
Fuel Cycle Safety (WGFCS)
Chair: P. Nocture
Secretary: R. Rehacek
Risk Assessment (WGRISK)
Chair: N. Siu
Secretary: A. Amri
Working Group on Analysis and Management of Accidents (WGAMA)
Chair: T. Nitheanandan
Secretary: A. Amri
Working Group on Integrity of Components and Structures (WGIAGE)
Chair: A. Blahoianu
Secretary: A. Huerta
Working Group on Human and Organisational Factors (WGHOF)
Chair: D. Tasset
Secretary: G. Lamarre
Working Group on Fuel Safety (WGFS)
Chair: T. Fukata
Secretary: R. Rehacek
Concrete Structures Ageing
Chair: L. De Marneffe
Seismic Behaviour of Structures
Chair: A. Murphy
Integrity of Metal Components and Structures
Chair: C. Faidy
OECD-NEA
Joint Research Projects
DIDELSYS2 Group
Chair: J. Bickel
Secretary: A. Huerta
Sump Clogging Group
Chair: D. Guzonas
Secretary: A. Amri

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Current Officers

WGIAGE Chairman ***Andrei Blahoianu (CNSC)***

WGIAGE Vicechair ***Claude Faidy (EDF)***

WGIAGE Seismic Subgroup Chair ***Andrew Murphy (NRC)***

WGIAGE Concrete Subgroup Chair ---------------

WGIAGE Concrete Subgroup Vice-chair ***Syed Ali (NRC)***

WGIAGE Metal Subgroup Chair ***Claude Faidy (EDF)***

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Mandate

- Forum to exchange views, information and experience on generic technical aspects of integrity and ageing of components and structures, and review national and international programmes concentrating on research, operational aspects and regulation.
- Stimulate, in relevant technical areas, new research and recommend possible international co-operative projects.
- Develop technical positions on specific integrity issues of operating & new NPPs and research reactors covering the entire life cycle
- Identify areas where further work is needed.
- Discuss the potential impact of ageing and other challenges to integrity on the safety, regulation and operability of operating and new nuclear power plants.
- IAGE products: workshops, SOARs, topical opinion papers

NEA OECD

Highlights from the WGIAGE meetings April 04-08, 2011

- Round the table information on current issues
- Review of status of current activities (PoW)
- WGIAGE Integrated Plan updated in April 2011 in order to take into account:
  - the current status of programme of work
  - alignment with the new CSNI OP
  - member's proposal for expansion of mandate scope
- Presentation of preventive measures/lessons learned taken in each organization following Fukushima event.
- WGIAGE participation at and support for international events
- Proposal for new CAPS

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NEA OECD

Planning of WGIAGE next annual meeting

- Next IAGE Meetings (Main Group, Concrete, Metal, and Seismic) 16-20 April 2012
  - Synergy workshop to discuss the Fukushima safety implications
- Election of the Concrete subgroup Chair

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NEA OECD

Status of WGIAGE Current PoW

Title (activity)/ Objectives	Remarks and Status
Fatigue of Components and Structures Objective: To assess fatigue data transferability practices in member countries from standard specimen to structures and components including environmental effects.	Preliminary review of the responses to the questionnaire was review at the April 2011 meeting. A delay of one year in the task.
Probabilistic structural integrity of a PWR reactor pressure vessel Objective: Round Robin exercise on deterministic analysis of Pressurized thermal shock (PTS) on a cracked RPV, and on probabilistic approach of crack initiation on a cracked RPV to issue some recommendations of best practices in this area and to assure an understanding of the key parameters of this type of approach, like transient description and frequency, material properties, defect type and distribution, fracture mechanics methodology.	Significant delay in the preparation of the final report. Comments were provided to the final report in April 2011. Plan to submit it to CSNI in December 2011.

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Status of WGIAGE Current PoW

Title (activity)/ Objectives	Remarks and Status
Long-Term Operations Research Objective: To identify other technical areas of mutual interest related to age-related degradation of materials in safety-related systems, structures, components (SSCs) during long-term operation of nuclear power plants and capture operating experience associated with degradation in buried tanks and piping.	Survey developed and submitted to the group for comments. Answers are to be provided by September 2011.
Hydro-Proof Pressure Test Objective: To identify the different approaches that are followed in the different countries in the performance or non performance of hydro-proof tests and the rationale that lies behind each approach and to determine if further technical knowledge is needed to support either option and if so what kind of research could be appropriate.	Questionnaire was issued and answers received. First synthesis report for October 2011, including metallic containment.

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Status of WGIAGE Current PoW

Title (activity)/ Objectives	Remarks and Status
Leak-Before-Break Research Objective: To identify technical areas of mutual interest related to 1) the structural integrity evaluation of piping systems using deterministic and/or probabilistic methods and 2) the demonstration that flaws in piping systems will exhibit leaks prior to failure.	Survey was issued for comments. Answers are expected by the end (December 2011).
Study on Post-tensioning Methodologies in Containments Objective: To develop a systematic study that investigates the comparative advantages and disadvantages of various post-tensioning techniques in reactor containments	First expert meeting of the task group held on 20-21 April 2011. Activities are under way in the Construction, monitoring and conventional design and on the modelling and consequences analysis.

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Status of WGIAGE Current PoW

Title (activity)/ Objectives	Remarks and Status
Soil Structure Interaction (SSI) knowledge and effect on the seismic assessment of NPPs structures and components Objective: To improve the understanding of the SSI effects on the seismic behaviour of the NPPs buildings that affects the dynamic response of the internal components and structures.	The workshop was hosted by CNSC on October 6-8, 2010 in Ottawa, Canada. Workshop proceedings are being submitted to the CSNI in June 2011.
Survey on nuclear facilities that have experienced an earthquake Objective: Compile a catalogue of seismic events recorded by NPPs around the world, describing the level of the earthquake and the consequences on the plant, answering the question: 'have nuclear facilities experienced a real earthquake?'	The survey was re-activated in the framework of the IAEA Extra Budgetary Programme on the Seismic Safety of Existing NPPs.

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Status of WGIAGE Current PoW

Title (activity)/ Objectives	Remarks and Status
Program to Study Earthquake Measurements in Deep Wells and their Application Objective: Improve the accuracy of the design basis ground motion by investigating the latest information about the following topics in a Workshop format: earthquake recording techniques that can withstand the environment of a deep borehole that reaches seismic bedrock – 3 km; evaluation and analysis of ground motion measurements at deep boreholes and their use; and seismic ground motion amplification by surface strata overlying seismic bedrock.	Workshop was held on 24-26 November 2010 in Kashiwazaki, Japan, embedded in the 1st Kashiwazaki International Symposium on Seismic Safety of Nuclear Installations. Workshop proceedings under preparation.
Improving robustness assessment methodologies for structures impacted by missiles (IRIS_2010) Objective: To develop guidance that outlines effective methods of evaluating the integrity of structures impacted by missiles. It is proposed that various methods be compared in a round-robin study of impact data.	Workshop to present and discuss the benchmark results was held on 13-15 December, 2010. Workshop proceedings are under preparation.

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Status of WGIAGE Current PoW

OECD

Title (activity)/ Objectives	Remarks and Status
Task on High Energy Arcing Events (HEAF) Objective: To provide the basis for deterministic correlations to predict damage and to establish a set of input data and boundary conditions for more detailed modelling which can be agreed to by the international community.	Fourth meeting held on March 2011. HEAF final report content discussed at the meeting. HEAF research project discussed at the meeting. CSNI will be requested to endorse the project.
Seismic Engineering Knowledge Transfer Seminar Objective: Knowledge transfer and education of young generation that will be responsible for the seismic upgrading of existing NPPs and the seismic safety review of new plant designs.	The seminar programme was discussed at the April 2011 meeting and approved by the participants. Seminar to be held at the end of 2011.

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Discussion on Fukushima Event

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- Plant configuration issues (impact on units 1-4) versus units 5-6
- Issues related to multi-unit vs. single unit
- Consideration of common cause failure
- Importance of passive systems
- Assessment of current knowledge in SSI
- Underestimation of earthquake ground motion (beyond design ground motion) and the earthquake induced events (tsunami)
- Behaviour of SSCs important to safety under extreme events (BDBE)

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NEA

Discussion on Fukushima Event

OECD

- Safety reassessment of external events (combination of events/loading superposition)
- Need to address BDBE and provide quantifiable limits in national and international engineering codes and standards (ASME, ASCE, ACI, etc.)
- Role of Civil engineering measures to mitigate the consequences of SA
- Impact of extreme events upon current approach for LTO
- WGIAGE Synergy workshop to discuss the Fukushima safety implications (April 2012)

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WGIAGE proposals

- Proposal to expand the scope of the WGIAGE Seismic subgroup to include all natural external events (earthquake, tsunami, tornados, floods, etc.) in order to:
  - get a better understanding/characterization of the external events
  - provide the necessary input to concrete and mechanical WGIAGE subgroups to assess the impact of external events on the integrity of SSC.
- Increase the synergy between the WGIAGE, WGRISK and WGFCS working subgroups
- Increase the collaboration with other NEA WGs and international organizations (IAEA-ISSC, EU)

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International collaboration

- Participation at and support to SMiRT21, New Delhi, India (Nov. 6-11, 2011) - WGIAGE one of the main organizers
  - Presentation of a general paper describing WGIAGE activities relevant for the topics of SMiRT21
  - Presentation in a special session of several papers dedicated to IRIS_2010 WGIAGE research activity
  - 5 plenary workshops
    - ***Challenges of Long Term Operation***
    - ***Codes Harmonization, a challenge for Nuclear Renaissance***
    - ***Issues related to small power reactors***
    - ***Leak before break approach for operating new reactors***
    - ***Issues related to prediction of structural response under missile impact***

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QUESTIONS?

THANK YOU.

IAEA SAFETY REQUIREMENTS FOR SAFETY ASSESSMENT OF FUEL CYCLE FACILITIES AND ACTIVITIES

G. Jones
IAEA Division of Nuclear Installations Safety
Department of Nuclear Safety and Security, Austria

Abstract – The IAEA's Statute authorises the Agency to establish standards of safety for protection of health and minimisation of danger to life and property. In that respect, the IAEA has established a Safety Fundamentals publication [1] which contains ten safety principles for ensuring the protection of workers, the public and the environment from the harmful effects of ionising radiation. A number of these principles require safety assessments to be carried out as a means of evaluating compliance with safety requirements for all nuclear facilities and activities and to determine the measures that need to be taken to ensure safety. The safety assessments are required to be carried out and documented by the organisation responsible for operating the facility or conducting the activity, are to be independently verified and are to be submitted to the regulatory body as part of the licensing or authorisation process.

In addition to the principles of the Safety Fundamentals, the IAEA establishes requirements that must be met to ensure the protection of people and the environment and which are governed by the principles in the Safety Fundamentals. The IAEA's Safety Requirements publication "Safety Assessment for Facilities and Activities" [2], establishes the safety requirements that need to be fulfilled in conducting and maintaining safety assessments for the lifetime of facilities and activities, with specific attention to defence in depth and the requirement for a graded approach to the application of these safety requirements across the wide range of fuel cycle facilities and activities. Requirements for independent verification of the safety assessment that needs to be carried out by the operating organisation, including the requirement for the safety assessment to be periodically reviewed and updated are also covered.

For many fuel cycle facilities and activities, environmental impact assessments and non-radiological risk assessments will be required. The assessment of these aspects will, in general, have many commonalities with the safety assessment that is carried out to address associated radiation risks. However, it should be noted that the Safety Requirements publication [2], does not establish requirements for combining these non-radiological assessments with the radiological safety assessment, or make recommendations for how to assess non-radiological hazards; consequently these aspects are not addressed in the paper.

The requirements of the safety assessment Safety Requirements publication [2] are discussed below.

1. Overall Requirements

Graded approach

Owing to the very different levels of possible radiation risks associated with the facilities and activities across the nuclear fuel cycle, a graded approach to determining the scope and level of detail of the safety assessment is required.

The main factor to be taken into consideration in the application of a graded approach is that the safety assessment has to be consistent with the magnitude of the possible radiation risks arising from the facility or activity and their amenability to control. The term "possible radiation risks" relates to the maximum possible radiological consequences that could occur when radioactive material is released from the facility or in the activity, with no credit being taken for the safety systems or protective measures in place to prevent this.

The maturity and complexity of the facility or activity are also factors to be considered. Maturity relates to the use of proven procedures and designs, data on operational performance of similar facilities or activities, uncertainties in the performance of the facility or activity. Complexity relates to the extent and difficulty of the effort required to construct a facility or to implement an activity, the number of related processes for which control is necessary, the extent to which radioactive material has to be handled, and the reliability and complexity of systems and components, and their accessibility for maintenance, inspection, testing and repair.

The application of the graded approach needs to be reassessed as the safety assessment progresses and a better understanding is obtained of the radiation risks arising from the facility or activity. The scope and level of detail of the safety assessment are then modified as necessary and the level of resources to be applied is adjusted accordingly.

Purpose and scope of the safety assessment

The primary purposes of the safety assessment are to determine whether an adequate level of safety has been achieved for a facility or activity and whether the basic safety objectives and safety criteria established by the designer, the operating organisation and the regulatory body have been fulfilled.

These requirements include the protection of workers and the public against radiation exposure, and any other requirements for ensuring the safety of the facility or activity. The safety assessment has to include an assessment of the provisions in place for radiation protection, to determine whether radiation risks are being controlled within specified limits and constraints, and whether they have been reduced to a level that is as low as reasonably achievable. The safety assessment has to address all radiation risks that arise from normal operation and from anticipated operational occurrences and accident conditions, in which failures or internal or external events have occurred that challenge the safety of the facility or activity.

The safety assessment determines whether adequate measures have been taken to control radiation risks to an acceptable level and whether the structures, systems, components and barriers incorporated into the design fulfil the safety functions required of them. It is also determined whether adequate measures have been taken to prevent anticipated operational occurrences and accident conditions, and whether any radiological consequence can be mitigated if accidents do occur.

The safety assessment has to address all the radiation risks to individuals and population groups that arise from operation of the facility or conduct of the activity. This includes the local population and also population groups that are geographically remote from the facility or activity.

The safety assessment has to determine whether adequate defence in depth has been provided, as appropriate, through a combination of several layers of protection (e.g. physical barriers, systems to protect the barriers, and administrative procedures) that would have to fail or to be bypassed before there could be any consequences for people or the environment.

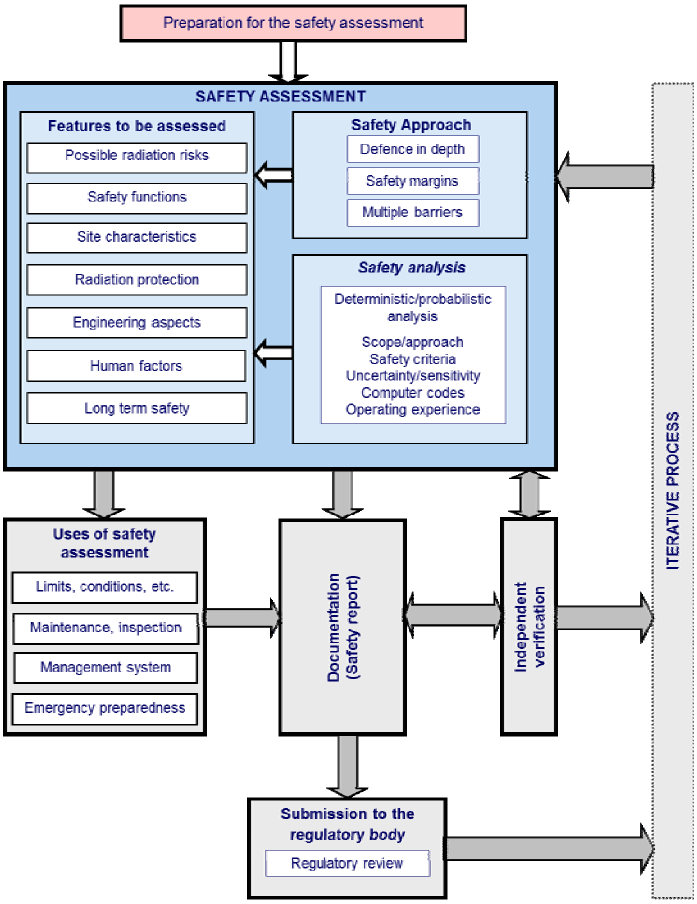
Responsibility for the safety assessment

The responsibility for carrying out the safety assessment rests with the responsible legal person; that is, the person or organisation responsible for the facility or activity — generally, the person or organisation authorized to operate the facility or to conduct the activity. The operating organisation is responsible for the way in which the safety assessment is carried out and for the quality of the results. If the operating organisation changes, the responsibility for the safety assessment has to be transferred to the new operating organisation.

2. Specific Requirements

Figure 1. Shows the main elements of the process for safety assessment and verification

Figure 1. **Overview of the safety assessment process**



The requirements associated with the main elements of safety assessment and verification are discussed below.

Preparation for the safety assessment

The necessary preparations have to be made to ensure that:

- There are a sufficient number of people with the necessary skills and expertise available to carry out the work and adequate funding is available.
- Background information relating to the location, design, construction, commissioning, operation, decommissioning and dismantling of the facility or activity, as relevant, is available, together with any other evidence that is required to support the safety assessment.
- The necessary tools for carrying out the safety assessment are available, including the necessary computer codes for carrying out the safety analysis.
- The safety criteria defined in national regulations or approved by the regulatory body to be used for judging whether the safety of the facility or activity is adequate have been identified. This could include applicable industrial safety standards and associated criteria.

Assessment of the possible radiation risks

The possible radiation risks associated with the facility or activity include the level and likelihood of radiation exposure of workers and the public, and of the possible release of radioactive material to the environment, that are associated with anticipated operational occurrences or with accidents that lead to a loss of control over a nuclear chain reaction, radioactive source or any other source of radiation.

Assessment of safety functions

All safety functions associated with a facility or activity are to be specified and assessed. This includes the safety functions associated with the engineered structures, systems and components, any physical or natural barriers and inherent safety features as applicable, and any human actions necessary to ensure the safety of the facility or activity. This is a key aspect of assessment, and is vital to the assessment of the application of defence in depth. An assessment is undertaken to determine whether the safety functions can be fulfilled for all normal operational modes, all anticipated operational occurrences and the accident conditions.

In the assessment of the safety functions, it has to be determined whether the structures, systems, components and barriers that are provided to perform the safety functions have an adequate level of reliability, redundancy, diversity, separation, segregation, independence and equipment qualification, as appropriate.

Assessment of site characteristics

An assessment of the site characteristics relating to the safety of the facility or activity has to cover:

- The physical, chemical and radiological characteristics that will affect the dispersion or migration of radioactive material released in normal operation or as a result of anticipated operational occurrences or accident conditions.
- Identification of natural and human induced external events in the region that have the potential to affect the safety of facilities and activities. This could include natural external events (such as extreme weather conditions, earthquakes and external flooding) and human induced events (such as aircraft crashes and hazards arising from transport and industrial

activities), depending on the possible radiation risks associated with the facilities and activities.

- The distribution of the population around the site and its characteristics with regard to any siting policy of the State, the potential for neighbouring States to be affected and the requirement to develop an emergency plan.

Assessment of the provisions for radiation protection

In the assessment of radiation risks, it has to be determined whether adequate measures are in place to control the radiation exposure of workers and members of the public within relevant dose limits and whether protection is optimized so that the magnitude of individual doses, the number of people exposed and the likelihood of exposures being incurred have all been kept as low as reasonably achievable, economic and social factors having been taken into account.

Assessment of engineering aspects

The design principles that have been applied for the facility have to be identified in the safety assessment, and it has to be determined whether these principles have been met. The design principles applied will depend on the type of facility but could give rise to requirements to incorporate defence in depth, multiple barriers to the release of radioactive material, and safety margins, and to provide redundancy, diversity and equipment qualification in the design of safety systems.

The safety assessment shall determine whether a facility or activity uses, to the extent practicable, structures, systems and components of robust and proven design. Where innovative improvements beyond current practices have been incorporated into the design, it has to be determined in the safety assessment whether compliance with the safety requirements has been demonstrated by an appropriate programme of research, analysis and testing complemented by a subsequent programme of monitoring during operation.

Within the safety assessment, all structures, systems and components, including software for instrumentation and control (I&C), that are items important to safety need to be identified and classified on the basis of their function and significance with regard to safety. The adequacy of the safety classification scheme and its application to the structures, systems and components has to be demonstrated in the safety assessment. It has to be determined whether the safety classification scheme adequately reflects the importance to safety of structures, systems and components, the severity of the consequences of their failure, the requirement for them to be available in anticipated operational occurrences and accident conditions, and the need for them to be adequately qualified.

The external events that could arise for a facility or activity have to be addressed in the safety assessment, and it has to be determined whether an adequate level of protection against their consequences is provided. This could include natural external events, such as extreme weather conditions, and human induced events, such as aircraft crashes, depending on the possible radiation risks associated with the facility or activity. Where there is more than one facility or activity at the same location, account has to be taken in the safety assessment of the effect of a single external event, such as an earthquake or a flood, on all of the facilities and activities, and of the potential hazards presented by each facility or activity to the others.

The internal events that could arise for a facility also have to be addressed in the safety assessment, and it has to be demonstrated whether the structures, systems and components are able to perform their safety functions under the loads induced by normal operation and the anticipated operational occurrences and accident conditions.

The safety assessment should determine whether the materials used are suitable for their purpose with regard to the standards specified in the design, and for the operational conditions that arise during normal operation and following anticipated operational occurrences or accidents that were taken into account in the design of the facility or activity.

It has to be addressed in the safety assessment whether preference has been given to a fail-safe design or, if this is not practicable, whether an effective means of detecting failures that occur has been incorporated wherever appropriate.

It has to be determined in the safety assessment whether any time related aspects, such as ageing and wear, or life limiting factors, such as cumulative fatigue, embrittlement, corrosion, chemical decomposition and radiation induced damage, have been adequately addressed. This includes the assessment of ageing management programmes for nuclear facilities.

Assessment of human factors

The safety of facilities and activities will depend on actions carried out by the operating personnel, and all such human interactions with the facility or activity are to be assessed. It has to be evaluated in the safety assessment whether personnel competences, the associated training programmes and the specified minimum staffing levels for maintaining safety are adequate. It has also to be determined in the safety assessment whether requirements relating to human factors were addressed in the design and operation of a facility or in the way in which an activity is conducted. This includes those human factors relating to ergonomic design in all areas and to human–machine interfaces where activities are carried out.

Assessment of safety over the lifetime of a facility or activity

A safety assessment is carried out at the design stage for a new facility or activity. However, the safety assessment has to cover all the stages in the lifetime of a facility or activity in which there are possible radiation risks. The assessment needs to include those activities that are carried out over a long period of time, such as the decommissioning and dismantling of a facility, the long term storage of radioactive waste, and activities in the post-closure phase of a repository for radioactive waste.

3. Defence in Depth and Safety Margins

Assessment of defence in depth

In the assessment of the application of defence in depth, it has to be determined whether adequate provisions have been made at each of the levels of defence in depth to ensure that the facility can:

- Address deviations from normal operation;
- Detect and terminate safety related deviations from normal operation should deviations occur;
- Control accidents within the limits established for the design;
- Specify measures to mitigate the consequences of accidents that exceed design limits;
- Mitigate radiation risks associated with possible releases of radioactive material.

To determine whether defence in depth has been adequately implemented, it has to be determined in the safety assessment whether:

- Priority has been given to: reducing the number of challenges to the integrity of layers of protection and physical barriers; preventing the failure or bypass of a barrier when challenged; preventing the failure of one barrier leading to the failure of another barrier; and preventing significant releases of radioactive material if failure of a barrier does occur.

- The layers of protection and physical barriers are independent of each other as far as practicable.
- Special attention has been paid to internal and external events that have the potential to adversely affect more than one barrier at once or to cause simultaneous failures of safety systems.
- Specific measures have been implemented to ensure reliability and effectiveness of the required levels of defence.

The safety assessment should determine whether there are adequate safety margins in the design and in the operation of the facility, or in the conduct of the activity, in normal operation and in anticipated operational occurrences or accident conditions. Safety margins are typically specified in codes and standards as well as by the regulatory body. It has to be determined in the safety assessment whether acceptance criteria for each aspect of the safety analysis are such that an adequate safety margin is ensured.

4. Safety Analysis

Scope of the safety analysis

The consequences arising from all normal operational conditions and the frequencies and consequences associated with all anticipated operational occurrences and accident conditions have to be addressed in the safety analysis. This includes accidents that have been taken into account in the design (referred to as design basis accidents) as well as beyond design basis accidents (including severe accidents) for facilities and activities where the radiation risks are high. The analysis has to be performed to a scope and level of detail that correspond to the magnitude of the radiation risk associated with the facility or activity, the frequency of the events included in the analysis, the complexity of the facility or activity, and the uncertainties inherent in the processes that are included in the analysis.

Anticipated operational occurrences and accident conditions that challenge safety are to be identifie in the safety analysis. This includes all internal and external events and processes that may have consequences for physical barriers for confining the radioactive material or that otherwise give rise to radiation risks. The features, events and processes to be considered in the safety analysis are to be selected on the basis of a systematic, logical and structured approach, and justification has to be provided that the identification of all scenarios relevant for safety is sufficiently comprehensive.

Relevant operating experience has to be taken into account in the safety analysis. This includes operating experience from the actual facility or activity, where available, and operating experience from similar facilities and activities.

Deterministic and probabilistic approaches

Deterministic and probabilistic approaches have been shown to complement one another and can be used together to provide input into an integrated decision making process. The extent of the deterministic and probabilistic analyses carried out for a facility or activity has to be consistent with the graded approach.

The aim of the deterministic approach is to specify and apply a set of conservative deterministic rules and requirements for the design and operation of facilities or for the planning and conduct of activities. When these rules and requirements are met, they are expected to provide a high degree of confidence that the level of radiation risks to workers and members of the public arising from the facility or activity will be acceptably low. This conservative approach provides a way of compensating for uncertainties in the performance of equipment and the performance of personnel, by providing a large safety margin.

The objectives of a probabilistic safety analysis are to determine all significant contributing factors to the radiation risks arising from a facility or activity, and to evaluate the extent to which the overall design is well balanced and meets probabilistic safety criteria where these have been defined. The probabilistic approach uses realistic assumptions whenever possible and provides a framework for addressing many of the uncertainties explicitly.

Criteria for judging safety

Criteria for judging safety, sufficient to meet the requirements of the designer, the operating organisation and the regulatory body, have to be defined for the safety analysis.

Uncertainty and sensitivity analysis

The safety analysis incorporates, to varying degrees, predictions of the circumstances that will prevail in the operational or post-operational stages of a facility or activity. There will always be uncertainties associated with such predictions that will depend on the nature of the facility or activity and the complexity of the safety analysis. These uncertainties have to be taken into account in the results of the safety analysis and the conclusions drawn from it.

Uncertainties in the safety analysis have to be characterized with respect to their source, nature and degree, using quantitative methods, professional judgement or both. Uncertainties that may have implications for the outcome of the safety analysis and for decisions made on that basis are to be addressed in uncertainty and sensitivity analyses. Uncertainty analysis refers mainly to the statistical combination and propagation of uncertainties in data, whereas sensitivity analysis refers to the sensitivity of results to major assumptions about parameters, scenarios or modelling.

Use of calculation methods and computer codes

Any calculation methods and computer codes used in the safety analysis have to undergo verification and validation. Model verification is the process of determining that a computational model correctly implements the intended conceptual model or mathematical model; that is, whether the controlling physical equations and data have been correctly translated into the computer code. System code verification is the review of source coding in relation to its description in the system code documentation. Model validation is the process of determining whether a mathematical model is an adequate representation of the real system being modelled, by comparing the predictions of the model with observations of the real system or with experimental data.

Use of operating experience data

If warranted by the possible radiation risks associated with a facility or activity, data on operational safety performance have to be collected and assessed, including records of incidents such as human errors, the performance of safety systems, radiation doses, and the generation of radioactive waste and effluents. The scope of the data to be collected for facilities and activities has to be in accordance with the graded approach. For complex facilities, data are to be collected on the basis of a set of safety performance indicators that have been established for the facility.

5. Documentation

Documentation of the safety assessment

The results and findings of the safety assessment are to be documented, as appropriate, in the form of a safety report that reflects the complexity of the facility or activity and the radiation risks associated with it.

The safety report presents the assessments and the analyses that have been carried out for the purpose of demonstrating that the facility or activity is in compliance with the requirements established by the designer, the operating organisation and the regulatory body, and any other safety requirements as established in national laws and regulations.

The quantitative and qualitative outcomes of the safety assessment form the basis for the safety report. The outcomes of the safety assessment are supplemented by supporting evidence for and reasoning about the robustness and reliability of the safety assessment and its assumptions, including information on the performance of individual components of systems as appropriate. The safety report has to document the safety assessment in sufficient scope and detail to support the conclusions reached and to provide an adequate input into independent verification and regulatory review. The safety report includes:

- A justification for the selection of the anticipated operational occurrences and accidents considered in the analysis.
- An overview and necessary details of the collection of data, the modelling, the computer codes and the assumptions made.
- Criteria used for the evaluation of the modelling results.
- Results of the analysis covering the performance of the facility or activity, the radiation risks incurred and a discussion of the underlying uncertainties;
- Conclusions on the acceptability of the level of safety achieved and the identification of necessary improvements and additional measures.

The safety report is to be updated as necessary. The safety report has to be retained until the facility has been fully decommissioned and dismantled or the activity has been terminated and released from regulatory control.

6. Independent Verification

Independent verification

The operating organisation is to carry out an independent verification to increase the level of confidence in the safety assessment before it is used by the operating organisation or submitted to the regulatory body.

The independent verification is performed by suitably qualified and experienced individuals or a group different from those who carried out the safety assessment. The aim of independent verification is to determine whether the safety assessment has been carried out in an acceptable way.

The independent verification has to combine an overall review, to determine whether the safety assessment carried out is comprehensive, with spot checks in which a much more detailed review is carried out that focuses on those aspects of the safety assessment that have the highest impact on the radiation risks arising from the facility or activity. It also has to be considered in the independent verification whether there are any contributions to the radiation risks that have not been taken into account.

In addition, the regulatory body has to carry out a separate independent verification to satisfy itself that the safety assessment is acceptable and to determine whether it provides an adequate demonstration of whether the legal and regulatory requirements are met. The verification by the regulatory body is not part of the operating organisation’s process and is not to be used or claimed by the operating organisation as part of its independent verification.

7. Management, Use and Maintenance of the Safety Assessment

Management of the safety assessment

The processes by which the safety assessment is produced shall be planned, organised, applied, audited and reviewed.

Use of the safety assessment

The safety assessment provides one of the inputs into defining the limits and conditions that are to be implemented by means of suitable procedures and controls. These procedures and controls have to include a means for monitoring to ensure that the limits and conditions are complied with at all times.

The results of the safety assessment have to be used to specify the programme for maintenance, surveillance and inspection to be established, which will use procedures and controls that are auditable to ensure that:

- All necessary conditions are maintained;
- All structures, systems and components maintain their integrity and functional capability over their required lifetime.

The results of the safety assessment have to be used to specify the procedures to be put in place for all operational activities significant to safety and for responding to anticipated operational occurrences and to accidents. The safety assessment is also to be used as an input into planning for on-site and off-site emergency response and accident management.

The results of the safety assessment are to be used to specify the necessary competences for the staff involved in the facility or activity, which are used to inform their training, control and supervision.

The results of the safety assessment have to be used to make decisions in an integrated, risk informed approach, by means of which the results and insights from the deterministic and probabilistic assessments and any other requirements are combined in making decisions on safety matters in relation to the facility or activity.

Maintenance of the safety assessment

The safety assessment in itself cannot achieve safety. Safety can only be achieved if the input assumptions are valid, the derived limits and conditions are implemented and maintained, and the assessment reflects the facility or activity as it actually is at any point in time. Facilities and activities change and evolve over their lifetimes (e.g. through construction, commissioning, operation, and decommissioning and dismantling or closure) and with modifications, improvements and effects of ageing. Knowledge and understanding also advance with time and experience. The safety assessment has to be updated to reflect such changes and to remain valid.

The safety assessment has to be periodically reviewed and updated at predefined intervals in accordance with regulatory requirements. Periodic review may need to be carried out more frequently to take into account:

- Any changes that may significantly affect the safety of the facility or activity;
- Significant developments in knowledge and understanding (such as developments arising from research or operating experience);

- Emerging safety issues due to a regulatory concern or a significant incident;
- Safety significant modifications to the computer codes, or changes in the input data used in the safety analysis.

8. Summary

The IAEA's Safety Requirements publication "Safety Assessment for Facilities and Activities" [2], establishes the safety requirements that need to be fulfilled in conducting and maintaining safety assessments for the lifetime of facilities and activities.

Safety assessment is the systematic process that is carried out throughout the lifetime of the facility or activity to ensure that all the relevant safety requirements are met by the design. It is required to be conducted within a management system. The processes by which it is produced have to be planned, organised, applied, audited *and reviewed in a way that is in accordance with the graded approach. The safety assessment has t*o determine whether adequate defence in depth has been provided through a combination of several layers of protection.

9. References

[1] European Atomic Energy Community, Food and Agriculture Organization of The United Nations, International Atomic Energy Agency, International Labour Organization, International Maritime Organization, Oecd Nuclear Energy Agency, Pan American Health Organization, United Nations Environment Programme, World Health Organization, Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1, IAEA, Vienna (2006).

[2] International Atomic Energy Agency, Safety Assessment for Facilities and Activities, IAEA Safety Standards Series No GSR Part 4, IAEA, Vienna (2009).

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CSNI Workshop on Safety Assessment of Fuel Facility Facilities
Regulatory Approaches and Industry Perspectives

IAEA Safety Requirements for Safety Assessment of Fuel Cycle Facilities and Activities

Mr G Jones (NSNI)

IAEA
International Atomic Energy Agency

GSR Part 4 "Safety Assessment" CONTENT

- Structure of IAEA Safety Standards
- GSR Part 4 "Safety Assessment for Facilities and Activities"
  - Objective
  - Scope
  - Requirements:
    - Overall requirements
    - Specific requirements
- Conclusion

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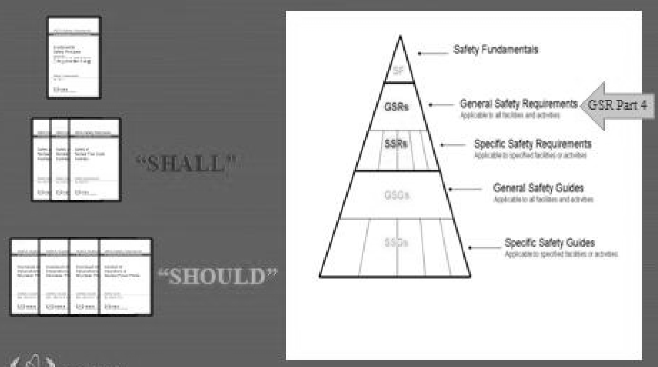
STRUCTURE OF IAEA SAFETY STANDARDS



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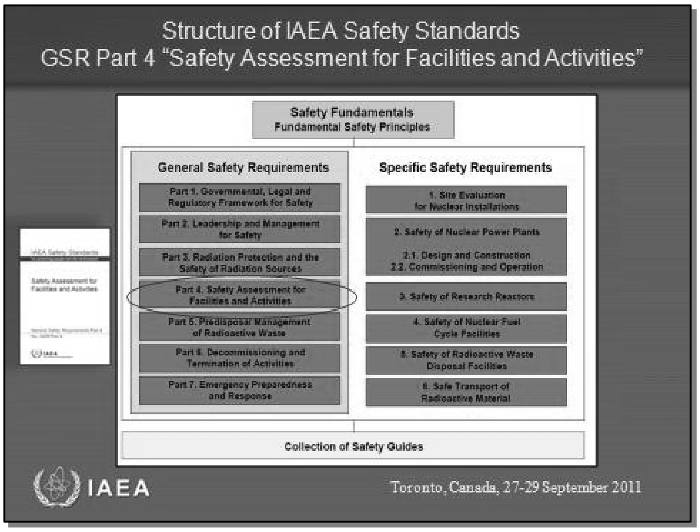
Structure of IAEA Safety Standards
GSR Part 4 "Safety Assessment for Facilities and Activities"
Safety Fundamentals
Fundamental Safety Principles
General Safety Requirements
Part 1. Governmental, Legal and Regulatory Framework for Safety
Part 2. Leadership and Management for Safety
Part 3. Radiation Protection and the Safety of Radiation Sources
Part 4. Safety Assessment for Facilities and Activities
Part 5. Predisposal Management of Radioactive Waste
Part 6. Decommissioning and Termination of Activities
Part 7. Emergency Preparedness and Response
Specific Safety Requirements
1. Site Evaluation for Nuclear Installations
2. Safety of Nuclear Power Plants
2.1. Design and Construction
2.2. Commissioning and Operation
3. Safety of Research Reactors
4. Safety of Nuclear Fuel Cycle Facilities
5. Safety of Radioactive Waste Disposal Facilities
6. Safe Transport of Radioactive Material
Collection of Safety Guides
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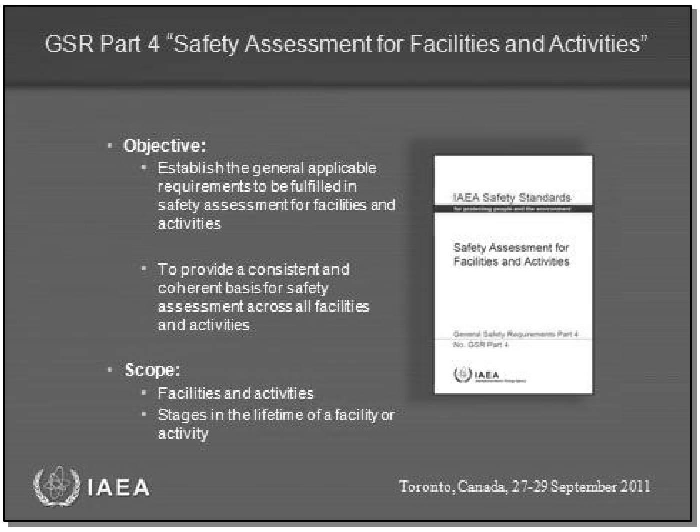
GSR Part 4 "Safety Assessment for Facilities and Activities"
Objective:
Establish the general applicable requirements to be fulfilled in safety assessment for facilities and activities
To provide a consistent and coherent basis for safety assessment across all facilities and activities
Scope:
Facilities and activities
Stages in the lifetime of a facility or activity
IAEA Safety Standards
Safety Assessment for Facilities and Activities
IAEA
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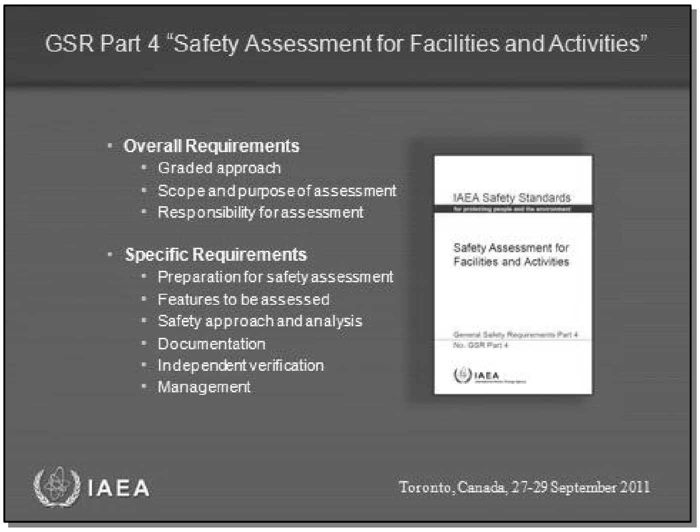
GSR Part 4 "Safety Assessment for Facilities and Activities"
Overall Requirements
Graded approach
Scope and purpose of assessment
Responsibility for assessment
Specific Requirements
Preparation for safety assessment
Features to be assessed
Safety approach and analysis
Documentation
Independent verification
Management
IAEA Safety Standards
Safety Assessment for Facilities and Activities
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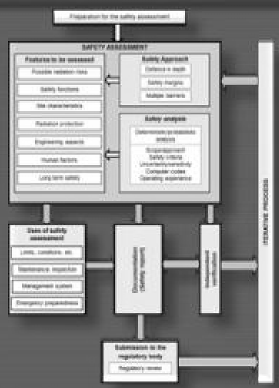
GSR Part 4 "Safety Assessment for Facilities and Activities"

- **Safety Assessment:**
  - Is a systematic process that is carried out throughout the lifetime of the facility or activity to ensure that all the relevant safety requirements are met by the design. Safety assessment includes, but is not limited to, the formal safety analysis.



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GSR Part 4 "Safety Assessment"
OVERALL REQUIREMENTS – Graded approach

- **Requirement 1: Graded approach**
  - *"A graded approach **shall** be used in determining the scope and level of detail of the safety assessment carried out in a particular State for any particular facility or activity, consistent with the magnitude of the potential radiation risks arising from the facility or activity."*

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GSR Part 4 "Safety Assessment"
OVERALL REQUIREMENTS – Responsibility for assessment

- **Requirement 3: Responsibility for safety assessment**
  - *"The responsibility for carrying out the safety assessment **shall** rest with the responsible legal person, i.e. the person or organization responsible for the facility or activity."*

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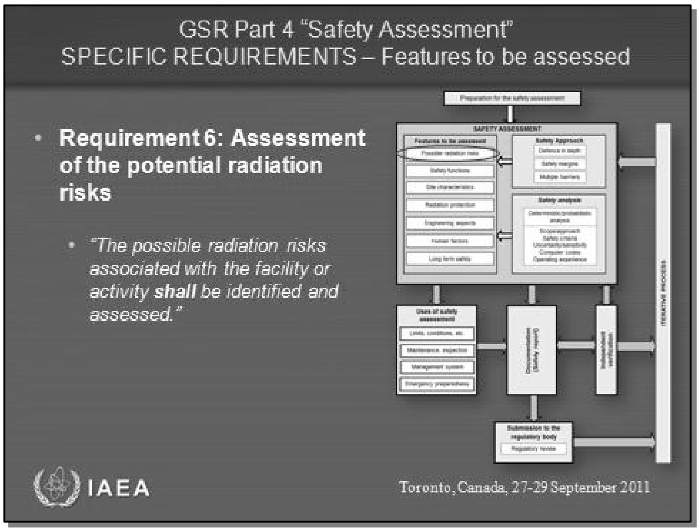
GSR Part 4 "Safety Assessment"
SPECIFIC REQUIREMENTS – Features to be assessed
Requirement 6: Assessment of the potential radiation risks
"The possible radiation risks associated with the facility or activity **shall** be identified and assessed."
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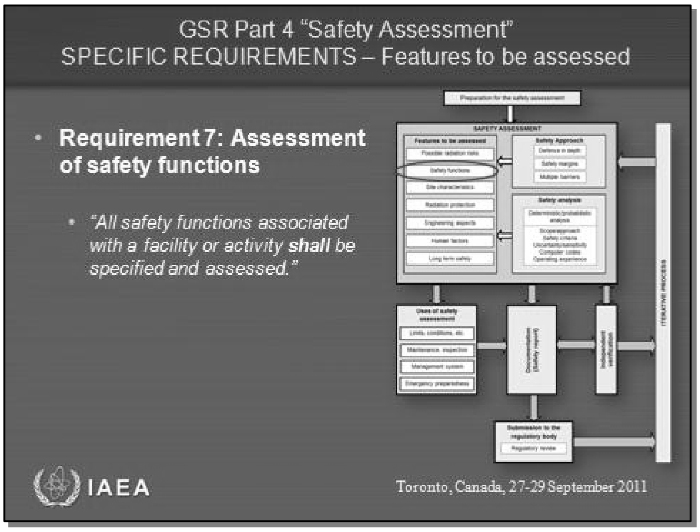
GSR Part 4 "Safety Assessment"
SPECIFIC REQUIREMENTS – Features to be assessed
Requirement 7: Assessment of safety functions
"All safety functions associated with a facility or activity **shall** be specified and assessed."
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GSR Part 4 "Safety Assessment"
SPECIFIC REQUIREMENTS – Features to be assessed
Requirement 9: Assessment of the provisions for radiation protection
"It **shall** be determined in the safety assessment for a facility or activity whether adequate measures are in place to protect people and the environment from harmful effects of ionizing radiation."
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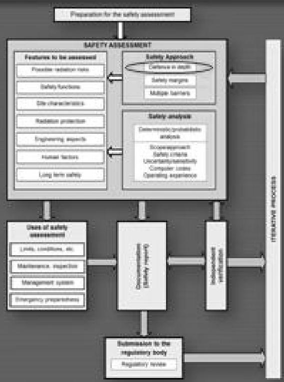
GSR Part 4 "Safety Assessment"
SPECIFIC REQUIREMENTS – Safety approach/analysis

- **Requirements 13: Assessment of defence in depth**
  - *"It **shall** be determined in the assessment of defence in depth whether adequate provisions have been made at each of the levels of defence in depth."*



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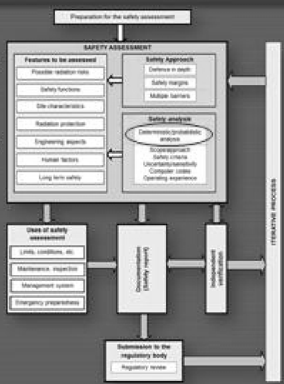
GSR Part 4 "Safety Assessment"
SPECIFIC REQUIREMENTS – Safety approach/analysis

- **Requirement 15: Deterministic and probabilistic approaches**
  - *"Both deterministic and probabilistic approaches **shall** be included in the safety analysis."*



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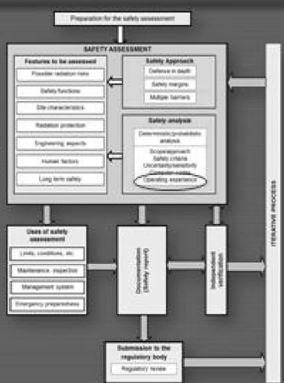
GSR Part 4 "Safety Assessment"
SPECIFIC REQUIREMENTS – Safety approach/analysis

- **Requirement 19: Use of operating experience data**
  - *"Data on operational safety performance **shall** be collected and assessed."*



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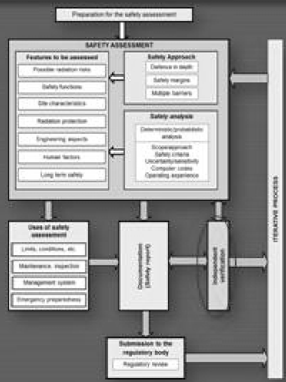
GSR Part 4 "Safety Assessment"
SPECIFIC REQUIREMENTS – Independent verification

- **Requirement 21: Independent verification**
  - *"The operating organization **shall** carry out an independent verification of the safety assessment before it is used by the operating organization or submitted to the regulatory body."*



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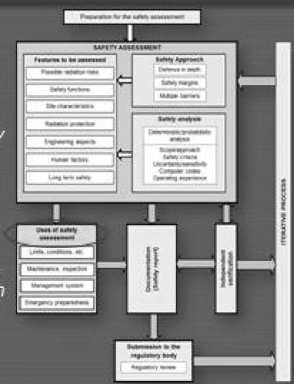
GSR Part 4 "Safety Assessment"
SPECIFIC REQUIREMENTS – Use of safety assessment

- **Requirement 23: Use of the safety assessment**
  - *"The results of the safety assessment shall be used to specify the programme for maintenance, surveillance and inspection; to specify the procedures to be put in place for all operational activities significant to safety and for responding to anticipated operational occurrences and accidents; to specify the necessary competences for the staff involved in the facility or activity and to make decisions in an integrated, risk informed approach."*



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GSR Part 4 "Safety Assessment"
CONCLUSION

- The IAEA's Safety Requirements publication GSR Part 4 "Safety Assessment for Facilities and Activities" establishes the safety requirements that need to be fulfilled in conducting and maintaining safety assessments for the lifetime of facilities and activities.
- Safety assessment is the systematic process that is carried out throughout the lifetime of the facility or activity to ensure that all the relevant safety requirements are met by the design.
- It is required to be conducted within a management system. The processes by which it is produced have to be planned, organized, applied, audited and reviewed in a way that is in accordance with the graded approach.
- The safety assessment has to determine whether adequate defence in depth has been provided through a combination of several layers of protection.

IAEA

Toronto, Canada, 27-29 September 2011

SESSION ONE

GENERAL APPROACH INCLUDING HUMAN ASPECTS

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CONFIRMING COMPETENCE OF OPERATORS - A REGULATORY APPROACH TO FUEL CYCLE FACILITIES

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Abstract – For the past 40 years the Canadian Nuclear Safety Commission (CNSC), formerly the Atomic Energy Control Board, has certified workers in nuclear facilities. The requirement for certified personnel has ensured that workers assigned to positions that have a direct impact on the safe operation of the facility are fully qualified to perform their duties. This certification regime is defined in the regulatory framework under which the CNSC operates.

Traditionally, this certification regime has been applied to Reactor Operators, Shift Supervisors and Health Physicists in Nuclear Power Plants and research reactors as well as to Exposure Device Operators who use nuclear substances for the purposes of industrial radiography.

Stemming from progress made in implementing risk-informed regulatory oversight activities as well as a formal suggestion from the International Atomic Energy Agency – International Regulatory Review Service (IRRS) conducted on the CNSC in 2009, a regulatory approach to confirming the competence of Operators at Fuel Cycle Facilities has been initiated by CNSC staff. In the first stage of the implementation of this new regulatory approach, the CNSC had Cameco Corporation implement a formal internal qualification programme for the UF_6 Operators at its Port Hope Conversion Facility (PHCF) in Port Hope, Ontario.

In the future, following a review of the results of the qualification programme at the PHCF, the CNSC staff will evaluate the need for the application of a similar regulatory approach to confirm the competence of the Operators at other Fuel Cycle Facilities in Canada.

1. Background

The CNSC has a long history of certifying workers at nuclear power plants and non-power research reactor facilities. The requirement for certified personnel at these facilities has ensured that workers assigned to positions that have a direct impact on the safe operation of the facility are qualified to perform their duties.

Prior to the coming into force of the *Nuclear Safety and Control Act* (NSCA) [1] in 2000, the CNSC issued certifications to workers that remained valid until the worker ceased to be employed at the facility. When the NSCA came into force, the *Class I Nuclear Facilities Regulations* [2], made pursuant to the NSCA, defined the requirement that a certification for a worker at a Class IA nuclear facility, i.e., a nuclear power plant (NPP) or a non-power research reactor facility, is valid for a period of five years. In addition to this new requirement to renew a person's certification every five years, in 2003 the CNSC introduced the requirement for the knowledge- and performance-based requalification testing of all certified shift workers.

In addition to the certification of workers at nuclear reactor facilities, pursuant to the *Nuclear Substances and Radiation Devices Regulations* [3], the CNSC also certifies Exposure Device Operators who use nuclear substances for the purposes of industrial radiography. Since industrial radiography has the potential for high radiation doses, prior to operating an exposure device, all workers must complete each phase of a multi-step certification process to ensure that they possess the required knowledge and skills to safely operate the device.

The *Class I Nuclear Facilities Regulations* do not require the certification of any Fuel Cycle Facilities (FCF) workers. However, the *General Nuclear Safety and Control Regulations* [4] require all FCF licensees to ensure that they employ a sufficient number of qualified workers to safely perform the activity permitted by their FCF operating licence. In addition, these regulations require FCF licensees to train their workers in order to ensure that they can perform the duties of their positions safely. As a result, it is the responsibility of the licensee to ensure that all of their workers are trained, qualified and competent. The methods used by the FCF licensees to ensure that their workers are trained, qualified and competent vary significantly between the various facilities.

2. Introduction

Over the past ten years, the CNSC has been implementing improvement programmes in order to continue to ensure that the health and safety of Canadians is protected. These improvement initiatives stem from two external audits of the CNSC: an audit by the Office of the Auditor General of Canada and an audit by an International Regulatory Review Service.

Between 1999 and 2000, the Office of the Auditor General of Canada (OAG) conducted an audit of the CNSC's approach to the regulation of Power Reactors. The OAG audit report [5] recommended areas where the CNSC needed to improve its regulatory regime for power reactors to ensure that the CNSC continued to protect the health and safety of Canadians. One of the findings of the audit was that the CNSC's regulatory activities were "not based on a rigorous, well-documented system of risk analysis".

In response to the OAG audit, the CNSC committed to implement a risk-informed approach to all of its regulatory activities [6]. The CNSC then embarked on an ambitious multi-year improvement

[1] Nuclear Safety and Control Act (NSCA), Canada Gazette Part III, May 9, 1997.

[2] Class I Nuclear Facilities Regulations, Extract Canada Gazette, Part II June 21, 2000.

[3] Nuclear Substances and Radiation Devices Regulations, Extract Canada Gazette, Part II June 21, 2000.

[4] General Nuclear Safety and Control Regulations, Extract Canada Gazette, Part II June 21, 2000.

[5] 2000 December Report of the Auditor General of Canada, Chapter 27 – Canadian Nuclear Safety Commission - Power Reactor Regulation, December 2000. http://www.oag-bvg.gc.ca/internet/English/parl_oag_200012_27_e_11214.html

[6] Canadian Nuclear Safety Commission Action Plan – Report of the Auditor General of Canada – Power Reactor Regulation, February 2001. http://www.nuclearsafety.gc.ca/eng/pdfs/OAG_Report.pdf

programme in all regulatory areas to improve its policies and programmes to ensure that the resources allocated to regulatory activities were based on a risk-informed approach [[7], [8]]. In a follow-up audit conducted in 2003 and 2004, the OAG found that the CNSC had made significant progress toward implementing a risk-managed approach to the allocation of resources which included the development of a systematic, risk-informed approach for the regulation of FCFs [[9]].

This systematic, risk-informed approach to FCF resource allocation was developed by CNSC staff following a series of internal workshops. These workshops produced a matrix that ranked the risk of each FCF in each of the 14 CNSC Safety and Control Areas. The CNSC Safety and Control Areas (SCA) are the technical topics that CNSC staff use across all regulated facilities and activities to evaluate, verify and report on regulatory requirements and performance. The risk ranking for the FCFs is then used by CNSC staff to ensure that the regulatory effort allocated to each facility is based on the facility's risk ranking in each SCA. The use of this structured risk-informed approach to the planning of regulatory activities for FCFs has significantly improved how resources are allocated within the CNSC.

The second major CNSC improvement initiative for FCFs stemmed from an external audit by an International Regulatory Review Service (IRRS). In 2009, the CNSC welcomed a peer review of its regulatory regime and regulatory processes by an international team of experts selected by the International Atomic Energy Agency (IAEA). This IRRS team reviewed the CNSC regulatory operations and identified opportunities for improvement as suggestions, recommendations or good practices. One of the suggestions presented by the IRRS team was that CNSC staff should review their current approach and continue to adopt a consistent process for confirming the competence of operators at FCFs commensurate with the risks and hazards posed by the facilities [[10]].

The risk-informed approach to the regulation of FCFs recommended by the OAG report and the recent IRRS suggestion led CNSC staff to initiate the development and implementation of a new regulatory approach to confirm the competence of operators at fuel cycle facilities.

3. The regulatory approach to confirming competence of operators at fuel cycle facilities

In December 2010, CNSC staff in the Training Program Evaluation Division and the Personnel Certification Division initiated the development of a regulatory approach that would provide additional assurance that the operators at FCFs are competent to safely perform the duties of their positions. To accomplish this goal, CNSC staff determined that operator qualification programmes are to be implemented at FCFs and decided that these qualification programmes are to be based on the following principles:

[[7]] 2002 Report of the Canadian Nuclear Safety Commission in response to the Report of the Auditor General entitled Canadian Nuclear Safety Commission – Power Reactor Regulation, February 2002. http://www.nuclearsafety.gc.ca/eng/pdfs/oag_report_02.pdf

[[8]] 2003 Report of the Canadian Nuclear Safety Commission in response to the Report of the Auditor General entitled Canadian Nuclear Safety Commission – Power Reactor Regulation, February 2003. http://www.nuclearsafety.gc.ca/eng/pdfs/OAG_Report_03.pdf

[[9]] 2005 February Status Report of the Auditor General of Canada, February 15, 2005. http://www.oag-bvg.gc.ca/internet/English/parl_oag_200502_06_e_14926.html

[[10]] Report - International Regulatory Review Service (IRRS) – Report to the Government of Canada, 31 May to 12 June *2009*, IAEA-NS-IRRS-2009/02, October 2009.

The degree of rigour of the operator qualification programme will be defined by the risk posed by the facility and the impact the operators have on the safe operation of the facility;

The degree of CNSC involvement in the operator qualification programme (i.e., the requirement for CNSC examinations or certification) will be defined by the risk posed by the facility and the impact the operators have on the safe operation of the facility; and

The operator qualification programme will align with the main components of the well-defined certification process at NPPs.

Depending on the risk posed by the FCF in the Human Performance Management SCA and on the impact the operators have on the safe operation of the facility, the qualification programme could encompass anything from a simple qualification process administered by the licensee at a lower-risk facility, to a very rigorous, multi-step process that includes CNSC examinations and the requirement for CNSC certification at a higher-risk facility.

As the first stage of the implementation of this new regulatory approach to FCFs, CNSC staff decided to implement the approach at one of the higher-risk FCFs. Following this initial implementation, CNSC staff will evaluate the results of the qualification programme and implement the approach for all FCFs. CNSC staff will then conduct periodic evaluations and will adjust the regulatory approach based on the lessons learned and any changes to the risk rankings of the facilities.

3.1 The application of the regulatory approach at the PHCF

A CNSC staff review of the FCF risk rankings in the area of Human Performance Management identified that one of the highest ranked facilities was Cameco's Port Hope Conversion Facility (PHCF). The PHCF, owned and operated by Cameco Corporation, is a Class IB facility that produces UO_2 and UF_6. The facility is located on Lake Ontario in the town of Port Hope, Ontario. Due to the nature of its activities, its location and the legacy of the PHCF site, the facility is allocated substantial CNSC resources for licensing and compliance activities. Furthermore, the role that operators play in the safe operation of the facility and in the protection of the environment is significant.

In addition, in the three months following an extended shutdown of almost two years, Cameco reported that seven minor events had occurred at the UF_6 plant. A Cameco investigation of these events discovered that operator error was a contributing factor in many of the incidents. Following an analysis by a Cameco consultant, it was reported that the training activities in support of the restart of the facility did not meet the needs of the workers and the job and tasks that they performed. The consultant's report also noted that Cameco had underestimated the skill that was lost by the operators during the extended shutdown.

Consequently, based on the risk ranking of the facility and the recent events involving operator errors, CNSC staff determined that a more rigorous approach to ensure the competence of operators was needed at the PHCF. Following a series of meetings between PHCF and CNSC staff, it was agreed that the PHCF would be subject to an internal qualification programme in line with the CNSC's new regulatory approach.

Commensurate with the risks and hazards posed by the PHCF, and the impact that operators have on the safe operation of the facility, Cameco and CNSC staff determined that this internal qualification programme would be overseen by Cameco and that the facility operators would not require either CNSC examinations or CNSC certification. Since the production of UF_6 constitutes the highest risk activity on site, the operators of the UF_6 plant at the PHCF were chosen for the internal qualification programme.

CNSC staff then collaborated with Cameco to develop an internal qualification process for UF_6 operators at the PHCF site that would align with the key aspects of the CNSC's certification process for NPP workers.

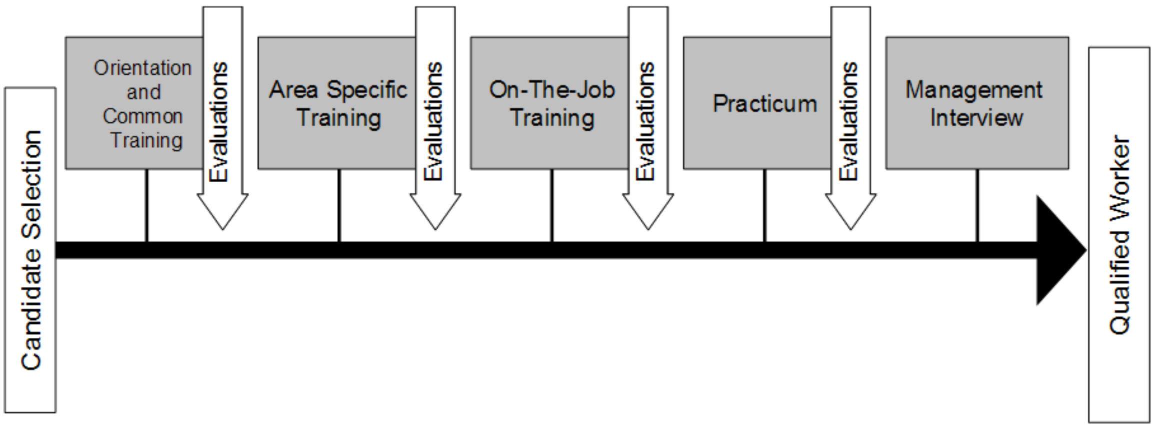
3.2 PHCF internal qualification process

The new internal qualification process implemented at the PHCF contains three main qualification processes for its UF_6 operators: the process for initial qualification, the process for the renewal of qualification and the process to maintain a qualification during an extended plant shutdown.

3.2.1 The process for initial qualification

The initial qualification process is the process that the PHCF follows to provide a new candidate with the knowledge and skills required to be a qualified worker. An outline of this qualification process is provided in figure 1.

Figure 1. **The process for the initial qualification of a worker**



This initial qualification process begins with the PHCF selecting a candidate that meets the entry level education and experience requirements. To be eligible to enter the qualification programme, a candidate requires a High School Diploma and a minimum of three years of relevant chemical process operating experience.

In-class training programmes are then completed to provide the candidate with an overview of the safe operation of the facility and in-depth knowledge focusing on the area of the facility where the candidate will be operating. These in-class sessions include formal written evaluations that each candidate is required to successfully complete prior to commencing the next phase of the training programme.

Following the successful completion of the in-class training, the candidate then applies their knowledge in an on-the-job training programme which covers the operation of the equipment, the performance of routine job tasks and the safety practices required to work safely in the area of the facility where they will be working. The candidate is required to complete on-the-job evaluations to ensure that they have achieved the appropriate level of knowledge and skill to safely perform each task.

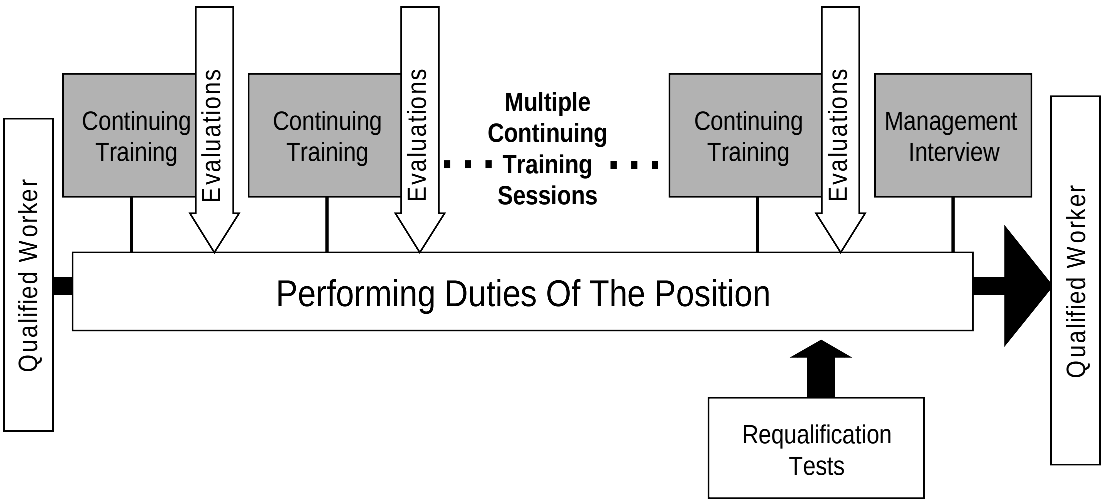
To ensure that all of the training programmes adequately impart the required knowledge and skills to the candidates, these training programmes are required to be based on a systematic approach to training (SAT). A SAT is a methodology that provides a logical progression from the identification of the competencies required to perform a job, to the design, development, implementation, and maintenance of the training programmes to achieve these competencies, and to the subsequent evaluation and continuous improvement of the training programmes. This methodology minimizes the risk that important elements of training are omitted and ensures that successful candidates possess all of the required competencies.

The candidate is then required to put into practice all of the training that they received by completing a requisite number of hours on shift under the supervision of a qualified worker. Following the completion of these supervised hours, or "practicum", the candidate completes a final scenario-based oral evaluation and an area walk-through. PHCF management then reviews the training and evaluations that the candidate has completed and interviews the candidate to confirm that he or she has the knowledge and skills required to perform the duties of a qualified worker. Cameco management then issues each candidate a qualification to perform the duties of their position for a period of five years.

3.2.2 The process for the renewal of qualification

The process for the renewal of qualification ensures that a qualified worker continues to have the required knowledge and skills to perform the duties of their position. A diagram of the process for the renewal of qualification is provided in figure 2.

Figure 2. **The process for the renewal of a worker's qualification**



During the five year validity period of their qualification, each qualified worker is required to complete SAT-based continuing training. This continuing training consists of two components: update training and refresher training. The update training component addresses the changes in the facility, including modifications to the facility and the changes to the plant policies, standards and procedures. The refresher training is the training to maintain the proficiency of each qualified person. This training covers, among other topics, a review of the knowledge acquired during the initial training, infrequently performed tasks or procedures and the response to emergency or upset conditions. Over a five year period, qualified workers complete continuing training sessions which include written and practical evaluations to ensure that each worker continues to have the requisite knowledge and skills to continue to be capable of safely performing the duties of the position.

In addition to successfully completing the continuing training, each qualified worker is required to successfully complete a knowledge-based written requalification test, an oral scenario-based test and an area walk-through during the five year period of their qualification. These requalification tests are conducted to ensure that all qualified personnel maintain the knowledge and skills required to perform their duties.

A failure of a requalification test results in the person being immediately removed from their duties for remedial training. This person cannot be returned to the duties of their position until they have successfully completed another requalification test and until PHCF management is completely satisfied that they possess the requisite knowledge and skills.

Following the successful completion of the requalification tests, PHCF management reviews the training and evaluations that the worker has completed and interviews the worker to confirm that he or she has the knowledge and skills required to continue to perform the duties of a qualified worker. Cameco management then renews the qualification of the worker for a period of five years.

3.2.3 The process to maintain a qualification during an extended plant shutdown.

The process to maintain a qualification during an extended plant shutdown ensures that qualified workers do not lose the required knowledge or skills to safely perform their duties.

During an extended plant shutdown, each qualified worker is required to complete a continuing training programme for each area of the plant in which they hold a qualification. This training programme includes a review of the startup procedures, the normal operating procedures, information concerning any changes to the facility that may have occurred during the shutdown and a procedural walk-through in the facility.

In addition to the completion of the training programme, each qualified worker is required to successfully complete a knowledge-based written requalification test to ensure that they have maintained the knowledge and skills required to safely perform their duties. Following the successful completion of the requalification test, each worker is then permitted to perform their duties.

3.2.4 Comparison of the PHCF internal qualification process to the NPP certification process

The regulatory approach to the process for internal qualification and the renewal of qualification were formulated based on the lessons learned from the certification processes for shift workers at Canadian NPPs. In line with the principles of the regulatory approach listed in section 3.0, since the risk posed by the PHCF and the impact the PHCF operators have on the safe operation of the facility is lower than that for NPPs, the qualification processes for PHCF were designed to be less rigorous and have less CNSC involvement than the certification process for NPPs.

Additional information concerning the processes used for the initial certification and the renewal of certification of workers at Canadian NPPs is provided in the appendix to this paper.

There are two main differences between the initial certification process at NPPs and the qualification process implemented at the PHCF. The first difference is that at NPPs, in addition to successfully completing the formal evaluations during the SAT-based in-class and hands-on training programmes, each candidate is required to successfully complete knowledge-based written certification examinations and a performance-based simulator certification examination which are overseen by CNSC staff. At NPPs, the additional assurance provided by certification examinations is necessary to ensure that each candidate has the knowledge and skills required to safely perform their duties. At the PHCF, due to the lower risk significance of the facility in comparison to that of NPPs, certification examinations are not required in the initial qualification programme.

The second difference is that NPP licensees must submit an application for certification to the CNSC on behalf of the candidate that documents that the candidate has satisfied all of the regulatory requirements and that states the licensee is fully confident that the candidate is capable of performing the duties of the certified position. Based on the licensee's submission and the training and examinations completed, the CNSC may then certify the candidate allowing the candidate to perform the duties of their certified position. At the PHCF, an independent CNSC review of the candidate's training and examination is not required. Instead, since the FCF poses a lower risk than the NPPs, the PHCF management is allowed to perform this review and issue a candidate's qualification.

The process at the PHCF for the renewal of qualification is very similar to the process at NPPs for the renewal of certification. The difference between these processes is that at NPPs no management interviews are conducted prior to the renewal of certification. As described in section 3.2.2, at the PHCF, each worker is required to successfully complete a management interview prior to having their qualification renewed. At NPPs, instead of completing a management interview, following the successful completion of the requalification tests, the licensee applies to the CNSC to renew the certification of the certified person. Based on the licensee's submission and the training and tests completed, the CNSC may then renew the certification of the worker and the worker can continue to perform their duties. Since this independent CNSC review of the worker's training and testing is not performed for workers at the PHCF, the requirement to conduct a management interview was introduced in order to ensure that PHCF management provides sufficient assurance that all workers possesses the required knowledge and skills.

4. The future implementation of the regulatory approach to all fuel cycle facilities

As discussed in sections 3.0 and 3.1, the first stage of the implementation of the regulatory approach to confirm the competence of FCF operators was to implement a formal qualification programme for the UF_6 operators at the PHCF. Once this qualification programme has been in place for at least two years, CNSC staff will evaluate whether the programme provides sufficient assurance that the UF_6 operators have and maintain the knowledge and skills required to perform their duties. Based on this evaluation, CNSC staff may refine and update the requirements for the PHCF qualification programme.

The other FCFs in Canada currently have risk rankings which are lower than that of the PHCF. Recent regulatory inspections conducted by CNSC staff from the Training Program Evaluation Division have focused on evaluating the qualification programmes for workers at these other FCFs. At this time, the qualification of operators at these facilities is managed by the licensee and is generally less onerous then the new qualification programme implemented at the PHCF. Based on the lessons learned from the implementation of the qualification programme at PHCF, CNSC staff will review the risk rankings of other FCFs and determine the level of rigour and the degree of CNSC involvement required for the implementation of similar qualification programmes at these facilities.

Due to the dynamic nature of the Canadian nuclear industry, CNSC staff periodically revisit their regulatory approaches. As a result, CNSC staff will conduct periodic evaluations of the qualification programmes at FCFs and will update the regulatory approach based on the lessons learned. In addition, due to changes in the industry or to changes in the operation of a facility, the risk rankings of FCFs are periodically reviewed by CNSC staff. Should a risk ranking for a facility change, CNSC staff will revisit its regulatory approach for the confirmation of the competence of operators.

5. Conclusion

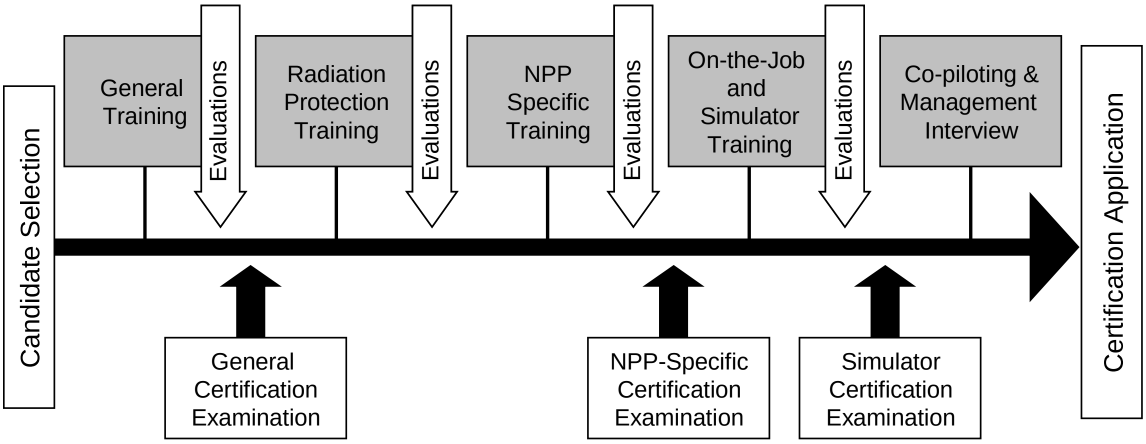
Over the past ten years, the CNSC has made significant progress in the allocation of its resources to regulatory activities based on a risk-informed approach. Using this experience with a risk-informed approach, CNSC staff developed a new regulatory approach to confirm the competence of operators at FCFs. CNSC staff is confident that the implementation of this approach will provide additional assurance that the operators at FCF have and maintain the required knowledge and skills to safely perform their duties.

APPENDIX – THE CERTIFICATION PROCESSES AT CANADIAN NUCLEAR POWER PLANTS

1. The process for initial certification

A typical initial certification process for a reactor operator at a Canadian Nuclear Power Plant (NPP) is shown in figure A.1.

Figure A.1. **A typical initial certification process for a reactor operator at a NPP**



The initial certification process at NPPs begins with a licensee selecting a candidate that meets the entry level education and experience requirements. In-class training programmes are then completed to provide the candidate with in-depth knowledge in the areas of science fundamentals for nuclear reactors, radiation protection, and the design, operation and interaction of the plant systems. Following the successful completion of the in-class training, the candidate then applies their knowledge in a hands-on simulator-based training programme and an on-the-job training programme which cover the operation and monitoring of the plant systems under normal, abnormal and emergency conditions. Each of these SAT-based training programmes includes formal evaluations that are conducted to ensure that each candidate is appropriately acquiring the requisite knowledge and skills.

The candidate is then required to put into practice all of the training that they received by performing a number of hours on shift under the supervision of a certified worker. Following the completion of these hours on shift, also known as a 'co-piloting period', NPP management reviews the training and examinations that the candidate has completed and interviews the candidate to confirm that he or she has the knowledge and skills required to perform the duties of a certified position.

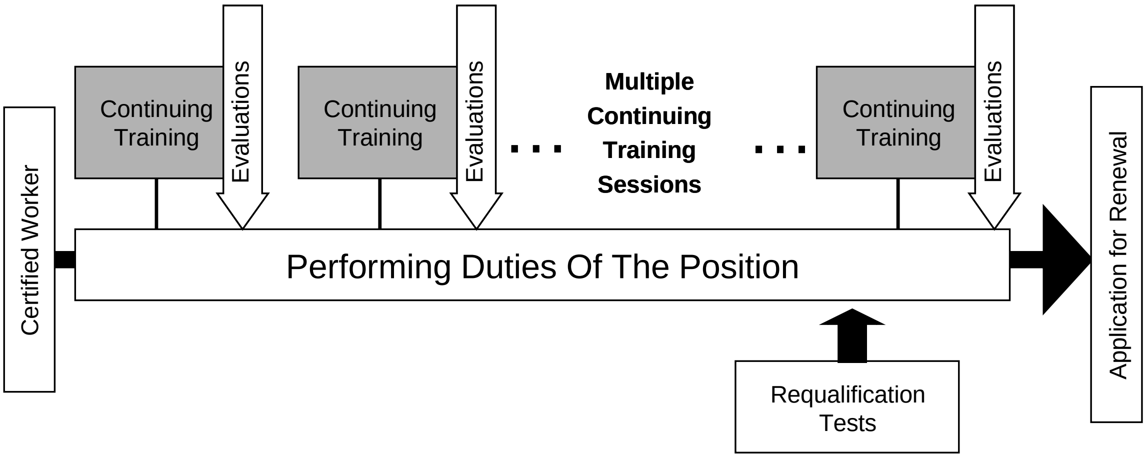
The NPP licensee then submits an application for certification to the CNSC on behalf of the candidate that documents that the candidate has satisfied all of the regulatory requirements and that states the licensee is fully confident that the candidate is capable of performing the duties of the certified position. If the submission meets the applicable regulatory requirements, the CNSC issues a certification to the candidate that is valid for a period of five years.

Following the selection of the candidate, it takes approximately 4 years for a candidate to complete all of the training programmes and examinations required in the initial certification process.

2. The process for the renewal of certification

The process for the renewal of certification for shift workers at a Canadian NPP is provided in figure A.2.

Figure A.2. **The process for the renewal of certification for shift workers at a NPP**



During the five year validity period of their certification, each certified person is required to complete SAT-based continuing training on a regular basis. This continuing training consists of two components: update training and refresher training. The update training component addresses the changes in the plant, including changes to the plant systems and the changes to the plant policies, standards and procedures. The refresher training is the training to maintain the proficiency of each certified person. This training covers, among other topics, a review of the knowledge acquired during the initial training and simulator-based exercises that challenge the person's diagnostic and decision-making abilities. Over a five year period, certified workers complete continuing training sessions which include formal evaluations that are conducted to ensure that each worker has retained the necessary knowledge and skills.

In addition to successfully completing the continuing training, each certified worker is required to successfully complete a knowledge-based written requalification test and performance-based simulator requalification tests during the 5 year period of their certification. These knowledge-based and simulator-based tests are conducted to ensure that all certified personnel maintain the knowledge and skills required to perform their duties.

A failure of a requalification test results in the person being immediately removed from their shift duties to be retrained. This person cannot be returned to shift duties until they have successfully completed another requalification test and until the plant management is completely satisfied that they possess the requisite knowledge and skills.

Following the successful completion of the requalification tests, the licensee then applies to the CNSC to renew the certification of the certified person. If the submission meets the applicable regulatory requirements, the CNSC renews the certification of the person for a period of five years.

Canadian Nuclear Safety Commission
Commission canadienne de sûreté nucléaire

Confirming Competence of Operators - A Regulatory Approach to Fuel Cycle Facilities

Presentation for NEA/CSNI Workshop on
Safety Assessment of Fuel Cycle Facilities

September 27-29, 2011

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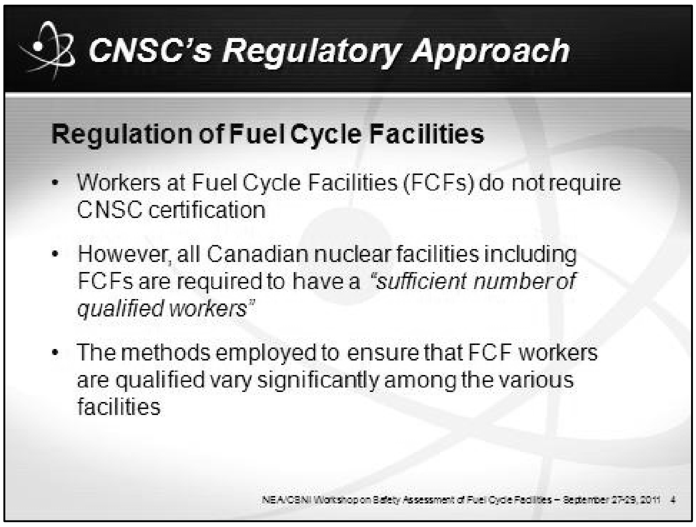
Presentation Outline

- Introduction
- CNSC's Regulatory Approach
  - Regulation at Fuel Cycle Facilities (FCFs)
  - Risk Informed Regulatory Approach
  - Confirming Competence of Operators at FCFs
  - Application of Regulatory Approach at the Port Hope Conversion Facility
- Port Hope Conversion Facility (PHCF) Qualification Processes
  - Initial Personnel Qualification
  - Renewal of Personnel Qualification
  - Maintenance of Personnel Qualification During Extended Plant Shutdowns
- Process Comparison: PHCF and Canadian Nuclear Power Plants
- Future Steps
- Conclusion

Introduction

The Canadian Nuclear Safety Commission (CNSC)

- Canada's Independent Nuclear Regulator
- Regulates in accordance with a well-defined legal framework:
  - Nuclear Safety and Control Act;
  - Regulations;
  - Licences; and
  - Regulatory Documents
- In accordance with the legal framework, the CNSC certifies:
  - Specific workers at Nuclear Power Plants (NPPs);
  - Specific workers at Non-Power Research Reactor Facilities; and
  - Exposure Device Operators


CNSC's Regulatory Approach
Regulation of Fuel Cycle Facilities
• Workers at Fuel Cycle Facilities (FCFs) do not require CNSC certification
• However, all Canadian nuclear facilities including FCFs are required to have a *"sufficient number of qualified workers"*
• The methods employed to ensure that FCF workers are qualified vary significantly among the various facilities
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CNSC's Regulatory Approach
Risk Informed Regulatory Approach
• Stemming from two external audits, the CNSC has been implementing a risk-informed approach to all regulatory activities
• These regulatory improvements include a systematic, risk-informed approach to the regulation of FCFs
• In line with these improvements, CNSC staff also initiated the development and implementation of a new regulatory approach to confirm the competence of operators at FCFs
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CNSC's Regulatory Approach
Confirming Competence of Operators at FCF
To provide additional assurance that operators at FCFs are competent, operator qualification programs will be required.
These qualification programs will be based on the following principles:
1. The **rigour** of the program will be defined by the risk posed by the facility and the impact operators have on the facility's safe operation;
2. The **degree of CNSC involvement** in the program will be defined by the risk posed by the facility and the impact operators have on the facility's safe operation; and
3. The program will be **aligned** with the certification process at NPPs.
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CNSC's Regulatory Approach

Application of Regulatory Approach at PHCF

- Fuel Cycle Facilities in Canada were ranked according to risk using the CNSC Safety and Control Areas
- Based on the risk rankings, the Port Hope Conversion Facility (PHCF) is one of the higher-risk facilities due to:
  - The nature of the activities at the PHCF;
  - The location of the facility; and
  - The legacy of the PHCF site
- Operators at the PHCF play a significant role in the safe operation of the facility and in the protection of the environment
- In addition, there were a series of minor events following a restart of the UF_6 plant at the PHCF

Based on these factors, CNSC staff determined that an Internal Qualification Program would be required for the operators at the UF_6 plant at the PHCF.

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Port Hope Conversion Facility Qualification Processes

Initial Personnel Qualification

Port Hope Conversion Facility Management:

- Selects UF_6 plant operator candidates who meet entry level education and experience requirements
- Trains candidates in accordance with a training program based on a Systematic Approach to Training (SAT)
- Confirms that candidates perform duties under supervision (practicum)
- Conducts management interviews
- Issues a qualification to the person that is valid for a period of 5 years

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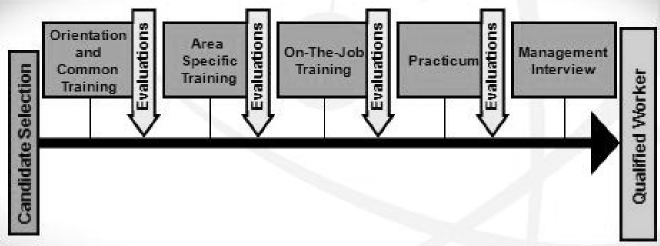
Port Hope Conversion Facility Qualification Processes

Initial Personnel Qualification



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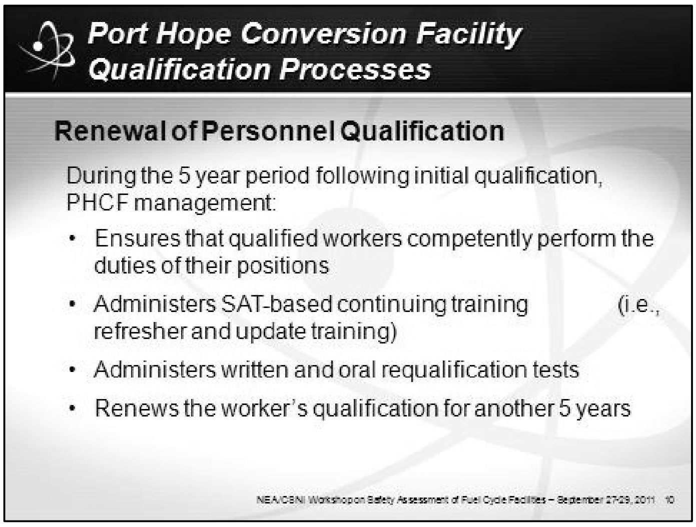
Port Hope Conversion Facility
Qualification Processes
Renewal of Personnel Qualification
During the 5 year period following initial qualification, PHCF management:
• Ensures that qualified workers competently perform the duties of their positions
• Administers SAT-based continuing training (i.e., refresher and update training)
• Administers written and oral requalification tests
• Renews the worker's qualification for another 5 years
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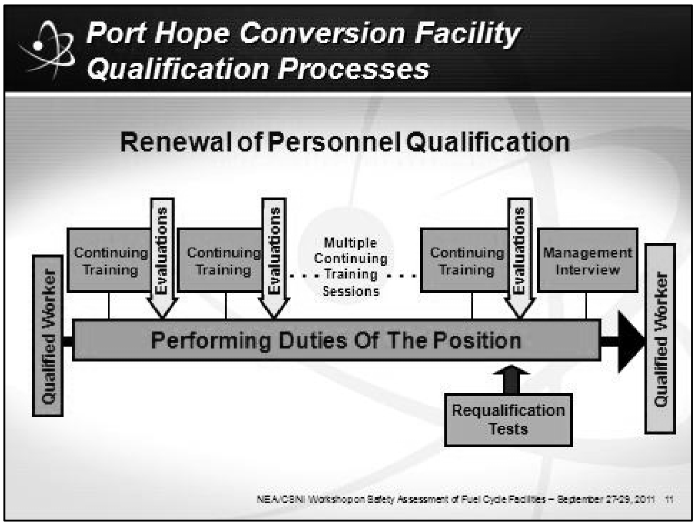
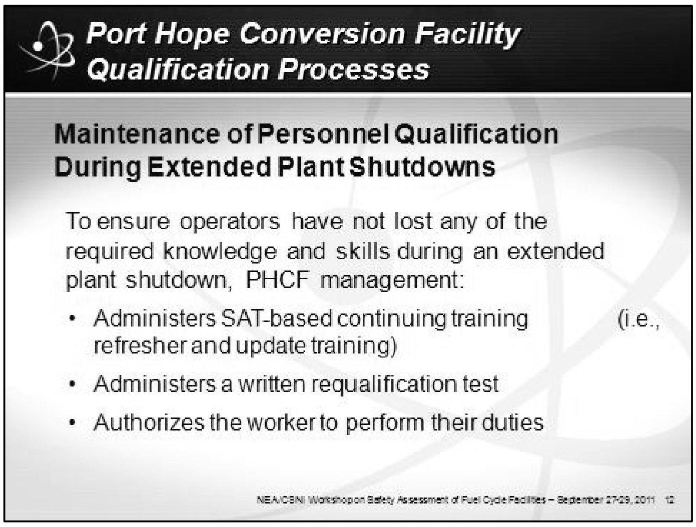

Port Hope Conversion Facility
Qualification Processes
Renewal of Personnel Qualification
Qualified Worker
Continuing Training
Evaluations
Continuing Training
Evaluations
Multiple Continuing Training Sessions
Continuing Training
Evaluations
Management Interview
Performing Duties Of The Position
Requalification Tests
Qualified Worker
NEA/CSNI Workshop on Safety Assessment of Fuel Cycle Facilities – September 27-29, 2011 11


Port Hope Conversion Facility
Qualification Processes
Maintenance of Personnel Qualification During Extended Plant Shutdowns
To ensure operators have not lost any of the required knowledge and skills during an extended plant shutdown, PHCF management:
• Administers SAT-based continuing training (i.e., refresher and update training)
• Administers a written requalification test
• Authorizes the worker to perform their duties
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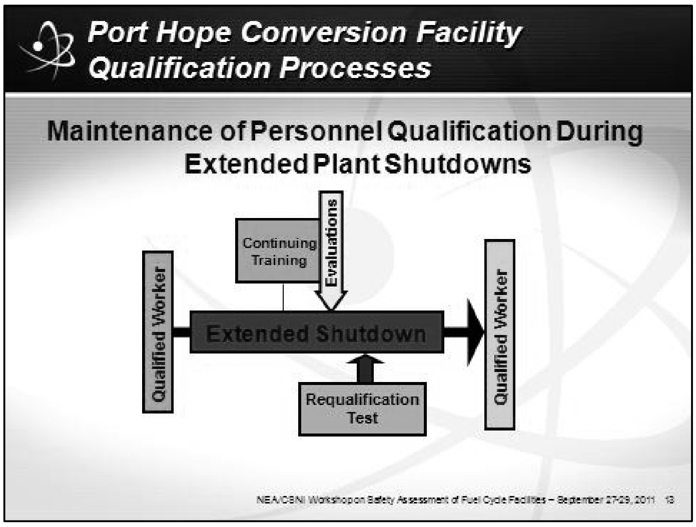
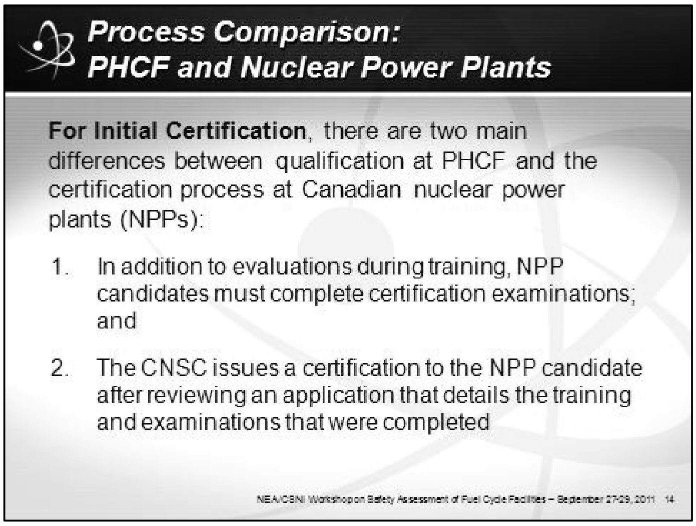

Port Hope Conversion Facility
Qualification Processes
Maintenance of Personnel Qualification During Extended Plant Shutdowns
Continuing Training
Evaluations
Qualified Worker
Extended Shutdown
Qualified Worker
Requalification Test
NEA/CSNI Workshop on Safety Assessment of Fuel Cycle Facilities – September 27-29, 2011 13


Process Comparison:
PHCF and Nuclear Power Plants
For Initial Certification, there are two main differences between qualification at PHCF and the certification process at Canadian nuclear power plants (NPPs):
1. In addition to evaluations during training, NPP candidates must complete certification examinations; and
2. The CNSC issues a certification to the NPP candidate after reviewing an application that details the training and examinations that were completed
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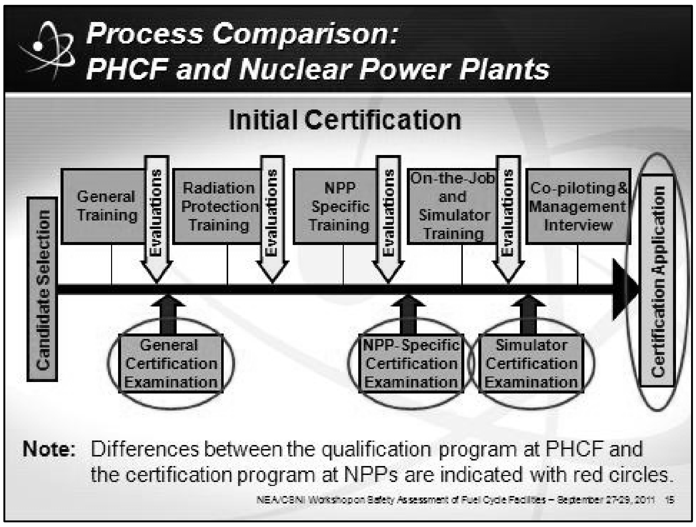
Process Comparison:
PHCF and Nuclear Power Plants
Initial Certification
Candidate Selection
General Training
Evaluations
Radiation Protection Training
Evaluations
NPP Specific Training
Evaluations
On-the-Job and Simulator Training
Evaluations
Co-piloting & Management Interview
Certification Application
General Certification Examination
NPP-Specific Certification Examination
Simulator Certification Examination
Note: Differences between the qualification program at PHCF and the certification program at NPPs are indicated with red circles.
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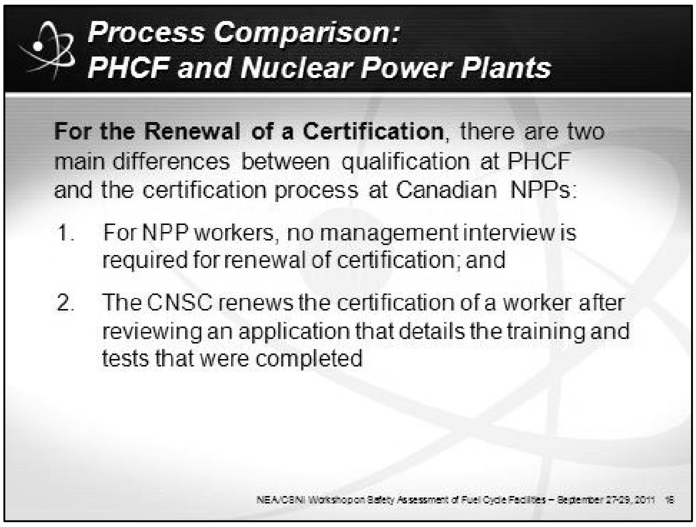
Process Comparison:
PHCF and Nuclear Power Plants
For the Renewal of a Certification, there are two main differences between qualification at PHCF and the certification process at Canadian NPPs:
1. For NPP workers, no management interview is required for renewal of certification; and
2. The CNSC renews the certification of a worker after reviewing an application that details the training and tests that were completed
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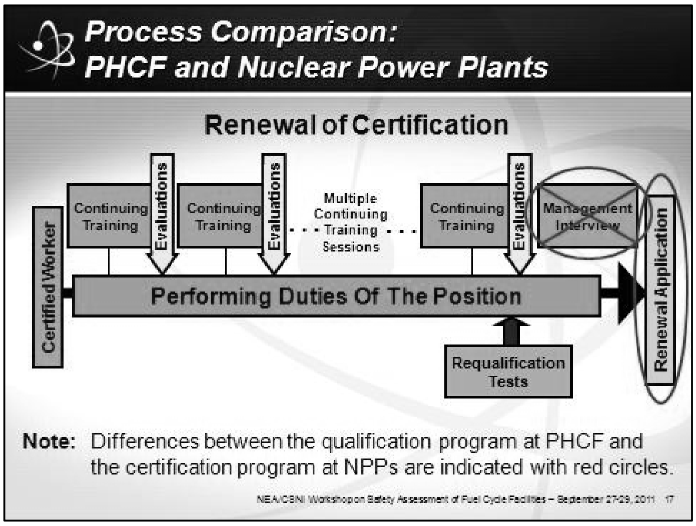
Process Comparison:
PHCF and Nuclear Power Plants
Renewal of Certification
Certified Worker
Continuing Training
Evaluations
Continuing Training
Evaluations
Multiple Continuing Training Sessions
Continuing Training
Evaluations
Management Interview
Performing Duties Of The Position
Requalification Tests
Renewal Application
Note: Differences between the qualification program at PHCF and the certification program at NPPs are indicated with red circles.
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Conclusion
CNSC staff:
• Developed a new regulatory approach to confirm the competence of operators at FCFs
• Implemented this new approach at the PHCF
• Will implement the approach at all FCFs in the future
Ultimately, this new approach will provide additional assurance that the operators at FCFs have and maintain the knowledge and skills required to safely perform their duties.
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BASIC NUCLEAR INSTALLATIONS PERIODIC SAFETY REVIEWS IN FRANCE

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Abstract – The French regulation, especially the law Nr 2006-686 of 13 June 2006 concerning transparency and nuclear safety, called the "TSN law", requires that basic nuclear installations (BNI) are submitted to a periodic safety review (PSR) every ten years. The same regulation requires an integrated approach for PSR, having the licensees to present all the risks, radiological and/or chemical, inherent in their installations, and to take into account human and organisational factors. PSR are also the opportunity to reassess bounding accidents. These accidents are assessed by a deterministic approach that can be completed by two methods: the "operating conditions" method, that is still a deterministic method introducing some probabilistic matters like failure frequencies; and the probabilistic safety analysis (PSA) which is only used in France for nuclear power plants. The main result of a PSR is to determine whether a facility can go on operating for 10 years more or not, and to determine the works and compensatory measures that must be done by the operator to reach the goals required by an update regulation.

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[11] ASN: Autorité de sûreté nucléaire = French nuclear safety authority
DRC: ASN directorate in charge of nuclear waste, research facilities (including research reactors) and fuel cycle facilities

[12] DCN: ASN directorate in charge of nuclear power plants

Introduction

The first French regulation dealing with basic nuclear installations (BNI)[13] was a decree from 1963, that settled down procedures to create or modify a BNI, and that was called by the 1961 law concerning olfactory nuisances and air pollution. This decree did not settle down anything concerning reassessment or periodic safety review (PSR). However the French nuclear safety authority tried to put in place in the 1980s periodic reassessment for BNI, because some of them were already ageing – the most ancient BNI in France were created in the late 1940s or the early 1950s. This practice has been introduced into the French regulation by being settled down in the law concerning transparency and nuclear safety (the TSN law) promulgated on the 13 June 2006 [1] and which one of its main application texts is the decree of 2 November 2007 known as the "Procedures decree" [2].

1. French regulation concerning PSR – the article 29-III of the TSN law and the article 24 of the Procedures decree

The TSN law [1] requires an integrated approach for creation authorisation, dismantling authorisation and periodic safety review (PSR). This leads the licensees to present all the risks, radiological and/or chemical, inherent in their installations, and to take into account human and organisational factors (HOF).

The TSN law [1] also requires that a facility performs a PSR every 10 years. This PSR should allow to assess the actual condition of the BNI, and its capability to carry on operating, according to the regulation in force at the time of the PSR, and on the basis of an update of the risks that the BNI presents for the interests listed in the article 28 of the TSN law (public security, public health, public healthiness, nature protection and environment protection) [1].

In application of the Procedures decree [2], the draft of the ASN regulatory decision concerning PSR indicates more precisely the documents that the licensee has to provide for a PSR:

The facility description when doing the PSR;

- Its compliance with the requirements and standards accepted for its creation, the analysis of the discrepancies ;
- Compliance of the BNI actual condition with requirements and standards accepted for its creation, and the analysis of the discrepancies ;
- Suggestions of compensatory measures or modifications, or justifications of if it is not possible to achieve these regulation and standards, regarding the life-time the facility is thought to continue operation, and regarding economic factors ;
- Possible evolutions of the plant foreseen for the 10 years to come (until the next PSR) ;
- Feedback from operating and of running of similar plants in the world ;
- Updating of the risks and hazards presented by the plant for the interests listed in the article 28 of the TSN law ;
- Updating of the BNI safety documents: safety report, general operating rules, internal emergency plan, waste management, dismantling provisional plan.

[13] Basic nuclear installation = facility containing amounts of radioactive substances or of fissile materials greater than the thresholds defined in the decree Nr 2007-830 of the 11 May 2007

As the TSN law [1] requires an integrated approach for each step in the life of the BNI, the description, the updating of the risks, the feedback and the suggestion of compensatory measures or modifications must include technical matters such as nuclear safety, radioprotection and/or chemical safety, but also HOF. This last requirement is completely new compared to the previous regulation, although the ASN and the IRSN (the technical support of the ASN) have systematically taken this point into account in their assessments since the late 1990s.

2. Assessing the risks

In France the risks assessment is first made with a deterministic approach. But to have more realistic results, French licensees can use two iterative methods that can indicate some weak points in civil engineering or in control systems:

The “operating conditions” method, that is the oldest complementary method used by operators ;

A probabilistic safety assessment (PSA), which is at the present time only used by the nuclear power plants (NPP) and by the EDF (Electricity of France) licensee.

Historically French operators of non-NPP facilities (including FCF) have used a deterministic envelop-type approach that determines only one “envelop” accident for both design and emergency management. This envelop accident was supposed to generate the greatest consequences among all the incidental and accidental situations for the concerned facility. But recent events or assessments have shown that the consequences of some incidental scenarios, defined as minor compared to the envelop accidents, were underestimated if these incidental situations could be combined, and that it could be useful to review in a systematic way the incidental and accidental scenarios to determine the most penalizing ones. In this point of view, the CEA (French Atomic Energy Council) has taken to using the “operating conditions” method since the middle of the 2000s.

2.1. Using the “operating conditions” method

This methodology is issued form the NPP safety assessments.

Previously, non-NPP nuclear installations, and in particular installations operated by CEA used the “three barriers” method to make their safety analysis. It was a deterministic approach. This method consisted in identifying three barriers, which should be materialized, leak proof, independent from each other and should separate the dangerous material from the public and the environment. But this method was limited since the barriers could not be systematically proved independent from each other. In addition the physical boundaries of the barriers could be difficult to set. That is why the CEA issued a “recommendation paper”, which sets the “operating conditions” method as a standard. This method consists in designing the installation and the components in regard of the incidental and accidental situations that need to be considered.

For this method, safety functions have to be defined. For the CEA, these functions concern generally:

- containment;
- neutron reaction;
- cooling;
- radiation protection;
- management of explosive gases produced by radiolysis.

The components operating directly for a safety function, the components for which a malfunction could result in the failure of a safety function or the components considered in the safety analysis are identified as safety important components (SIC).

The "operating conditions" method, which is a deterministic method, consists in defining an initial state of the installation, adding a single internal initiating event. These operating conditions are postulated.

They will be:

- listed;
- merged into series of initiating events in order to determine bounding scenarios to analyse;
- classified in four categories according to their annual rate of occurrence (ARO) and the accumulation of the accidental situations;
- examined in regard of the acceptability of their consequences.

Regarding the four categories:

- the first one corresponds to the normal operation;
- the second (10^{-2} < ARO), the third (10^{-4} < ARO < 10^{-2}) and fourth (10^{-6} < ARO < 10^{-4}) categories contain less and less probable situations;
- conditions with an ARO < 10^{-6} constitute the beyond design basis accidents. Nevertheless, the conditions with 10^{-7} < ARO < 10^{-6} are analysed, but the ones with an ARO < 10^{-7} are excluded.

The operating conditions could be classified according to:

- frequency, if it is defined by a feedback (that is less often determined for nuclear installations than for nuclear power plants);
- the number and the robustness of the lines of defence. These lines are defined as "strong" or "weak".

To examine the acceptability of the consequences of the accidental situations, general safety objectives have to be defined. If the analysis shows that these objectives are exceeded, lines of defence should be added. It is an iterative process.

At the final step of the operating conditions method, the SIC are defined. These SIC are organised according their importance. Requirements are defined to design, build and operate these SIC.

The CEA considers that the operating conditions method has to be applied to all the PSR and to new facility projects. It was applied to the PSR of the ORPHEE reactor[14], to the design of the RJH reactor[15], of the ITER installation[16] and of the MAGENTA plant[17].

[14] Research reactor located in the Saclay site, and producing neutrons for fundamental researches
[15] Research reactor located in the Cadarache site, still under construction, and planned to irradiate materials, for example to produce medical radionuclides
[16] Prototype of a fusion reactor, located next to the Cadarache site, and still under construction
[17] Fissile materials storage, located in the Cadarache site

The external hazards are studied, according to their probability, in order to maintain the safety functions if these hazards occur and in order to design the installation.

There is at the moment not enough feedback to evaluate clearly the limits of the “operating conditions” method for laboratories and research reactors. This method could be applied to all nuclear facilities in the future. It will be considered through the application of the “BNI order”, as it will require several methods to evaluate the risks and hazards presented by BNI, including probabilistic ones. Such requirements will be close to the ones already in force for installations classified for environment protection (ICPE – chemical industries), that have to use a “probabilistic” method similar to the “operating conditions” method.

2.2. Using PSA for French PWR (EDF operator)

For French PWRs, the deterministic approach is completed by **PSA**. This approach is described in the Fundamental Safety Rule (RFS) Nr 2002-01 [3] that was published on 26 December 2002.

PSA objectives:

PSA helps to assess whether the arrangements made by the plant operator are satisfactory. It can be used to prioritise the safety problems relating to the design or operation of reactors, and is a tool for dialogue between the plant operators and the authorities.

For operating reactors, PSA contributes to assessment of their overall safety level and highlights points for which design or operating changes can be examined or even judged necessary.

The main objectives of probabilistic analysis of events are the prioritisation of events according to the conditions probability of core damage and the assessment to the pertinence of the corrective actions.

In addition, identification of the main contributions to the core damage frequency highlights any weak points for which design and operation changes can be studied, or even judged necessary. They can be ordered so as to target the priority work.

The PSA helps to put in light uncertainties and limits that would be hidden by deterministic hypotheses.

Because of the systematic investigation of accidental scenarios, PSA can point out some scenarios that haven’t been thought of before, because they do not necessary involve equipments or actions classified as “safety important”, but that can have very serious consequences at last.

PSA definition:

PSA provide a risk assessment method based on systematic investigation of accident scenarios. They provide an overall view of safety, including both equipments and operator behaviour.

PSA considers a list of initiating events which is realistic and complete as possible. In practice, the events studied can include initiating events originating inside the installation (equipment or human failures, internal fire or flooding, etc.) or originating outside (earthquake, external fire or flooding, storm, etc.) associated with the different reactor states.

It highlights operating situations covering complex events and combinations of events, including situations involving the loss of redundant systems and depending on the scope (the nature of the consequences examined and by the events studied) those involving the occurrence of an internal or external hazards.

For each initiating event, PSA establishes accidental sequences resulting from the success or failure of the operation systems and actions involved to perform the safety functions, and assesses the frequency of an undesired event which depends on the type of PSA (below). By summing all the calculated frequency values, it estimates the total frequency of the undesired event, the contribution of each initiating event to the calculated frequency, and the importance for safety of equipment and operating actions.

Three types of PSA can be produced, depending on the consequences studies:

- a level 1 PSA identifies the sequences leading to core damage and determines their frequencies;
- a level 2 PSA assesses the nature, magnitude and frequencies of releases outside the containment;
- A level 3 PSA assesses the calculated frequencies of consequences expressed in dosimetry or contamination terms (or in terms of frequencies of cancers or other effects on health).

For French NPP, the level 3 PSA have not yet been developed to date.

Method:

During the first step of the PSR, the reference PSA is updated, incorporating the most recent operating experience (identification and frequency of initiating events, equipment reliability data, operating profile), the standard construction condition (design and operation) and new knowledge about the behaviour of the installation obtained from the most recent studies.

An acceptable method for highlighting and prioritising the principal contributions to the core damage frequency consists in grouping elementary sequences with similar functional characteristics into “functional sequences”, then assessing the hazard associated with the latter. The priority of the grouping method is to constitute “functional sequences” whose frequency and consequences could be reduced by implementing a given provision in order to optimise the identification of opportunities for improvement.

The scope of the reference PSA and the grouping into functional sequences are likely to change at each periodic safety review. In this context, assumptions, criteria, and date have to be justified. Reliability data have to be update and extended, considering in particular feedback of the similar plants in France and abroad; in this frame, particular attention has to be devoted to common mode failures as well as to instrumentation and control systems (hardware and software).

When the conditional probability of core damage associated with an event is greater than a defined reference value, the event is called a “precursor event” and is subject to a thorough analysis.

For the most important precursor events, the plant operator defines specific processing and lead times for the implementation of corrective measures. If possible the expected improvement is assessed.

The results obtained are not used on their own: they are only one of the elements contributing to the decision to implement a corrective measure.

In the safety analysis report compiled for each PSR, the plant operator includes a summary of the reference PSA consistent with the reference and operating conditions of the reactors. This summary includes the main study assumptions and the predominant contributions to the calculated core damage frequency.

The proposed changes are then evaluated and ranked according to their cost and benefit both in terms of probability of consequences. This decision support to achieve the safety objectives set.

Following the PSR, a new version of the reference PSA is produced, taking into account the changes decided on completion of the review process.

PSA contribution to the decision-making process:

PSA are a decision-making aid for assessing the importance for safety of systems and equipment.

Depending on the type of use, thresholds can be defined to identify:

- Systems playing an important role with regard to safety according to their contribution to the frequency of core damage.
- The critical failure modes of equipment.

Moreover, in the technical specifications, very long equipment unavailability times should be avoided if the equipment can be repaired in much shorted times.

The analysis must take into account the frequency of the sequences, the possible consequences on containment integrity and the uncertainties.

After the review of any conservative assumptions of the PSA, this analyses results either in a status quo or in an indication of the usefulness of implementing design or operation changes. In the case where changes are made, PSA can be used to assess the advantages and drawbacks of the various solutions considered. The satisfactory character of such changes must be demonstrated by an analysis of their impact on the contributions to the core damage frequency and on the overall core damage frequency.

PSA limits:

Despite systematic determination of accident scenarios, PSA have identified limits in terms:

- Incompleteness for the scope (some aggressions, some human interventions processed are not taken into account).
- Uncertainties related to the PSA data and assumptions (especially the estimation of human actions, the estimations of the reliability of equipment operating beyond its qualification conditions).

The uncertainties concerning reliability data, common cause failures and human reliability have to be dealt. As the PSR, the PSA are extended to new aggressions (internal explosion, earthquake, external flooding…).

2.3. Bounding accidents

Bounding accidents are a conclusion of the safety analysis and one of the bases of the authorisation decree. So it is important to identify precisely these bounding accidents.

Moreover they are reassessed at every PSR of the concerned facility, in order to take into account new standards or regulations (example: new technical Fundamental Safety Rule concerning earthquakes to take into account for BNI assessment, edited in 2001).

They depend on parameters like:

- Age of the plant ;
- Ageing management ;
- Modifications the plant was subjected to ;
- Actual condition of its containment. As a global feedback, the containment of the facility confinement is related to the age of the plant and the ageing management made by the licensee ;
- Radioactive and chemical inventories. In fact the distinctive feature of FCF is that some of them use chemical products that can be very harmful for health or environment ;
- Form (liquid, solid, gas) of the radioactive and chemical materials, and constraints generated by their storage or use ;
- Number of activities operated on the site, and eventual interactions between them.

Examples of accidents that may be postulated as possibly bounding accidents:

- UF_6 storage or use : accidental release of HF. Example : Eurodif BNI (uranium enrichment);
- Storages or use of high-enriched uranium and/or plutonium powder or nuclear fuel (UOx, MOX): criticality accident. Example : the Melox facility;
- Storages of highly radioactive waste or plants treating highly radioactive liquid waste : loss of containment and/or of cooling. Example : storage pool of spent nuclear fuel or of fission products tanks;
- Amounts of flammable or explosive materials (associated with radioactive materials or not) : fire or explosion. Example: the Superphenix reactor (fast breeder reactor cooled by liquid sodium, under decommissioning).

Most nuclear sites host several facilities, operating different activities linked to each other, at least because utilities are often common. So it is necessary to examine the bounding accident for each facility, as well as the most serious in terms of consequences, to determine perimeters for public protection or evacuation in case of serious events.

Example: the FBFC site in Romans-sur-Isère:

- There are 2 BNI on this site: the BNI Nr 63 that manufactures uranium fuels for research reactors; and the BNI Nr 98 that manufactures uranium fuels for French NPP.
- The bounding accident for BNI Nr 63 is a criticality reaction, because of the storage and the use of highly-enriched uranium. The bounding accident for BNI Nr 98 is an accidental release of HF, because of the storage and the use of great amounts of UF_6.

The most penalizing bounding accident, because generating the widest perimeter for public protection or evacuation, is the one concerning BNI Nr 98.

But, because the two bounding accidents generate different consequences requiring different types of protection measures for the public, the bounding accidents are both kept in the operator safety documents and in public-protection documents made by the local authorities.

The feedback of the Fukushima accident is hoped to cause reassessment of the bounding accidents, by taking into account new scenarios like events that are not bounding accidents for the concerned activities, but which consequences on the other activities can result in the bounding accident for the site.

3. PSR – practical point of view

The PSR allows having a global view on the facility, and making it easier to take decisions for both the ASN and the licensee. However this approach is quite long and heavy to develop and to analyse : the feedback for non-NPP facilities is about 1 year for the operator development, and about 1 year also for the IRSN and the ASN analysis.

For French PWR which are standardized into 4 technological types (900 MW, 1300 MW, 1450 MW, EPR), a PSR is assessed as follow:

- A general PSR concerning one technological type;
- A specific PSR concerning a reactor, taking into account the results of the technological type PSR and the actual condition of the reactor. But this specific PSR mostly deals with technical matters ;
- A ten-yearly specific review of a reactor, where components crucial for safety are submitted to special inspections and tests. For example, the vessel and the primary coolant systems are submitted to visual examination and under-pressure tests.

PSA results are examined and criticized by the operator (EDF), IRSN and ASN during this PSR.

As PSA has to take into account the actual condition of the plant, and as facilities have to perform a PSR every 10 years, there is no specific use of PSA for justifying extension beyond design life for NPP.

The French PWR licensee (EDF) can be submitted to specific PSR on one particular subject. For example, it is submitted every 3 years to an operational feedback PSR, and it was submitted in 2010 to a PSR concerning specifically HOF. Results of these specific PSR can be taken into account while assessing PSR for a specific reactor.

PSR **for French non-PWR facilities** look more as integrated assessments, as they combine almost all subjects listed in the chapters above, except that safety is only examined through deterministic considerations. If the licensee is the CEA, the “operating conditions” method can be used.

For these facilities, there is no design life acted in their creation authorisations. Consequently every ten-yearly PSR concerning one plant has to examine whether the plant can go on operating or not.

4. Conclusion

PSR in France have looked like integrated approaches for more than 10 years, even though such approaches are of regulatory matters since the publication of the TSN law that explicitly requires taking into account all the risks inherent to a facility, including HOF. Assessing a PSR requires among other things, updating accidental scenarios. This updating is based on deterministic approaches completed by an iterative method: PSA for PWR, "operating conditions" method for some other facilities. The main result of a PSR is to determine whether a plant can go on operating for 10 years more or not, and to determine the works and compensatory measures that must be done by the operator to reach the goals required by an update regulation, or to improve the safety of his installation.

The French regulation body concerning PSR has to be completed by the approval of 3 texts:

- The order concerning general regulation for BNI (known as the "BNI order") ;
- The ASN decision concerning PSR ;
- The ASN guide concerning the use of PSA for French nuclear power plants.

References

[1] Law Nr 2006-686 of 13 June 2006 (the TSN law);

[2] Decree Nr 2007-1557 of 2 November 2007 (the Procedures decree);

[3] Fundamental Safety Rule Nr 2002-01 concerning the use of PSA for French PWR;

[4] Technical guidelines for the design and construction of the next generation of nuclear power plants with pressurized water reactors.

Bibliography

[a] Draft of the order concerning general regulation for BNI (known as the "BNI order");

[b] Draft of the ASN regulatory decision concerning PSR.

Nuclear installations periodic safety reviews in France

ASN / DRC
D. Conte, L. Tabard

WGFCS – 27-29 September 2011 – Nuclear installations periodic safety reviews in France 1

Brief history

The decree Nr 63-1228 of the 11 December 1963 :

❑ 1st text regulating nuclear safety

❑ nothing concerning reassessment or periodic safety review

However, in the 1980s, French nuclear safety authority put in place periodic reassessments, because some nuclear installations were already ageing :

❑ effective for NPP

❑ various results for other installations

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Regulation in force concerning PSR

The law Nr 2006-686 of the 13 June 2006 – the "TSN law" :

❑ requires an integrated approach for each step in the life of a facility and for PSR

❑ requires a PSR every 10 years for all facilities (NPP + non-NPP)

The PSR should allow to assess the state of the facility taking into account :
❑ international and national best practices
❑ norms and regulations in force at the time of the PSR
❑ actualization of the facility description
❑ actualization of risks that the facility can generate for public and environment
❑ feedback of operation

The PSR must also cover suggestions of compensatory measures or of modifications made by the licensee to improve its facility safety.

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Regulation in force concerning PSR

The TSN law requires an integrated approach that includes :

- ❑ nuclear safety
- ❑ when it is appropriate : chemical or biological safety
- ❑ radioprotection
- ❑ human & organizational factors

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Assessing the risks

The risks assessment is first made with a **deterministic approach.**
For non-NPP facilities in France, this method was historically used to determine one "envelop" accident for both design and emergency management. This envelop accident was supposed to generate the greatest consequences among all the incidental and accidental situations.

But events have shown that this method was not enough, because they have revealed that other incidental or accidental scenarios, not taken into account for design or even not considered at all, could lead to equal or greater consequences than the envelop accident.

=> The CEA has used a complementary method since the beginning of the 2000s at least for some of its reactors, and the ASN would like to extend this use.

The deterministic approach can be supplemented by :
❑ the "operating conditions" method => the oldest complementary method ; the one the CEA is using
❑ a probabilistic safety assessment (PSA) => only used by NPP

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Assessing the risks

The "operating conditions" method :

The CEA issued a "recommendation paper", which settles down the "operating conditions" method.

1. Definition of safety functions among the list :
   - Containment
   - Neutron reaction
   - Cooling
   - Radiation protection
   - Management of explosive gases produced by radiolysis

2. Identification of the safety important components (SIC)

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Assessing the risks

The "operating conditions" method :

3. Definition of an initial state of the facility, and then adding of a single internal initiating event = creation of an "operating condition"

4. Evaluation of the consequences of each "operating condition"

5. Assessment of the operating conditions regarding the acceptability of their consequences and regarding their probability of occurrence

asn

Assessing the risks

The "operating conditions" method :

4 classes of operating conditions linked to their probability of occurrence per year (ARO) :

- ❑ 1st class : normal operations
- ❑ 2nd class : 10^{-2} < ARO
- ❑ 3rd class : 10^{-4} < ARO < 10^{-2}
- ❑ 4th class : 10^{-6} < ARO < 10^{-4}

ARO < 10^{-6} : excluded situations (beyond design)

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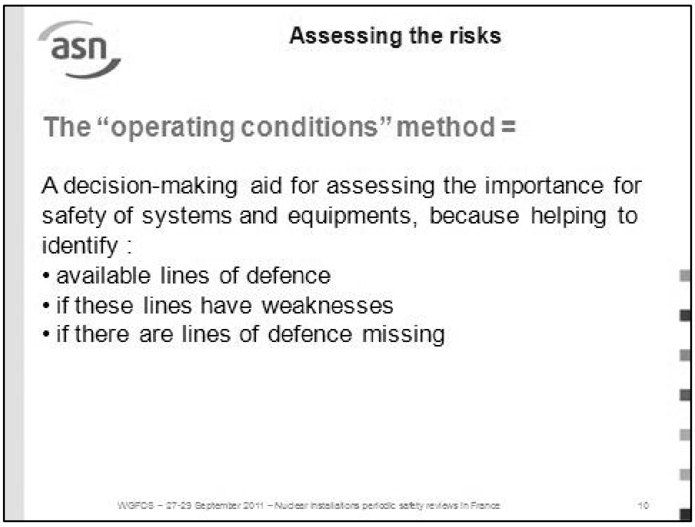
Assessing the risks

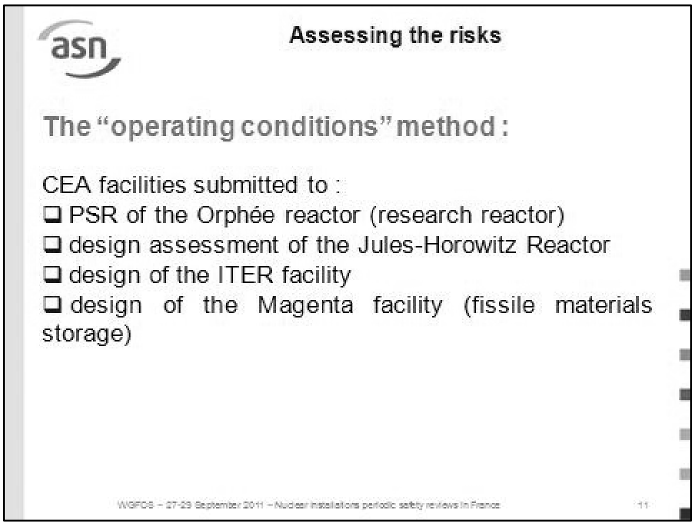
The "operating conditions" method :

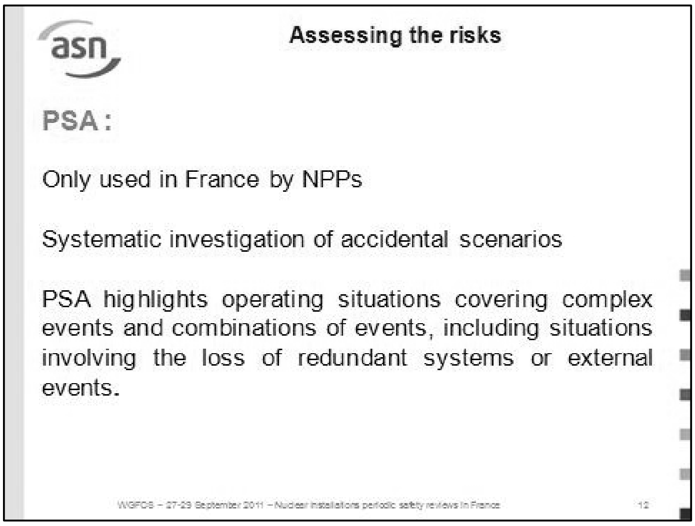
The class of the operating condition is compared to the acceptability of its consequences.
Acceptability is defined by general safety objectives, that depend on the ARO : the more frequently the operating condition can occur, the less its consequences can be.

If the general safety objectives are exceeded, lines of defence have to be added.

The process is iterative.


asn
Assessing the risks
The "operating conditions" method =
A decision-making aid for assessing the importance for safety of systems and equipments, because helping to identify :
• available lines of defence
• if these lines have weaknesses
• if there are lines of defence missing


asn
Assessing the risks
The "operating conditions" method :
CEA facilities submitted to :
❑ PSR of the Orphée reactor (research reactor)
❑ design assessment of the Jules-Horowitz Reactor
❑ design of the ITER facility
❑ design of the Magenta facility (fissile materials storage)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Assessing the risks
PSA :
Only used in France by NPPs
Systematic investigation of accidental scenarios
PSA highlights operating situations covering complex events and combinations of events, including situations involving the loss of redundant systems or external events.

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Assessing the risks

PSA :

1. Choice of an initiating event
2. Determination of accidental sequences resulting from the success or failure of the operation systems and actions involved to perform the safety functions
3. Assessment of the frequency of an undesired resulting event

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Assessing the risks

PSA :

3 types of PSA :

- ❑ PSA level 1 : identification of the sequences leading to core damage, and determination of their frequencies
- ❑ PSA level 2 : assessment of the nature, magnitude and frequencies of releases outside the containment
- ❑ PSA level 3 : assessment of the calculated frequencies of consequences expressed in dosimetry or contamination terms (or in terms of frequencies of cancers or other effects on health)

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Assessing the risks

PSA :

PSA are a decision-making aid for assessing the importance for safety of systems and equipments :

- ❑ Identification of systems contributing to frequency of core damage
- ❑ Determination of the critical failure modes of equipments

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Assessing the risks

Bounding accidents :

Bounding accidents are a conclusion of the safety analysis.

They are reassessed at every PSR of a facility, in order to take into account new standards or regulations.

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Assessing the risks

Bounding accidents :

They depend on parameters like :

- Age of the plant
- Ageing management
- Modifications the plant was subjected to
- Actual condition of its containment
- Radioactive and chemical inventories
- Form (liquid, solid, gas) of the radioactive and chemical materials, and constraints generated by their storage or use
- Number of activities operated on the site, and eventual interactions between them. The interactions can modify accidental scenarios and bounding accidents.

=> Bounding accidents are determined case by case.

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Assessing the risks

Bounding accidents : example of the FBFC site

2 facilities on the site :

- FBFC (facility Nr 98) manufacturing nuclear fuel for NPP => uranium enriched up to 5% ; great amounts of UF_6
- CERCA (facility Nr 63) manufacturing nuclear fuel for research reactors => uranium under metallic form, enriched up to 93.5%

The buildings of the 2 facilities are nested. But the main buildings where the fuel are manufactured are clearly separated.

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Assessing the risks

Bounding accidents : example of the FBFC site

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Assessing the risks

Bounding accidents : example of the FBFC site

Each facility has its bounding accident :

- ❑ FBFC : air crash on its UF_6 storage => major leak of HF
- ❑ CERCA : criticality accident

Theoretical bounding accident of the site : the one of FBFC, because it leads to the largest evacuation zone

But both bounding accidents are taken into account in the operator emergency documents and in emergency and public protection plan, because they do not have similar consequences.

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Assessing the risks

Bounding accidents : perspectives :

The feedback of the Fukushima accident is hoped to cause reassessment of the bounding accidents, by taking into account new scenarios like :

- ❑ interactions between buildings
- ❑ chemical events occurring inside buildings that are not classified as parts of nuclear installations…

The consequences of such events could lead to a bounding accident for the site.

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PSR : practical point of view

- ❑ global view on a facility
- ❑ helps to take decisions for both the ASN and the licensee
- ❑ quite long to perform : about 2 to 3 years between the beginning of the writing by the operator and the ASN demand letter concerning the assessment
- ❑ heavy to develop and to analyse

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PSR : practical point of view

For NPP :

PSR are assessed in 3 steps :

- ❑ A general PSR concerning one technological type (900 MW, 1300 MW, 1450 MW, EPR)
- ❑ A specific PSR concerning the reactor itself => technical matters
- ❑ A ten-yearly specific review of a reactor, where components crucial for safety are submitted to special inspections and tests

PSA results are examined and criticized during PSR.

NPP can also be submitted to specific PSR on one particular subject (HOF, feedback from events…)

The creation authorization of one NPP settles down a precise lifetime. PSR helps to examine whether the NPP can go on operating or not beyond this initial lifetime.

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PSR : practical point of view

For non-NPP facilities :

PSR look more as integrated assessments

no design life acted in the creation authorizations
=> every ten-yearly PSR concerning one plant has to examine whether the plant can go on operating or not

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THE SAFETY EVALUATION GUIDE FOR LABORATORIES AND PLANTS A TOOL FOR ENHANCING SAFETY

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Abstract – The Institute for Radioprotection and Nuclear Safety (IRSN) acts as technical support for the French government Authorities competent in nuclear safety and radiation protection for civil and defence activities.

In this frame, the Institute's performs safety assessments of the safety cases submitted by operators to these Authorities for each stage in the life cycle of a nuclear facility, including dismantling operations, which is subjected to a licensing procedure. In the fuel cycle field, this concerns a large variety of facilities.

Very often, depending on facilities and on safety cases, safety assessment to be performed is multi-disciplinary and involves the supervisor in charge of the facility and several safety experts, particularly to cover the whole set of risks (criticality, exposure to radiation, fire, handling, containment, human and organisational factors…) encountered during facility's operations.

Taking these into account, and in order to formalize the assessment process of the fuel cycle facilities, laboratories, irradiators, particle accelerators, under-decommissioning reactors and radioactive waste management, the "Plants, Laboratories, Transports and Waste Safety" Division of IRSN has developed an internal guide, as a tool:

- To present the methodological framework, and possible specificities, for the assessment according to the "Defence in Depth Concept" (Part 1);
- To provide key questions associated to the necessary contradictory technical review of the safety cases (Part 2);
- To capitalise on experience on the basis of technical examples (coming from incident reports, previous safety assessments…) demonstrating the questioning (Part 3).

The guide is divided in chapters, each dedicated to a type of risk (dissemination of radioactive material, external or internal exposure from ionising radiation, criticality, radiolysis mechanisms, handling operations, earthquake, human or organisational factors…) or to a type of safety file (safety options file, general operating rules, on site emergency plan, periodic safety review documents, incident analysis…). In each chapter, the aforesaid Parts 1, 2 and 3 are developed.

A first draft of the guide was published in March 2010 for use by assessment's teams of IRSN, and to obtain an operational feedback to improve it.

Beyond that, the guide is also intended to be, on the topic of safety assessment for the fuel cycle facilities, laboratories, irradiators, particle accelerators, under-decommissioning reactors and radioactive waste management, a tool for tutoring (inside and outside the IRSN) and a reference to make available, outside of the IRSN, the approach of expertise and the "know-how" of IRSN. In this context, the IRSN's methodology of assessment regarding "criticality" and "fire" have been put on-line, on the IRSN's website.

The paper presents the purpose and the structure of the guide and its interest for the safety assessment of fuel cycle facilities; in this frame, the chapters "Assessment of the risk from handling operations" and "Assessment of the periodic safety review documents" are presented in details as illustrations. It gives also information about its others uses.

1. Introduction

In the field of the nuclear fuel cycle facilities, France has more than 120 facilities (civil and defence activities), classified as basic nuclear installations (BNI) according to the French regulation, falling under the "Laboratories, plants, facilities being dismantled, waste processing or interim storage facilities or disposal facilities" category (called LUDD facilities from now on).

Given the diversity of LUDD facilities, the type and significance of risks associated with their operation, and the potential consequences for the environment or for workers, differ from one facility to the next. The risks associated with these facilities are normally grouped into the following major categories:

- The risk due to radioactive materials in the facility: risk of these materials being disseminated both inside and outside the facility, risk of internal and external exposure to ionising radiation, criticality risk, risk of explosion caused by radiolysis and risk due to the heat releases induced by the radioactivity;
- The risk originating inside a facility which may disseminate radioactive materials, exposure to ionising radiation or a criticality accident, such as the risks of fire, explosion, overpressure, falling load, loss of auxiliary fluid (especially power supply), etc.;

In France, the LUDD facilities are extremely diverse, but with regard to the type of activity they are mainly:

Facilities belonging to the front-end of the cycle: conversion and enrichment of uranium plants at the Tricastin site, uranium fuel manufacturing plant in Romans-sur-Isère, MOX fuel manufacturing plant at the Marcoule site;

Spent fuel reprocessing plants at La Hague site;

Laboratories and research facilities in several sites, essentially at Cadarache site, Marcoule site and Saclay site;

Liquid and solid waste treatment and conditioning facilities in many sites;

Waste storage facilities in different sites;

Waste disposal facilities at the Manche site and Aube site.

Unlike the nuclear power plants operated by EDF, the LUDD facilities are run by a number of plant operators, the main ones being AREVA, CEA, ANDRA and EDF.

- The risk originating outside a facility, either related to human activities (risks from neighbouring facilities, transport of hazardous materials nearby (gas pipes, tankers, etc.), aircraft crashes, etc.) or natural disasters (earthquake, flood, extreme climate conditions, etc.);
- The risk relating to organisational aspects and human factors.

With regard to the safety of these facilities, all plant operators must demonstrate that safety provisions in place to face the risks based on studies of safety and radiation protection are adequate and sufficient. These studies are reported in safety cases transmitted to the French Authorities competent in nuclear safety and radiation protection for civil and defence activities. The safety cases must also justify the provisions of controlling the risks. Various types of documents are made according to:

- The different states of the life of the facilities (creation, operation, modification, periodic safety review, final shutdown and dismantling…) that are subject to an authorisation procedure;
- The different aspects of plant operation under normal or accidental conditions (general operating rules, on-site emergency plan...).

The Institute for Radioprotection and Nuclear Safety (IRSN) assesses the files submitted by operators to the different competent authorities (Nuclear Safety Authority – ASN, Defence Nuclear Safety Authority – DSND). The safety of the facilities is reviewed on a case by case basis, taking account of the specificities of the facilities, their risks and the provisions proposed by the operators, and also the operational experience feedback. The safety assessment consists of a contradictory technical review of the safety cases; it aims to check the robustness of the proposed provisions under the principle of "defence in depth". The conclusions and recommendations of this assessment are provided to the Safety Authorities in the form of a technical report.

Inside IRSN, the "Plants, Laboratories, Transports and Waste Safety" Division (DSU), as an operational safety division of IRSN, examines in particular the safety cases of the LUDD facilities.

Very often, depending on the type of LUDD facilities and safety cases, safety assessment to be made is multi-disciplinary and involves the supervisor in charge of the facility, who is responsible for the coherence and consistency of the assessment and also links with the operator, and several safety experts from IRSN, especially to cover the whole set of risks of the facility (criticality, radiation exposure, fire, explosion, handling, containment, human and organisational factors ...). This assumes that each person who takes part in the safety assessment works in his own area of expertise, but also participates in technical discussions with colleagues from other areas or skills, and also with the operator, to elaborate the ultimate collective evaluation which is given in the form of a technical report of the IRSN. Per year, about 200 technical reports are sent to the ASN concerning civil LUDD facilities.

In 2004, it was decided to improve the process of safety assessment of LUDD facilities by formalizing it in a common tool that each participant of the assessment's team could use as a technical aid for his expertise.

So, in order to meet this goal, the DSU has developed the Safety Evaluation Guide for Laboratories and Plants, as a tool:

- To present the methodological framework, and possible specificities, for the assessment according to the "defence in depth" concept (Part 1);

- To provide key questions associated to the necessary contradictory technical review of the safety cases (Part 2);
- To capitalise on experience on the basis of technical examples (coming from incident reports, previous safety assessments…) demonstrating the questioning (Part 3).

The paper presents, in section 2.1, the purpose and the structure of the guide and its interest for the safety assessment of LUDD facilities. The first draft of the guide was published in March 2010 for use by assessment teams of IRSN, and to obtain an operational feedback to improve it; in connection with this version, the chapters “Assessment of the risk from handling operations” and “Assessment of the periodic safety review documents” are presented as demonstrations, respectively in sections 2.2.2 and 2.2.3.

It should be underlined that in its three-part configuration, the guide is intended only for internal use in the IRSN. In this respect, it also serves to transmit the safety assessment know-how to any “junior” staff or even to give a view of the safety approach on the overall risks to any staff member of the ISRN.

However, the guide is also designed:

- To be used as a tool for tutoring on the topic of safety assessment for the LUDD facilities. Indeed, IRSN has given considerations for several years to develop its capabilities to transmit, by the way of expertise schools, its safety assessment know-how, inside and outside IRSN;
- To serve as a reference to make available, outside the IRSN, for example to other technical support organisations (TSO), the approach of expertise and the "know-how” of IRSN in the field of safety assessment for the LUDD facilities.

Information about these uses is given in section 2.4.

Finally, it appears important to note that this paper is related to the one titled “IRSN Global Process for Leading a Comprehensive Fire Safety Analysis” to be presented by Y. Ormières, also fron IRSN.

2. Safety Evaluation Guide for Laboratories and Plants of IRSN

2.1. General description

In order to cover the vast field of the safety assessment of the LUDD facilities, the guide considers two key areas of expertise constituted by:

- The different types of events/risks (criticality, internal or external exposure to ionizing radiation, fire, handling...) associated with facilities operations or external hazards (earthquake);
- The different types of safety files (safety options report, safety report, on-site emergency plan, general operating rules...) that accompany the life of the facility.

The guide is divided into chapters corresponding to one of the above mentioned type of event/risk or file. Each chapter consists of three separate parts which are described below.

A Part 1 "Doctrine" is aimed at presenting the safety approach of the IRSN in the field of LUDD facilities. In this part, the general principles and the key points retained for the IRSN's expertise, for the type of risk or the type of file, are set out and any specific points are specified. This part is also used to introduce Part 2.

A Part 2 "Assistance for the expertise" is aimed at formalizing the basic questioning related to the expertise of the considered type of risk or file. The questions are those that people working on the safety assessment must consider systematically. This formalisation allows:

- To help to bring out issues that could not outlined in the safety case;
- To promote an independent questioning approach of the operator’s approach;
- To assist in the formulation of questions to the operator on the various topics assessed by the guide;
- To prevent from the discovery of “basic” questions at the end of instruction;
- To have a better perception by supervisors of the facility of the issues examined by experts and, ultimately, to make sure to get a common vision between the various participants involved in the safety assessment...

Questions are presented in diagrams defining several successive steps that follow the usual logic of the assessment in the concerned field (presented in Part 1) and prefigure the structure of the final document (technical report of the IRSN) that will provide assessment results. The usual procedure for the diagrams is:

- Identification and analysis of input data;
- Identification and analysis of safety requirements set by the operator;

Identification and analysis of provisions proposed by the operator to meet these requirements. The analysis is done for each proposed measure, but also for all measures with the examination of emergency situations.

In practice, studying the diagrams before reading the operator’s safety report is strongly encouraged. Indeed, starting an assessment with a critical reading from the first to the last page of the operator’s report means taking the risk of being influenced by its contents and structure thereby failing to identify issues that are not addressed in the report.

A list of “support documents” is included just before the diagrams in each chapter. The list has been intentionally reduced to include documents only useful for the assessment.

A Part 3 "Capitalisation of experience" is aimed at maintaining and disseminating knowledge and experience. In this part, the issues considered to be essential to the field of expertise are shown in an explicit form. The demonstrations may derive from examples as previous safety assessments and particular events or incidents (based on an existing data base (SAPIDE LUDD) which, currently, contains more than 4 900 incidents that occurred in non-NPP BNI, mainly in France).

However, it should be noted that "this guide is only a guide". Indeed, in the process of safety assessment as advocated by the IRSN:

The expertise of the safety of nuclear installations is not carried out by an exhaustive list of predefined questions;

The safety assessment of LUDD facilities should be adapted in each case, given the great diversity of the facilities, their location and the external hazards.

Also, this guide does not replace the dialogue between all the participants involved in the safety assessment, and it does not provide the conclusions of the expertise (which are the result of the collective judgment of the aforesaid participants)...

Finally, it should be recalled that, in its use as a tool for safety assessment, the guide is part of the documentary set associated with the quality assurance process of DSU "Provide support and technical assistance to public authorities."

2.2. Presentation of the current guide

2.2.1. General content

Initially, it was chosen to draft a first version of the guide limited to 13 chapters:

- for key events/risks encountered:
  - ~ spread of radioactive materials;
  - ~ internal or external exposure to ionizing radiations;
  - ~ criticality;
  - ~ fire;
  - ~ radiolysis;
  - ~ handling;
  - ~ earthquake;
  - ~ human and organisational factors;
- for the main files:
  - ~ safety options report;
  - ~ general operating rules;
  - ~ on-site emergency plan;
  - ~ safety review;
  - ~ incident analysis.

The other types of risk (explosion, loss of auxiliaries, aircraft crash, internal or external flooding and external hazards) and assessments (modification of a facility or a facility's operating range) will be covered in a next version of the guide.

For each chapter:

The Part 1, is, up until now, just an introduction page that describes the steps to follow to achieve the expertise of the field and highlights the key points to be considered in each step. This Part 1 will be completed in the future to present the doctrine for all chapters;

Part 2 is written in-extenso;

Part 3 is partially completed, excepted for the chapter "Assessment of criticality risk", as writers of this chapter, which were all criticality experts, developed this part to include the feedback of events related to the criticality risk which occurred in the French LUDD facilities. For this purpose, a specific database, derived from the SAPIDE database, gathers all events that could have affected the nuclear criticality safety. Each incident of this database is analyzed and connected to one of the questions listed in the diagrams of Part 2. At this time, this database contains about 400 events.

Currently, this version of the guide exists only in a "paper" form.

2.2.2 Chapter "Assessment of risk relating to handling operations"

The Part 1 "Doctrine" of this chapter presents, in an introductory page, the approach used by IRSN to assess the risk related to handling operations as following.

Initially, it is necessary to assemble the basic data required for the analysis. This involves taking an inventory of handling equipment, stating its areas of operation. This also includes listing each load handled by each piece of equipment, including the specifications as well as any safety targets in the danger zone underneath the load.

One basic point in the assessment of risk related to handling is the identification of risky configurations in using handling equipment. For the various scenarios, it is necessary to determine the situations for which potential damage to the load and/or safety target (structure, system and components – SSC) are considered significant.

For each of the previously identified cases with a significant safety issue, the examination should continue with an analysis of technical and/or human measures taken into account by the operator to prevent, detect, monitor and limit the consequences. For each case, the defense barriers are evaluated to determine if they are adequate given the safety requirements. Finally, risk related to handling are examined more generally through analysis of handling accidents considered by the operator, such as accidents that could require other provisions (containment measures, radiological protection, etc.). After examining the relevance of selected scenarios (deterministic approach), the next step is to insure that consequences for any employees, the facility, and the environment have been correctly assessed by the operator. Examination of accidents should lead to an assessment of the overall design basis of the facility with regard to handling risks.

In Part 2 "Assistance for the expertise", diagrams follow the order of the phases described in the above mentioned introductory page:

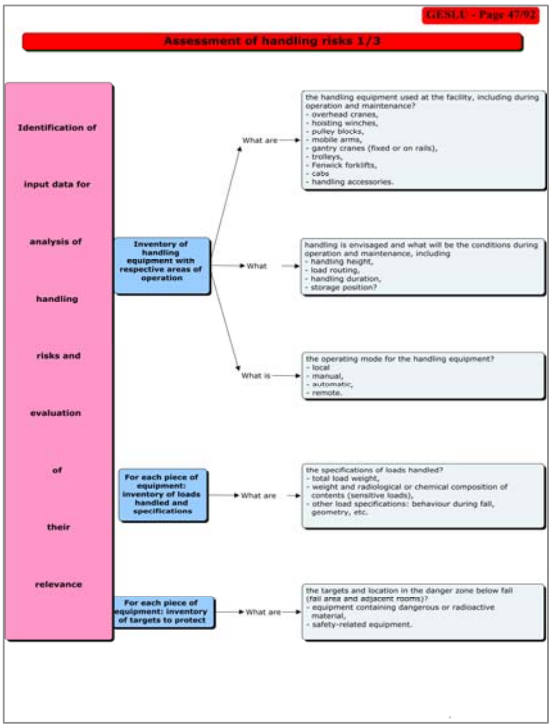
In rectangles on the left side of the diagrams, basic topics related to the assessment (identification of inputs data, identification of safety issues, analysis of risk control measures, analysis of accident situations) are identified;

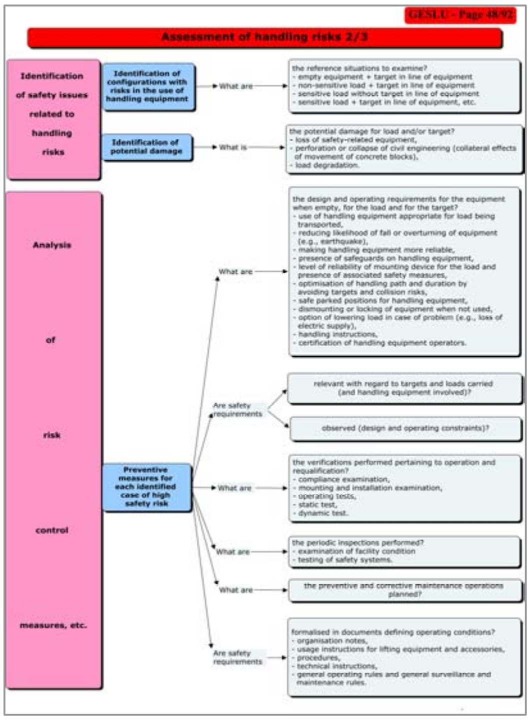
In rectangles on the middle of the diagrams, each sub-theme related to the basic topics is mentioned (for example, the sub-themes "measures for preventing the risk", "measures for monitoring the risk" and "measures for limiting consequences" which are linked to "the analysis of risk control measures" basic topic), in order to introduce the questioning. Actually, these sub-themes are those that must be in the operator's safety files regarding the risk related to handling operations;

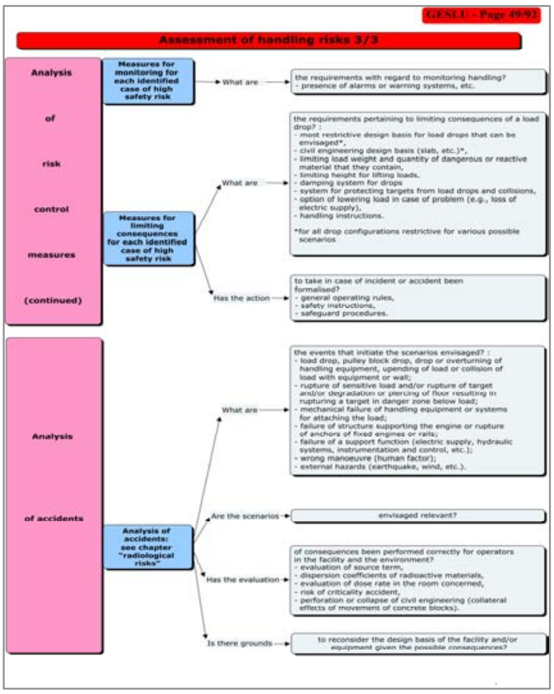
In rectangles on the right side of the diagrams, the questioning itself (necessary issues to be addressed) is detailed.

This is illustrated in the diagrams given in the Figure 1 hereafter.

Figure 1**. Chapter "Assessment of risk relating to handling operation"**
Part 2**. "Assistance for the expertise"**







2.2.3 Chapter "Assessment of a safety review"

According to the principle of introducing the methodology of IRSN, the methodology for the assessment of a safety review is presented as follows in the Part 1 of the corresponding chapter.

Section III of article 29 of the law promulgated on June 13, 2006 pertaining to transparency and nuclear safety states: "the operator of a regulated nuclear facility will perform regular safety reviews of the facility while taking the best international practices into account… Safety reviews will take place every ten years."

The ASN's decision in preparation specifies conditions in which a nuclear plant operator performs safety reviews of basic nuclear installations and issues to be covered in the report required by article 24 of the order of November 2, 2007. The decision requires that the operator submit a safety review preparation file to the ASN at least two years before the date intended for doing and submitting the complete safety review.

The safety review preparation file:

- Presents the major changes planned over the next decade regarding the facility;
- Gives the safety review objectives;
- Justifies the priorities of the issues handled as part of the safety review;
- Justifies the methodology to be used for the facility's safety review with conditions adopted for performing the compliance review;
- Presents an organisation, an estimate of human and financial resources and the schedule for the safety review.

The safety review has two parts: the compliance review and the reassessment review, and a conclusion which is the operator's proposed action plan.

Initially, operating experience feedback from the facility and similar plants are examined to identify issues for further assessment.

Examination of the facility's compliance with its updated safety baseline, and its operation, must ensure that changes (due to modifications or ageing) do not put into question observance of applicable safety requirements. It is advisable to ensure that the on-site verification programme, operating documents and associated criteria are sufficient. Also, the management of facility ageing must be assessed.

The safety review is intended to evaluate and improve the facility's level of safety with regard to regulations and the latest French and international safety practices. All risks have to be assessed, especially risks that may have been impacted by modifications at the facility that were authorized internally.

Finally, the last part of the assessment consists in analyzing the operator's conclusions and any proposed improvements by evaluating their adequacy and the schedule for carrying them out in light of the future of the facility (at least for the next ten years).

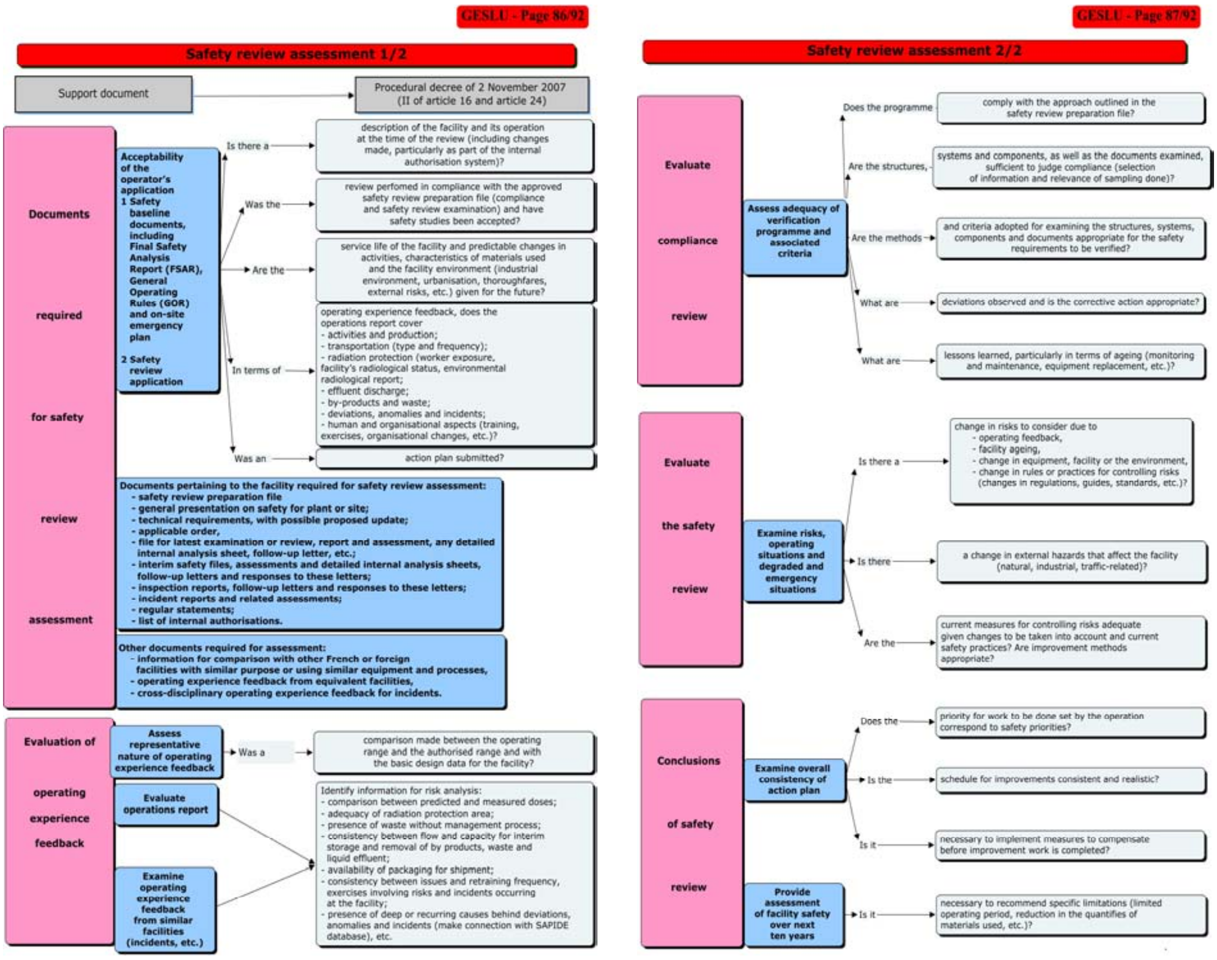
Consequently, in the Part 2:

- The basic topics identified in the rectangles on the left side of the diagrams concern the documents required for the assessment, the evaluation of the experience feedback, the evaluation of the compliance review, the evaluation of the safety review, and also the operator's conclusions;
- For the sub-themes related to the basic topics, and identified in the rectangles on the middle of the diagrams, we can highlight, for example, that "to examine the experience feedback from

similar facilities" is one of the topics asked to the operator when speaking about the "evaluation of the experience feedback", as well as "to examine risks, operating situations and degraded and emergency situations" when speaking about the "evaluation of the safety review".

This is illustrated in the diagrams given in the Figure 2 below, which also show, in rectangles on the right side of the diagrams, the minimum issues to be addressed.

Figure 2 - Chapter "Assessment of a safety review**" – Part 2 "Assistance for the expertise"**



2.2.4. Drafting, implementation and perspectives

The drafting work has mobilized, between March 2006 and February 2009, about fifteen people confirmed or experts (one by type of risk and by type of file) belonging to four different divisions of the IRSN; in addition, one people was in charge of coordinating the work. A cross-validation of each drafted parts was carried out by a small group (5 persons) composed of experienced people and future users.

Ultimately, the set of this work involved about 3 man.year, on about four years.

In March 2010, the first edition was made available to potential users (supervisors in charge of the facilities, experts, etc.) to be used for safety assessments. This provision has been accompanied by the presentation of the objectives and limits of the guide to users of the guide, and examples on the way to use it.

The purpose of this release was to get "back", from users, comments, remarks and suggestions... constituting a first experience feedback. So, at the end of 2010, the main lessons learned were the following:

- The guide is used primarily by the personal "inexperienced" in the safety assessment;
- It is considered as an appropriate tool for the assessment of risks of the LUDD facilities and the structure of the Part 2 (questions) is generally appreciated;
- In some cases, only the rectangles of the left and of the middle are used. It depends particularly on the experience of the engineer and the type of assessment requested (preliminary analysis for example);
- It provides valuable assistance in the structuring of the analysis to conduct, especially in terms of identification of risks, as well as in the development of questionnaires transmitted to operators so that they could complete the safety file;
- Some chapters are particularly appreciated (assessment of general operating rules, assessment of risk relating to handling operations, for example), certainly because of the lack of other documents available to treat in detail the topics or of experts units of IRSN in charge of these issues;
- The “paper” shape of the guide is not convenient, and a “computer” shape is suggested.

An action-plan of 5 years has been established for completing and improving the guide on the basis of this feedback. This includes short-term actions to complete Part 1 and draw up Part 3 of existing chapters (on the model of what was done for the “criticality” chapter – see section 2.2.1), to write new chapters and to implement a computer tool for consulting the guide. This includes also developments regarding a systematic use of the guide for safety assessments of LUDD facilities, which may in particular be achieved by using it as a tool for tutoring “junior” staff of the IRSN. These actions, which involve once again various divisions of IRSN, will be driven by the “Competences, knowledge development and experience feedback” (COREX) directorate of DSU, which has skills that cross the topics covered in the guide.

2.3 Other uses of the guide

In parallel of what is described in section 2.1, the two main following objectives are pursued regarding the guide.

Firstly, Parts 1, 2 and 3 are intended to be used as a tool for tutoring on the topic of safety assessment for the LUDD facilities. Indeed, IRSN has given considerations for several years to develop its capabilities to transmit, by the way of expertise schools, its safety assessment know-how:

- Inside of IRSN, tutoring sessions are organised by the experts units in their area of competencies (fire, criticality, containment…) and an "expertise school" is about to be implemented;
- Outside IRSN, four TSOs (IRSN, GRS - Germany, UJV - Czech Republic, and LEI - Lithuania) have created, in 2010, the European Nuclear Safety Training and Tutoring Institute (ENSTTI, http://www.enstti.eu/Pages/Home.aspx). This Institute offers short applied training sessions and longer tutoring periods for university graduates and for those with some professional experience in the nuclear sector. Its focus is on transmitting European research and assessment know-how in the fields of nuclear safety and radiation protection.

Secondly, Parts 1 and 2 of the guide (we recall that Part 3 is intended only for internal use inside IRSN) are intended to serve as a reference to make available, outside IRSN, the approach of expertise and the "know-how" of IRSN in the field of safety assessment for the LUDD facilities. For example, the IRSN's methodology of assessment regarding "criticality" and "fire" risks have been put on-line, with an English version, on the IRSN's website (http://www.irsn.fr/FR/base_de_connaissances/librairie/publications_professionnels/Pages/guides_techniques.aspx), to provide, for professionals in the nuclear field, support to the implementation or assessment of a criticality or fire safety analysis. It is planed to do the same with the other chapters of the guide. Beside, IRSN suggests that it could be a support to exchange bout methodology, safety approaches… particularly with other TSOs like those of the European TSO Network (ETSON), on the common area of safety assessment (e.g. to improve the TSOs ability to assess the licensees capacity for accident prevention and mitigation).

In addition, to inform the international community involved in nuclear safety, presentations are done at international technical conferences. For example, a poster of the "criticality" chapter was presented at the last TSOs conference (Tokyo, October 2010) and will be at the next EUROSAFE Forum (Paris, 7-8 November 2011), and a full paper will also be presented at the International Conference on Nuclear Criticality 2011, in Edinburgh (19-22 September 2011).

3. Synthesis and conclusion

One TSO concern is to maintain its capability to perform highly relevant safety assessment in order to contribute to enhance the safety of nuclear facilities, particularly in the light of a wide variety of facilities. In France, IRSN has given considerations for several years to develop tools, like safety guides or training, to achieve this goal, taking into account national and international matters.

In practice, a safety assessment of a nuclear facility begins with issues that arise for the assessment team (supervisor of the facility, experts…) during examination of the operator's file. Responses to these issues become the subject of technical discussions between the members of this team and operator, with further input from those who validate the assessment. Technical content of the assessment thus depends both on the nature of issues that have emerged and the quality of discussions among those participating in the assessment.

Experience has shown that issues vary for each assessment but, when a risk is assessed (criticality, fire, etc.), certain issues tend to recur. This also applies to an assessment of a given type of report (safety options report, general operating rules, etc.). Moreover, experience has also shown the need to get a common vision between the various participants involved in a safety assessment.

Taking this into account, the "Safety Evaluation Guide for Laboratories and Plants" has been developed, as a fruitful exercise of collegial work at IRSN, for the safety assessment of facilities falling under the "Laboratories, plants, facilities being dismantled, waste processing or interim storage facilities or disposal facilities" category.

The guide, which is divided into chapters corresponding to a type of risk or file, intends to:

- Express the IRSN's methodology to be followed in performing an assessment (Part 1 "Doctrine");
- Give the necessary issues to be addressed, in order to raise issues that could have not been addressed in the operator's safety file (Part 2 "Assistance for the expertise");
- Present illustrations (from previous assessments or incidents experience feedback) about points considered essential in the assessment (Part 3 "Capitalisation of experience").

The three parts complement each other, and relate to each other, but can be considered separately if necessary.

A simple and user-friendly computerized version has been developed to facilitate its use; an English version will be available.

However, as a tool for safety assessments, this guide is only a guide, and it may apply completely, partially or not at all. Indeed, the safety assessment of a nuclear installation is not carried out by an exhaustive list of predefined questions. In addition, other issues not raised in the guide may arise. Finally, it does not provide the judgement necessary to respond to these issues, since the final collective judgement is the product of technical discussions between various participants in the assessment.

In parallel to this primary objective, the guide is also intended to transmit, inside and outside IRSN, the knowledge and the know-how of IRSN on the topic of safety assessment of fuel cycle facilities, in providing:

- A support for tutoring, e.g. in the frame of "expertise schools";
- A point of reference on the approach of expertise, about which the IRSN would like to exchange, e.g. with other TSOs or professionals in the nuclear field.

Clearly, the goal of this guide is to be a tool for enhancing safety.

IRSN
INSTITUT DE RADIOPROTECTION ET DE SÛRETÉ NUCLÉAIRE

Enhancing safety

The Safety Evaluation Guide for Laboratories and Plants
A Tool for Enhancing Safety

V. LHOMME and J.P. DAUBARD
IRSN - France

The Safety Evaluation Guide for Laboratories and Plants

Content

The Safety Evaluation Guide for Laboratories and Plants

1. Context

In France, more than 120 nuclear facilities (civil and defence activities), falling under the "Laboratories, plants, facilities being dismantled, waste processing or interim storage facilities or disposal facilities" category (LUDD facilities hereafter)

- **very diverse** (type of activity, type of risks (very specific for some facilities), type of provisions implemented to control risk, importance of consequences that may result from risks)
- **operated by** various operators (main ones: AREVA, CEA, ANDRA and EDF), **with different safety culture**

Marcoule site (CEA & AREVA)

La Hague site (AREVA)

Saint Laurent site (EDF)

Soulaines site (ANDRA)

The Safety Evaluation Guide for Laboratories and Plants

1. Context

Safety approach for LUDD

- Essentially deterministic approach:
- various situations, and in particular accidents, considered to be plausible, have been taken into account
- the monitoring systems and engineered safety systems implemented will be capable of ensuring the containment of radioactive materials
- Based on the "defence-in-depth" concept (successive barriers for prevention, detection and limitation of consequences of any events)

Safety assessment of LUDD

- Safety of the facility examined on a case by case basis: depending on the specificities of the facility, its risks, provisions proposed by the operator, OEF, environment
- Contradictory technical review of the operator's safety case:
- for each stage in the life cycle of a facility which is subjected to a licensing procedure
- for all risks within an evaluation of the "defence-in-depth" concept set up by the operator
- Take into account the various operation conditions: normal, incidental and accidental

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The Safety Evaluation Guide for Laboratories and Plants

1. Context

IRSN's safety assessment's team

- supervisor in charge of the facility, responsible for the coherence and the consistency of the assessment and linking with the operator and the safety authority
- safety experts (criticality, radiation exposure, fire, explosion, containment, human and organizational factors...) to cover the whole set of risks of the facility
- hierarchy in charge of the validation of the assessment

Technical content of the assessment depends both on:
- the nature of issues that have emerged
- the quality of discussions among each participant of the multi-disciplinary assessment's team and also with the operator

- issues vary for each assessment but certain questions recur
- a common methodology between the various participants is essential

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The Safety Evaluation Guide for Laboratories and Plants

1. Context

Formalization of the safety assessment methodology

Safety Evaluation Guide for Laboratories and Plants

tool that each participant to the assessment's team can use as a technical aid for his expertise

- 2004 — Decision to establish the guide
- March 2006 — Drafting of the parts of the guide (15 people, confirmed and experts) and cross-validation of each drafted parts (5 people, experienced and future users) *about 3 man.year, from 4 divisions of the IRSN*
- Feb. 2009 / March 2010 — First edition of the guide (« current guide ») available for users
- March 2010 / Dec. 2010 — First operational feedback from users
- Dec. 2010 — Further improvements and developments
- time

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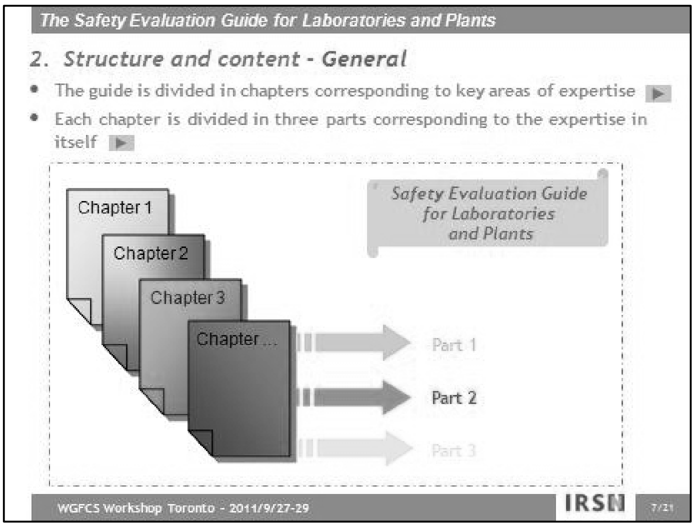
The Safety Evaluation Guide for Laboratories and Plants
2. Structure and content - General
The guide is divided in chapters corresponding to key areas of expertise
Each chapter is divided in three parts corresponding to the expertise in itself
Safety Evaluation Guide for Laboratories and Plants
Chapter 1
Chapter 2
Chapter 3
Chapter ...
Part 1
Part 2
Part 3
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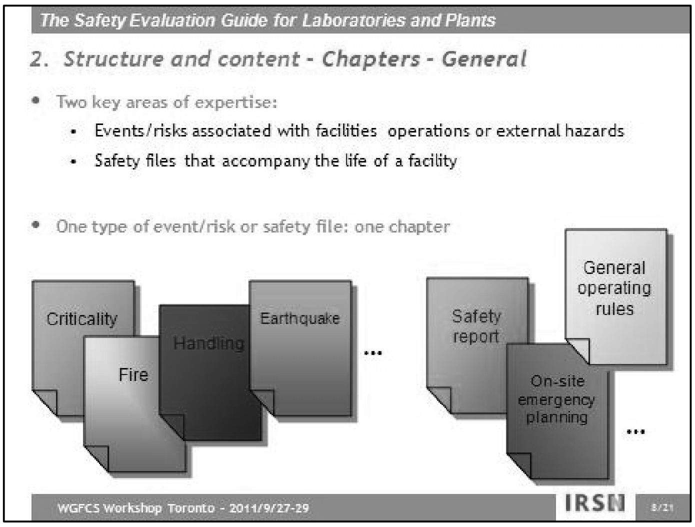
The Safety Evaluation Guide for Laboratories and Plants
2. Structure and content - Chapters - General
Two key areas of expertise:
Events/risks associated with facilities operations or external hazards
Safety files that accompany the life of a facility
One type of event/risk or safety file: one chapter
Criticality
Fire
Handling
Earthquake
...
Safety report
General operating rules
On-site emergency planning
...
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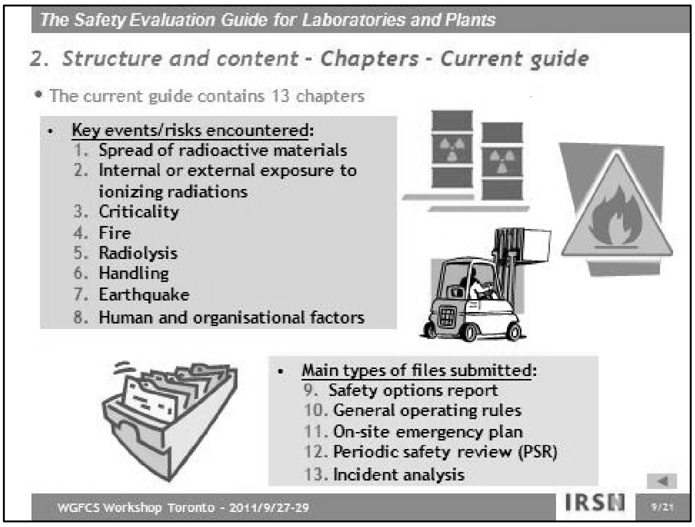
The Safety Evaluation Guide for Laboratories and Plants
2. Structure and content - Chapters - Current guide
The current guide contains 13 chapters
Key events/risks encountered:
1. Spread of radioactive materials
2. Internal or external exposure to ionizing radiations
3. Criticality
4. Fire
5. Radiolysis
6. Handling
7. Earthquake
8. Human and organisational factors
Main types of files submitted:
9. Safety options report
10. General operating rules
11. On-site emergency plan
12. Periodic safety review (PSR)
13. Incident analysis
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The Safety Evaluation Guide for Laboratories and Plants

2. Structure and content - Parts - General

Part 1 « Doctrine »	Part 2 « Assistance for the expertise »	Part 3 « Capitalization of experience »
Presentation of the safety approach of the IRSN: general principles and key points for the assessment according to the "defence in depth" concept	*Formalization of the basic questioning related to the expertise: questions that people working on the safety assessment must consider systematically*	*Dissemination of knowledge and experience feedback: illustrative examples based on existing data from previous safety assessments, technical reports, incidents data bases, etc*

introduction

demonstration

The three parts complement each other, and relate to each other, but can be considered separately if necessary

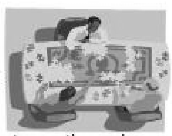
The Safety Evaluation Guide for Laboratories and Plants

2. Structure and content - Part 2 - General

The formalization allows

- To help to bring out safety issues that could not be outlined in the safety case
- To promote an independent questioning approach not linked to the operator's safety demonstration
- To assist in the formulation of questions to the operator

- To prevent from forgetting "basic" questions
- To have a better perception, by IRSN's supervisor of a facility, of the issues examined by experts



- To make sure to get a common methodology between the various participants of the safety assessment's team

The Safety Evaluation Guide for Laboratories and Plants

2. Structure and content - Part 2 - General

- Questions are presented in diagrams:
  - defining several successive steps that follow the usual logic of the assessment in the concerned field (presented in Part 1)
  - prefiguring the structure of the technical report of the IRSN

Identification and analysis of input data

- Identification and analysis ...
- ... of safety requirements set by the operator
  - Identification and analysis of provisions ...
  - ... proposed by the operator to meet these requirements

The Safety Evaluation Guide for Laboratories and Plants

2. Structure and content - Part 2 - General

The guide is only a guide

- The expertise of the safety of nuclear installations is not carried out by just an exhaustive list of predefined questions
- Other issues not raised in the guide may arise
- The safety assessment should be adapted given the specificities of the facilities, their location, the external hazards...
- It does not replace the dialogue between all the participants involved in the safety assessment
- It does not provide the conclusions of the expertise (which are the result of the collective judgment of the aforesaid participants)...

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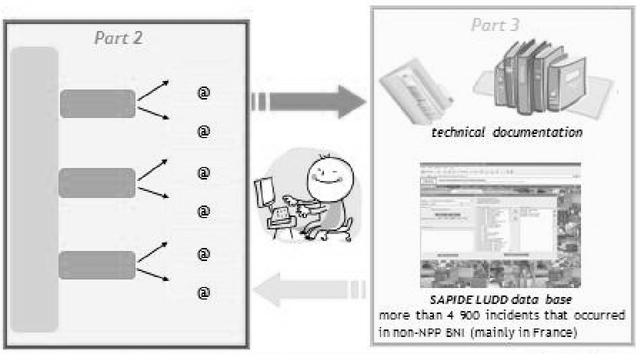
The Safety Evaluation Guide for Laboratories and Plants

2. Structure and content - Part 3 - General

- Illustrative exemples related to the field of expertise are given plainly



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The Safety Evaluation Guide for Laboratories and Plants

2. Structure and content - Current guide

- For the 13 existing chapters

Part 1 « Doctrine »	*One introduction page for all the chapters* *Additions being made to detail the IRSN's doctrine*
Part 2 « Assistance for the expertise »	*Written in-extenso for all the chapters*
Part 3 « Capitalization of experience »	*Only the chapter "Assessment of criticality risk" is fully completed* *Additions being made for the 12 other chapters*

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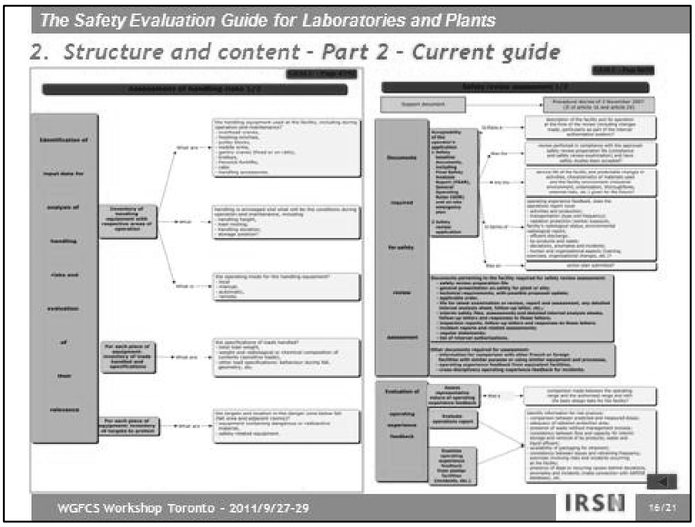
The Safety Evaluation Guide for Laboratories and Plants
2. Structure and content - Part 2 - Current guide
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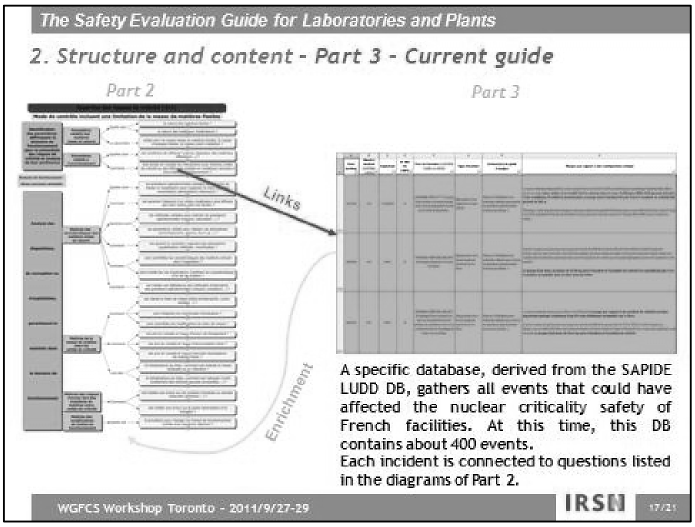
The Safety Evaluation Guide for Laboratories and Plants
2. Structure and content - Part 3 - Current guide
Part 2
Part 3
Links
Enrichment
A specific database, derived from the SAPIDE LUDD DB, gathers all events that could have affected the nuclear criticality safety of French facilities. At this time, this DB contains about 400 events.
Each incident is connected to questions listed in the diagrams of Part 2.
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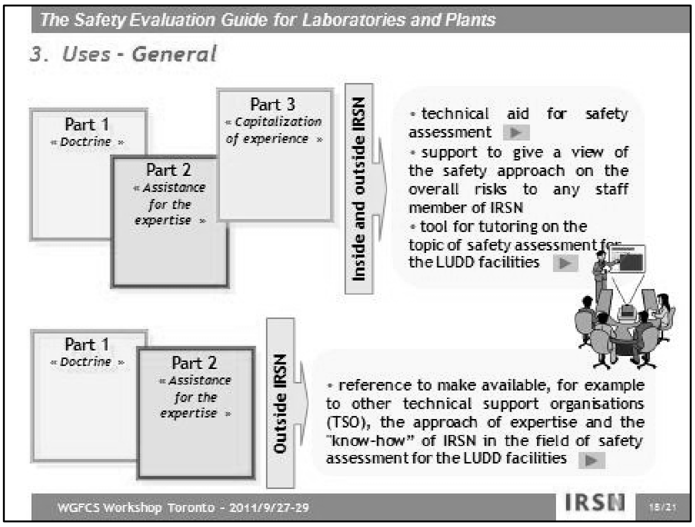
The Safety Evaluation Guide for Laboratories and Plants
3. Uses - General
Part 1
« Doctrine »
Part 2
« Assistance for the expertise »
Part 3
« Capitalization of experience »
Inside and outside IRSN
• technical aid for safety assessment
• support to give a view of the safety approach on the overall risks to any staff member of IRSN
• tool for tutoring on the topic of safety assessment for the LUDD facilities
Part 1
« Doctrine »
Part 2
« Assistance for the expertise »
Outside IRSN
• reference to make available, for example to other technical support organisations (TSO), the approach of expertise and the "know-how" of IRSN in the field of safety assessment for the LUDD facilities
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The Safety Evaluation Guide for Laboratories and Plants

3. Uses - Current guide

First operational feedback from users of the guide as technical aid for safety assessment

- Used primarily by the less experienced person in safety assessment
- Appropriate tool for the assessment of risks of the LUDD facilities. The structure of the Part 2 (questions) is generally appreciated
- Some chapters are particularly appreciated (assessment of general operating rules, assessment of risk relating to handling operations, for example), certainly because of the lack of other documents available to treat the topics
- The "paper" shape of the guide is not ergonomic, and a "computer" shape has to be developed

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The Safety Evaluation Guide for Laboratories and Plants

3. Uses - Current guide

Tool for tutoring — Parts 1, 2 and 3

- Inside IRSN:
  - tutoring sessions organized by the experts units in their area of competencies (fire, criticality, containment...)
  - "expertise school" about to be implemented
- Outside IRSN:
  - tutoring sessions in the European Nuclear Safety Training and Tutoring Institute (ENSTTI), created by four european TSO

Reference — Parts 1 and 2

- Outside IRSN:
  - "criticality" and "fire" risks chapters available on the IRSN's website (with an English version)

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The Safety Evaluation Guide for Laboratories and Plants

4. Conclusion

- The "Safety Evaluation Guide for Laboratories and Plants", which is divided into chapters corresponding to a type of risk or safety file, intends to:
  - express the IRSN's methodology to be followed in performing an assessment (Part 1 "Doctrine")
  - provide the necessary issues to be addressed (Part 2 "Assistance for the expertise")
  - illustrate (on the basis of previous assessments or incidents experience feedback) points considered essential in the assessment (Part 3 "Capitalization of experience")
- Two key roles regarding the IRSN's approach on the topic of the safety assessment of fuel cycle facilities:
  - aid for safety assessment's team of IRSN (but "this guide is only a guide")
  - transmission, inside and outside IRSN, of the knowledge and the know-how of IRSN, as:
    - ✓support for tutoring, e.g. in the frame of "expertise school"
    - ✓support for exchanges with e.g. other TSOs or professionals in the nuclear field
- Already operational, but has to be completed and improved (electronic version)

The guide is a tool for enhancing the safety of LUDD

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SAFETY ASSESSMENT OF HUMAN AND ORGANIZATIONAL FACTORS IN FRENCH FUEL CYCLE FACILITIES

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Sophie BEAUQUIER
IRSN, France

Abstract - According to the French law, each nuclear facility has to provide a safety demonstration every ten years. The assessment of this demonstration supports the decision of the French Safety Authority regarding the authorisation of operating for the ten years to come. In addition, transversal topics, which are linked with safety performance, such as safety management, management of competencies, maintenance's policy are periodically evaluated.

One aspect of these assessments relates to Human and Organizational Factors (HOF) and their contribution to safety.

Our communication will describe the assessment of the HOF-related part, performed by the Institute for Radioprotection and Nuclear Safety Institute (IRSN) the Technical Support Organisation of the French Safety Authority). It will focus on the methodological framework, the tools which are developed and used for assessing the integration of HOF in safety demonstration, and the main difficulties of this kind of assessment. Each situation will be illustrated by concrete examples coming from safety assessments concerning fuel cycle's plants: Areva's plants dedicated to uranium conversion, uranium enrichment, fuel manufacturing, spent fuel reprocessing, treatment facilities and CEA's laboratories dedicated to research and development and to interim spent fuel storage.

The methodological framework for assessing HOF currently implements three main steps which will be precisely described:

- checking that the nuclear plant has made an exhaustive analysis of the risks linked with HOF. Regarding to HOF, the Licensee safety demonstration is based on the description of the main human activities which are considered as hazardous regarding safety. These activities are accomplished with a human contribution and they require a safe realisation.
- assessing the human, organisational and technical barriers that the nuclear plant have planed in order to make the operations safe, to avoid, prevent or detect an incident and to limit its potential consequences. Different topics have especially to be assessed like the working organisation and the structuring of tasks, the training and the competencies, the knowledge and the operational experience of workers and licensees, the management of operating documentation, the control of subcontractors and externalized workers.

- assessing the results of operating experience feedback (OEF) management, in particular human and organisational aspects.

In conclusion, we will highlight the lessons learned from HOF's assessment, in particular regarding the methodological requirement to collect and analysis data, in order to perform relevant assessment.

1. Introduction

The origin of the word "expert" refers to experience or testing. The experience of the expert is based on comprehension of a problematic situation requiring the knowledge of a specialist, which results in a recommendation as requested by a principal (the Safety Authority) so that it can reach a decision (Trepos, 2006). In order to better understand assessment practices, we propose starting from the assessment situations with which the human and organisational factors (HOF) expert is most often confronted in the context of safety evaluations: preparation of a safety review for a facility in operation, preparation for dismantling a facility, preparation of an over-arching theme for an operator: safety management.

For each of these assessment situations, we will present the methods and tools of investigation, the specific features of HOF assessment with regard to the lifecycle phase of the facility (design, commissioning, operation, dismantling), the principal difficulties encountered during the assessment, and the lessons that this type of assessment allows to be highlighted, based on examples of assessment by the IRSN.

2. Evaluating the safety of a nuclear facility from the HOF perspective in the context of a safety review

We discuss in detail below the approach adopted by the IRSN during the safety review of a nuclear facility to evaluate the way in which risks related to human and organisational factors are taken into account.

2.1 General methodology for the evaluation of risks related to human and organisational factors

To evaluate the way in which risks related to human and organisational factors are taken into account, the IRSN examines whether the organisational measures implemented by plant licensees ensure control of the human activities presenting a safety issue. Schematically, one can distinguish provisions in facilities that do not involve intervention of operators (safe geometry of a tank, servomechanism, radiological protection of an armoured door, etc.) and provisions that include one or more operator activities (operation, maintenance, etc). Operations requiring human intervention such that non-performance or poor performance can have safety consequences for the facility are designated as "safety-sensitive activities". Two criteria thus determine the sensitive nature of an activity: the activity must have an impact on safety (potential impact on the facility, the workers or the environment), and the human role in this activity must be preponderant.

The three evaluation plans

A <u>first step</u> in the safety evaluation with regard to human and organisational factors consists of examining the "safety-sensitive activities" that have been identified by the licensee : activities that must be performed by the operators to ensure the safety of the facility, possible human errors (incomplete, late, or unsuitable performance, omission, inadequate performance of activities), and possible consequences.

The exploration of sensitive activities can be conducted by the IRSN according to two angles. The first angle concerns the human activities that are part of the process, defective performance of which can have safety consequences. This can involve for example system alignment, introduction of a chemical reagent, handling of fuel, non-automated sampling of uranium hexafluoride from a container, etc. The second angle involves human activities that are part of a line of defence, failure in which can damage its effectiveness.

By way of example we cite welding a vinyl sleeve, weighing material, performance of periodic maintenance equipment testing, or setting the threshold for an alarm.

The objective of the evaluation at this stage is to ensure that all activities with an immediate or latent impact on safety have undergone in-depth analysis. It is then advisable to examine whether the activities that present a safety issue (sensitive activities) identified by the licensee are:

- pertinent: their identification should rely on a structured process of analysis of the potential failures of the activities and their consequences on safety;
- sufficient: the analysis of the licensee should be exhaustive. The analysis conducted by the IRSN should not therefore lead to identifying sensitive activities that have not been previously identified and analysed by the licensee.

Taking account of the principle of in-depth defence[18], the licensee should define and establish provisions allowing performance of safety-sensitive activities to be made reliable. This is the second step of the evaluation.

In this step, the IRSN seeks to evaluate the measures provided by the licensee to allow the operators to develop the expected activities, that is, how the licensee deals with the various components of work situations, notably with regard to technical systems, training those involved, operating interfaces and technical tools or operating documentation (instructions, operating methods, etc.), and work organisation (definition of assignments, structuring services, sharing tasks and responsibilities, etc.).

This analysis is performed in part on the basis of the safety standards of the licensees and the productions of the organisation (activity assessments for units, audit reports, assessments drawn up at the conclusion of specific activities, organisational memos, instructions, procedures, certain operating documents, etc.). This analysis also necessitates getting closer to the “field” to examine specifically the operational provisions established for control of sensitive activities and their effectiveness. The operation of the organisation is then considered by conducting case studies, targeted activity situations that allow examination of how the organisation established by the licensee behaves and (re)acts faced with specific events in the everyday functioning of the facility.

Case studies are particularly important in the approach of the IRSN, as they reveal the time constraints of the operators, their current practices, the skills used, the technical or socio-organisational features that form an obstacle or contribute to the efficiency of the collective work and performance of certain types of tasks, the communication problems between teams and/or with the hierarchy (deficiency in feedback, failure to take into account “bottom-up” comments and warnings), etc., in the target activity situations. They allow data to be collected on the performance of activities, modality of interaction, features of the context, technical and organisational means employed and the operational uses made of them.

The analysis conducted by the IRSN in the third step is an analysis of organisational processes and support processes for design and management of work situations that provide a framework for performing safety-sensitive activities. Analysis of the processes for designing work situations or defining provisions for control of sensitive activities has the aim of understanding how the licensee organises safety considerations within his facility, how he elaborates and develops the operating requirements and tools necessary for performing activities (man-machine interfaces or equipment), and how he identifies and allows acquisition of the skills required for operators to carry out their activities. These processes of design and organisation

[18] In-depth defences are mechanisms integrated into the system that allow production or propagation of failures to be limited. These defences can be material (alarms, indicators, controls, etc.) or intangible (management, recruitment, training, procedures, supervision, regulation, etc.). Achieving a proper level of safety relies in general on several defences of these two types arranged in series. For an accident to occur, it is necessary that cumulative failures take place in several defences in the system.

of work situations should incorporate both safety issues and the characteristics of the worker in order to specify provisions adapted to the constraints of the activities to be performed.

The systems/processes generally analysed at this stage are the processes for:

- safety management,
- management of working documents,
- training, management of skills and professionalising practices,
- management of subcontracting,
- management of operational experience feedback (OPEX).

Evaluation of the safety management process is described in paragraph 0 of this article.

Below, we specify the items investigated by the IRSN to analyse the process of training and skills management and the professional training programmes implemented by a licensee in a nuclear facility during a periodic safety review (PSR). The IRSN seeks to ensure that the licensee follows the development of the available internal safety skills and anticipates development of the skills required (depending on the appearance/disappearance of new tasks, outsourcing of certain activities, transfer of personnel, etc.). The IRSN is also mindful of the professionalisation practices of personnel (training, simulations, peer training), in particular those involved in performing sensitive activities, and the various forms of skills evaluation implemented by the licensee. Through this evaluation, the IRSN seeks to understand how safety skills are defined and identified by the licensee, what systems are used to identify skills needs (skills mapping, sensitive skills), what the systems are for developing and maintaining skills (prospective management of skills, management of "rare" skills, formalisation of peer training, recourse to subcontracting, etc.), what the systems are for evaluating skills and training and management of professional training programmes, and what is the effectiveness of all these systems.

Another process analysed at this stage is the process for managing operational feedback. To conduct this analysis, the IRSN seeks to find out whether there is a structure dedicated to the process of managing event feedback, including clearly identified personnel who are charged with the functioning of this process and the application of suitable event analysis methods, and allowing identification of the root causes of the events. The IRSN is also interested in the distribution of the results and analyses (is it ensured with regard to the facility personnel likely to be involved?) and evaluation of the actions taken after an incident (have they been analysed? and if they have not proven effective, have they been readjusted?)

Data collection methods

The IRSN performs evaluations on the basis of safety files sent by the licensee to the safety authority and supplementary information obtained from the nuclear plant operator during technical meetings, interviews with various staff in the facilities, or "in situ" observations of the performance of certain operations (case studies).

The interviews conducted during assessment rely on formalised techniques. They may be exploratory (informative, carried out closer to the beginning of assessment) or directed, individual or collective; they may be conducted at the workstation or in an office. This is one of the main tools used during the third step of the analysis.

Observations of work situations aim to study the operators in their actual working environment in order to reveal the determining features of the work activity. They allow in particular difficulties and constraints in performance, practices, interactions, etc. necessary for carrying out a work activity to be identified. They are non-intrusive.

In IRSN instructions, these different methods of data collection are “mixed”, as they provide data complementary to the others, notably during the second step of the analysis. For example, observations made in the working environment then allow operating staff to be questioned effectively.

We note that the methods of data collection through interviews and observations of activities are always negotiated between the IRSN and the licensee, and can result in a protocol being drafted to guide the data collection practices of the IRSN in the nuclear facility during the assessment.

2.2 The critical points of the safety evaluation from the HOF perspective

In this paragraph we draw attention to certain critical points in the evaluation and indicate the measures that currently exist for managing them.

The first involves the question of safety issues: when one considers a particular action that can prove to be deficient and can contribute to an incident-accident scenario, the relation between human action and safety can be established. When a number of actions are considered which support the proper functioning of the lines of defence, but for which it cannot be shown that a single failure contributes to a scenario, the demonstration is much more difficult to establish.

The second involves the question of support data for the evaluation: to conduct an HOF assessment in a rigorous manner, data on work activities are necessary. But access to these data is highly limited, due to deficiency or even absence of data in the files of licensees, transmission of globalized notes, serious constraints on data collection by IRSN specialists (duration, individual interviews, etc.).

The last is that of generalisation: the passage from observation of several particular cases to general conclusions can give rise to debate and challenge by the licensee. Nonetheless, the IRSN believes that conclusions resulting from targeted case studies have “meaning”, and that to this extent (not in the statistical sense) they are representative of sensitive work situations and thus can be generalised.

2.3 Example of a case study of handling operations in a nuclear facility

The objective of this paragraph is to illustrate a point in the methodology outlined in paragraph 0, that of case studies. During the periodic safety review of a nuclear sanitizing and reconditioning treatment facility, the IRSN carried out a targeted case study on handling and docking activity with transport packaging (“flasks”) in the rear area. The context is as follows.

The operation of handling and docking flasks is one of the sensitive activities of the facility due to the undersized flooring in the facility, related to risks of confinement and “external exposure” in the event the load is dropped during the operation.

Therefore, an analysis of handling operations was conducted by the licensee from the perspective of human factors, with the aim of evaluating whether the existing provisions for handling satisfactorily prevent the risk of dropping the load, and of proposing compensatory measures (mainly organisational). In this study, the potential deficiencies identified by the licensee in handling and docking flasks are connected with the risks of confinement and external exposure, and lie in non-observance of the rules for handling and docking, inappropriate actions or risks of errors related to the frequency with which the same operation is performed by the same operator. As for the risks inherent in handling operations, they are controlled by a system of regulatory habilitations, as well as organisational measures such as personnel operating in pairs, operator training and periodic inspection of the lifting units.

During the assessment, the IRSN was able to study how the organisational measures taken by the licensee were implemented during follow-up of a reception, transfer and storage operation for a container in the rear area. The IRSN was then able, once the operation had ended, to individually interview the three persons involved, including a subcontractor, in order to collect additional information.

The IRSN was able for example to observe that the various systems for preventing and limiting risks were in fact operational. With regard to the environment at the workstation, the handling areas were free of all obstacles, facilitating location and storage of apparatus.

With regard to work organisation, there was a hold point before the container was lifted by an authorised person, independently of the operator (a subcontractor with little experience) and the shift supervisor, who were moreover already involved in handling the container.

The IRSN was also able to observe that the line of defence constituted by the skills of the operators and the team, in the sense of cooperation between the participants and behaviours in the working situation, was particularly sensitive in this type of operation. The skills and habilitations of the overhead crane, sling and supervisory operators, as well as observance of the procedures for use of the handling equipment, contribute to control of handling risks. In addition, the working conditions with regard to work environment on the site (area cleared, orderly storage of apparatus) help to control the risk of error and are favourable to carrying out handling operations.

On the other hand, with regard to training for handling operations, the IRSN was able to note that the least experienced participant had not been trained "on the job" by peer training. Yet the IRSN believes that peer training and simulation situations are necessary training methods as they allow personnel to be confronted with the working situation.

In conclusion, the IRSN believes that case studies allow a better knowledge of actual situations and practices implemented in the nuclear facilities of various licensees to be developed.

3. Evaluating the safety of nuclear facilities from the HOF perspective: the case of dismantling a nuclear facility

3.1 Issues and risks related to dismantling operations

Dismantling includes all the technical operations, after the definitive end of operation of a nuclear facility, to remove the hazardous materials present, then to decontaminate and sanitize the equipment and structures, and finally to disassemble and remove them.

There is a substantial safety and radioprotection issue related to supervision of dismantling as well as the methodology for integrating human and organisational factors, both in the approach to designing and organising dismantling situations and in planning, preparation, follow-up and control of the activities.

Risks different from those of operation

One of the first operations conducted at the end of the operating phase of a facility (of fuel cycle plant type) is removal of the radioactive materials present and easily accessible. The quantities of radioactive materials present in the facility to be dismantled will decrease as sanitizing and dismantling operations progress. Therefore, the level of risk from the point of view of nuclear safety is generally lower during the dismantling phase than during that of operation of the facility.

Nonetheless, the first step of dismantling is critical from the point of view of safety and radioprotection. Operations at this stage are carried out very close to the radioactive material and hence they present risks of dissemination and exposure to ionising radiation for those involved. The risks will also be greater because of the works on disassembly of electrical facilities, handling, cutting and deconstruction, etc.

In addition, depending on the state of progress of the works on the worksites, working conditions in certain areas of equipment to be dismantled can become very constrained. These conditions can also make strict application of safety or radioprotection instructions difficult for those involved. Moreover, in this context, there is the risk of discrepancy at some point between what the safety reference authorises and the working methods on the worksites. Management of this risk requires, once again, coordination of all those involved in the dismantling project.

Risks of concurrent activities

The licensee supervises dismantling by relying on a specific organisation in charge of planning, preparation, follow-up and control of the operations, with preservation of safety and radioprotection as major objectives.

One of the characteristics of dismantling projects is that concurrent activity is a recurring situation, either due to the coexistence of areas being dismantled and areas in operation managed by different teams, or due to the operation of several teams in the same location or the use of common utilities (liquids, electricity, etc.) and/or equipment, with different objectives and points of view. Another special feature of dismantling worksites is the frequent shift of schedules, related notably to the occurrence of technical difficulties in access to equipment or the discovery of a configuration that was not as expected.

Under these conditions, coordination between the project, operating and subcontracting teams is essential in taking into consideration these new contextual features and deciding upon measures to be taken to prevent the potential risk of concurrent activity.

Massive recourse to subcontracting

Decontamination, deconstruction or disassembly operations often require special skills and tools that the licensee does not always have, but that service provider companies have developed to the point of making them, for some, a speciality. As a consequence, recourse to subcontracting, both for upstream engineering studies (project ownership assistance) and in carrying out dismantling operations is very widespread among nuclear licensees, sometimes extending to 100% outsourcing of the dismantling works to specialised companies. The subcontractor may also subcontract, according to the specialities. This technique is called “cascade subcontracting”. Consequently, the number of first- and second-rank subcontracting companies on a dismantling project can be high (up to fifty).

3.2 The principal questions in the course of the analysis conducted by the IRSN

The experience gained by the IRSN in analysis of several dismantling projects conducted by various nuclear licensees has revealed a certain number of lessons on assessment of the conditions for taking into account risks related to HOF in this field.

To conduct its analysis, the IRSN first examines the organisational structure (roles and missions, manpower, etc.) established by the licensee to manage dismantling operations. In this connection, the terms for taking into account future work combining operators and the management in specifying dismantling situations, the terms for planning and preparation of activities, the terms for coordination between services and with subcontractors, the terms of cooperation between the various participants in dismantling operations, and the terms for monitoring and checking subcontractor activities will be especially studied. Provisions to allow performance follow-up, monitor performance of works and coordinate activities are lines of defence to which the IRSN is especially attentive.

In addition, the IRSN is interested in the mobilisation of HOF skills throughout the dismantling project, as they allow HOF to be taken into account in the various steps of designing dismantling situations (from the planning phase to that of "pre-job briefing" before the works), and allow HOF failure scenarios and specific provisions for control of these risks to be identified.

In the study phase, the IRSN seeks to find out whether a prior analysis of safety-sensitive work activities, from the perspective of human and organisational factors, has provided assurance that the risks related to the dismantling operations considered are controlled.

On dismantling worksites, the analysis conducted by the IRSN seeks to reveal the strengths and weaknesses of the organisation set up on the worksites by examining the lines of defence actually implemented on the site : the organisation set up, and if possible the adjustments planned and implemented by the teams to handle non-nominal or even disrupted situations, and the tools (and their interfaces) from the perspective of their suitability for the tasks and practices of those involved. The analyses may rely on observations of dismantling worksites and interviews with the participants.

The IRSN also evaluates the set up of an organisation to draw up operational experience feedback (OPEX), notably the features that allow the effectiveness of the organisational measures made to be assessed. According to the IRSN, to be effective the operational experience feedback process requires dedicated resources, a formalised process (known to those involved), collection of information identified in part before the works began, and finally an overview of the features of the feedback collected for use on future worksites.

Finally, control of risks related to recourse to subcontracting will also necessitate the establishment of systems by the licensee to ensure that the service providers have the required skills to supervise and monitor the worksites, and to evaluate the services provided (feedback loop). These systems are also subject to evaluation by the IRSN.

Given the importance of subcontracting in the sector, the IRSN may also examine the technical specifications sent to the service providers to evaluate how HOF requirements related to the conditions for performing activities on the worksites and worksite organisation are integrated.

These requirements may possibly concern the development of special tools easy to use and adapted to the requirements for performance of the tasks. They also involve initial training and periodic training of the participants. Issues of concern are also the practices for providing detailed information to the participants before they are assigned to a work location (detailed dismantling scenario, adequate familiarity of the participants with the safety standard of the facility, verification of the applicability of instructions, etc.), the organisation provided on the worksite and the practices for setting up preliminary meetings for security/safety coordination of risk activities (management of the risk of concurrent activities, OPEX, etc.). At last, daily follow-up of hazards situations by the site foreman has also to be dealt with.

4. Evaluating the safety of nuclear facilities from the HOF perspective: the case of evaluation of the safety management system of a nuclear licensee

In the field of nuclear facilities, the safety management system must in particular observe the regulatory requirements, notably the "quality" decree of the 10th of August 1984. Its principal objective is to ensure the protection of individuals, the public and the environment. The safety management system of nuclear licensees, and the provisions on which it relies, must thus allow the safety requirements to be taken into account and the safety of the facilities to be continually improved. In the framework of its assessment missions on behalf of the Nuclear Safety Authority, the Institute for Radioprotection and Nuclear Safety Institute (IRSN) carries out evaluations of the safety management systems of various nuclear licensees.

After presenting some features of definitions of safety management, we will present the methodology for investigation that is used in evaluation of the safety management system of a nuclear licensee.

4.1 What is a safety management system?

The IAEA and WENRA consider safety management to be a “framework including well-defined requirements specifying the responsibilities and activities necessary to ensure safety and to observe the legal and regulatory requirements as well as those of the operating company”. This framework has the function of determining the principal responsibilities and activities required to ensure safety. The functions of this system are presented as: defining safety organisation and objectives; planning; implementing; supporting; controlling; and modifying. The licensee must establish and implement an applicable safety management system as of the design phase and during all the subsequent phases of existence of the facility, allowing in particular control of activities involving safety to be ensured in observance of the defined requirements. The principal objective of any safety management system is defined as “ensuring that safety is in the forefront of the objectives of the organisation”.

A safety management system refers to all the policies, objectives, roles, responsibilities, codes, standards, communications, procedures, processes, tools, data and documents that contribute to risk management in an organisation. This system is not simply documentation, a set of procedures and rules, but involves actual execution of the processes, procedures and practices. It reflects the safety culture of the organisation (Ross, 2004).

The precondition for implementation of a management system is to identify its essential functions. These functions are generally grouped into six main categories: the functions of management, orientation and arbitration; organisation and coordination; activities and involvement; regulation and negotiation; control and evaluation; adaptation and improvement.

We deduce from all these definitions that a safety management system can be defined as a set of policies, strategies, practices, procedures, roles and functions, people and resources related to safety that interact in an organised way with the aim of ensuring safety and reducing incidents and accidents (Fernandez-Muniz et al., 2007). To be effective and efficient, this system must be incorporated into the daily life of the organisation and promote safety objectives among all personnel. It relies in essence on a safety policy that reflects the organisational principles and values of the licensee with regard to safety, activities aiming to promote safety among those involved, to inform them of the risks to which they are exposed, to give them a share and to motivate them,training systems ensuring that the necessary skills to provide the expected level of safety are present in the organisation and are used by those involved, a feedback system, control actions to check and evaluate the implementation of the safety policy at the various levels of the organisation, and corrective actions.

Starting from these defining features, how can we carry out a relevant evaluation of a safety management system?

4.2 Methodological aspects

To analyse the application of the safety management system on a day-to-day basis, it is necessary to rely on collection and analysis of operational practices. This approach has an immediate methodological impact: it assumes access to the practices at the various levels of the organisation.

The efficiency of a safety management system should be evaluated in this context on the basis of reliable information on the contribution of the safety policy defined by the licensee to the resolution of difficulties and problems that arise for those involved, and the efficient allocation of the resources necessary to do so.

Such an evaluation should take into account the productions of the management system: the operating practices, decisions, feedback data, etc.

In-depth qualitative empirical investigations combining document analysis, questionnaires, interviews and in-situ observations constitute the principal steps in this evaluation.

One of the principal drawbacks of any safety management system is the difficulty in understanding its performance (Foucault and Kopp, 2005). This consequently necessitates, in particular, diversifying the evaluation criteria.

The criteria that seem to us the most important to evaluate in this context are the following: efficiency (matching the means to the objectives and aims); pertinence (matching the results to the expectations of the recipients: the Safety Authorities, industrial customers, staff, and the public); efficacy (matching the results to the objectives); balance (notably between the formal and operational dimensions of the system); improvement in and reliance on feedback; and dynamics (the adaptation of the management system to developments, internal and external).

In this multicriteria evaluation, it is mainly the effectiveness of the management system and the learning dynamic associated with it that are questioned: in what way does this system, intended to settle the (safety) problems with which those involved are confronted, constitute a resource for acting and aiding them in making certain that safety takes precedence over any other priority? Conversely, to what degree do the practices implemented in the organisations studied participate in the development of the safety management system and the learning dynamics of the organisation as a whole?

Moreover, the formal level (objectives, rules, procedures), organisational level (participants, systems, structures, means), and the operational level (practices, day-to-day managerial supervision) of the safety management system should be simultaneously examined. This should be done taking into account in the analysis the existing processes and systems (doctrines, rules, references, charters, training systems, etc.), those in charge of the managerial animation (their resources, their legitimacy, their strategies, etc.) and the practices implemented, in particular by the first-line operators (impacts of the management system on practices, variation of these practices over time, possible deviations). It is thus a matter of using, as Vaughan (1996) has done in his analysis of the Challenger shuttle incident, three levels of analysis: the environment of the organisation, the organisation itself, and the point of view of those involved. The approaches of political sciences, management sciences, sociology and social psychology are all especially applicable in this context.

The multilevel approach assumes that the interviews conducted at the organisational level involve the entire hierarchy, from the top management that defines and orients the safety policy to the first-line operator who implements the operating practices to promote safety.

4.3 Illustration by means of a case

The evaluation conducted was principally thematic, through analysis of the provisions implemented involving safety policy at the national and local level (policy documents, objectives, methods of integration into processes, communication and supervisory tools, reporting systems, etc.); the deployment and implementation of this policy in all the facilities; the degree of assimilation and use of the associated tools and management systems; the lessons drawn from feedback; the evaluation criteria used to evaluate the performance of the safety management system; and the activities aiming to improve consideration of safety and to anticipate the impacts of organisational and strategic developments on consideration of safety issues.

At the end of this evaluation, a certain heterogeneity was noted in particular in the operational implementation of the safety management system of this licensee. The day-to-day management safety practices and tools are relatively variable from one facility to another. The contacts encountered attribute this heterogeneity mostly to the diversity in the facilities of the licensee. This diversity in practices is also related to the autonomy and leeway available to the facilities in implementing the safety policy and establishing managing practices for safety management. In the present management system of this licensee, the management level defines national directives and recommendations, with the operating methods for application most often left to the initiative of the facilities. The result is a diversity in the level of implementation of national recommendations or directives. This autonomy presents advantages, put forth by the contacts encountered (consideration of local characteristics, flexibility in management), but it also presents drawbacks and risks: lack of guidance, failure to learn from and coordinate local practices, national standards and references unsuited to the actual conditions of performance of activities, failure to detect discrepancies in local practices, failure to take account of local objectives and the associated risks.

The principal difficulty of such a system lies in managing to specify a common orientation without driving the facilities apart on the grounds of their specific characteristics. One way to avoid this pitfall is to rely on the existing local practices and capitalise on them.

Furthermore, it appeared that the evaluation of the overall performance of the safety system management of this licensee relied on a reduced number of criteria (industrial performance indicators or quality indicators) and could be supplemented by additional criteria, such as those previously cited.

5. Conclusion

In the field of human and organisational factors, the evaluation methods of the IRSN with regard to nuclear licensees who principally operate laboratories and factories are varied. They are at the same time adapted to the characteristics of these facilities, which are mostly oriented toward research (laboratories) or production (factories). They are also adapted to the lifecycle phase of the facility (design, commissioning, operation, dismantling), examples of which we have discussed.

With regard to evaluation of an operating facility during its periodic safety review, we have seen that it involves principally a methodology distinguishing the microscopic (analysis of sensitive human activities) and macroscopic (analysis of the organisational processes) levels, and evaluating the safety requirements applicable. The whole difficulty of such an evaluation lies in managing to connect these two levels and analyse the way in which the work organisation, management of skills and manpower, technical systems and working environment contribute to increasing the reliability of the human activities carried out in the facilities and so improving safety.

With regard to the dismantling phase of a facility, we have seen that it presents a set of specific features important in the design and performance of the activities to be conducted. Analysis of these specific features and their concurrence allows certain issues related to the impact of the operational context on the activity of the participants and, even more, on safety and radioprotection to be identified. Moreover, the initial collection of feedback from dismantling operations currently conducted with several nuclear licensees shows that consideration of the risks related to human and organisational factors suffers from certain deficiencies, notably at the level of the phase of studying and contracting for activities subcontracted by the plant operator to service provider companies.

It would be advisable notably, in the light of specific studies and feedback from past operations, to improve comprehension of this in order to improve organisational provisions ranging from the design of dismantling situations to planning and preparation of activities, to ensure control of risks during upcoming dismantling activities.

With regard to evaluation of the safety management system of a nuclear licensee, we have seen that consideration of, and systems involving, management provisions seem however still in their infancy in human and social sciences. We have also described that on the part of the licensees, the importance of anticipation and consideration of the effects of safety management systems on human and organisational reliability is not yet acknowledged at the proper level. This consideration appears all the more important in the context of an increase in production pressures for some licensees. Thus, a recent report by the supervisory authority for nuclear licensees (2008) stresses that it is reasonable to examine more profoundly the managerial and organisational aspects, notably with regard to the debate on the pressure exerted on managers. The IRSN indicates, in its report on the status of the French nuclear power system (2008), that the increase in production pressures should be counterbalanced by managerial and organisational measures aiming to maintain a high level of safety, in the absence of which this increase could lead to deviations likely to favour the occurrence of an accident. It appears necessary therefore that licensees reinforce even more the managerial and organisational lines of defence and that they clarify the way in which safety requirements are jointly managed with all their other requirements.

Bibliographie

Autorité de Sûreté Nucléaire (2007), Convention sur la sûreté nucléaire : Quatrième rapport national établi en vue de la réunion d’examen de 2008, Juillet, 205 p,

Autorité de Sûreté Nucléaire (novembre 2008), Facteurs organisationnels et humains et démantèlement des installations nucléaires, in *Contrôle.*, n° 181,

http://www.asn.fr/index.php/content/download/15126/99581/Quatrieme_rapport.pdf

Institut de Radioprotection et de Sûreté Nucléaire, Le point de vue de l’IRSN sur la sûreté et la radioprotection du parc électronucléaire français en 2008, Rapport DSR n°316,

http://www.irsn.fr/FR/expertise/rapports_expertise/Documents/surete/IRSN_rapport_surete_du_parc_2008.pdf

Fernandez-Muniz, B., Montes-Peon, J., Vazquez-Ordas, C. (2007), Safety management system: Development and validation of a multidimensional scale, *Journal of Loss Prevention in the Process Industries* n°20, p. 52–68.

Foucault M., Kopp P. (2005), La performance dans les armées : définitions et perceptions, *Rapport pour le Centre d’Etudes en Sciences Sociales de la Défense.*

Institut de Radioprotection et de Sûreté Nucléaire, Le point de vue de l’IRSN sur la sûreté et la radioprotection du parc électronucléaire français en 2008, Rapport DSR n°316,

http://www.irsn.fr/FR/expertise/rapports_expertise/Documents/surete/IRSN_rapport_surete_du_parc_2008.pdf

Ross, A. (2004), What measures best indicate the success of a rail safety management system, *International Rail Safety Conference*, Perth.

Thuderoz, C. (2006), *Histoire et sociologie du management*, Presses polytechniques et universitaires romandes, Lausanne, 416 p.

Trépos, J.-Y. (2006), Savoirs professionnels et situations d'expertise, *Savoir, Travail, Société*, vol. 4, n°2.

Vaughan, D. (1996), The Challenger Launch Decision: Risky Technology, Culture and Deviance at NASA, University Chicago Press, Chicago, 600 p.

Safety assessment of human and organizational factors in french fuel cycle facilities

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1. Decenial HOF safety evaluation for fuel cycle facilities

General methodology : the 3 levels of assessment

- First step
  - Assessment of the safety-sensitive activities analysis from the licensee
- Second step
  - Assessment of work situations and implemented organisational provisions to carry out safety sensitive activities
- Third step
  - Assessment of the organisational process and support process for safety sensitive activities design and management

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The 3 levels of assessment lead by IRSN

Safety sensitive activities and work situations reveal the efficiency of the organisational and managerial provisions

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First step : IRSN assessment of the safety-sensitive activities analysis from the licensee

- What are safety sensitive activities?
  - Manual activities
  - Complex Organisational and intellectual activities
  - Non performance and poor performance has direct impact on nuclear safety
- What to evaluate for safety sensitive activities?
  - Possible errors or inadequate performance
  - Possible safety consequences
- The objectives for IRSN at this stage
  - The method of the licensee to analyse potential failures and safety consequences should rely on a structured process
  - The analysis of the licensee should be exhaustive
- Assessment of the safety-sensitive activities analysis report from the operator in combination with an assessment of work in the field

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First step : IRSN assessment of the safety-sensitive activities analysis from the licensee

- **What are safety sensitive activities?**
  - Manual activities
  - Complex Organisational and intellectual activities
  - Non performance and poor performance has direct impact on nuclear safety
- **What to evaluate for safety sensitive activities?**
  - Possible errors or inadequate performance
  - Possible safety consequences
- **The objectives for IRSN at this stage**
  - The method of the licensee to analyse potential failures and safety consequences should rely on a structured process
  - The analysis of the licensee should be exhaustive
- **Assessment of the safety-sensitive activities analysis report from the operator in combination with an assessment of work in the field**

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Second step : IRSN Assessment of the work situations implemented by the operator

- How does the operator deals with the various components of the work situation?
  - Technical systems
  - Training
  - Operating interfaces
  - Operating documentation (instructions, operating methods,etc.)
  - Work organisation (definition of assignments, structuring services, sharing tasks and responsabilities, etc.)
- The analysis performed by IRSN is based on
  - Safety standards of the licensee
  - Description and results in practice of the organisation
- Case studies
  - Targeted activity situations permitting to study how the established organisation behaves and reacts according to specific events in the every day functioning of the facility.

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Third step : IRSN Assessment of the organisational and support processes implemented by the licensee

- What is the aim of this analysis?
  - How the licensee organises safety consideration within the facility
  - How the licensee specifies the safety requirements and tools necessary for performing activities
  - How the licensee identifies and allows acquisition of the skills required to carry out the sensitive activities
- Processes generally analysed at this stage :
  - Safety management
  - Management of skills and profesionalising practices
  - Management of working documents
  - Management of subcontracting
  - Management of operationnal experience feed back

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IRSN data collection methods

- A combination of different methods to collect information
  - Safety files sent by the licensee
  - Technical meetings with representatives of the nuclear plant operator
  - Interviews with different staff in the facilities
  - « In situ » observations of the performance of a few operations (study cases)
- Methods of data collection through interviews and observations are negociated between the IRSBN and the licensee.
  - A protocol is drafted
  - Unfortunately! access to data on work activity is limited by the licensee : it's one of the critical point of the safety evaluation from a HOF perspective

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2. Evaluating safety from a HOF perspective : the case of dismantling a nuclear facility

Context of dismantling projects

- Important increase of dismantling operations in the coming ten years
- 50 to 60 nuclear reactors on the 157 in total in E.U are to be dismantled before 2025
- Long (several decades) and complex projects (scenarios to define and multiple actors)
- Massive recourse to subcontracting

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Issues and risks related to dismantling operations

- Risks change of nature
  - The nuclear plant is subjected to successive transformations during its dismantling, due partly to the evacuation of the remaining fuel and radioactive material
  - Dismantling operations take place very close to the radioactive material
  - A lot of manual activities : disassembly of electrical facilities, handling, cutting, etc.
  - Constrained working conditions
  - Risks management requires strong coordination of all those involved
- Major nuclear risks
  - Dissemination
  - Fire and explosion
  - Exposure to ionising radiation
- The HOF stakes
  - Conduct of the operations
  - Control of concurrent activities
  - Supervision of subcontracted activites

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IRSN major issues for evaluation

- Definition of the organisation
  - Organisational structure to manage the dismantling operations
  - Multiple interfaces production / maintenance / dismantling project
- Conduct of the project
  - Planification and preparation of the activities
  - Coordination between units and in combination with the sub-contractors
  - Organisation of the sub-contracted activities's supervision
- Up stream study phase
  - Identification of HOF failures scenarios and specific provisions for control of these risks
  - Identification of required skills for the works owner (to supervise and monitor) and project manager (to carry out the activities)
  - Set up of the operational experience feed-back
- Work in progress phase
  - Make sure the required skills are available
  - Control and supervision of sub-contracted activities
  - Adjustments to the to handle non-nominal situations or disrupted situations
  - Evaluation - Feed-back loop

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3. Evaluating the safety management system of a nuclear licensee : a HOF point of view

What is a safety management system?

What is safety management ?

- *IAEA, WENRA* : A framework including well-defined requirements specifying the responsabilities and the activities necessary
  - To ensure safety
  - To observe the legal and regulatory requirements as well as those of the operating compagny
- What is the objective of any safety management system?
  - Ensuring that safety is the main objectives of the organisation
  - The safety principle should be part of the global management system

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What are the essential functions of a safety management system?

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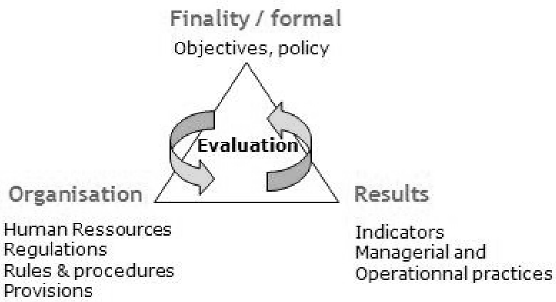
What is analysing a safety management system?



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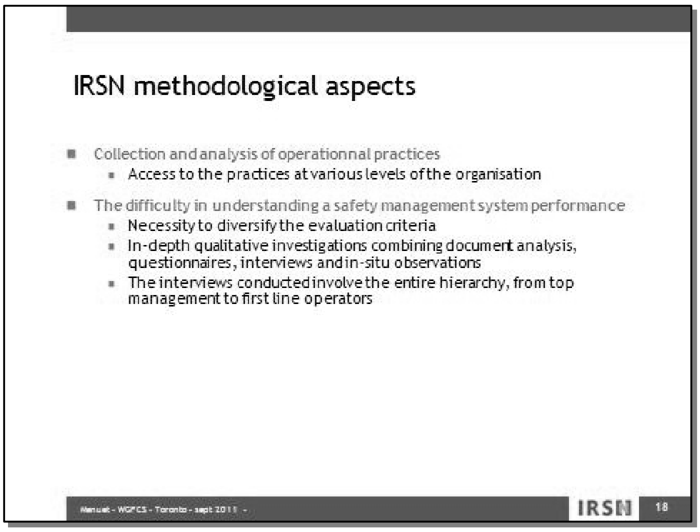
IRSN methodological aspects
Collection and analysis of operationnal practices
Access to the practices at various levels of the organisation
The difficulty in understanding a safety management system performance
Necessity to diversify the evaluation criteria
In-depth qualitative investigations combining document analysis, questionnaires, interviews and in-situ observations
The interviews conducted involve the entire hierarchy, from top management to first line operators
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IRSN
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BOW TIE METHODOLOGY: A TOOL TO ENHANCE THE VISIBILITY AND UNDERSTANDING OF NUCLEAR SAFETY CASES

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Office for Nuclear Regulation, UK

Abstract – There is much common ground between the nuclear industry and other major hazard industries such as those subject to the Seveso II regulations, e.g. oil, gas & chemicals.

They are all subject to legal requirements to identify and control hazards, and to demonstrate that all necessary measures have been taken to minimise risks posed by the site with regard to people and the environment.

This places a requirement on the Operators of major hazard installations, whether nuclear or conventional, to understand and identify the hazards of their operations, the initiating events, the consequences, the prevention and mitigation measures.

However, in the UK, nuclear and “Seveso” type facilities seem to adopt a different approach to the presentation of their safety cases.

Given the magnitude of the hazards, safety cases developed for nuclear fuel cycle facilities are rigorous, detailed and complex, which can have the effect of reducing the visibility of the key hazards and corresponding protective measures.

In contrast, on installations in the oil & gas and chemical industries, a real attempt has been made over recent years to improve the visibility and accessibility of the safety case to all operating personnel, through the use of visual aids / diagrams.

In particular, many Operators are choosing to use “bow tie methodology”, in which very simple overview diagrams are produced to illustrate, in a form understandable by all:

- what the key hazards are;
- the initiating events;
- the consequences of an incident;
- the barriers or “Layers of Protection” which prevent an initiating event from developing into an incident;
- the barriers or “Layers of Defence” which mitigate the consequences of an incident, i.e. which prevent the incident from escalating into major consequences.

The bow tie method is one of a number of methodologies that can be used to make safety cases more accessible. It is used in this paper to illustrate ways to improve the visibility and accessibility of complex

safety cases. The bow tie method is a purely qualitative technique, which could be successfully introduced (or similar methodologies) to the nuclear industry as an additional tool to improve the visibility and understanding of the safety case, and thus complement (not substitute) the more rigorous safety analysis techniques which are the norm in this industry.

By making the diagrams readily accessible in the control room, the operators of nuclear facilities could further improve their understanding of the safety significance of their role in preventing major accidents and mitigating consequences.

1. Introduction

The nuclear industry is by no means unique in its legal obligation to demonstrate that hazards associated with its operations are properly managed.

In Europe for example, various high profile accidents in the chemical industry have resulted in legislation (Council Directive 96/82/EC [9], also known as the Seveso II Directive) aimed at the prevention of major accident hazards, but also aimed at limiting the consequences of such accidents to people and the environment.

One of the objectives of the Directive is to prevent or reduce accidents caused by management factors, or safety management systems, which have proved to be a significant cause of accidents. [9]

The Directive, implemented via national legislation, places a general duty on the operators of hazardous installations to prepare a safety report demonstrating that “major accident hazards have been identified, and that the necessary measures have been taken to prevent such accidents and to limit their consequences for persons and the environment.” [1] “The complexity of the demonstration presented in the safety report is expected to increase with the complexity of the facility concerned.” [1]

Similarly, the nuclear industry operates in an extremely highly regulated environment, where national legislation requires the operators of nuclear facilities to ensure that “all practical efforts [are] made to prevent and mitigate nuclear or radiation accidents.” [12] “The fundamental safety objective is to protect people and the environment from [the] harmful effects of ionising radiation.” [12]

Therefore, in general terms, national and international legislation requires the operators of major hazard installations (chemicals, oil & gas, nuclear, etc …) to understand and identify:

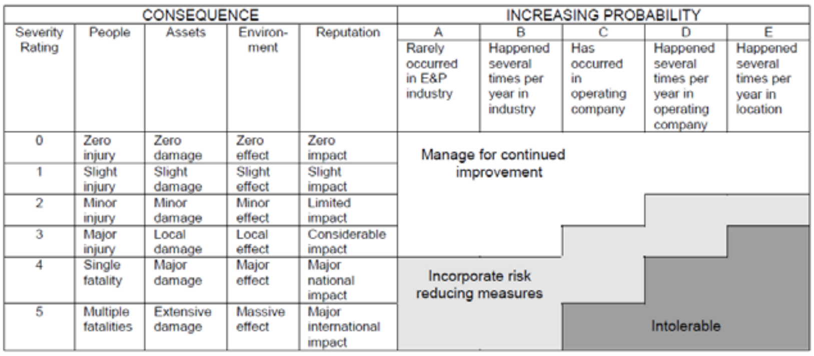
- The hazards of their operations;
- The initiating events which could result in an accident;
- The consequences of such accidents;
- The prevention measures which they have in place to prevent an initiating event developing into an accident;
- The protective measures which are in place to limit the effects of an accident.

This report considers to what extent methodologies such as bow tie could be applied to the nuclear industry, in order to enhance the communication, visibility and understanding of the safety case. Bow tie methodology has been selected for examination in this report over other similar techniques, due to its increasing popularity and widespread use throughout the world by the operators of major hazard installations.

2. Risk Analysis and Safety Critical Events

There are numerous ways in which risk analyses may be conducted, from simple qualitative approaches to a fully quantitative risk analysis. In order to select the most appropriate technique, UK guidance to the "Major Hazard Industries" indicates that "the depth and sophistication of the analysis should be proportionate to the hazard and risk present." [10]

UK guidance [13] also provides a useful definition of safety critical events, i.e.: "Safety critical events are those that dominate the contribution to risk, so they should be identified by your risk analysis."



CONSEQUENCE					INCREASING PROBABILITY				
Severity Rating	People	Assets	Environ-ment	Reputation	A Rarely occurred in E&P industry	B Happened several times per year in industry	C Has occurred in operating company	D Happened several times per year in operating company	E Happened several times per year in location
0	Zero injury	Zero damage	Zero effect	Zero impact					
1	Slight injury	Slight damage	Slight effect	Slight impact					
2	Minor injury	Minor damage	Minor effect	Limited impact					
3	Major injury	Local damage	Local effect	Considerable impact					
4	Single fatality	Major damage	Major effect	Major national impact					
5	Multiple fatalities	Extensive damage	Massive effect	Major international impact					

Figure 1 – risk matrix

The risk matrix [fig 1] is an easy and effective way to qualitatively assess and represent the risks to people, assets, environment and reputation. [2]

A qualitative assessment is made of the probability and the severity of the consequences of each event without taking credit for protective measures. A more detailed analysis is then carried out for all safety critical events.

3. Prevention and protective measures

The principle of defence in depth is described in detail in the IAEA's Safety Standard NS-R-1 [14], the UK nuclear regulator's safety assessment principles [11], and research papers commissioned by the UK regulator for the Major Hazard Industries [8]. The various protective measures are illustrated as concentric "lines of defence" (LODs) or "layers of protection" (LOPs) around the basic process. [8, 11, 14, fig 2] The function of the inner layers is to prevent the process from deviating from its normal design

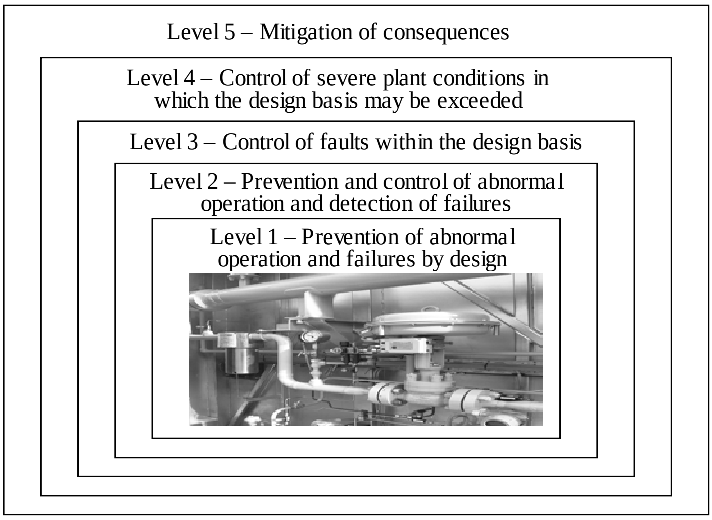


Figure 2 – The 5 general levels of defence in depth

parameters. The severity of the deviation from normal operation increases as the internal layers of protection are progressively breached and challenges are placed on outer layers.

The "Bow Tie" model [fig 3] provides an effective illustration of the different functions of the layers of protection, by placing them into two separate categories: [7]

- those which provide primary barriers to prevent hazards from arising;
- those which mitigate the consequences of hazards which arise if prevention measures identified in (a) are breached.

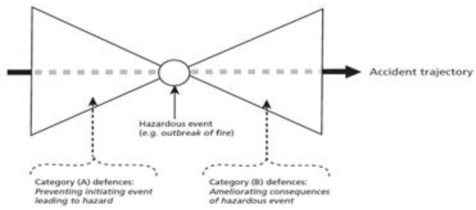


Figure 3 – the bow tie model

The name given to the resulting diagram [fig 3] comes from its very obvious resemblance to the bow tie, i.e. the short knot tie worn by gentlemen on formal occasions.

4. Safety cases in the nuclear industry

The principle of proportionality, described in previous paragraphs for major hazard installations, also applies to nuclear facilities, i.e. "*the depth and sophistication of the analysis should be proportionate to the hazard and risk present.*" [1] Nuclear installations fall into the "high hazard" category, and should therefore use "the most developed and sophisticated techniques." [11]

In accordance with this principle, the safety cases developed for nuclear facilities are detailed and can appear complex to non-specialists.

Safety inspections of nuclear fuel cycle facilities routinely identify that the necessary rigour and depth of analysis of nuclear safety cases can reduce the clarity with which key hazards and safety measures are communicated to the operators of those facilities. There is scope for improving plant operators' understanding of the safety significance of the operations and checks which they routinely carry out.

The typical hierarchy [fig 4] of safety case documents on nuclear fuel cycle facilities in the UK shows three generic categories of documents, i.e.:

- At the highest level, the detailed safety case documents which are normally reserved for safety case specialists and plant management.
- An intermediate level, which translates the detailed safety case requirements into rules and instructions for the controlled operation and management of the

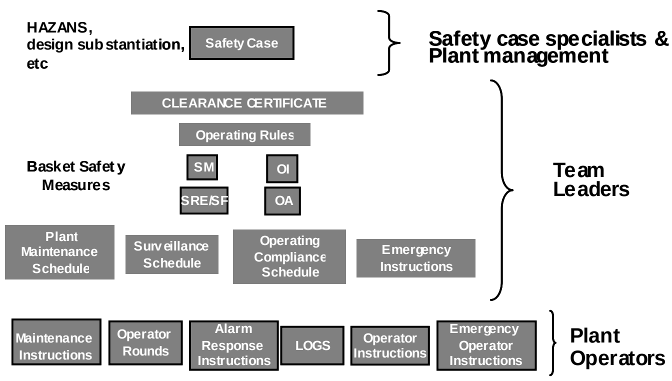


Figure 4 - Typical hierarchy of safety related documents

plant. This level of documentation is typically accessed by team leaders.

- Finally, detailed instructions are provided for plant operators and maintainers, to ensure that the nuclear facility remains within a tightly controlled operating envelope, and that nuclear emergencies are adequately managed.

This hierarchy of safety documentation has evolved over a number of years, and has been successful in managing and controlling hazards on nuclear fuel cycle facilities.

However, this approach leads to limited visibility of the overall safety case, and further efforts need to be made by nuclear licensees to ensure plant operators understand why it is important to perform the specific tasks which they routinely carry out.

What seems to be missing is a simple overview diagram, which illustrates in simple terms:

- What the key hazards are;
- What could initiate those hazards (the initiating events);
- The consequences of an incident, once it has occurred;
- The barriers or "Layers of Protection" which prevent an initiating event from developing into an incident;
- The barriers or "Layers of Defence" which mitigate the consequences of an incident, i.e. which prevent the incident from escalating into major consequences.

5. Bow tie methodology

Bow tie methodology has been increasingly used throughout different areas of business and industry since the early 1980s, principally to provide corporate assurance that major risks are adequately identified and controlled. [4]

Bow tie diagrams [fig 5] combine into a single diagram the possible causes and consequences of a potential accident. The left hand side represents the fault tree, which links the accident to possible causes, whilst the right hand side represents the event tree, linking the accident with possible consequences. [6]

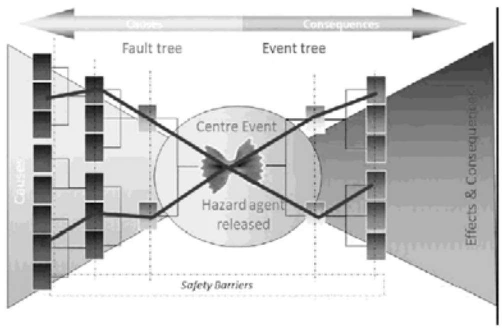


Figure 5 - The bow tie diagram

The development of the diagram [fig 6] starts with the identification of all potential threats which could lead to an event, i.e. the realisation of the hazard. The next step is to identify, for each threat, the barriers which prevent those threats from developing into the event. The right hand side of the diagram is then developed in a similar manner. All consequences are identified, and, for each one, the barriers which prevent the escalation of the event are identified and included. [5]

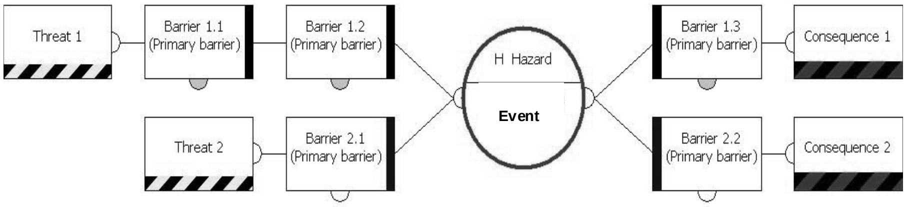


Figure 6 – Construction of bow tie diagram

This systematic step by step approach results in the construction of a complete bow tie diagram, which includes all protective measures or barriers.

6. Types of barriers

Barriers are variously classified into different categories:

- Primary and secondary barriers [5]
  Primary barriers prevent or reduce the development of threats into accidents, and mitigate or prevent the accident from escalating into major consequences. Secondary barriers prevent or reduce the erosion (referred to as "decay") of the effectiveness of the primary barriers. Primary barriers are the primary means of preventing and controlling hazards, whilst secondary barriers are put in place to ensure that the effectiveness of the primary barriers is maintained.

Typical examples of primary and secondary barriers are listed below: [5]

- Primary barriers:
  - ~ Active barriers: shut-down valves, deluge system;
  - ~ Passive barriers: fire wall, blast wall, containment, separation;
  - ~ Control barriers: fire and gas detection, alarms;
  - ~ Organisational / procedural barriers: inspection and monitoring;
  - ~ Human / operator barriers: process control operator.

- Secondary barriers:
  - ~ Human / operator barriers: supervision;
  - ~ Procedural barriers: design reviews, operational reviews, competenceassurance;

- Passive, active and behavioural barriers [6].
  Passive barriers prevent or reduce the transmission of threats, or mitigate the consequences of accidents without any requirement for automatic or human intervention. Retention bunds are a typical example of passive barriers.

  Active barriers require human intervention or are activated automatically. An automatic shut down system is a typical example of active barriers.
  Finally, behavioural barriers require human judgement and intervention. A typical example would be the requirement for a plant operator to base his decision to shut down a piece of equipment on instrument readings.

- Full or partial barriers [8]

A full barrier is designed to completely stop a cause from developing into a consequence, unless it fails to operate. A partial barrier offers a degree of protection, but does not fully prevent a cause from generating a consequence. An alarm is a typical example of partial barrier.

7. Barrier decay – barrier failure modes

There are many reasons why the integrity of protective barriers may be lower than originally intended, such as inadequate design or maintenance, poor procedures and communication, insufficient training, etc… [5]

In bow tie methodology, barrier decay modes, i.e. the underlying causes of barrier failure, are identified and included on the diagram. [fig 7] The secondary barriers, i.e. those which prevent the degradation or decay of the primary barriers, are also identified and included on the same diagram. As illustrated [fig 7], there could be several possible decay modes for each primary barrier, which need to be protected by corresponding secondary barriers. [5]

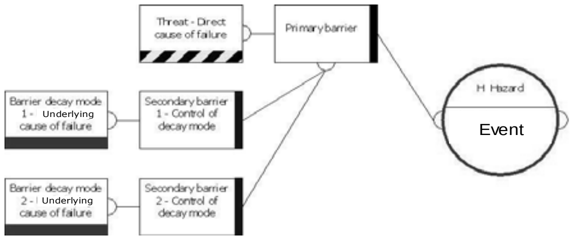


Figure 7 – Barrier decay modes

This approach brings together in a single diagram all relevant protective barriers, their modes of degradation, and all measures taken (secondary barriers) to prevent the decay of those barriers. [5]

Reference 6 proposes a hierarchy for the implementation of barriers, and suggests useful questions for the investigation of incidents and accidents, i.e.:

- Provide barrier (question: “Was the barrier provided?”);
- Use barrier (question: “Was the provided barrier used?”);
- Maintain barrier (question: “Was the provided barrier maintained?”);
- Monitor barrier (question: “Was the state of the barrier monitored?”);

“For an operating plant, the key controls are in making sure that the installed equipment keeps working properly” [5], and “that barriers are operational at all times.” [3]

Reference 5 proposes a “barrier rating scheme” to rate the effectiveness and complexity of barriers.

8. Compatibility of the bow tie method with fault analysis techniques commonly used in the nuclear industry

This section examines the compatibility of bow tie methodology with fault analysis techniques routinely used in the nuclear industry, i.e. Design Basis Analysis (DBA), Probabilistic Safety Analysis (PSA) and Severe Accident Analysis (SAA).

“Design basis analysis is a robust demonstration of the fault tolerance of the facility, and of the effectiveness of its safety measures.” [11] DBA uses appropriate conservatism to assess the suitability and sufficiency of the safety measures. [11] Risks are not quantified in this approach. However, “DBA should provide a clear auditable link between initiating faults, fault sequences and safety measures.” [11] The bow tie method, which is also qualitative, could act as a useful complement (but not a substitute) to the more rigorous DBA approach, by providing a diagrammatic illustration of fault progression and protective measures.

Probabilistic Safety Analysis is a quantitative approach, which addresses the overall risk presented by facilities. It “enables complex interactions to be identified and examined, and provides a logical basis for identifying any relative weaknesses.” [11] This aspect of safety analysis is recognised as one of the principal limitations of the bow tie technique. (see paragraph 10)

Severe Accident Analysis addresses severe but very unlikely faults. Indeed, “DBA should ensure that severe accidents are highly unlikely. Nevertheless, the principle of defence in depth requires that fault sequences leading to severe accidents are analysed and provision made to address their consequences.” [11] Severe Accident Analysis is designed to identify protective measures required for severe accident scenarios. Also, “severe accident analysis may also identify that providing further plant and equipment for accident management is reasonably practicable.” [11] As outlined in para 0, the bow tie method could be used as a purely qualitative diagrammatic aid to help identify additional measures of protection.

9. The benefits of bow tie diagrams

a. Communication: visualisation and improved understanding of major hazards

“A picture paints a thousand words.” Bow tie diagrams are an extremely user-friendly tool for providing a readily understandable visualisation of the main hazards, their causes and consequences, and the prevention and protective measures. Bow ties could be introduced in addition to the more rigorous safety analysis techniques currently used in the nuclear industry to improve visibility and understanding. Ideally, the bow tie diagrams should be displayed on posters in the control room to allow plant personnel at all levels to understand the importance of their role in preventing accidents, and where their work fits in the overall picture.

“Understanding why and how something has to be done on the barrier facilitates appreciation of the barrier function and its failure.” [5]

Removing a barrier or a set of barriers for the purpose of maintenance immediately highlights the resulting weakening of the system, and the importance of making a risk informed decision whether to shut down the facility and / or to put in place additional protection. [5]

b. Workforce involvement and ownership

The benefits of bow tie are from applying the approach and involving the workforce in the development of the diagrams. [3] Operational personnel typically have limited knowledge of the safety case, but a high level of experience of its practical implementation and of any related incidents. [3]

When people feel involved and action is taken based on what they say, they tend to “buy in” to the process and take ownership. [3]

c. Focus on Critical Systems

Bow ties are an extremely effective tool for keeping sight of the big picture, i.e. top level safety critical events and corresponding control measures. They clearly illustrate, to plant personnel and stakeholders such as regulators, how hazards are managed and controlled. “What is important is that critical tasks have been identified, and that people know they need to do them and why.” [4]

Bow ties pull together into one single diagram not only the hardware, but also the actions and controls conducted by real people on the plant. A visual examination of the diagrams provides a useful overview of the protective measures, and allows users to qualitatively identify weak areas where there appear to be gaps or insufficient protection.

d. Continuous Improvement and demonstration of ALARP

The bow tie approach fits well with the legal requirement to demonstrate that all reasonably practicable controls have been identified and implemented in order to ensure that risks are As Low As Reasonably Practicable (ALARP). [5]

However, it must be recognised that bow tie is a purely qualitative method, which cannot be used to quantitatively estimate the improvement afforded by an additional Independent Protective Layer.

The following questions must always be asked in order to achieve sustained and continuous improvements in safety standards: “What additional, practical controls can we implement?” or “Is there anything more we can reasonably do?” [3]

10. Limitations of bow tie diagrams

Bow tie diagrams provide a clear qualitative graphical representation of system failure logic and the role of the various layers of protection (barriers) in place. [8] “However, the method avoids any explicit calculation of risk. Therefore, barrier diagrams could be used in circumstances where a qualitative approach was justified, but would not be appropriate in situations where the use of a semi-quantitative or quantitative approach was demanded.” [8]

Bow tie methodology also has a tendency to over-simplify the real underlying safety challenges of complex facilities. For example, it tends to hide dependencies and is not able to adequately model complex inter-relationships between various risk controls.

Also, in common with most other hazard management techniques, the successful application of the bow tie approach depends on the experience of personnel involved.

11. Conclusions

In accordance with the “high hazard” categorisation of the nuclear industry, safety cases developed for nuclear facilities are rigorous and detailed, which can reduce the clarity with which key hazards and safety measures are communicated to the operators of those facilities. There is scope for improving plant operators’ understanding of the safety significance of the operations and checks which they routinely carry out.

The bow tie method is one of several methods which has been successfully used in numerous areas of business and industry to enhance the visibility of the safety case. It successfully illustrates in a single user-friendly diagram the hazards, the initiating faults, the consequences, and the prevention and mitigation barriers.

The bow tie method is a purely qualitative technique, which could be successfully introduced to the nuclear industry as an additional tool to improve the visibility and understanding of the safety case, and thus complement (not substitute) the more rigorous safety analysis techniques which are the norm in this industry. By making the diagrams readily accessible in the control room, the operators of nuclear facilities could further improve their understanding of the safety significance of their role in preventing major accidents and mitigating consequences.

References:

[1] Application of QRA in operational safety issues, Andrew Franks, Richard Whitehead, Phil Crossthwaite and Louise Smail, Det Norske Veritas, 2002, http://www.hse.gov.uk/research/rrpdf/rr025.pdf

[2] Marine risk assessment, Det Norske Veritas, 2002, http://www.hse.gov.uk/research/otopdf/2001/oto01063.pdf

[3] Lessons Learned from Real World Application of the Bow tie Method, Steve Lewis and Kris Smith, Risktec Solutions, March 2010, http://www.risktec.co.uk/knowledge-bank/publications.aspx

[4] Bow tie: an Elegant Solution?, Steve Lewis and Sheryl Hurst, Risktec Solutions Ltd, November 2005. http://www.risktec.co.uk/knowledge-bank/publications.aspx

[5] Optimising Hazard Management by Workforce Engagement and Supervision, Vladimir M Trbojevic, prepared by Risk Support Ltd for the Health and Safety Executive, Research Report 637, 2008, http://www.hse.gov.uk/research/rrpdf/rr637.pdf

[6] Feasibility of Storybuilder Software Tool for Major Hazards Intelligence, prepared by the Health and Safety Laboratory for the Health and Safety Executive, Diego Lisbona & Mike Wardman, Research Report 778, 2010, http://www.hse.gov.uk/research/rrpdf/rr778.pdf

[7] The Nimrod Review, An independent review into the broader issues surrounding the loss of the RAF Nimrod MR2 Aircraft XV230 in Afghanistan in 2006, Charles Haddon-Cave QC, 28th October 2009, London: The Stationery Office, ISBN: 9780102962659

[8] Lines of Defence / Layers of Protection Analysis in the COMAH Context, Andrew Franks, prepared by Amey VECTRA Ltd for the health and Safety Executive, http://www.hse.gov.uk/research/misc/vectra300-2017-r02.pdf

[9] Council Directive 96/82/EC on the control of major-accident hazards (also known as the Seveso II Directive), http://ec.europa.eu/environment/seveso/index.htm

[10] Health & Safety Executive (HSE 1999a). A Guide to the Control of Major Accident Hazard Regulations 1999. L111, HSE Books.

[11] Safety Assessment Principles for Nuclear Facilities, Health and Safety Executive, 2006 Edition, Revision 1, http://www.hse.gov.uk/nuclear/saps

[12] Fundamental Safety Principles, International Atomic Energy Agency, Vienna, 2006, http://www-pub.iaea.org/MTCD/publications/PDF/Pub1273_web.pdf

[13] Health & Safety Executive (HSE 1999c). Preparing Safety Reports: Control of Major Accident Hazard Regulations 1999. HSG 190, HSE Books.

[14] IAEA Safety Standards Series, Safety of Nuclear Power Plants: Design, Safety Requirements, No. NS-R-1, International Atomic Energy Agency, Vienna, 2000, http://www-pub.iaea.org/MTCD/publications/PDF/Pub1099_scr.pdf

Bow Tie Methodology

A tool to enhance the visibility and understanding of nuclear safety cases

Workshop on the Safety Assessment of Fuel Cycle Facilities – Sep 2011

Marc Vannerem
UK - Principal Nuclear Inspector

Office for Nuclear Regulation
An agency of HSE

Structure of presentation

- **Safety cases nuclear industry:** rigorous, in-depth, complex
- **Reduce visibility / understanding** (workforce)
- **Bow tie method: widespread use in the oil & gas and chemical industries**
- **Comparison / common ground: nuclear and other major hazard industries**
- **Incl. legislative framework (high level)**
- **Introduction to the bow tie method**
- **Simple example for nuclear fuel cycle facilities**
- **Finally, conclusions as to how it could be applied to nuclear fuel cycle facilities**

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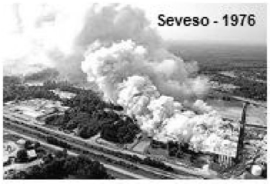
Nuclear: not unique in managing hazards



Major hazard industries: oil & gas, chemicals

- E.g. Seveso 1976, Bophal 1984
- Europe: Seveso II directive (1996): control of major-accident hazards involving dangerous substances
- Implemented via national legislation
- UK: COMAH Regs 1999

Seveso Directive: safety report demonstrating

- major accident hazards have been identified
- all necessary measures have been taken to prevent such accidents and to limit their consequences for persons and the environment

The complexity of the demonstration presented in the safety report is expected to increase with the complexity of the facility concerned

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Similarly - Nuclear industry:
highly regulated environment

IAEA's Fundamental Safety Principles
(implemented via national legislation)

require the operators of nuclear facilities to:

- **ensure that "all practical efforts are made to prevent and mitigate nuclear or radiation accidents."**
- **"The fundamental safety objective is to protect people and the environment from the harmful effects of ionising radiation."**

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Nuclear / major hazard: much common ground

Nuclear industry Major Hazard Industries (oil & gas, chemicals)	Legal requirement to: •identify & control hazards •minimise consequences to persons & environment

Legislation requires the operators of nuclear and other major hazard installations (oil & gas, chemicals) to understand and to have identified:
- The hazards of their operations
- The initiating events which could result in an accident
- The consequences of such accidents
- The barriers which they have in place to prevent an initiating event becoming an event
- The various layers of protection which are in place to limit the effects of an accident

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What are the Regulators' expectations?
•depth and detail of the 'safety analysis' by the Operators of hazardous facilities

Principle of proportionality

Nuclear: IAEA's 'Fundamental Safety Principles'
(applied by UK's ONR)
Conventional: UK guidance to Major Hazard Industries

The depth and sophistication of the safety analysis should be proportionate to the hazard and risk present.

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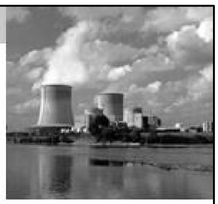
Depth and rigour of safety cases in the nuclear industry



Nuclear installations: high hazard category

Proportionality principle: safety analysis should use "the most developed and sophisticated techniques"

- Nuclear safety cases:
  - Typically detailed, rigorous, in-depth analysis
  - Complex and lengthy (100s / 1000s pages)
- This can reduce: clarity, visibility, communication
  - key hazards and safety measures
- UK's ONR is always favourable to improvements:
  - visibility of safety case to plant personnel.
  - understanding of safety significance of operations and controls

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Typical hierarchy of safety case documents – UK NFC facilities

Highest Level : Detailed Safety Case Documents	Safety case specialists & plant management	•Evolved / refined over many years •Successful in managing and controlling hazards on nuclear fuel cycle facilities
Intermediate Level : converts "detailed safety case requirements" into "rules and instructions". Controlled operation / management of plant.	Team Leaders	
Detailed instructions: Operation, Maintenance and Emergencies	Plant Operators and Maintainers	

This approach: scope for improving visibility of the overall safety case.
UK's ONR encourages nuclear operators to further improve:
- visibility of safety case to all plant personnel.
- understanding of safety significance of operations and checks carried out.

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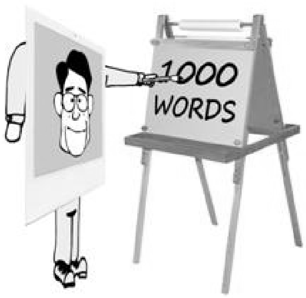
Simple overview diagram

illustrate in simple terms

- Key hazards
- Faults: initiate events
- Consequences of events
- Prevention measures: barriers which prevent faults developing into events
- Mitigation measures: Barriers which mitigate / reduce the consequences of events



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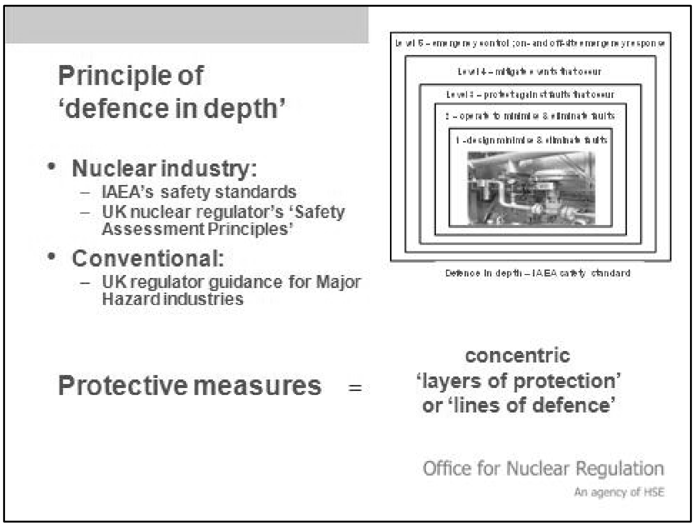
Principle of
'defence in depth'
• Nuclear industry:
– IAEA's safety standards
– UK nuclear regulator's 'Safety Assessment Principles'
• Conventional:
– UK regulator guidance for Major Hazard industries
Defence in depth – IAEA safety standard
Protective measures = concentric 'layers of protection' or 'lines of defence'
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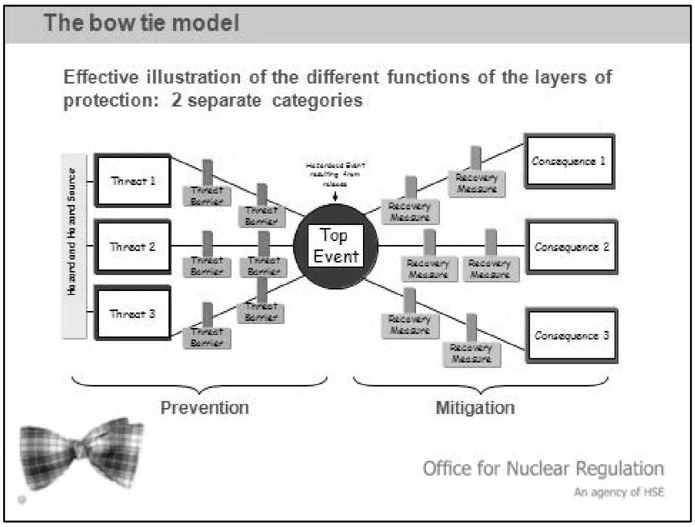
The bow tie model
Effective illustration of the different functions of the layers of protection: 2 separate categories
Hazard and Hazard Source
Threat 1
Threat 2
Threat 3
Threat Barrier
Top Event
Recovery Measure
Consequence 1
Consequence 2
Consequence 3
Prevention
Mitigation
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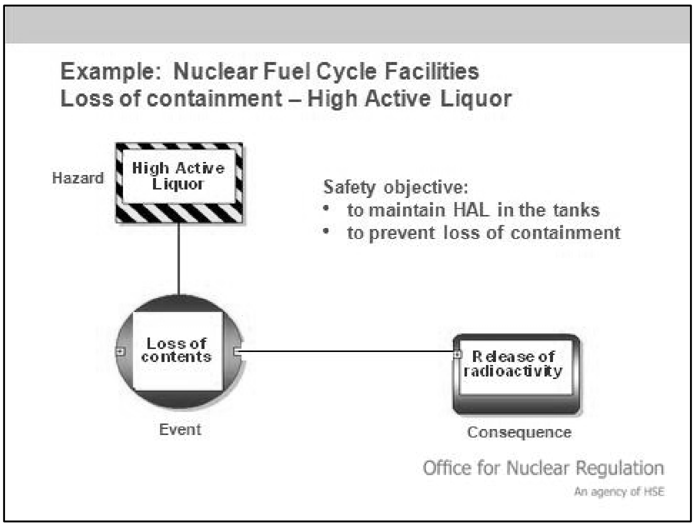
Example: Nuclear Fuel Cycle Facilities
Loss of containment – High Active Liquor
Hazard
High Active Liquor
Safety objective:
• to maintain HAL in the tanks
• to prevent loss of containment
Loss of contents
Event
Release of radioactivity
Consequence
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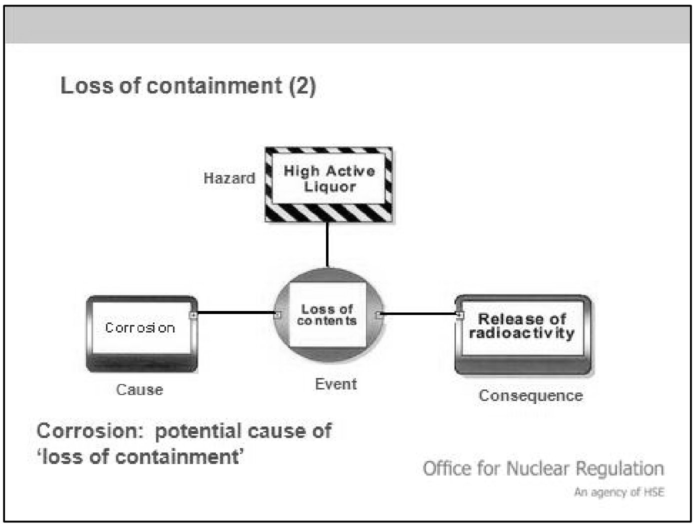
Loss of containment (2)
Hazard
High Active Liquor
Corrosion
Loss of contents
Release of radioactivity
Cause
Event
Consequence
Corrosion: potential cause of 'loss of containment'
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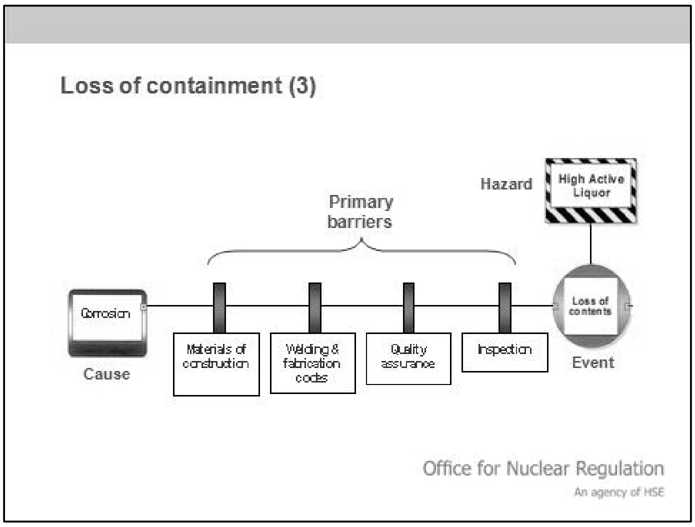
Loss of containment (3)
Hazard
High Active Liquor
Primary barriers
Corrosion
Cause
Materials of construction
Welding & fabrication codes
Quality assurance
Inspection
Loss of contents
Event
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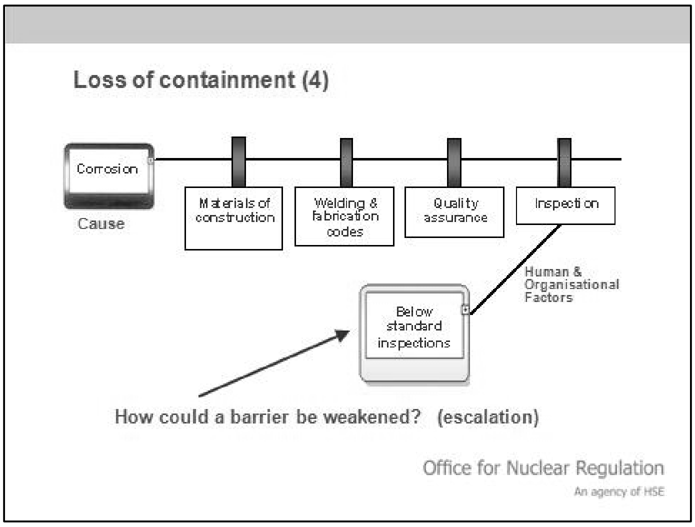
Loss of containment (4)
Corrosion
Cause
Materials of construction
Welding & fabrication codes
Quality assurance
Inspection
Human & Organisational Factors
Below standard inspections
How could a barrier be weakened? (escalation)
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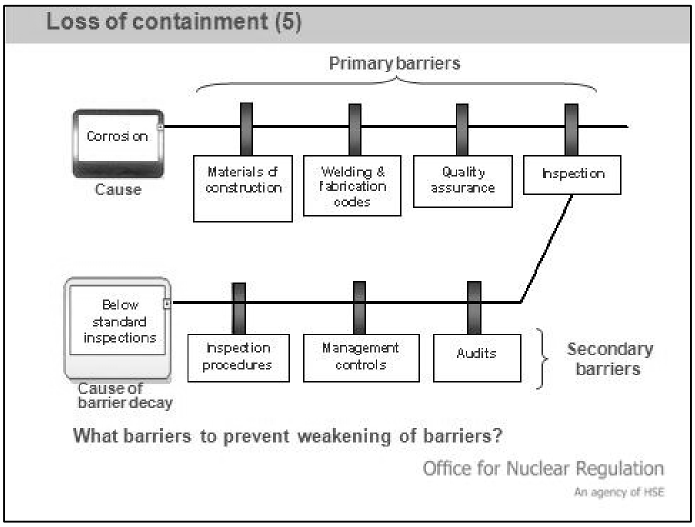
Loss of containment (5)
Primary barriers
Corrosion
Cause
Materials of construction
Welding & fabrication codes
Quality assurance
Inspection
Below standard inspections
Cause of barrier decay
Inspection procedures
Management controls
Audits
Secondary barriers
What barriers to prevent weakening of barriers?
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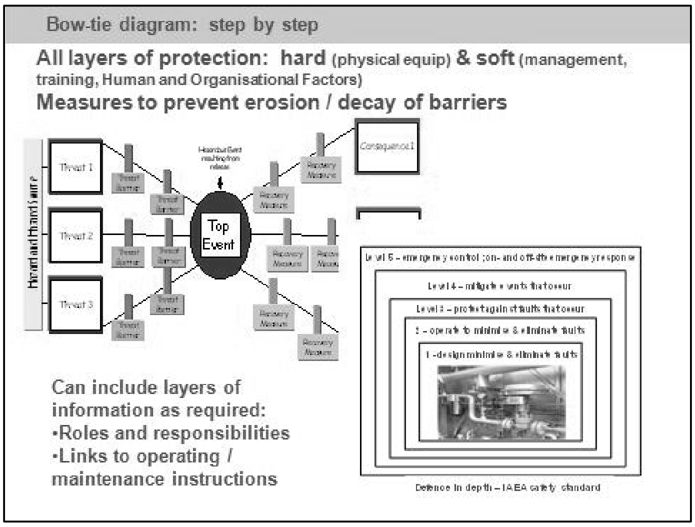
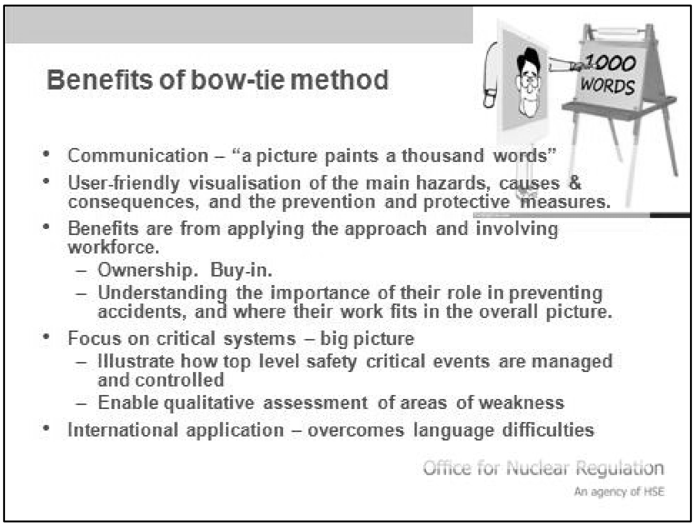

Bow-tie diagram: step by step
All layers of protection: hard (physical equip) & soft (management, training, Human and Organisational Factors)
Measures to prevent erosion / decay of barriers
Threat 1
Threat 2
Threat 3
Top Event
Consequence 1
Can include layers of information as required:
•Roles and responsibilities
•Links to operating / maintenance instructions
Defence in depth – IAEA safety standard


Benefits of bow-tie method
1000 WORDS
• Communication – "a picture paints a thousand words"
• User-friendly visualisation of the main hazards, causes & consequences, and the prevention and protective measures.
• Benefits are from applying the approach and involving workforce.
– Ownership. Buy-in.
– Understanding the importance of their role in preventing accidents, and where their work fits in the overall picture.
• Focus on critical systems – big picture
– Illustrate how top level safety critical events are managed and controlled
– Enable qualitative assessment of areas of weakness
• International application – overcomes language difficulties
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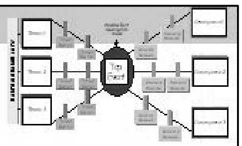

Limitations of bow-tie

- Purely qualitative graphical representation
- Tends to over-simplify
  - –Underlying safety challenges of complex facilities
  - –Inter-dependencies / inter-relationships between various risk controls

<u>Hence</u>:

Bow ties: in addition (<u>not substitute</u>) / complement

–the more rigorous safety analysis techniques currently used in the nuclear industry (DBA, PSA, SAA)

–to improve visibility and understanding.

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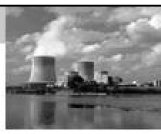
Conclusions

Nuclear Fuel Cycle Facilities: 'high hazard'
Safety cases: rigorous, in-depth, complex



- Can reduce: visibility & understanding of key hazards and safety measures
- Bow-tie method:
  - successful in other 'Major Hazard' industries (oil & gas, chemicals)
  - could be introduced to the nuclear industry as an <u>additional tool</u>
    - diagrams readily accessible in the control room
    - provides purely "qualitative illustration"
    - improve visibility and understanding of the safety case
    - + role of operators in preventing major accidents and mitigating consequences
- <u>Not a substitute</u>, but a <u>complement</u> to the more rigorous safety analysis techniques currently used in the nuclear industry
- Purpose of presentation: awareness - a tool amongst others in the tool box

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Questions

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AN OVERVIEW OF THE UK REGULATORY EXPECTATION FOR DESIGN BASIS ACCIDENT ANALYSIS

Andy Trimble
Office for Nuclear Regulation, UK

Abstract - The UK Health and Safety Executive published its most recent regulatory expectations in the 2006 version of it's safety assessment principles (SAPs). This built on experience regulating the full range of facilities for which it is responsible. Thus the principles underpinning all regulatory safety case assessment are the same but the implementation differs depending on the application.

This paper will describe the published design basis accident analysis (DBAA) logic in context with other technical aspects of the regulatory expectation for safety cases. It will further illustrate the DBAA methodology with practical examples from actual experience on reprocessing plant gained over the last 15 years or so. Among the examples will be the relevance of conventional safety fault initiators to nuclear safety assessment. It will further demonstrate the derivation of facility limits and conditions necessary for nuclear safety.

Introduction

In the UK's nuclear regulatory regime, the Office for Nuclear Regulation (ONR) (formerly, The Health and Safety Executive's, Nuclear Installations Inspectorate) does not specify what should and should not be in a safety case [1]. However, the regulatory goals are set out in our Safety Assessment Principles (SAPs) [2]. These Principles were originally written for nuclear plant in design and were also used to inform periodic safety case reviews required under licence conditions. They are applied, proportionately, to all types of facility including power reactors, chemical plant, fuel fabrication facilities and waste stores.

In common with the goal setting principles of safety regulation in the UK, this is regulatory guidance which sets the regulatory expectation, not a rigid set of rules that need to be followed [6]. Rather, it is intended to assist both Nuclear Inspectors in setting consistent standards and to inform our licensees as to the benchmark they are expected to meet. In line with the injunction in UK safety law, all this is subject to the test "so far as is reasonably practicable" better known as As Low As Reasonably Practicable – ALARP (broadly equivalent to ALARA).

The Standard Licence

The heart of the regulatory control system is the licence and its attached conditions. The regulator can, at any time, attach to a licence conditions which appear necessary or desirable in the interest of safety. However, a standard set of licence conditions (LCs) [13] has evolved with the aim of producing consistent safety requirements which are largely non-prescriptive, flexible and apply to all facilities. The most relevant here include:

a. LC23. OPERATING RULES
   (1) The licensee shall, in respect of any operation that may affect safety, produce an adequate safety case to demonstrate the safety of that operation and to identify the conditions and limits necessary in the interests of safety. Such conditions and limits shall hereinafter be referred to as operating rules;

b. LC1. INTERPRETATION
   (1) In the conditions set out in this Schedule to this licence, unless the context otherwise requires, the following expressions have the meanings hereby respectively assigned to them, that is to say -........"operations" includes maintenance, examination, testing and operation of the plant and the treatment, processing, keeping, storing, accumulating or carriage of any radioactive material or radioactive waste and "operating" and "operational" shall be construed accordingly;

c. LC27. SAFETY MECHANISMS, DEVICES AND CIRCUITS
   The licensee shall ensure that a plant is not operated, inspected, maintained or tested unless suitable and sufficient safety mechanisms, devices and circuits are properly connected and in good working order.

In making its regulatory decisions, inspectors must make judgments about compliance with LCs. This is, in part, achieved using the relevant Safety Assessment Principles [2]. For example Principle FA.9 and the following SAPs paragraphs state that the purpose of Design Basis Accident Analysis is to provide information relevant to trip settings, plant operational limits (Operating Rules), plant operating instructions for fault conditions and the availability requirements for the sufficiency of safety systems. These form some of the fundamental safety controls on the relevant operations.

Technical SAPs - Definitions

In order to ensure proportionate protection for people and the environment and to set DBAA in context the following shows:

a. the key aspects and relationships DBAA has with other forms of analysis;
b. the key relationship with the engineering that delivers the safety function.

It is important that safety provisions reflect this holistic approach – which extends further than the aspects considered here.

The SAPs explicitly state that the technical (as opposed to organisational or cultural) aspects are fundamentally important to engineering a demonstrably safe, fault tolerant plant. The aspects considered here are:

a. Design Basis Accident Analysis (DBAA);
b. Probabilistic safety analysis (PSA sometimes known as QRA);
c. Severe accident analysis (SAA);
d. Good Radiological practice (GRP or Good Engineering Practice - GEP);
e. Waste Management.

Dealing with each of these broad areas in turn:

DBAA: is a robust demonstration of fault tolerance. It is intimately linked to the engineering that delivers the safety function by providing proportionate levels of protection. It also links directly to the engineering principles which call for a preferred series of responses to faults. These vary from designs that are inherently safe to those that may require operator intervention in the fault sequence. The important feature of DBAA is that any uncertainty is allowed for by conservatism. Often this conservatism is in the input data and requires expert judgments about the degree of conservatism appropriate to any particular case. Thus the conservatism becomes a constraint or input on the engineering. DBAA is concerned with internal faults with larger harm potential and not normally with more minor events which are governed by radiological control arrangements [e.g. 9].

PSA: The main purpose of PSA is to demonstrate a balanced design (i.e. where there is no undue reliance on any particular safety feature) and that risks are minimised. The great strength of PSA is this overview. The real value is in the modeling itself and the insights this gives e.g. subtle cross plant interactions can be revealed. This is not covered by DBAA which deals with faults on a fault by fault basis.

It is very tempting to believe the figures produced by PSA. However, there is no basis for this and the numbers, useful in comparative terms, are little more than a crude estimate of the overall risks from the operation under consideration for comparison with criteria. It is carried out best estimate as far as possible.

SAA: A severe accident is one which is not necessarily expected in a plant lifetime but has the potential for high doses or environmental damage. It is not necessary for this potential to be realised (in these definitions, Three Mile Island was a severe accident but there was no significant radiological release). The prime difference between DBAA and SAA is in the way that data is used. SAA is based on best estimates and as such may well be bounded by the DBAA if the level of conservatism is high. However, a sound understanding of the underlying phenomena during such accidents avoids the need for introducing unnecessary conservatism and hence unfruitful expenditure. The main aim of SAA is to provide an input to emergency planning and to identify reasonably practical design improvements that can be implemented at reasonable cost.

GRP {GEP}**:** Because we deal with radiological hazards it is referred to here as Good Radiological Practice (GRP) although it is more commonly known as Good Engineering Practice (GEP). In every industry there are both pressures to reduce costs and increase cost effectiveness. However, most companies and most industries set basic standards below which any design should not fall. This ensures that, for harm potentials smaller than would be covered by DBAA, the learning experience of the company and/or the industry are taken into account. Often GRP is embodied in design manuals or company standards. Quality engineering should only stray outside this standard with sound reason. There is also the concept of the “modern standard” which simply asks “what would the facility look like if it were designed today”. This includes changes to engineering standards as well as progress in safety thinking nationally and internationally. The modern standard is used as a benchmark against which ALARP may be judged. Thus, the engineering is required to provide appropriate engineered provisions to deliver the safety function, reliably and with an appropriate, proportionate capability.

Waste Management: There are major additional waste management constraints as well as those normally required for safety. Much of the regulation is concerned with implementing government policy and GRP. Plainly, much of this reflects public opposition to careless waste accumulation and storage (disposal is dealt with under Environmental Legislation administered by the Environment Agencies). These aspect is becoming more and more relevant in the UK with its legacy of facilities designed before independent nuclear regulation.

DBAA

The detailed guidance for nuclear inspectors is found in T/AST/006 [3]. This is currently in revision following the revision to SAPs although there is nothing that detracts from the underpinning thinking in the current guide. There is a cross reference table on the web that allows the new SAPs principles to be read across to the previous ones as an interim measure [4].

This engineering fault analysis that the guide refers to as "Deterministic Safety Assessment", includes DBAA and forms the bedrock on which the safety case is built. The rigour in such analysis is directly linked to the harm potential or hazard. This is a key idea in deterministic analysis as harm potential is related to the inherent characteristics of the material with that potential. This is a measure of the activity (broadly equivalent to inventory), radio toxicity, mobility and driving force under the prevailing conditions. Thus a mobile, highly active material which can undergo self heating e.g. high level liquid waste, has a higher harm potential than intermediate level solid waste encapsulated in cement. Although the guide gives broad classes of harm potential the reality is that harm potential is a continuous variable and we judge each case on its merits. The guide contains guidelines on the rigour and conservatism appropriate to the classes or categories of nuclear plants.

It is important to understand that DBAA deals exclusively with faults and does not consider normal operation except as the state from which fault sequences develop. Therefore, the only consideration or constraint DBAA puts on normal operation is this "fault starting" condition. The initiating fault frequency lower bound for DBAA is given in principle FA.5 as 10E-5 p.a. and the dose boundary in principle FA.2. However, it is accepted that it is disproportionate for all sequences to have the same rigour in their analysis resulting in extensive, highly diverse and redundant quality safety systems, and so, for infrequent faults (initiating frequencies <10E-3) a single line of defence is adequate provided doses are not excessive whereas, at higher initiating frequencies two independent lines of defence are more appropriate. If potential doses are extremely high, then each of these lines of defence may themselves be diverse and / or redundant to improve their reliability (capability is an engineering function). In addition, the higher the hazard or potential dose, then the more regulators will insist on a robust case which infers engineered provisions in preference to human interaction (subject to ALARP). However, should time be a factor, then is may be acceptable to argue that to achieve a timely hazard reduction, the balance may favour other approaches [10].

The DBAA technique is conceptually simple and can follow logic in the flowchart at Figure 1. Certain decisions must be taken and in most cases the order in which they are taken is not vitally important. There is one exception to this. There is a decision node labeled "low consequence". The intention is to remove the analysis burden where the consequences are low. However, this decision must only be carried out after the harm potential or hazard has been judged. The intention is not to place high reliance on mitigation (often filtration on nuclear chemical plant) but rather to soundly engineer the process for defence in depth in the first place. The engineering inherent safety hierarchy tends to drive towards this approach with a preference for prevention over termination and finally mitigation (see later). Therefore, this decision must only be taken in the light of the overall assessment. In case of doubt, we would expect the decision to be prudently based.

In order to carry out the analysis on a process it is essential to have a sound technical understanding of that process and the associated plant. Much of the basic information is either identical to that needed for design or closely related to it. There is an ongoing iteration between the designer and safety analyst in the search for both a suitable and sufficiently safe design, one which is economic, environmentally acceptable and operable. In DBAA the main concern is safety in the fault condition. The outcome is that the options for the underlying processes are assessed and an informed decision made about the preferred option (optioneering). There are similar considerations for existing plant in periodic review but the options for change in order to achieve ALARP will be limited by what already exists.

The foundation for all this work is fault identification. The main characteristics we seek are that this has been carried out in a structured and comprehensive way. Such techniques might include HAZOP (Hazard and Operability studies) and FMEA (Failure Mode and Effect Analysis). In each case it is important that the individuals involved understand the underlying processes in the plant.

The outcome of the fault identification should be a compilation of all potential faults for the plant (which may be grouped). As the design evolves the balance of faults changes and so further fault identifications are carried out. In addition, the act of analysing the fault may identify further faults or knock on effects. These should also be analysed. It is very important to ensure that a change on one part of a complex plant does not have an unanalysed consequential effect on another part. One of the key aspects of this type of work is the iteration between the analysts and the designers or operators in the search for improvements to meet ALARP. The faults so identified become the “Fault Schedule”. The analysis takes each fault or groups of faults and analyses them in a technique very akin to event tree analysis. The technique simply assumes the fault initiation occurs and examines how the plant responds (usually without any safety systems other than high reliability passive features such as shielding). Depending on the harm potential of the sequence being considered, the safety systems are then put in place as part of the design and their quality constraints flow from their safety function (see later).

The options for dealing with faults during iteration are prioritised on what is known as the inherent safety hierarchy (relating to Principles EKP.1 - 5 in SAPs) which may be summarised:

The design should be such that hazards are avoided (intrinsic or inherent safety);

- The design should use passive features without undue reliance on control or safety systems;
- Any failure or fault should produce no significant deviation other than an indication that the fault has happened;
- The plant should be brought to a safe state by continuously available safety measures or, if not practical, safety measures that need to be brought into operation*;
- Administrative safety measures are an option where there is no reasonable alternative;
- Finally, mitigation is then taken into account.

Note: Filtration falls into the final category not the 4th (*). The aim is to be as near the top as possible.

In other words, these SAPs say faults should be avoided by safe passive means if possible and that that the sensitivity to faults should be minimised. This hierarchy should be at the front of every engineers mind when designing or analysing designs. This provides a driver towards inherently safer facilities. It is difficult to overestimate the importance of this hierarchy and this has been the thrust of several initiatives for some years [5]. Intrinsic and inherent safety should be the target for all facilities. The practical outcome is to drive the safety systems closer to the part of the operation where the faults initiate before considering safety systems that act later or further on in the fault sequence.

For plants which already exist (especially nuclear plants where access is often either difficult or impossible) the response to this hierarchy can be different to that for plants in design. At this stage the Reasonably Practicable or ALARP principle takes effect. Whilst the ALARP principle is conceptually simple, in a DBAA the concept is not so easy to apply. The surrogate developed from many years of experience has been to establish the "modern standard". The modern standard simply asks “what would the facility look like if designed today?”. Thus, it involves not just changes to published engineering codes and standards but also advances in safety thinking, nationally and internationally. This is compared with what exists (a gap analysis) and those modifications that improve safety are highlighted. The judgment about what to implement is a combination of the balance of plant life, the potential hazard, the current deficit in performance, costs and benefits. The judgments in nuclear plant are often made on the basis of experience both national and international. It is important to note, that it may be acceptable to partly meet the safety shortfall where a safety gain can be made at reasonable cost. Equally, a case based on cost-benefit analysis alone is unlikely to be sufficient.

In summary the fundamental DBAA technique is simple:

a. assume the fault occurs with the worst possible harm potential (usually qualitatively). Often this will be a design limit for the facility or a parameter derived from the design limit;

b. assume the worst allowable plant state in terms of feeds, impurities, plant availability and other conditions including start up and shut down;

c. develop a technical description of how a fault develops and the engineering calculations which demonstrate how the system or plant behaves under that fault condition. Do not assume any control or safety provision operates correctly. Often this will be a transient analysis. Put in place the safety systems;

d. determine if the safety systems meet the inherent safety hierarchy;

e. determine if these meet the characteristics of quality safety systems e.g. single failure proof, diverse, redundant, segregated, capable of detecting the fault under fault conditions, provide sufficient defence in depth and so on. For more frequent faults (>10E-3 initiating frequency), single failures in the safety system are assumed. This is one route for deciding how many redundant trains will be needed in some safety systems. In particular, safety related items which are maintained on line should be assumed to be in the worst maintenance state;

f. judge the adequacy against the SAPs Target 4 criteria of no dose and at least one barrier intact except in the most severe cases and, ideally, having an accident rate less than 10^{-7} per annum for major accidents. For lower consequence faults such a frequency is likely to be both unnecessary and expensive given the potential harm from that fault. It is often the case that surrogate or subordinate rules can be developed to help engineers and analysts demonstrate adequate reliability.

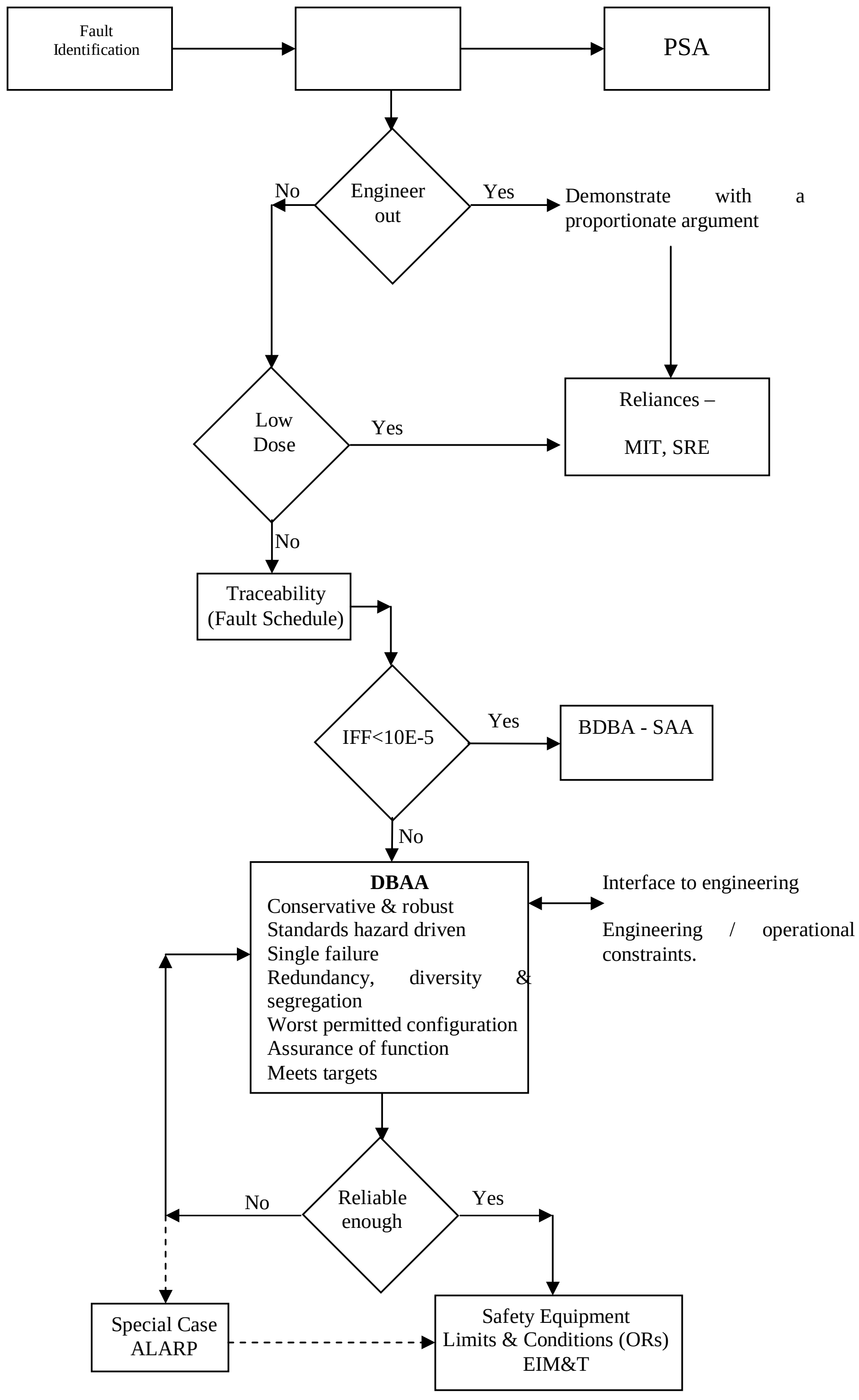

Figure 1: DSA Logic diagramme

In many cases faults can be considered as transients from steady state and modeling the time variation of some parameters can vary from simple to extremely complex. The more complex calculations are often carried out with computer codes e.g. Computational Fluid Dynamics (CFD). If such codes are used, they should be validated (ensure the code models plant behavior as accurately as possible with due conservatism) and verified (ensure that both the code and the input data are as correct as possible).

Uncertainties which lead to undue constraints on operations can often result in research and development either to look at ways of better preventing or terminating the fault or to reduce conservatism in the analysis by increasing confidence in the underpinning data. Also, the conservatism in the analysis helps develop a design that is robust and can tolerate unforeseen faults e.g. Three Mile Island's containment was not designed for the potential hydrogen ignition insult but, because the design was conservative, it would have tolerated it.

The results of such analyses give outputs that put constraints on the operations in question. These are referred to in total as the Safe Operating Envelope for the plant or operation. Licence Condition 23 calls for Limits and Conditions and these should be derived from the DBAA as shown in Figure 1.

LC 23 defines Operating Rules (ORs) as Limits and Conditions.

a. Limits are operational parameters such as temperature, pressure or concentration that operations must be controlled within in order to remain within the safe operating envelope;
b. Conditions are plant configurations that must be complied with in order to ensure the safe operation of the plant.

Both of these are primarily derived from the DBAA although there are a number of exceptions. Most notable will be the feed specification for the facility. This forms the basis for both the underpinning design and the analyses on which the ORs are based. The regulatory guidance [11] is being developed to bring it further into line with good international practice [12] and should be available by the time this paper is published.

Practical Examples

Here we consider how “conventional hazards” affect the nuclear analyses. Most conventional safety matters that affect nuclear safety fall into one of two categories:

a. Failure prevents the safety function being delivered e.g. electric power or steam supplies;
b. A reactive chemical or mechanical device interacts with the facility to initiate or aggravate a fault.

There are a number of these which are generic to most reprocessing facilities, some of which are covered below.

Hydrogen: The UK convention on hydrogen is that it should be controlled to 25% of the lower flammable limit (LFL) – 1%. Similarly we have accepted that reaching the LFL is acceptable under fault conditions provided the nuclear safety case is robust and conservative. In other words, the LFL is unlikely to be breached in practice. These are broadly equivalent to the limits used in conventional industry. The difference in the nuclear field is that we require this robust safety case to demonstrate that such levels will be achieved. It is worth noting that there is no distinction between radiolytic hydrogen and hydrogen from other sources e.g. reagent hydrogen or from battery charging. Our licensees have developed methodologies for meeting these limits [e.g. 7].

In plant handling solutions of plutonium or other alpha emitters in closed tanks, it is often difficult to back fit a purge to control hydrogen concentration from radiolysis. It is acceptable to use lines intended for other functions e.g. pressure or level indication (provided the primary safety function is not compromised), to supply enough air to control the hydrogen below the LFL.

Cranes: As facilities develop and age there is often a need to either replace or add to the exiting plant and equipment. In common with most chemical plant this drives the need for construction cranes. The threat is either crane collapse or impact between the crane load and a sensitive operation. In the UK our sites are limited in area and so, particularly with older facilities, they tend to be relatively close together. This is particularly the case when decommissioning. Our view is that cranes near sensitive, high hazard should be avoided if possible as the consequences can be serious. To this end our licensees have gone to extreme lengths to avoid larger cranes, instead using jacking arrangements or lifting frames [8] and other means to minimise the use of mobile cranes.

There are some preferences for using cranes which include:

a. Remove or minimise the hazardous inventory in the potentially affected plants.

b. Avoid the use of cranes or, where necessary, minimise the size and reach of smaller cranes used to erect other lifting devices.

c. Where cranes must be used ensure they are used in the safest possible way e.g. limit travel, are operated by reputable contractors, are controlled on site according to good crane practice and are only used under well defined weather conditions.

Reactive chemicals: Probably the most interesting are the nitrogen based chemicals used in the salt free flow sheet for THORP. Hydrazine, hydroxylamine and hydrazoic acid are present either as reagents or as decomposition products. They represent a direct safety implication as reagents because their instability makes them potentially a fire hazard. This fire hazard could not only have the direct effects of the fire itself but also represent a loss of control with the potential for criticality in the separation plant.

Similarly, they decompose to ammonium compounds in the process which, with the nitric acid based flow sheet, produce ammonium nitrate. Ammonium nitrate in sufficient quantities and under the right conditions is a low grade explosive. Fortunately, most of today’s nuclear chemical plants are small enough for this to be no more than a minor concern.

In addition, the odourlesss kerosene (used in the solvent extraction process), when mixed with ammonium nitrate forms the industrial explosive ANFO (ammonium nitrate – fuel oil). This further goes to highlight the importance of good solvent control in reprocessing plant. However, the nuclear safety case does need to demonstrate adequate control over these phenomena and that the DBAA criteria are met.

The solvent control example highlights one other aspect of nuclear chemical plant that is not so prevalent in power reactors, namely, facility – facility interactions. On a multi facility site, such as Sellafield, faults initiated on one facility can propagate to another and there must be an adequate analysis to cover this resulting in suitable and sufficient safety systems to deal with it (SAPs principle ST.6).

Conclusion

In the UK, DBAA is a nuclear industry wide technique that is intended to demonstrate the robustness of nuclear facilities to tolerate relatively frequent, potentially serious faults. Although the form of analysis varies across the facilities we regulate the overall intent is the same. The technique is quite different to the fault trees used for PSA (QRA) which serve a different purpose. It requires a detailed and comprehensive professional knowledge of how the operations (plants and facilities) respond to faults. This can involve anything from simple hand calculation to complex computer models. The rigour and conservatism is a matter of professional judgment but increasing rigour and increasing conservatism is expected as harm potential and uncertainty increase.

The output of DBAA are both the operational and engineered controls necessary for robust safety in nuclear facilities. In addition, the level of examination, maintenance, inspection and testing to achieve appropriate capability and reliability should also flow from the same analysis and its associated engineering. Thus, there should be an assurance that facilities operated in compliance with these parameters should be safe from internal faults in all reasonably foreseeable circumstances.

References

[1] Technical assessment guide T/AST/051, Guidance on the purpose, scope and content of nuclear safety cases. http://www.hse.gov.uk/nsd/tast051.htm

[2] Safety Assessment Principles for Nuclear Facilities 2006 Edition, Revision 1 http://www.hse.gov.uk/nuclear/saps/saps2006.pdf

[3] Deterministic safety analysis and the use of engineering principles in safety assessment http://www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/tast006.pdf

[4] 1992 to 2006 SAPs Cross-reference Table http://www.hse.gov.uk/nuclear/saps/crossreference.pdf

[5] The storage of Liquid High Level Waste at BNFL Sellafield – an updated review of safety. A report by HM Nuclear Installations Inspectorate. HSE February 2000.

[6] Trimble G A, Proc HAZARDS XX Symposium Series 154 P236 – 246, Background to and experience of using the NII's new safety assessment principles – Learning for the high hazard sector? I Chem E 2008

[7] ID Kempsell, MJ Wakem, MP Fairclough, JM Ingram, Proc HAZARDS XVI, Symposium Series 148, P523 – 540, Hydrogen Explosions - An Example of Hazard Avoidance & Control, I Chem E 2001

[8] e.g. http://www.mammoet.com/Default.aspx?tabid=692

[9] Ionising Radiations Regulations 1999 (IRR99) SI 1999 No. 3232

[10] Trimble A, Proc HAZARDS XIX, Symposium Series 151 P792 ff , Summary of the current position on decommissioning safety cases and the control of operations. I Chem E

[11] T/AST/035 Issue 2, The Limits and Conditions for nuclear plant safety. http://www.hse.gov.uk/foi/internalops/nsd/tech_asst_guides/tast035.pdf

[12] IAEA "Operating Limits and Conditions and Operating Procedure for Nuclear Power Plants", IAEA Safety Standards Series, Safety Guide No. NS-G-2.2 http://www-pub.iaea.org/MTCD/publications/PDF/Pub1100_scr.pdf

[13] Licence Condition Handbook http://www.hse.gov.uk/nuclear/silicon.pdf

Acknowledgment

Thanks go to many in ONR for help and advice in developing this paper and the guide it describes. The opinions here are those of the author.

Disclaimer

No part of this paper should be taken as definitive interpretation of policy, UK law or their application.

SAPs
Engineering & DBAA

Dr Andy Trimble
HM Superintending Inspector
(Nuclear Installations)

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CONTENTS

- Context setting & Underlying thinking
- Interpretation
- Conclusions
- Closing thought

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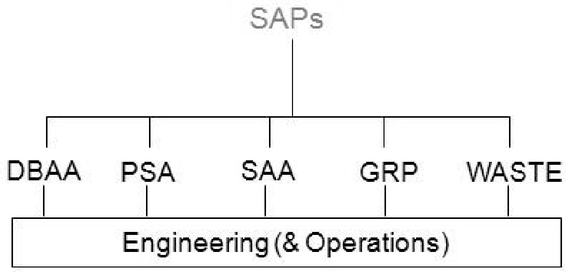

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Proportionality

Based on Hazard: Hazard = Harm Potential

Harm Potential is a function of the intrinsic material properties under the prevailing conditions:

- radioactive inventory,
- radio toxicity
- "driving force" and
- mobility.

For example: these are all High for Highly Active Liquor – Hence rigorous & robust case,

conversely

- Mobility is LOW for Glass and the plant is passively safe (walk away)

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Inherent Safety Hierarchy (see also EKP 1 – 5)

- The design should be such that hazards are avoided (intrinsic or inherent safety);
- The design should use passive features without undue reliance on control or safety systems;
- Any failure or fault should produce no significant deviation other than an indication that the fault has happened;
- The plant should be brought to a safe state by continuously available safety measures or, if not practical, safety measures that need to be brought into operation;
- Administrative safety measures are an option where there is no reasonable alternative;
- Finally, mitigation is then taken into account.

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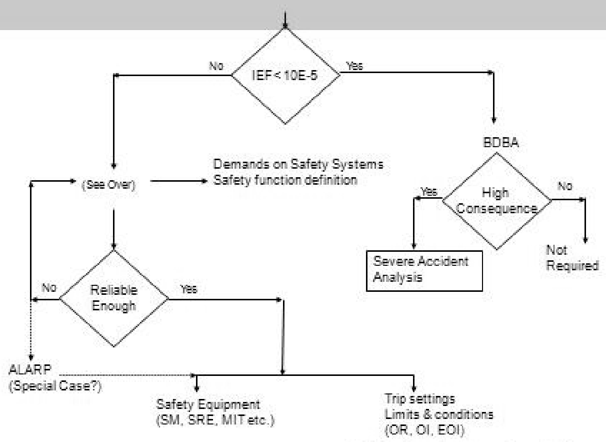

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DBAA

- Conservative & robust
- Standards related to hazard (ALARP)
- Safeguards as high up Inherent safety hierarchy as possible
- Single failure
- Redundancy, diversity, segregation
- Multiple physical barriers
- Worst permitted plant configuration
- No breach & at least one barrier intact (Target)
- No dose except in most severe cases (Target)
- Assurance of continued function

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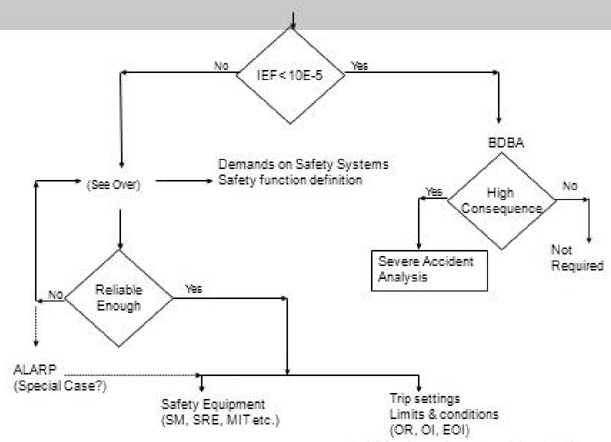

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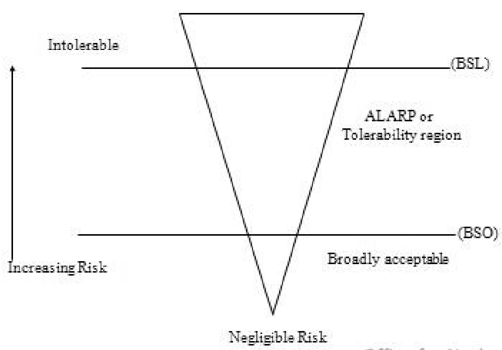
TOR Framework



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CONCLUSIONS

The case should show the prime DBA characteristics being ROBUST and FAULT TOLERANT (infers optioneering) thus, a sound technical justification will be required to underpin the case

So ALARP (SFAIRP) can be achieved and thereby compliance with the law.

Thus meeting HSE's policy of securing compliance with the law in line with the principles of proportionality, consistency, transparency and targeting on a risk related basis.

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CLOSING THOUGHT

"HSE {ONR} does not advocate relying solely on quantified risk assessment, particularly as this may be misused to justify poor practice when factors relating to good engineering practice in design or construction may be more meaningful. It is our view that probabilistic estimates, particularly with such low numbers, must always be treated with caution as there are inevitably high levels of uncertainties in both the data they are based upon and the calculational models which produce them. Safety cases must therefore be primarily based on other elements such as defence in depth and good engineering practice. Probabilistic or quantified risk assessment should only be on input in the overall case. The normal approach used for nuclear installations of robust engineering design, defence in depth and the use of deterministic conservative assessments of both normal operation and fault behaviour, with probabilistic risk assessments to judge the significance of uncertainties, should be sufficient to ensure public protection both now and in the future".

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SEVERE ACCIDENT ANALYSIS AND MANAGEMENT IN NUCLEAR FUEL CYCLE FACILITIES

Mina Golshan
Office for Nuclear Regulation, UK

Abstract – Within the UK regulatory regime, assessment of risks arising from licensee's activities are expected to cover both normal operations and fault conditions. In order to establish the safety case for fault conditions, fault analysis is expected to cover three forms of analysis: design basis analysis (DBA), probabilistic safety assessment (PSA) and severe accident analysis (SAA).

DBA should provide a robust demonstration of the fault tolerance of the engineering design and the effectiveness of the safety measures on a conservative basis.

PSA looks at a wider range of fault sequences (on a best estimate basis) including those excluded from the DBA.

SAA considers significant but unlikely accidents and provides information on their progression and consequences, within the facility, on the site and off site. The assessment of severe accidents is not limited to nuclear power plants and is expected to be carried out for all plant states where the identified dose targets could be exceeded.

This paper sets out the UK nuclear regulatory expectation on what constitutes a severe accident, irrespective of the type of facility, and describes characteristics of severe accidents focusing on nuclear fuel cycle facilities. Key rules in assessment of severe accidents as well as the relationship to other fault analysis techniques are discussed. The role of SAA in informing accident management strategies and off-site emergency plans is covered. The paper also presents generic examples of scenarios that could lead to severe accidents in a range of nuclear fuel cycle facilities.

1. Introduction and relationship to other fault analysis techniques

The Office for Nuclear Regulation's (ONR) Safety Assessment Principles (SAPs, Ref. 0) and their supporting Technical Assessment Guidance are used by the ONR inspectors to guide their regulatory decision making. Underpinning such decisions is the legal requirement on nuclear site licensees in the UK to reduce risks so far as is reasonably practicable, and the use of SAPs should be seen in that context. The SAPs also provide nuclear site duty holders with information on the regulatory principles against which their safety provisions will be judged. However, they are not intended or sufficient to be used as design or operational standards, reflecting the non-prescriptive nature of the UK's nuclear regulatory system. This section explains a brief background and justification for the Numerical Targets and Legal Limits provided to inspectors as guidance in SAPs in order to provide clarification for use of these limits and targets in relation to severe accidents.

The Basic Safety Levels (BSLs) and Basic Safety Objectives described in Safety Assessment Principles translate guidance in the Health and Safety Executive's (HSE) Reducing Risks Protecting People (R2P2, Ref. 0) into a framework and guide for decision making by inspectors. HSE policy is that the BSLs indicate doses/risks which new facilities should meet and provide benchmarks for existing facilities. BSOs are set at a level where HSE does not consider seeking further improvements to be consistent with a proportionate regulatory approach. The numerical targets described in SAPs are for guidance only except for those identified as authorised limits in Ionising Radiation Regulations 1999 and Radiation Emergency Public Preparedness Information Regulations, REPPIR (2001).

In contrast, licensees have a duty to consider whether they have reduced risks to as low as reasonably practicable (ALARP) on a case by case basis irrespective of whether the BSOs are met. As such, it will in general be inappropriate for licensees to use the BSOs as design targets, or as surrogates to denote when ALARP levels of dose or risk have been achieved.

Assessment of risks arising from licensee's activities are expected to cover normal operations and fault conditions. In order to establish the safety case for fault conditions, fault analysis should cover three forms of analysis: design basis analysis (DBA), probabilistic safety assessment (PSA) and severe accident analysis (SAA). The purpose of this paper is to cover severe accidents with a special focus on nuclear fuel cycle facilities.

The 2006 SAPS (Para 543) defines a "severe accident" to be: "…a fault sequence which leads either to consequences exceeding the highest radiological doses given in the BSLs of Target 4, or to a substantial unintended relocation of radioactive material within the facility which places a demand on the integrity of the remaining physical barriers. A substantial quantity of radioactive material is one which if released could result in the societal risk target (Target 9 – Para 623f)…". This means where such a beyond design basis fault sequence can be identified with consequences greater than those in Target 4 (i.e. >500mSv on-site and >100mSv off-site), then severe accident analysis is expected to have been carried out. This is expected to include determination of the magnitude and characteristics of the radiological consequences, including societal effects (see SAPs Para 622ff); and a demonstration that there is no sudden escalation of consequences just beyond the design basis ("cliff edge"). SAPs Para 544 states "Rigorous application of DBA should ensure that severe accidents are highly unlikely. Nevertheless suitable and sufficient severe accident analysis is still required to ensure that risks are reduced so far as is reasonably practicable."

As a whole, the three form of analysis listed in para. 0 (DBA, PSA and SAA) should provide qualitative and complementary inputs to the design, operation and emergency preparedness of the facility.

DBA should provide a robust demonstration of the fault tolerance of the engineering design and the effectiveness of the safety measures on a conservative basis. PSA looks at a wider range of fault sequences (on a best estimate basis) including those excluded from the DBA.

SAA considers significant but unlikely accidents and provides information on their progression and consequences, within the facility, on the site and off site. Therefore, the analysis contributes to identification of any additional reasonably practical preventative or mitigating measures. Furthermore, it provides information needed to support the development of severe accident management strategies and emergency preparedness plans. However, in SAA (unlike DBA) a best estimate approach should be used so far as is reasonably practicable..

For reactor facilities, severe accident analysis is an established concept but there are other nuclear facilities that have the potential for a severe accident, and where this is so then SAA should be expected to form part of the facility's risk assessment. ONR's SAPs (Para 549) provides a list of applications where information obtained from SAA could be used:

- to assist in the identification of any further reasonably practicable preventative or mitigating measures beyond those derived from the design basis. (This is generally most relevant for a new build);
- to form a suitable basis for accident management strategies;
- to support preparation of emergency plans for the protection of people; and;
- to support the PSA of the facility's design and operation.

2. Basis and characteristics of severe accident analysis in the context of UK regulatory regime

The general features of SAA as described in this Section can be applied to the analysis of severe accidents in a wide spectrum of facilities. There are 3 types of scenarios that can be addressed by SAA. These are:

i. Low frequency / high consequence events beyond design basis;
ii. Design basis events but where the safety provisions are assumed to fail; and;
iii. Scenarios omitted from the safety case such as malevolent acts (or known exclusions outside of the duty holder's control or additional scenarios excluded inadvertently).

Analysis of all these events will be addressing "level 4" in the defence in depth levels of protection and is expected to provide answers to questions such as "How quick is the accident progression?" and thus "What accident management strategies work best?".

The key rules to bear in mind for analysis of severe accidents are: (a) that possible initiating events should not be ruled out just because they are of very low frequency and (b) the focus should be on conceivable plant states not on fault sequences. Thus, the analysis should provide information on "what could happen if the control of a hazard were lost and who should be protected".

As discussed earlier, the basis for the analysis of the magnitude and characteristics of radiological consequences and of failure modes in barriers or shielding should normally be on a best estimate basis supported by sensitivity analysis. A key objective of sensitivity analysis is to check that there is no sudden escalation of consequences just beyond the design basis ("cliff edge effect"). If there are significant uncertainties that might undermine confidence in the results then a more conservative or bounding case approach should be considered.

It is the necessity of realism that distinguishes SAA from the analysis of transients or fault conditions within the plant design basis. Incorporation of conservatism into the analysis of design-basis faults adds a measurable margin to calculated results. This is possible because the effects of bias in selected values of modelling parameters or other claims in the analysis on plant response can be demonstrated with confidence to produce "bounding" or more restrictive outcomes. Such trends can rarely be demonstrated in severe accident analysis as each scenario could potentially be unique. Furthermore, Interactions among operating phenomena are often diverse, complex and counter-intuitive, and situations may occur in which a "conservative" assumption in one area produces a non-conservative outcome in another area. Indeed, what is "conservative" in DBA may become "non-conservative" once the accident is happening. As a result, each analysis should be approached with the intention of evaluating the progression of events in as realistic a manner as possible.

3. Severe accident analysis for nuclear fuel cycle facilities

There is much greater variability in the types of severe accident that could occur in nuclear chemical plants (compared with reactors), since the process and technologies are much more varied. Severe accident analysis carried out for reactors is generally concerned with core damage and often a change of phase of materials (*e.g.* fuel melt and gaseous release of fission product, pond fire *etc*). The resulting severe accident analysis has to deal with significant uncertainties in the basic physical and chemical properties of damaged fuel and core components in addition to the range of potential damage states that containment systems may suffer. For nuclear fuel cycle facilities, severe accident could be high-energy driven faults that lead to release or unintended relocation of nuclear material. Because of the nature of the processes involved, there is a lower degree of uncertainty associated with physical and chemical properties of the material that could be released.

The general principles for analysis of severe accidents discussed in Section 3 equally apply to nuclear fuel cycle facilities. One approach that could be adopted in determining candidate scenarios for severe accidents is considering the critical safety functions and determining which is relevant to the process. Using the screening criteria discussed in Para. 0, conceivable plant state that could result in a release approaching Target 4 doses could be identified. The next steps are to analyse the escalation, determine how the safety functions can be affected and what further control measures could be put in place in support of the safety function(s). A key factor of the analysis is establishing the amount of material that could be released from the process containment systems (leakage or airborne). It should be noted that because of the variety in processes and technologies in nuclear chemical plants, there may be a need for further research to determine effective control measures.

The reactivity of released materials also should be considered; *e.g.* the potential for reaction between acidic liquors and cements used in construction materials. Some indication of the mobility of the material (*e.g.* sludge/ supernate) should be provided to assist with determining recovery plans.

Another important factor is establishing the degree of conservatism in the assumptions used and the extent of simplifications made for deriving parameters such as the time to respond to temperature or pressure rises. In a safety case, these could be based on bounding arguments and are therefore not suitable for use when decisions need to be made based on realistic data to determine what measures or actions are need to be taken.

In multi-facility sites, an event in one plant can have an effect on surrounding facilities (domino effects). The primary event may not be in itself a nuclear accident but could lead to an event in other facilities on site with potential for consequences exceeding severe accident dose targets. Particular care should be given to identification of such scenarios to ensure these events can be dealt with in a coordinated manner. Therefore, consideration should be given to establishing a central site-wide control centre to coordinate response to emergencies and severe accidents.

Resilience to loss of services is one of the most discussed topics in recent months following the events in Fukushima. As for any multi-facility site, consideration should be given to scenarios where loss of services in isolation or combined with other faults, could affect a number of plants within the site. SAA supported by PSA would enable identification of common emergency equipment relied upon by various facilities on site and aide development of an appropriate strategy for deployment of these equipment.

4. Generic examples of potential severe accidents in nuclear fuel cycle facilities

The aim of this section is not to consider details of a particular design but to focus how severe accident analysis could be performed to determine conceivable plant states following loss of a critical safety function.

Loss of cooling in processes involving high-level waste

High-level waste is a type of nuclear waste created by reprocessing of spent nuclear fuel and exists in two main forms; raffinate (and other waste streams) or vitrified solid waste. The focus of this example is the raffinate form. High-level waste contains many of the fission products and transuranic elements generated in the reactor core. It is very active and requires shielding during handling and transport as well as cooling as it is self-heating.

As discussed, storage and processing of highly active raffinate requires cooling. As for any hazardous nuclear process, there are protective measures to insure cooling is maintained and identification of these measures is part of the design basis accident analysis. Considering that total loss of cooling could lead to a release exceeding the dose targets described in Section 0 , consideration should be given to severe accident analysis to determine whether further measures could be put in place to mitigate the consequences and terminate the release. Modelling and simulation of behaviours of raffinate in time and availability of other plant systems or services that may be needed for recovery at each point of the accident progression time line is part of the analysis. These could range from removal of raffinate to another location where cooling is available, provision of temporary hoses to deliver cooling water from other sources or any suitable combination of these measures. The affect of such an event on other areas of the plant or, in the case of multi-facility sites, other plants on the site would be considered as part of the analysis.

Runaway reactions

In the following generic scenario the aim is to draw attention to both on-site and off-site consequences. Where reprocessing of raffinate involves operating below the atmospheric pressure, a runaway reaction can occur following a pressure surge (loss of depression within the process vessel and immediate reinstatement to the required pressure whilst other process parameters remain unchanged). This could result in eruption of superheated raffinate and it's relocation to a vessel not intended for its containment. This means that some of the design parameters such as shielding or provision of cooling within the secondary containment may not be adequate with the potential to lead to significant doses to workers (>500 mSv) or members of the public (>100 mSv). This and similar scenarios are most akin to the second criterion for assessment of severe accidents discussed in para. 0. This is where there are adequate measures in place to prevent the fault but the severity of consequences warrant further analysis to determine what additional measures may be needed to mitigate the consequences. Again, the analysis would provide information to answering questions such as what could be the effect on other facilities or operations on site.

5. Conclusions

The background to risk assessment within the UK regulatory regime, the definition and purpose of severe accident analysis and its relationship to other fault analysis techniques is discussed in this paper.

SAA should provide information on survivability, vulnerability, withstand and location of safety equipment. Furthermore, the analysis helps identifying additional reasonably practicable measures that can be put in place to mitigate the consequences of severe accidents. Similar to other kind of analysis techniques, the focus of SAA should be protecting people and the environment rather than meeting a specific target and therefore it is a technology-neutral analysis tool that can be used for all types of nuclear facilities.

References

[1] HSE Safety Assessment Principles for Nuclear Facilities. First edition, 2006. Web version: www.hse.gov.uk/nuclear/saps/

[2] Reducing risks, protecting people: HSE's decision making process. HSE Books ISBN 0 7176 2151 0. Web version: www.hse.gov.uk/risk/theory/r2p2.pdf

Severe Accident Management Strategies in Nuclear Fuel Cycle Facilities

Dr Mina Golshan, Nuclear Inspector

Office for Nuclear Regulation

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Overview

- Brief overview of nuclear fuel cycle facilities in the UK
- Risk assessment→ Fault analysis→ Severe Accidents and analysis of SAs
- Focus of severe accident analysis and out-put of the analysis
- A methodology proposed for SAA
- Examples
- Parallels across the wider (chemical) high-hazard industry
- Conclusion

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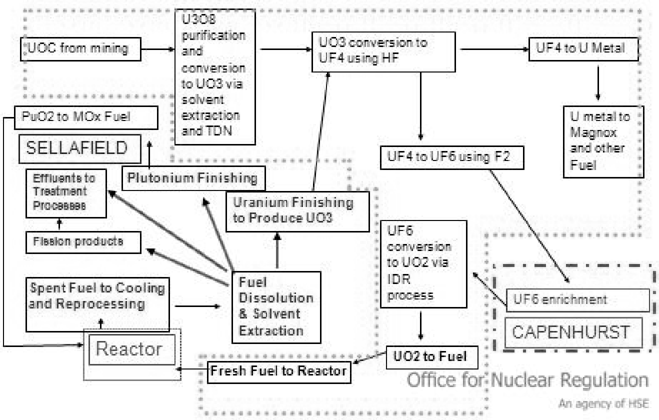
The Civil Nuclear Fuel Cycle in the UK on a Slide !



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Risk Assessment

UK Law:

HSWA 1974, sections 2(1) and 3(1) impose duties upon employers to consider risks (to workers and others) arising from their operations. Legal precedent gives interpretation of the law (case law).

…A material risk simply means that the risk must not be trivial or fanciful.";

"[The HSWA sections] are not limited, in the risks to which they apply, to risks which are obvious. They impose, in effect, a duty on employers to think deliberately about things which are not obvious."

ONR's Safety Assessment Principles

- Guidance provided to inspectors on risk assessment in Safety Assessment Principles (SAPs)
- Assessment of risks arising from nuclear facilities
  - Normal operation
  - Fault condition

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Three Pillars of Fault Analysis

- DBAA- Design Basis Accident Analysis:

  Providing a robust demonstration of the fault tolerance of the engineering design and the effectiveness of safety measures …FA10

- PSA- Probabilistic Safety Assessment

  help ensure the safe operation of the site and its facilities e.g. investigation of significant abnormal occurrences, developing and changing operating procedures and associated training…FA12

- SAA- Severe Accident Analysis

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What is a Severe Accident?

IAEA NS-G-2.15:

A Beyond Design Basis Accident comprises accident conditions more severe than a design basis accident, and may or may not involve core degradation, such accidents are termed severe accidents.

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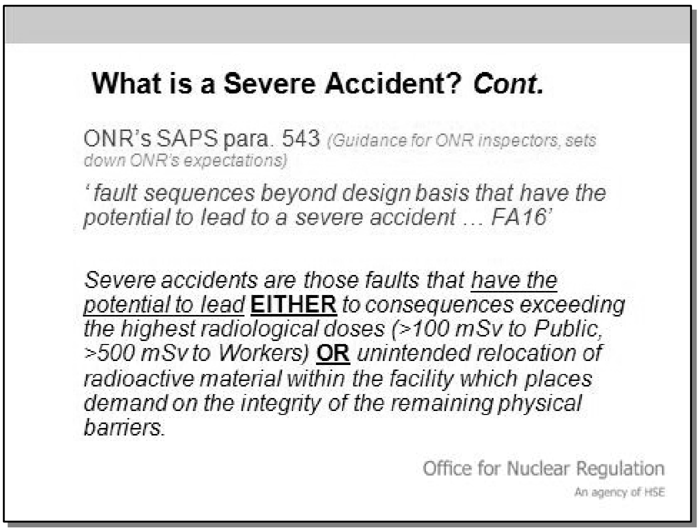
What is a Severe Accident? Cont.
ONR's SAPS para. 543 (Guidance for ONR inspectors, sets down ONR's expectations)
'fault sequences beyond design basis that have the potential to lead to a severe accident … FA16'
Severe accidents are those faults that have the potential to lead EITHER to consequences exceeding the highest radiological doses (>100 mSv to Public, >500 mSv to Workers) OR unintended relocation of radioactive material within the facility which places demand on the integrity of the remaining physical barriers.
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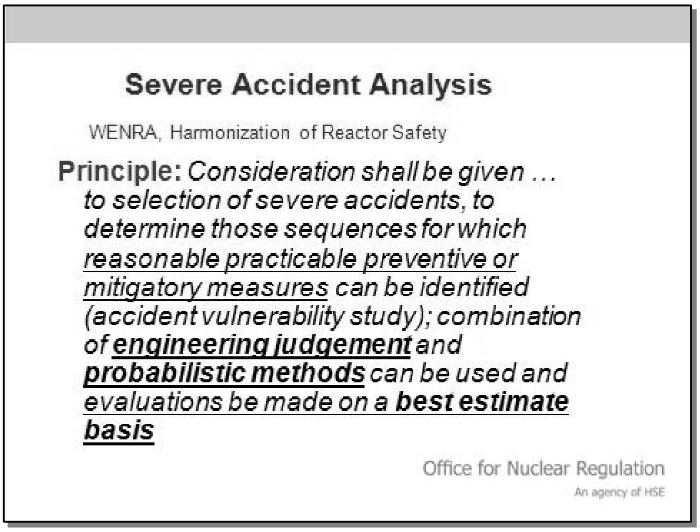
Severe Accident Analysis
WENRA, Harmonization of Reactor Safety
Principle: Consideration shall be given … to selection of severe accidents, to determine those sequences for which reasonable practicable preventive or mitigatory measures can be identified (accident vulnerability study); combination of engineering judgement and probabilistic methods can be used and evaluations be made on a best estimate basis
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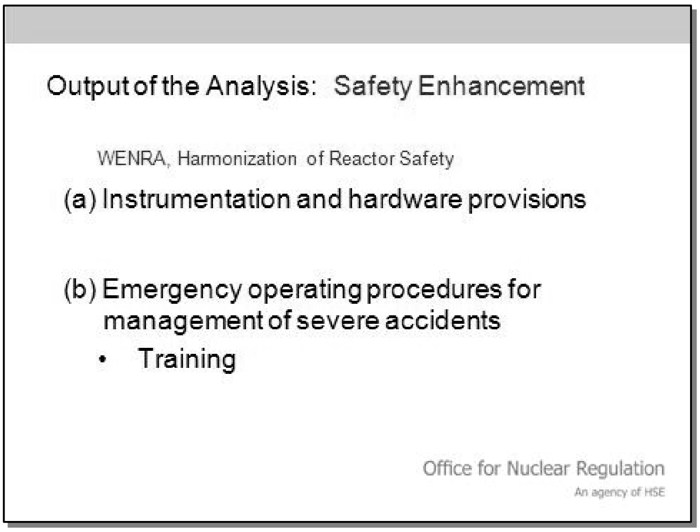
Output of the Analysis: Safety Enhancement
WENRA, Harmonization of Reactor Safety
(a) Instrumentation and hardware provisions
(b) Emergency operating procedures for management of severe accidents
• Training
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Safety Enhancement Cont.

SAPs:

Using best estimate approach analysis should determine:

(a) Magnitude and characteristics of consequences

(b) Consider cliff edge effects

TO

1- Identify reasonably practicable preventive or mitigating measures for BDBA

2- Provide a basis for emergency plans and severe accident management strategies

3- Support PSA of the facility

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Safety Enhancement Cont.

- Facility's accident management strategy
  - Development of a strategy for maintenance and examination of equipment required for emergencies
  - Development of strategies across the plant for specific fault groups such as loss of containment
  - Development of relevant training for personnel involved in responding to a severe accidents/emergency.
- For multi-facility sites, development of a strategy for dealing with external hazards at a site (rather than facility) level
- For multi-facility sites, development of a strategy for dealing with domino effect where an incident in one facility/area may have an impact on others
- Assisting in identification of vulnerable plant areas/systems

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Summary: SAA

- Applicable to design and operation phases of the plant
- Measures identified do not need to follow conservative engineering practice
- The analysis informs additional instructions that cover beyond design basis scenarios and identifies training needs for plant personnel involved in responding to such events

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A Methodology

- Identification of critical safety functions from the safety case/ other sources of technical data
  - Containment
  - Cooling
  - Criticality
  - Shielding
  - Control
- Identify candidate fault scenarios

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A Methodology (2)

- Three types of scenarios that should be addressed by SAA
  - Low frequency/ high consequence event beyond design basis
  - Design basis events where safety provisions have assumed to have failed
  - Scenarios omitted from the safety case
- Identifying control measures in place in support of the safety functions discussed

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Relevant examples

- Real Example: TOMSK 7
- Generic Scenarios with the potential to lead to SAs

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Real Example: TOMSK 7 incident

- Accident occurred in a chemical reprocessing facility in a closed town of TOMSK 7 in Siberia
- Explosion of a vessel containing U and Pu solution followed by a fire caused by short circuited electrical system
- The explosion was so violent that it led to collapse of the walls on two floors of the building and spread contamination over 120 km^2

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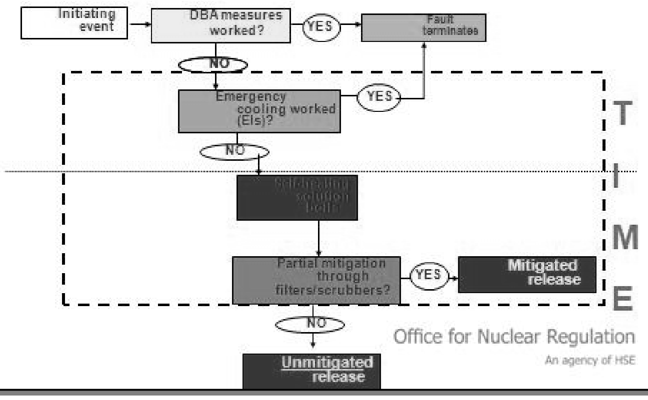
Example 1: Loss of Cooling



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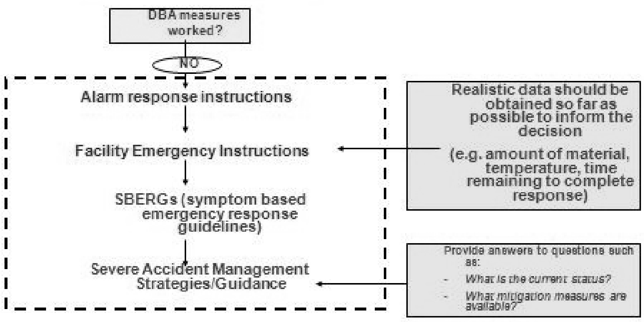
Layers of emergency instructions



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Example 2: runaway reactions

- Eruption in a superheated state

  → Relocation of material

  → off-site AND on-site consequences

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Domino effects

Faults in services or non-nuclear facilities on site

- – Utilities on site
- – Other non-nuclear chemical facilities on site

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Consideration given to Severe Accidents in other (non-nuclear) high-hazard industries

- UK's response to council directive (SEVESO II)
- COMAH Regs. require all duty holders to have a *Safety Report* in which all relevant Major Accident Hazard (MAH) scenarios are assessed (Reg. 7) **AND**
- Have a Major Accident Prevention Policy (MAPP) that specifically addresses the MAHs (Reg. 5)

Assessing the consequences of a major accident...
'This enable the operator (duty holder) to determine the depth of demonstration needed

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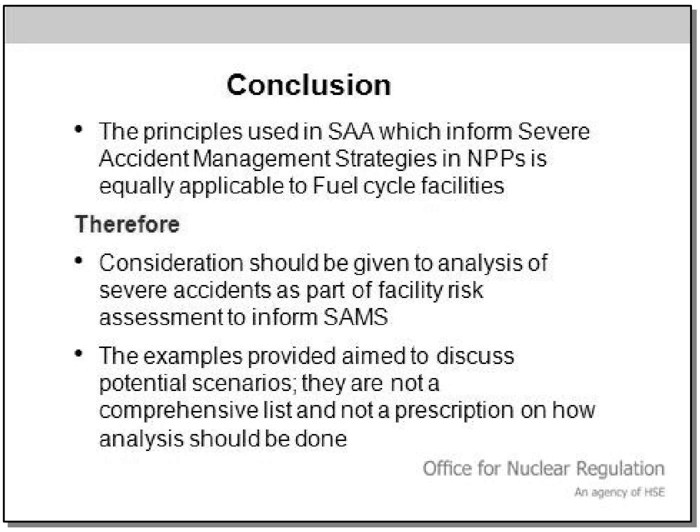
Conclusion
• The principles used in SAA which inform Severe Accident Management Strategies in NPPs is equally applicable to Fuel cycle facilities
Therefore
• Consideration should be given to analysis of severe accidents as part of facility risk assessment to inform SAMS
• The examples provided aimed to discuss potential scenarios; they are not a comprehensive list and not a prescription on how analysis should be done
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A COMPARISON OF INTEGRATED SAFETY ANALYSIS AND PROBABILISTIC RISK ASSESSMENT

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Kevin S. Mattern
Nuclear Regulatory Commission, United States

Abstract – The U.S. Nuclear Regulatory Commission conducted a comparison of two standard tools for risk informing the regulatory process, namely, the Probabilistic Risk Assessment (PRA) and the Integrated Safety Analysis (ISA). PRA is a calculation of risk metrics, such as Large Early Release Frequency (LERF), and has been used to assess the safety of all commercial power reactors. ISA is an analysis required for fuel cycle facilities (FCFs) licensed to possess potentially critical quantities of special nuclear material. A PRA is usually more detailed and uses more refined models and data than an ISA, in order to obtain reasonable quantitative estimates of risk. PRA is considered fully quantitative, while most ISAs are typically only partially quantitative. The extension of PRA methodology to augment or supplant ISAs in FCFs has long been considered. However, fuel cycle facilities have a wide variety of possible accident consequences, rather than a few surrogates like LERF or core damage as used for reactors. It has been noted that a fuel cycle PRA could be used to better focus attention on the most risk-significant structures, systems, components, and operator actions.

ISA and PRA both identify accident sequences; however, their treatment is quite different. ISA’s identify accidents that lead to high or intermediate consequences, as defined in 10 *Code of Federal Regulations* (CFR) 70, and develop a set of Items Relied on For Safety (IROFS) to assure adherence to performance criteria. PRAs identify potential accident scenarios and estimate their frequency and consequences to obtain risk metrics. It is acceptable for ISAs to provide bounding evaluations of accident consequences and likelihoods in order to establish acceptable safety; but PRA applications usually require a reasonable quantitative estimate, and often obtain metrics of uncertainty. This paper provides the background, features, and methodology associated with the PRA and ISA. The differences between the approaches are enumerated and their potential use in regulating fuel cycle safety is discussed. A critical evaluation of the application to FCFs including, hazards, completeness, adequacy, interactions, common causes, and personnel is performed. The application of both methodologies to various inspection and assessment tools is discussed. The regulatory advantages of the PRA, namely, the ability to quantify uncertainty and provide importance measures, are provided. The paper concludes that, while the ISA method is sufficient to establish an adequate safety basis, PRA is able to provide additional insights such as risk significance, uncertainty assessment, and prioritisation of safety features.

I. Integrated safety analysis background and description

A. Definition of integrated safety analysis

In 10 CFR 70.62(c), the NRC defines ISA as a systematic analysis, required for major fuel cycle facilities, that identifies hazards, accident sequences, their consequences, likelihoods, and IROFS. The rule does not mandate specific methods for performing such analysis, but guidance appears in NUREG-1520[19] and NUREG-1513[20].

B. Regulatory uses of integrated safety analyses

1. Performance requirements

ISAs are directly used for compliance with the performance requirements in 10 CFR 70.61. The ISA is used to identify all event sequences that could lead to high- or intermediate-consequence events, as defined in the regulation. The regulation specifies that high-consequence events must be highly unlikely, and intermediate-consequence events must be unlikely. The terms "highly unlikely" and "unlikely" must be defined by the licensee and reviewed and approved by NRC staff in accordance with the Standard Review Plan (SRP)[1]. This regulatory use of ISA differs from PRAs, which are used to inform decisions but not directly to demonstrate compliance with criteria specified by regulation.

2. Identification of items relied on for safety

The ISA process identifies a set of IROFS. When a structure, system, or component (SSC) is designated as an IROFS, certain regulatory requirements become applicable. These requirements include that the IROFS be sufficient to meet the likelihood and consequence requirements of 10 CFR 70.61. In addition, management measures must be applied to assure that IROFS are available and reliable. Changes to IROFS must be reported to the NRC annually.

3. Other applications of integrated safety analysis results

ISA results have sometimes been used for applications other than compliance with the regulation, a licensing function. For example, ISA results were used by the staff to prioritize IROFS to be inspected during the operational readiness reviews of the gas centrifuge enrichment plants. In addition, the licensees provide annual updates to their ISA summaries with IROFS lists, and maintain failure logs that are useful in guiding regular inspections.

C. Technical features of an integrated safety analysis

1. End states

End states of accident sequences are defined in 10 CFR 70.61 as high or intermediate consequences. Specifically, "high" and "intermediate" are defined in terms of rem for radiation doses, and by qualitative criteria, such as "endanger the life," for chemical health effects. Most accident sequences that are identified in ISAs as exceeding these consequence thresholds involve consequences to the workers rather than the public. Given such onsite events, ISAs typically assume that high consequences result and apply IROFS sufficient to make the event highly unlikely, rather than calculating consequences realistically. Offsite,

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[19] U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility," NUREG-1520, Rev. 1, May 2010.

[20] U.S. Nuclear Regulatory Commission, "Integrated Safety Analysis Guidance Document," NUREG-1513, May 2001.

consequences are more likely to be evaluated quantitatively. These offsite consequence evaluations are typically "worst case" rather than realistic estimates. In ISAs, total frequencies of fatality to individuals are not summed over all accidents.

2. Accident sequences leading to end states

ISAs must identify all potential accident sequences that could result in the end state consequences defined in 10 CFR 70.61. This is accomplished by using a variety of methods, including hazard and operability analysis, what-if checklists, fault trees, and event trees. Licensees list all of these sequences in the ISA summary submitted to the NRC and update them annually. Any credible event exceeding the consequence levels of the rule, whether hardware failure or human error, must be addressed in this accident identification task. All hazards, both internal and external to plant processes, must be considered. Event sequences may be screened out of the eventual list submitted to the NRC on the grounds that they cannot produce the consequences specified in the rule or are not credible.

3. Hardware failures and human errors

ISAs model both hardware failures and human errors. Hardware IROFS are usually identified at the subsystem rather than component level. For example, an IROFS could be defined as "an automatic control that stops a process given detection of a temperature out of range." ISAs using the risk index method generally assign indices based on simple qualitative criteria, such as passive, active, or administrative control (human error). Quantitative ISAs use more specific hardware descriptions, such as internal valve leaks, to assign failure and error frequencies and probabilities of failure on demand. These values are typically taken from generic data sources[21] [22]. Human error probabilities might also be estimated based on plant experience. Human reliability modeling is typically not applied.

4. Physical and chemical phenomena

All phenomena that could produce the consequences specified in 10 CFR 70.61 must be considered. However, except for calculating chemical and radiation exposures, physical and chemical phenomena involved in fuel cycle accidents usually do not require modeling or calculation to achieve the purposes of the ISA. For example, the magnitudes of criticality accidents can vary, depending on the initiating sequence of events. However, the ISA usually assumes that, if a criticality occurs, high consequences could result. Calculating a realistic estimate of total risk to individuals, as in a PRA, would require more detailed quantitative modeling of such phenomena, including estimating probabilistic variations in the magnitude and locations of the accidents.

5. Fires and external hazards

Fires and external hazards are evaluated as accidents in ISAs as initiating events potentially leading to either a radiological or chemical release. Fire safety is one of the technical disciplines normally represented on each ISA team. By rule, ISAs must consider external hazards as well as fire. The impact of fires, chemical releases, explosions, and similar events on the safety of processes other than those in which the event occurred must also be considered.

[21] Alber, T. G., et al., "Idaho Chemical Processing Plant Failure Rate Database", INEL-95/0422, August 1995.

[22] H.C. Benhardt, et al., "Savannah River Site Human Error Data Base Development for Nonreactor Nuclear Facilities," WSRC-TR-93-581, February 1994.

6. Plume dispersion

For most scenarios, ISAs use worst-case dispersion to determine if the offsite radiological or chemical thresholds of 10 CFR 70.61 are exceeded. Typical assumptions include stability class F, low wind speed, no heavy gas model, and no plume rise. Consequently, the magnitude of the doses is not an average or typical case but a worst case. Probabilistic weather averaging, as in the MELCOR Accident Consequence Code System (MACCS) code used for PRAs, is not used. This conservatism would have to be removed in order to obtain realistic risk significance. In many cases releases are simply assumed to produce high-consequence doses as defined in 10 CFR 70.61.

7. Quantification of accident sequences

Two of the approved ISAs quantify accident sequence frequencies. One ISA has no form of quantification but applies qualitative criteria to assure that IROFS are suitably reliable. The rest use a risk index method, which could be called semi-quantitative[20]. Worker doses, if not calculated, are often conservatively assumed to be high consequences. Offsite doses are often calculated conservatively using computer codes in order to determine if the regulatory thresholds are exceeded. These calculations are not probabilistically averaged over weather conditions. They are typically for worst-case source terms and weather. Not all ISA assessments are conservative, but, if so, are acceptable for assurance of safety under 10 CFR Part 70, Subpart H.

8. Uncertainties in physical and chemical phenomena

ISAs usually handle uncertainties in accident phenomena by making conservative assumptions. These uncertainties are not modeled probabilistically to estimate known variations. Epistemic uncertainties, as opposed to variations, exist in the initiation of some types of chemical accidents, such as unanticipated chemical reactions, gas evolution, or precipitations. Thus, rare events of these types are difficult to assess.

9. Importance measures

In an ISA, licensees do not routinely calculate importance measures, such as relative change in risk given that an IROFS failure probability is set to 1.0. Importance measures have been evaluated and used by NRC staff in a few applications, such as prioritizing which IROFS should receive more attention in inspections. A risk-significance metric has also been considered for use in determining the risk significance of inspection findings.

II. Probabilistic assessment background and description

A. Definition of probabilistic risk assessment in the reactor context

PRA is a systematic methodology to evaluate risks. Risk, in this context, refers to both probabilities and consequences of unintentional adverse events (i.e. accidents). In the NRC context, PRA has been applied to some NRC regulated nuclear activities, including all nuclear power reactors. PRA involves identifying potential accidents and quantifying the magnitude of their consequences and their probability or frequency of occurrence. Consequences are expressed numerically (e.g., the number of early fatalities or dollar cost impacts of the accident), and likelihoods of occurrence are usually expressed as frequencies. Collective risk metrics, such as the expected value of cost impacts, are calculated by summing the products of each accident's consequences (dollars) and its frequency.

B. Technical features of a probabilistic risk analysis

The end states and scope of PRAs vary, depending on the application of the results. The scope of a particular PRA application requires analysis of all operating modes and initiators that significantly affect the required risk metric. PRAs model accident sequences leading to the end states within the scope of the particular application that is using the PRA results. Within a particular scope, reactor PRAs aim to be complete in terms of the spectrum of potential initiating events and accident scenarios. This includes consideration of hardware failure rates and human error probabilities at a level of detail sufficient for quantification. Human reliability methods developed under NRC auspices have been applied to estimate operator error probabilities in scenarios requiring operator action.

Certain physical phenomena in reactor accident sequences need to be modeled in PRA sufficient to allow quantification of outcomes. For example, pressure and temperature challenges that accidents pose to containment must be quantified. For example, if releases from containment occur in a hypothetical sequence, the timing and amounts of isotopes released need to be quantified in order to determine offsite doses. Fires and other external challenges to safety systems are typically modeled in complete reactor PRAs. Plume dispersion is modeled realistically, considering probabilistic variations of weather in Level 3 PRAs.

PRAs quantify frequencies of accident sequences using computer codes that incorporate a variety of probabilistic models, such as event trees, fault trees, and reliability equations. Event trees and fault trees will later be referred to as they are used in some ISAs. This is not to equate event tree/fault tree modeling with PRA, but these are one pair of PRA techniques that can be useful for ISA. Applicable input for quantifying these modes is available from the extensive database of hardware failures for the existing reactor fleet. More recently, quantitative probability models other than the standard event tree/fault tree approach have been applied.

Typically, PRAs search for potential dependencies, common-cause failures, and systems interactions. Explicit methods and data for modeling dependencies in hardware have been developed and applied in PRAs. Similarly, human error models developed for reactor applications have explicit consideration of dependency between human errors. Uncertainty analyses have been performed for PRAs, but not universally. Importance measures have been developed and applied in some cases to facilitate such insights as identifying dominant risks or vulnerabilities. In sum, PRAs of reactors strive to provide a realistic quantitative calculation of risk metrics appropriate to their application and scope.

C. Probabilistic risk assessment in fuel cycle facilities

No facility wide PRAs have been carried out for fuel cycle facilities in the United States; however, some limited analysis has been performed focusing on particular accidents that identified common-cause failures and human errors as major contributors. Compared to nuclear power plants, a wider range of hazards is posed by fuel cycle plants, including toxic chemicals, explosions, hazardous chemical reactions, radiological releases, and inadvertent nuclear criticality accidents. In most fuel cycle accident scenarios, facility workers are the receptors. The fuel cycle facility geometry of multiple sources and multiple receptors differs from the reactor geometry.

Dedicated standby safety systems, as in reactors, are not the most common type of controls. Instead, process safety designs rely more on normal operating systems, operator actions, and passive features to cope with abnormal conditions. This is more analogous to nuclear power plants in low-power shutdown mode. Individual processes are characterized by many unique process and operations aspects, especially with respect to the diversity of human actions that are involved. Since PRA has not been performed for these plants, it remains to be seen what difficulties might arise in attempting to represent the system's processes and functions in sufficient detail to quantify end states realistically.

While PRA methodology can be used to estimate the overall likelihood of undesirable consequences (as defined in the PRA model), an additional important strength of PRA is the ability to better understand and rank the relative importance of each modeled component, system, or event. Such understanding can aid in several regulatory processes, including prioritizing licensing reviews, focusing inspection and routine oversight, and evaluating the significance of equipment failures or other events. It should also be noted that the traditional use in PRA of event trees and fault trees can present challenges to modeling process systems like those that exist at fuel cycle facilities.

III. Critical evaluation of integrated safety analysis and probabilistic risk assessment for safety under 10 CFR part 70

As previously indicated, some licensees use some PRA techniques in the ISAs. In principle, the desired results of the first phase of ISA or PRA are the same: identifying all relevant accident sequences. Once the first phase (accident identification) is complete, ISAs must evaluate compliance with the performance criteria of 10 CFR 70.61. The objective is to attain reasonable assurance that the set of IROFS limiting the likelihood or consequence of each accident sequence is adequate. To provide this assurance, a quantification of sequence frequencies may or may not be used. Some PRA methods are useful for quantification of accident frequencies and consequences in the ISA context.

1. Hazards at fuel cycle facilities

The nature and magnitude of hazards at fuel cycle facilities governed by the ISA requirement differ markedly from nuclear reactors. The designs of some types of safety controls are also quite different both from reactors and among processes within a facility. Principal hazards include toxic chemicals and fissile materials with the potential for inadvertent criticality. Radiological sources are, except for plutonium facilities, of very low magnitude. Thus, except for a few large chemical sources, most hazards do not pose a significant risk to members of the public offsite. Toxic chemicals are typically controlled through careful and robust containment. Criticality is often controlled by use of passive safe geometry equipment. For low-enriched uranium facilities, criticality can be controlled by independent controls on mass and moderation. Automatic controls are less common, as is dependence on power or other active support systems.

2. Completeness in identifying accident sequences

One potential problem in ISA or PRA is overlooking a potential accident. Instances of this have occurred in fuel cycle ISAs because the analysts either had not thought of a particular scenario or had incorrectly screened it out as not credible. However, these instances have not usually been a result of methodological differences between ISAs and PRAs. Under 10 CFR Part 70, the objective is to identify sequences and apply IROFS sufficient to limit risk, not to estimate risk per se. NRC staff reviews and oversight of ISAs have, so far, concluded that the ISAs have accomplished this objective overall and so have performed their function in the safety regulatory programme required by 10 CFR Part 70.

3. Establishing adequate controls for safety

Although ISAs do not necessarily provide quantitative estimates of IROFS failure rates and probabilities, the regulation does state that likelihoods of consequential events are to be made appropriately unlikely, hence acceptably safe. The ways that accidents and IROFS identified in ISAs are managed under the requirements of the rule provide this assurance of safety. In 10 CFR Part 70, the NRC requires that accident sequences be evaluated and shown to comply with the performance requirements of 10 CFR 70.61. 10 CFR 70.62, "Safety Program and Integrated Safety Analysis," requires that "management measures" be applied to each IROFS to ensure that it is sufficiently reliable and available. Required practices beyond management measures are listed as "baseline design criteria" in 10 CFR 70.64, "Requirements for New Facilities or New Processes at Existing Facilities," which are made mandatory for safety designs of new facilities or processes.

4. Process interactions

One challenge to assuring safety is to identify interactions between processes that may cause problems. This may happen when an upset in one process impacts other processes, or when safety features that address different hazards interact. The classic cases are (1) fire suppression that uses water providing moderator that could facilitate a criticality accident, and (2) chemical accidents affecting adjacent processes. The regulations explicitly require that ISAs analyze these types of interactions. In fact, this is what is meant by "integrated" in the phrase ISA.

5. Common cause and dependencies

For redundant hardware safety controls, the risk index method described in the original SRP had not explicitly recommended a method of common-cause correction like the beta factor method used in PRAs. However, the issue of the independence of controls arose early during performance of the ISAs, and NRC staff provided guidance in ISG-1, which has now been incorporated into Chapter 3 of the revised SRP[1]. Facility methods of modeling identical redundancy vary, from taking no credit for the second control to applying a dependency factor, as in the beta factor method. Licensees are very aware of common-cause and dependency issues because of the prominence of the "double contingency principle" in the basic American National Standards Institute/American Nuclear Society (ANSI/ANS) criticality safety standard, ANSI/ANS 8.1[23]. A commitment to apply the double contingency principle is often part of a fuel facility license.

6. Integrated safety analysis personnel issues

One licensee who applied PRA techniques to ISAs discussed this process in a paper, "Applying Nuclear PRA to a Nuclear Fuel Cycle Facility Integrated Safety Analysis," presented at Probabilistic Safety Assessment and Management Conference 10 in June 2010[24]. The paper points out the challenge that plant staff familiar with the safety design of processes are usually not familiar with PRA or ISA techniques. On the other hand, it takes time for PRA experts to become familiar with fuel facility hazards and processes because of their large number and diversity. This dichotomy of personnel experience may have more influence on ISA results than purely methodological ISA and PRA issues.

Table 1, summarizes each of the ISA and PRA technical features in the context of whether a more PRA-like analysis would produce a better ISA result with respect to the ISA's function of assuring safety.

[23] American Nuclear Society, Nuclear, "Criticality Safety in Operations with Fissionable Materials Outside Reactors", ANSI/ANS 8.1, 1998.

[24] Matthew Warner and Jim Young, "Applying Nuclear PRA to a Nuclear Fuel Cycle Facility Integrated Safety Analysis," presented at Probabilistic Safety Assessment and Management Conference 10, June 2010.

Table 1**. Evaluation of ISA-PRA differences for fuel cycle safety**

Technical Features or Topics	ISA	Hypothetical PRA for Fuel Cycle	Implication for Safety or Compliance
End states	high or intermediate consequences (see 10 CFR 70.61)	could use more refined consequences than found in the ISA	10 CFR Section 70.61 acceptable for current facilities, may need to be supplemented for risk significance determinations
Completeness of accident sequences	Uses various systematic methods	uses various systematic methods	In principle no difference.
Quantification of accident sequences	a few ISAs are quantified, most use risk index method	Quantified accident sequences frequencies	ISAs generally acceptable. Quantification might be helpful in marginal cases.
Modeling of physical/chemical phenomena	ISAs often use conservative assumptions	PRA could quantify some phenomena	ISAs generally conservative, which is acceptable. Some accidents may be mis-categorized due to lack of quantitative understanding of a phenomenon.
Offsite consequences	ISAs use bounding weather assumptions	Level 3 PRAs use realistic statistical consequences	conservative approach is adequate for safety, PRA might allow relaxations in some cases
Internal fire modeling	ISAs always consider fire scenarios and interactions	PRA not necessarily different	in principle no difference, except ISA assessment is usually not quantitative.
Level of detail in modeling	ISAs often use simplified models	PRA could have more detail	detail is not usually needed for safety; but detail may lead to better understanding.
Treatment of hardware failures	hardware failures are addressed at subsystem level.	often more detail in models	detail may provide better understanding of failure likelihood.
Treatment of human errors	some ISAs are simplistic and have only one value for human error	PRA could attempt modeling, but basis may not exist for many scenarios	this is an undeveloped area for some situations that occur at fuel facilities.
Completeness of safety control systems analyzed	some ISAs do not take credit for all safety controls as IROFS	PRA would credit additional controls besides those credited as IROFS	not crediting all controls is acceptable for assuring at least minimal safety under 70.61, but other safety principles may apply.

Technical Features or Topics	ISA	Hypothetical PRA for Fuel Cycle	Implication for Safety or Compliance
Treatment of dependency and system interactions	dependencies considered in double contingency[23] analysis, sometimes quantitatively	PRA explicitly model dependencies across multiple systems.	in principle, no difference, but risk index method does not have dependency analysis built-in, but must be added via double contingency or other analysis.
Risk metrics	ISAs assess individual accident sequences, not risk to individuals	PRA could sum risk to individuals	avoids problem of number of sequences, but excessive numbers not a common problem with ISAs
Uncertainty and importance measure evaluation	ISAs do not quantify uncertainty or importance, but ISA results have been used for importance evaluation	PRAs often include uncertainty analysis and can produce several types of importance measures for modeled events.	Uncertainty assessment might be important for cases where safety is marginal, or there is very large uncertainty. Understanding the relative importance of plant systems, components, and events can aid regulatory focus and priority.

IV. Potential application of ISA and PRA methods in significance determination for fuel cycle oversight

If the NRC were to revise the oversight process for fuel cycle facilities to be risk-informed and systematic, one required element would be a realistic and predictable process for assessing the risk significance of inspection findings. This process could use qualitative and quantitative risk insights to evaluate the significance of licensee performance deficiencies. The determination of risk significance within the process could be conducted in phases. The initial phase would be a screening review, based on qualitative criteria, to identify those findings that would clearly not result in a significant increase in risk. Based on a test analysis of past inspection findings, the NRC staff anticipates that a majority of findings would be screened out by this initial qualitative process. For the remaining set of inspection findings, the effect on the likelihood and consequences of accident sequences could be evaluated in more detail.

A quantitative (or more detailed qualitative) risk assessment would categorize the findings into broad categories based on its risk impact. Fuel cycle facility processes are capable of producing accidents with a variety of consequences, such as direct doses from nuclear criticality, doses from exposure to radioactive material, and various effects from exposure to toxic chemicals. Because worker safety plays a large role in the NRC's regulation of fuel cycle facilities, accidents at these facilities can affect multiple categories of receptors, the public outside the controlled area, as well as workers. Thus multiple risk significance metrics will, in principle, have to be used. In practice however, a single safety deficiency typically only affects one type of accident in one unit process. For example, a retention dike under a process containing fissile solution assures that a leak or overflow from the process assumes a sub-critical geometry, thus preventing a criticality accident. If such a dike were disabled for some period of time, an additional risk of large radiation dose from a criticality beyond that planned in the design would be imposed on nearby workers. One metric of risk significance would be this additional risk of large dose imposed on the nearest worker. The metric would be the increase in frequency of criticality with the dike disabled times the duration of the disabled condition. This quantity is thus a probability of the accident that was incurred by the worker because of the disabled dike. Typically, such a deficiency only affects a few accident sequences, in this case those causing a leak or overflow of fissile solution into the dike. The limited scope of processes and accident sequences affected by a single deficiency may thus make it feasible for NRC staff to perform such risk significance evaluations for deficiencies on a case-by-case basis using realistic risk evaluations, even though the ISA for the process may not have been fully quantitative, or may have been very conservative. A full safety significance determination process and quantitative risk significance approach remains to be developed and tested. It could involve a mixture of qualitative and quantitative methods, depending on the situation to be analyzed and availability of information.

Conclusion

Fuel cycle ISAs and reactor PRAs are performed for different purposes. Some ISAs have used several PRA methods extensively, and other ISAs have used them selectively, as recommended in NRC guidance[20]. ISAs were not performed to estimate risk as PRAs do. ISAs were performed to identify potential accident sequences, designate IROFS to prevent or mitigate them, and describe management measures to be applied to assure IROFS reliability and availability. As a result of substantial reviews of ISAs which have been approved, NRC staff has concluded that the ISA methods and processes have succeeded in meeting this objective and are acceptable for assuring safety under 10 CFR Part 70. This does not preclude that ISAs of specific processes may contain one of the potential deficiencies previously mentioned, thus PRA methods should continue to be explored during future efforts in assessment of fuel cycle safety.


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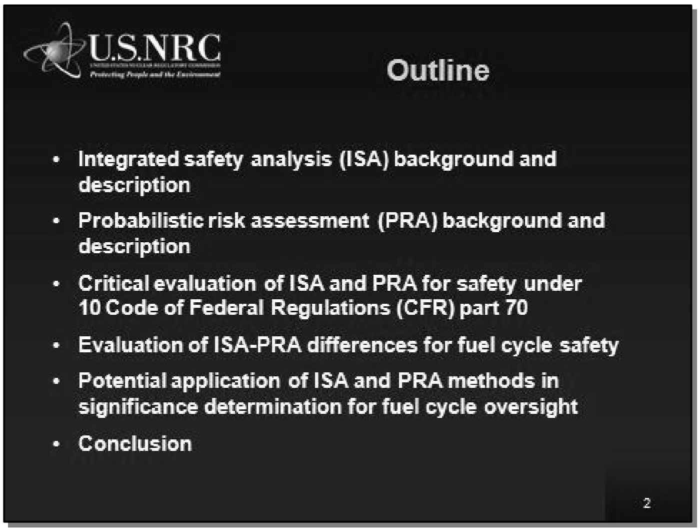
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Outline
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• Evaluation of ISA-PRA differences for fuel cycle safety
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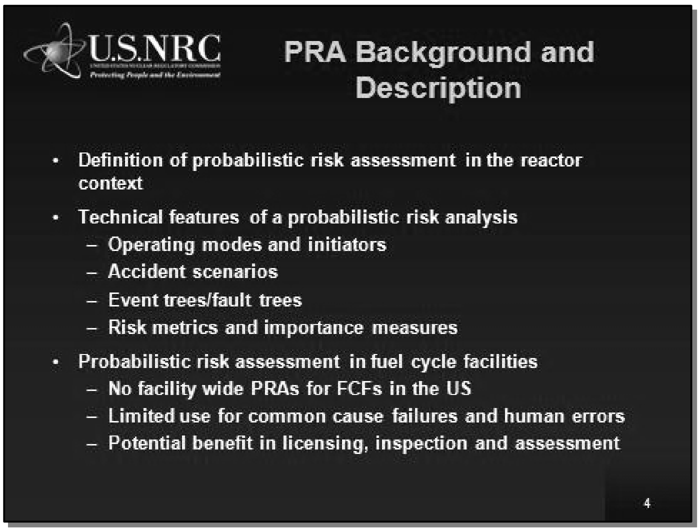
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– No facility wide PRAs for FCFs in the US
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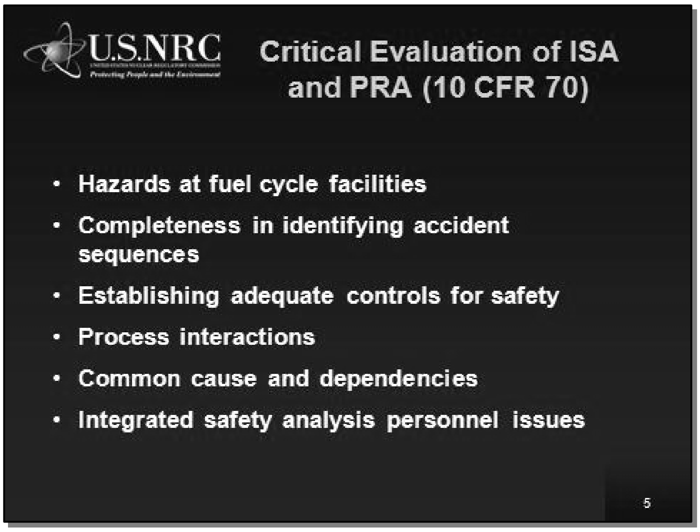
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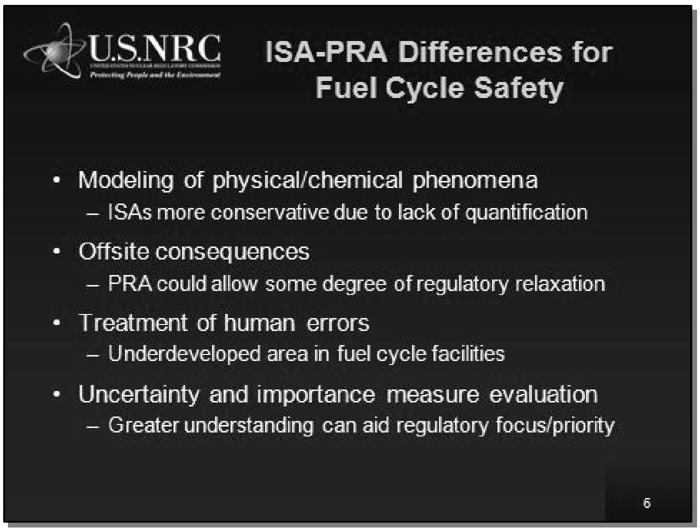
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ISA-PRA Differences for Fuel Cycle Safety
• Modeling of physical/chemical phenomena
– ISAs more conservative due to lack of quantification
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ISA and PRA in Significance Determination

- Realistic and predictable process for assessing the risk significance of inspection findings
- Use qualitative and quantitative risk insights to evaluate the significance of licensee performance deficiencies
- Risk significance evaluations for deficiencies on a case-by-case basis using realistic risk evaluations
- A mixture of qualitative and quantitative methods, depending on the situation to be analyzed and availability of information

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Conclusion

- Fuel cycle ISAs and reactor PRAs are performed for different purposes
- ISAs were not performed to estimate risk as PRAs do
- ISAs identify potential accident sequences, designate IROFS to prevent or mitigate them, and describe management measures to be applied to assure IROFS reliability and availability
- ISA methods and processes have succeeded in meeting their objective and are acceptable for assuring safety under 10 CFR Part 70
- ISAs of specific processes may contain some potential deficiencies
- PRA methods should continue to be explored during future efforts in assessment of fuel cycle safety, including significance determination

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Questions?

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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[25], 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[26] 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).

[25] U.S. Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," *Federal Register*, Vol. 60, August 16, 1995, p. 42622 (60 FR 42622).
[26] NRC Web Glossary : http://www.nrc.gov/reading-rm/basic-ref/glossary.html (accessed on July 13, 2011)

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." [2]

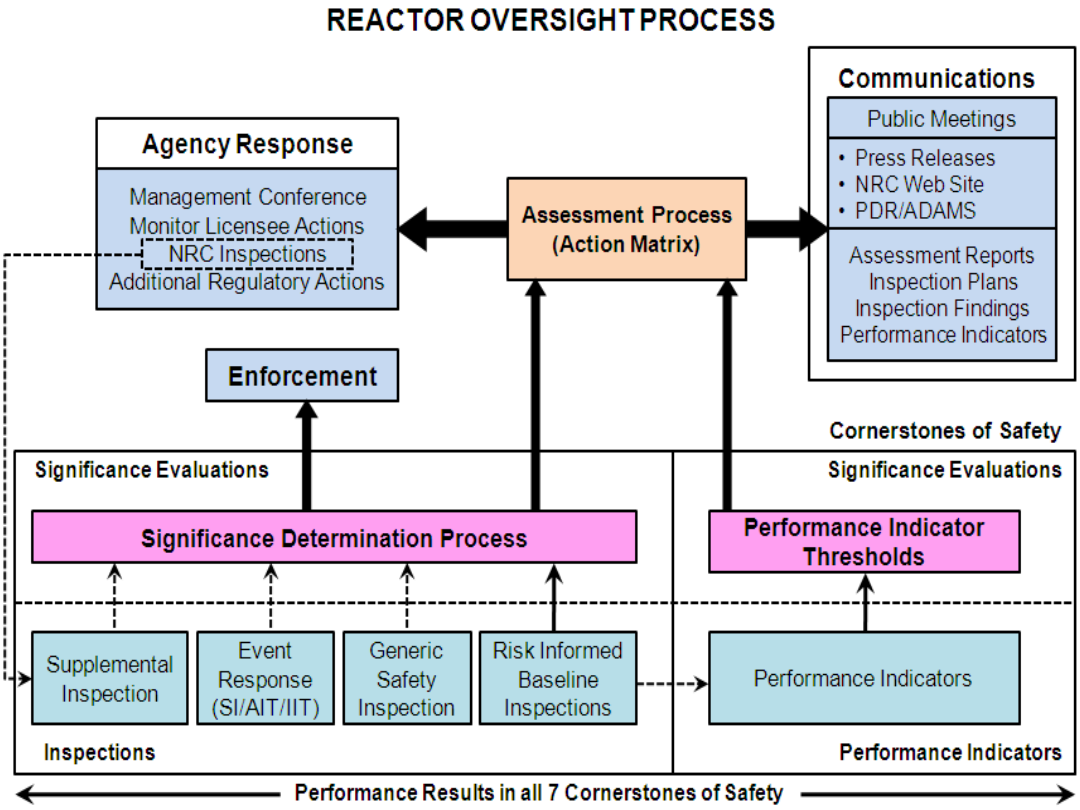
For nuclear power plants there are three levels of PRA: Level 1 PRA, Level 2 PRA and Level 3 PRA. Each level considers a sequential step in risk assessment processes as described below:

- "A Level 1 PRA models the various plant responses to an event that challenges plant operation." [2] 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.

Figure 1. **Reactor Oversight Process Overview**[27]



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:

[27] Reactor Oversight Process (ROP) website: http://www.nrc.gov/NRR/OVERSIGHT/ASSESS/index.html (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[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

[28] U.S. Nuclear Regulatory Commission; "NRC Inspection Manuals" http://www.nrc.gov/reading-rm/doc-collections/insp-manual/manual-chapter/ (accessed on July 13, 2011)

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.[4] 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[2]; 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[29].

Some of the tools used for the SDP are inspection manual guidance[4], 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 trough 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.

[29] ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RA-S-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.

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

[30] Hickman, J.W.; et al., "PRA Procedures Guide: A Guide to Performance of Probabilistic Risk Assessments for Nuclear Power Plants"; NUREG/CR-2300; January 1983.

[31] Science Applications International Corporation; "Nuclear Fuel Cycle Facility Accident Analysis Handbook"; NUREG/CR-6410; March 1998.

[32] U.S. Nuclear Regulatory Commission, "Risk-Informed Decision Making for Nuclear Material and Waste Applications" Revision 1, February 2008; ML080720238.

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.

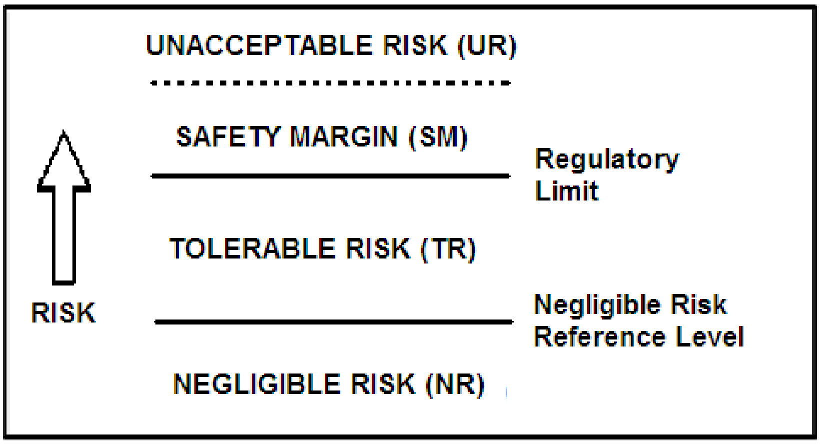
6. Available Guidance

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.

Figure 3. **Four region risk diagram**[6]



[33] NUREG-1520, "Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility NUREG-1520", Revision 1, May 2010.

[34] U.S. Nuclear Regulatory Commission, "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement; Republication." *Federal Register*, Vol. 51 (4 August 1986): 28044/30028.

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[35].

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).

[35] U.S. Nuclear Regulatory Commission; “Revising the Fuel Cycle Oversight Process”; SECY 10-0031; March 19, 2010.

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 storage[37], wet storage[38], and transportation of nuclear waste[39]. Other work includes limited preliminary work that has been performed in FCF oversight[40, 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.

[37] U.S. Nuclear Regulatory Commission; "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant"; NUREG-1864; March 2007; ML071340012.

[38] U.S. Nuclear Regulatory Commission; "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants"; NUREG-1738; February 2001; ML010430066.

[39] Sprung, J.L.; Ammerman, D.J.; Breivik, R.J.; ET. Al; Sandia National Laboratories for U.S. Nuclear Regulatory Commission; "Reexamination of Spent Fuel Shipment Risk Estimates"; NUREG/CR-6672; March 2000; ML003698324.

[40] U.S. Nuclear Regulatory Commission; "Summary of Meeting between the U.S. Nuclear Regulatory Commission Staff, Nuclear Energy Institute and Fuel Cycle Facilities Representatives Concerning Enhancements to the Fuel Cycle Oversight Process"; June 22, 2011; ML111710119.

[41] NRC Memorandum, "Paper Comparing Integrated Safety Analysis for Fuel Cycle Facilities and Probabilistic Risk Assessment for Reactors," dated December 15, 2010 (ML103330474).

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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

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Introduction

- Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities (1995)
- 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".

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Probabilistic Risk Assessment

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

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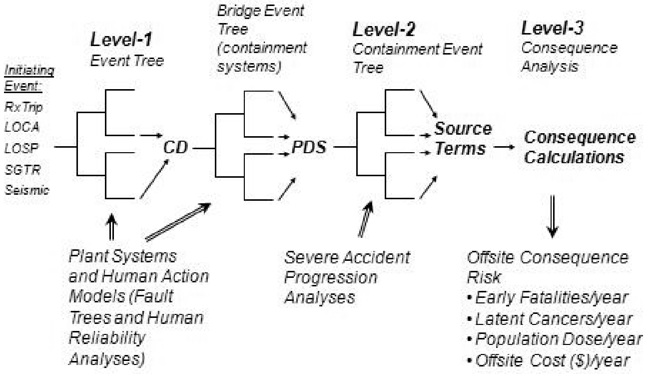
Probabilistic Risk Assessment



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Reactor 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

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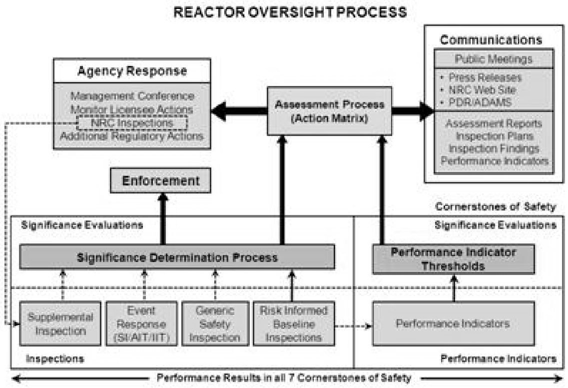

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Reactor Oversight Process

- 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

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Reactor Oversight Process

- 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

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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

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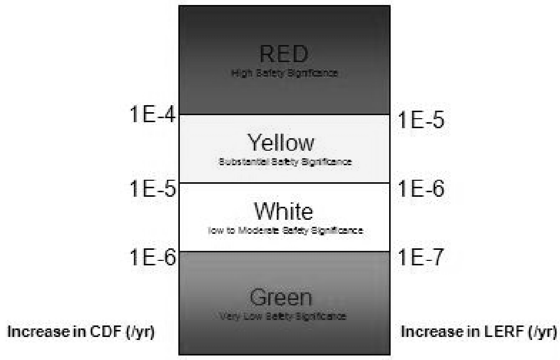
Significance Determination Process (SDP)



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Non-Reactor Nuclear PRA

- Storage and transportation
- Fuel cycle facilities (FCFs)
- Advanced reactors
- Waste applications

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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.

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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.

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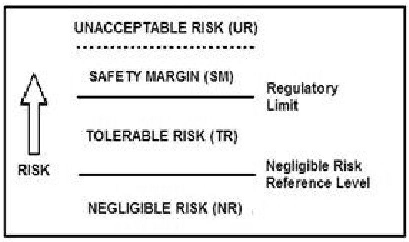
Non-Reactor Nuclear PRA

Four region risk diagram



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Oversight Process Application

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

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Challenges

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

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Available NRC Guidance

- NUREG-1520: *Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility**
- *Risk informed Decision Making for Nuclear Material and Waste Applications (RIDM)**
- NUREG/CR-2300: *PRA Procedures Guide: A Guide to Performance of Probabilistic Risk Assessments for Nuclear Power Plants**

*Available at www.nrc.gov

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Questions??

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Acronyms:

- CD- Core damage
- FCF- Fuel cycle facility
- LOCA- Loss of coolant accident
- LOSP – Loss of offsite power
- PDS- Plant damage state
- ROP- Reactor oversight process
- SDP- Significance determination process
- SGTR- steam generator tube rupture

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SESSION TWO

FRONT END FACILITIES

LESSONS LEARNED FROM RECENT SAFETY RELATED INCIDENTS AT A CANADIAN URANIUM CONVERSION FACILITY

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Abstract – This paper presents the Canadian Nuclear Safety Commission's (CNSC) regulatory requirements for nuclear fuel facility licensees to report any situation or incident that results or is likely to result in a hazard to the health or safety of any person or the environment and to submit its incident investigation report with cause(s) of the incident and corrective actions taken or planned. In addition, the paper presents two recent safety-related incidents that occurred at a uranium conversion facility in Canada along with their consequences, causes, corrective actions and any lessons learned. The first incident resulted in a release of uranium hexafluoride (UF_6) inside the UF_6 cylinder filling station and the second one resulted in a spill of uranium tetrafluoride (UF_4) slurry inside the UF_6 plant. Both incidents had no impact on the workers or the environment.

1. Introduction

The Canadian Nuclear Safety Commission (CNSC) is a federal agency whose mandate is to regulate the use of nuclear energy and associated materials in Canada. CNSC's regulatory philosophy is founded upon two principles:

Those persons and organisations that are subject to the *Nuclear Safety and Control Act* and its associated *Regulations* are directly responsible for ensuring that the regulated activities that they engage in are managed so as to protect the health, safety and security of Canadians and the environment; and to implement Canada's international commitments on the peaceful use of nuclear energy.

The CNSC is responsible to the public for regulating persons and organisations that are subject to the *Nuclear Safety and Control Act* and its associated *Regulations* in order to assure that they are properly discharging their obligations.

The CNSC's Directorate of Nuclear Cycle and Facilities Regulation regulates Canada's uranium mining, processing and fuel fabricating facilities, and other nuclear facilities. The uranium mining facilities are located in the province of Saskatchewan, and the uranium refining, conversion and fuel fabrication facilities are located in the province of Ontario, in Canada.

Each fuel cycle facility licensee has regulatory reporting requirements which include the obligation to report to the CNSC any situation or incident that results or is likely to result in a hazard to the health or safety of any person or the environment. The licensee submits a preliminary report immediately and a final report within 21 days after becoming aware of the incident or situation. In all cases, the licensee is required to remedy the hazardous situation (or non-compliance) in a timely manner and to take all necessary measures to prevent recurrence. CNSC staff reviews these event reports to ensure that the licensee has effectively implemented corrective actions and lessons learned in the established time period. Depending on the safety significance of the reported incident and efforts made by the licensee to achieve compliance, additional enforcement actions may be taken by the CNSC.

2. Two Recent Safety Related Incidents at a Canadian Uranium Conversion Facility

2.1 First Incident: Release of UF_6 during Filling of a 48X UF_6 Cylinder

Description of Processes

The UF_6 production plant consists of a chemical process that converts uranium trioxide to the final UF_6 product.

The final transportation packages are generally one of three types, a 48X (10 tonne UF_6 capacity), a 48Y (14 tonne UF_6 capacity) or a 30B (2 tonne UF_6 capacity). All of these transportation packages are designed, inspected and maintained in accordance with ANSI N14.1: American National Standard for Packaging of Uranium Hexafluoride for Transport. The cylinder type involved in this event was a 48X.

Transportation cylinders are inspected upon receipt and transferred into the UF_6 plant for filling. There are three side-by-side loading stations in the cylinder filling room. At the filling station, the cylinder is placed on a carriage and is connected to the UF_6 drop line by means of a threaded flexible connection referred to as a pigtail, which is insulated and steam heat traced. After the pigtail has been connected to a cylinder, it is pressure tested to ensure that there are no leaks. Remotely operated drop line valves and a cylinder valve closer can be activated in the cylinder filling control room in the event of a leak at the cylinder filling area.

Description of Incident

On June 20, 2010 at approx 02:00 hours, while transferring liquid uranium hexafluoride (UF_6) from a cold trap into a 48X cylinder at the filling station A, the solid piping connection to the flexible pigtail connection leaked. The leak was evident as smoke which is formed when UF_6 comes into contact with moisture in the air. The cylinder filling operator immediately stopped the transfer, switched the cylinder to a cold trap that was under vacuum to draw any material in the line back into the cold trap. Although the transfer was stopped, the smoke in the cylinder filling bay area exited into the adjacent cylinder filling room.

The emergency ventilation was activated and contact made with plant supervision and management to notify them of the incident. A decision was made to shut down the UF_6 plant until the cause of the leak could be determined and corrective actions, if necessary, could be implemented. Recognizing that additional resources would be required to disconnect the pigtail using A-suits, the Emergency Response Team (ERT) was placed on stand-by and the emergency response vehicles were brought to the UF_6 plant as a precautionary measure

A plan was developed and implemented to have the ERT enter the cylinder filling bay wearing A-suits and tighten the nut on the connection of the pigtail to the hard-piped drop line. The drop line and pigtail would then be pressurized and the pressure would be observed for two minutes. If the pressure did not drop, then it could be assumed that the tightening of the connection had eliminated the leak. This activity was successfully completed; the pigtail was disconnected from the cylinder and a metallurgical engineer was called in to assess the faulty connection.

The metallurgical engineer examined the connection and concluded that the problem was related to the tightening of the pigtail connection at the drop line end.

A review of the events was conducted and it was established that after the initial pressure test was performed, the cylinder carriage was moved to ensure proper operation of the cylinder valve closer. This movement occurred with the pigtail connected. There was no pressure test performed after this activity was completed to ensure that this had no impact on the integrity of the pressure connections.

Based on the examination of the pigtail connection and the review of events, additional work instructions were developed to ensure that both ends of the pigtail were pressure tested before a transfer and the plant was restarted.

Implications and Actual Consequences of the Incident

In this instance, the immediate detection and response by the operator limited the impact to the in-plant uranium levels. The system contained the uranium and fluoride materials and directed them through the emissions control systems. Based on the following monitoring data, no impacts to the environmental and the health and safety of workers are expected:

(a) Uranium and fluoride emissions from the main facility stack for this period were approximately 5 g U/h and 29 g HF/h. These are significantly below the licensed action levels of 50 g U/h and 330 g HF/h.

(b) In-plant airborne uranium levels immediately after the event were elevated and ranged from 3.8 to 6.4 Derived Air Concentrations (DAC). A DAC is equivalent to100 µgU/m^3. With the application of emergency ventilation system, these levels returned to normal levels within a few hours as confirmed by the sampling results of the next shift.

(c) Post-shift urinalysis results for operating and ERT personnel were below the urinalysis screening levels of 13 µg U/L and 4 mg F/L. The highest results were 5 µg U/L and 2.3 mg F/L. These are normal levels indicating no impact on workers.

Causes of Incident

Upon successfully completing the pressure test, the operator attempted to install the automatic valve closer and felt that it would not be fully engaged if it was required to be operated. The operator then moved the cylinder carriage with the pigtail connected. It is most likely that the pigtail to drop line connection was loosened at this point. No pressure test was performed after this move. A review of the operating procedure indicates that the procedure does not require a leak test be conducted as the last step prior to filling a cylinder.

Upon examination of the connection, it was concluded that the problem was related to the tightening of the pigtail connection at the drop line piping end,

Human performance (operator’s care) for high risk activities like cylinder filling was also identified as one of the causes of this incident.

Corrective Actions Taken to Prevent such Incidents

As requested by CNSC staff, the cylinder filling operating procedure was revised by the licensee to require the operators to conduct a leak check on both connections to the pigtail and to stipulate that any adjustments to the cylinder, carriage or pigtail after the leak check will necessitate that a leak test be repeated prior to filling the cylinder. This remedial action is considered adequate to prevent a recurrence of such a leak.

The preventive maintenance routine for installing a new pigtail was revised to ensure that it is re-tightened after it is installed and allowed to heat up to operating temperature conditions.

In addition, detailed operating instructions for high risk activities in the plant were developed by the licensee and posted in the field to improve human performance (operator's care).

Lessons Learned from the Incident

Based on the above incident investigation, the following lessons were learned by the licensee:

1. All high risk activities that require additional controls and operator's care are clearly identified and effectively communicated to operators by the licensee management.
2. Licensees provide redundant controls for high-risk operations and ensure that such controls are maintained as designed.
3. Development and posting of work instructions near the high risk operational activities will reduce the likelihood of operator error while performing these activities.
4. Operating procedures for high risk activities are maintained up to date and followed by the operators. Any deviations from the procedure are to be first discussed with and approved by the supervisor before proceeding.
5. Human performance issues are considered during the incident investigation and corrective action process.
6. Operating procedures or work instructions are improved based on human performance issues discovered during incident investigations

2.1 Second Incident: Release of UF_4 Slurry from a Diaphragm Valve

Brief description of the UF_4 production system:

Uranium tetrafluoride slurry (UF_4) is produced as an intermediate product in the uranium hexafluoride (UF_6) plant.

The piping material specification for the lines conveying this slurry specifies carbon steel components with polypropylene lining because of corrosive nature of the slurry. The slurry lines are designed for a pressure of 150 psig at 220^o F. The slurry tank is a rubber lined, carbon steel vessel.

Description of Incident

During normal operating conditions, on March 12, 2010 at approximately 17:45, an UF_4 slurry release was detected by an operator during the shift changeover. The UF_4 slurry leak occurred due to failure of a diaphragm valve in a slurry line. Three indicator alarms associated with the UF_4 slurry production system were triggered during the period when the spill incident occurred. However, the production operations of the production circuit were only shutdown after an area operator discovered the slurry leak during a routine round. The areas affected by the slurry leak were roped off and posted as respirator areas. The spilled slurry was allowed to be dried up naturally before initiating and completing the clean-up.

Implications and Actual Consequences of the Incident

There were no environmental impacts from this event. No worker received an uptake of uranium or fluoride as a result of this event.

- Causes of Incident;
- The investigation of the failed valve indicated the following causes of the event;
- The slurry leak occurred due to a process created leak path in the body of a diaphragm valve;
- The failed valve body was the result of the erosion action of the normal UF_4 slurry being conveyed through the piping system;
- No preventive maintenance (PM) was performed on the valve to ensure its integrity;
- Corrective Actions Taken to Prevent Such Event.

As requested by CNSC staff, the licensee reviewed the findings and recommendations of the event investigation and took the following corrective actions:

Created a PM procedure to routinely inspect piping and valves used in the UF_4 slurry service.

All short radius elbows in current piping handling the UF_4 slurry have been removed.

Installed additional HF detectors in the UF_4 slurry production circuit to early detect and alarm any leaking equipment.

Lessons Learned from the Incident:

Based on the above investigation of the incident, the following lessons were learned and implemented by the licensee as requested by CNSC staff:

1. The licensee should apply its existing in-service inspection requirements for safety critical piping and equipment to the piping and equipment in the UF_4 slurry production circuit as well, even though all piping and vessels in UF_4 service are lined with plastic because of the corrosive and abrasive nature of the UF_4 slurry.
2. The licensee should revise its in-service inspection and PM procedures to include periodic inspection of piping and equipment handling UF_4 slurry at the conversion facility.
3. Any piping and equipment handling UF_4 slurry should be designed with a particular focus on performance and leak detection.

3. CONCLUSIONS

From CNSC's perspective as a regulator, these 2 incidents demonstrated that:

Preventive maintenance and in-service-inspection procedures must be applied uniformly to all piping and equipment used in the UF_6 production circuits.

Systematic verification of leak checks is made after making any changes to piping connections in the UF_6 production circuits.

Where there is a potential for leaks of hazardous substances in the UF_6 production circuits, early leak detection and alarm device must be provided for early intervention by a control room operator or emergency response to stop the leak.

Human performance issues must be considered during incident investigations and corrective action processes at the conversion facility.

Lessons learned from each incident must be implemented in a timely manner as part of continuous improvement process at a conversion facility.

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OECD/NEA-WGFCS International Workshop on Safety Assessment of Fuel Cycle Facilities
Toronto, Canada
(27-29 September, 2011)

Lessons Learned from Recent Safety Related Incidents at a Canadian Uranium Conversion Facility

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September 27-29, 2011

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OUTLINE OF PRESENTATION

- Introduction to Canadian Nuclear Regulator and its Reporting Requirements
- Introduction to Canadian Fuel Cycle Industry
- Recent Safety-Related Incidents (2) at a Uranium Conversion Facility
- CNSC's Requirements and Expectations
- Licensee's Practices before the Incidents
- What Caused the Incidents (GAPs)
- Licensee's Practices after the Incidents
- Overall Lessons Learned from the Incidents

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Introduction to Canadian Nuclear Safety Commission - Regulator

- The Canadian Nuclear Safety Commission (CNSC) is a federal agency whose mandate is to regulate the use of nuclear energy and associated materials in Canada
- CNSC's Directorate of Nuclear Cycle and Facilities Regulation regulates Canada's uranium mining, processing and fuel fabricating facilities, and other nuclear facilities

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Nuclear Regulator (Cont. 2)

- Nuclear Fuel Cycle Facilities are regulated by the CNSC under the *Nuclear Safety and Control Act and its Regulations:*
  - the *Class I Nuclear Facilities Regulations*
  - the *General Nuclear Safety and Control Regulations*

  (For all activities from site preparation, construction, operation, decommissioning, and abandonment phases)
- Reporting requirements are specified in the CNSC's *General Nuclear Safety and Control Regulations* and the Operating Licence for each nuclear fuel facility
- Both incidents were reported under a licence condition to report an incident that is likely to result in a hazard to health and safety of any person or the environment

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Introduction to Canadian Fuel Cycle Industry

- Currently, five CNSC Licensed Fuel Facilities in Canada
- Cameco Corporation (Cameco) owns and operates three facilities:
  - Uranium Refining Facility in Blind River, ON
  - Uranium Conversion Facility in Port Hope, ON
  - Uranium Fuel Fabrication Facility in Port Hope, ON
- GE-Hitachi Nuclear Energy Canada Inc. (GE-Hitachi) owns and operates the other two fuel facilities:
  - Uranium Fuel Pellets Production Facility in Toronto, ON
  - Fuel Bundles Production Facility in Peterborough, ON

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Cameco Port Hope Conversion Facility Location

- uranium conversion facility located in Port Hope, Ontario
- 100 kilometres east of Toronto on Lake Ontario
- produces uranium hexafluoride (UF_6) and uranium dioxide (UO_2) from (UO_3) powder received from Cameco's Blind River Refinery

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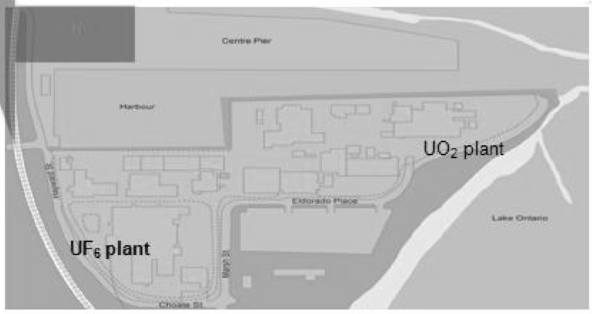
Cameco's Port Hope Conversion Facility's Layout



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Cameco's Port Hope Conversion Facility's Layout (Aerial View)

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Two Recent Safety-Related Incidents at a Canadian Uranium Conversion Facility

- Both incidents occurred at Cameco's Uranium Hexafluoride (UF_6) Conversion Plant in Port Hope
- 1st Incident on June 20, 2010: Release of uranium hexafluoride (UF_6) during cylinder filling
- 2nd Incident on March 12, 2010: Release of uranium tetrafluoride (UF_4) slurry from a diaphragm valve

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First Recent Incident and its Safety Significance

- Description of UF_6 Production Process and Safety
- On June 20, 2010, while transferring liquid UF_6 from a cold trap into a cylinder, a leak occurred at the flexible pigtail connection to the drop line piping in the UF_6 plant
- Operator immediately stopped the transfer from the local control room and Cameco voluntarily shutdown the UF_6 plant to complete an investigation of the incident
- Causes of the incident : Procedural non-compliance – inadequate tightening of the pigtail to the drop line piping end and improper leak testing
- The release was contained and mitigated with no adverse impact on health and safety of workers and the environment

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CNSC's Requirements

- Under section 12(1) of the *General Nuclear Safety and Control Regulations* every licensee shall:
  - have sufficient number of qualified workers
  - take all reasonable precautions to control the release of nuclear or hazardous substances within the site of the licensed activity and into the environment
- The licence for the conversion facility requires the licensee to:
  - report any unauthorized releases to CNSC and take corrective actions to prevent their re-occurrences
  - ensure compliance with its approved safety programs and associated procedures

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CNSC's Expectations

CNSC expects the conversion facility licensee to:

- Use the Defence-In-Depth approach
- Prefer preventive controls over mitigative controls
- Evaluate hazards continuously and act upon
- Detect early any Loss of Primary Containment (LOPC) incidents related to hazardous materials
- Implement in a timely manner any lessons learned from the LOPC incidents
- Document all LOPC incidents and update the Facility's Design, Safety Analysis and other documents based on lessons learned

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Licensee's Practices Before the UF_6 Leak incident

- Presence of an operator in the cylinder filling area control room during the entire transfer operation
- Adequate UF6 leak detection and prevention measures in place at the plant
- Smoke detectors/alarms activate emergency ventilation and shut down normal ventilation
- Pigtails PM inspected every 2 months and replaced after one year in service
- Pigtail connection pressure (leak) tested prior to each drop **(However, not as a final step)**
- Cylinder carriage moved in its correct position before drop line valves are opened by an operator

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What Caused the UF_6 Leak Incident

- Non-compliance with cylinder filling procedure: Conducting a pressure (leak) test after completing all piping connections and movement of the cylinder carriage
- After performing the leak test for pigtail connections, the operator moved the cylinder carriage to put it in its correct position
- **This move caused the connection between the pigtail and drop line to become loose and no final leak test done**
- Inadequate operator's **duty of care** for high risk activities (A pressure (leak) test should have been re-performed after moving the cylinder carriage – **Common sense**)

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Licensee's Practices After the UF_6 Leak Incident

- All past good practices maintained except the timing for conducting the pressure (leak) test
- Operating procedure for cylinder filling was revised and implemented to conduct a pressure (leak) testing of the pigtail connections after all preparations for filling a cylinder have been completed before opening the drop line valves to transfer liquid UF_6
- Detailed operating instructions for high risk activities at the UF_6 plant were developed and posted in the field by the licensee to improve human performance (Operator's duty of care)

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Second Recent Incident and its Safety Significance

- Description of UF_4 Production Process and Safety
- On March 12, 2010, an UF_4 slurry leak occurred due to failure of a diaphragm valve in the slurry line
- At that time, 3 indicator alarms associated with the UF_4 slurry production operations were triggered
- But the leak was only detected by an operator during a routine round of the UF_4 production circuit
- The UF_4 production circuit was shutdown, the spill was stopped and its clean-up was completed
- The failed diaphragm valve was replaced and cause of failure was a manufacturing defect
- No adverse impact on workers or the environment

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CNSC's Requirements/Expectations

- CNSC's requirements and expectations are same as discussed in previous slides nos. 11 and 12
- Additionally, the conversion facility licence requires the licensee to maintain the reliability and effectiveness of all UF6 plant equipment and safety systems in accordance with its quality assurance program accepted by the CNSC

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Licensee's Practices Before the UF_4 Leak Incident

- Operator's walkdowm of the UF_4 circuit every 2 hours
- **Small UF_4 slurry leaks were monitored but not corrected in a timely manner**
- High HF monitoring alarm activates emergency ventilation and shuts down normal ventilation (No such alarms triggered during this event)
- No PM procedure established to routinely inspect piping, valves and equipment (carbon steel lined with polypropylene or rubber) used in the UF_4 slurry service

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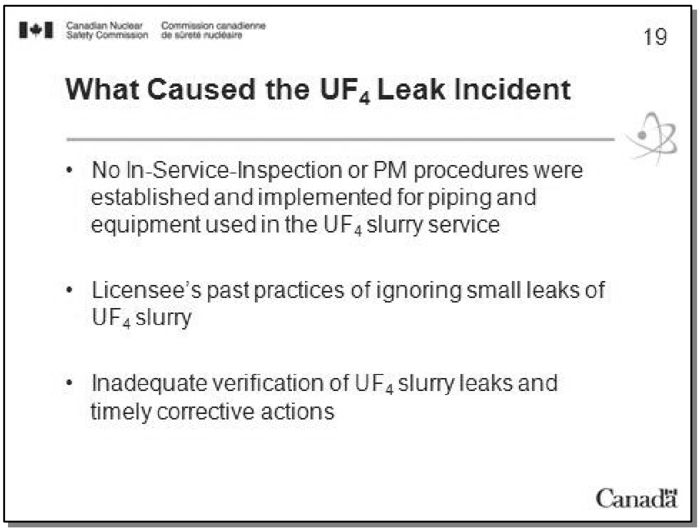
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What Caused the UF4 Leak Incident
• No In-Service-Inspection or PM procedures were established and implemented for piping and equipment used in the UF4 slurry service
• Licensee's past practices of ignoring small leaks of UF4 slurry
• Inadequate verification of UF4 slurry leaks and timely corrective actions
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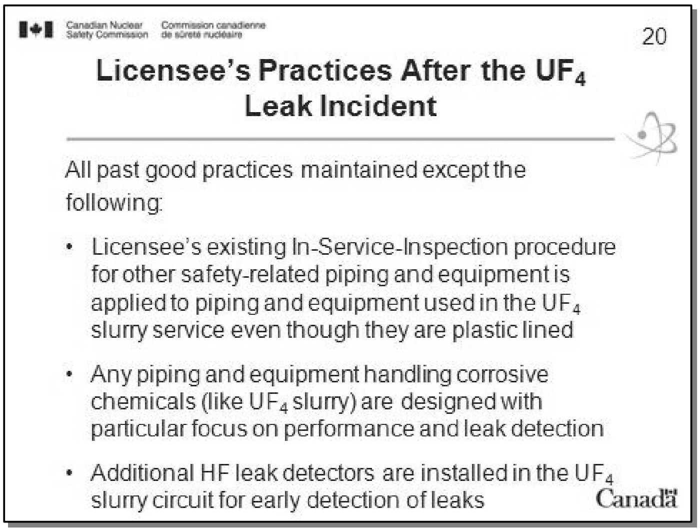
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Licensee's Practices After the UF4 Leak Incident
All past good practices maintained except the following:
• Licensee's existing In-Service-Inspection procedure for other safety-related piping and equipment is applied to piping and equipment used in the UF4 slurry service even though they are plastic lined
• Any piping and equipment handling corrosive chemicals (like UF4 slurry) are designed with particular focus on performance and leak detection
• Additional HF leak detectors are installed in the UF4 slurry circuit for early detection of leaks
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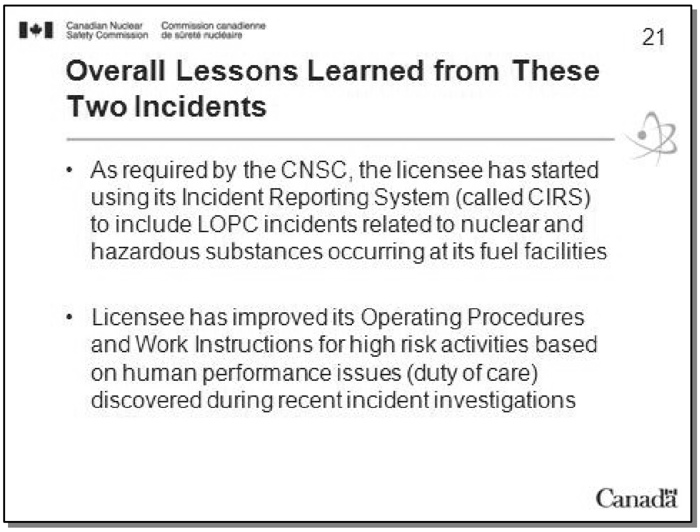
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Overall Lessons Learned from These Two Incidents
• As required by the CNSC, the licensee has started using its Incident Reporting System (called CIRS) to include LOPC incidents related to nuclear and hazardous substances occurring at its fuel facilities
• Licensee has improved its Operating Procedures and Work Instructions for high risk activities based on human performance issues (duty of care) discovered during recent incident investigations
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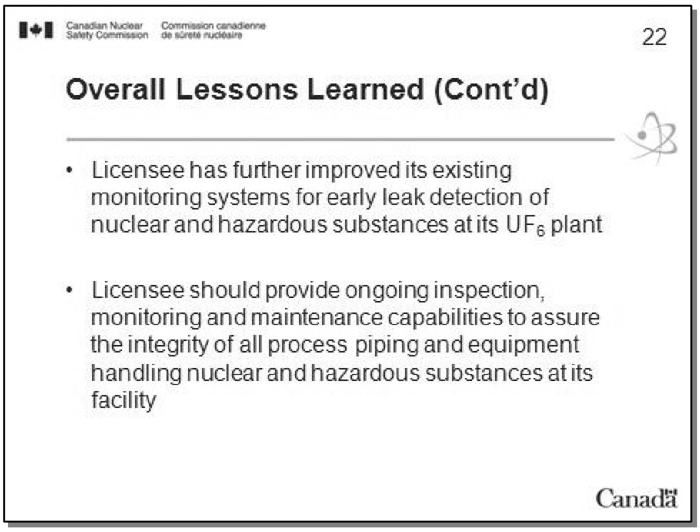
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Overall Lessons Learned (Cont'd)
• Licensee has further improved its existing monitoring systems for early leak detection of nuclear and hazardous substances at its UF_6 plant
• Licensee should provide ongoing inspection, monitoring and maintenance capabilities to assure the integrity of all process piping and equipment handling nuclear and hazardous substances at its facility
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Thank you
• Available for any questions
Visit CNSC Website: www.nuclearsafety.gc.ca for Canadian regulatory information
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DEVELOPING A SAFETY REPORT FOR AN EXISTING CONVERSION FACILITY

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Abstract - A review of the process used to meet the regulatory requirements for a Safety Report at an existing conversion facility is described. This paper will cover the establishment of the regulatory criteria, selection of appropriate methodologies, identification of events and modeling of credible events. Once established there is on-going maintenance to deal with design changes and the need for periodic reviews will also be discussed.

Challenges in dealing with the various phases, including incorporation of historical licensing documents, and lessons learned are presented. Of specific interest is the failure of the selected methodology to deal with infrastructure issues. One aspect of lessons learned that will be explored is the lack of an available mechanism for sharing information with similar fuel cycle facilities which is compounded by the fact that there are a small number of fuel cycle facilities compared to nuclear power plants. Possible approaches to dealing with this issue are also discussed.

1. INTRODUCTION/BACKGROUND

The Port Hope Conversion Facility (PHCF) occupies an area of approximately 10 hectares on the shore of Lake Ontario. Immediately to the east of the site is the Port Hope harbour, the centre pier (currently leased by Cameco) and the Ganaraska River. To the south is a beach, which is remote from the recreational activities of the inner harbor and is presently used for strolling, swimming and fishing. The VIA Rail station building sits just to the northwest of the PHCF. To the north of the PHCF are the CN and CP rail corridors whose tracks cross the Ganaraska River valley on two viaducts supported on masonry piers. Commercial and residential areas are located north of the tracks and east of the river. The site is believed to be the oldest operating nuclear facility in the world, predating the Canadian regulator, and because of the history had some unique challenges in developing a Safety Report. A photograph of the PHCF site is provided in Figure 1.

Figure 1**. Arial Photograph of the Port Hope Conversion Facility Site**

Cameco is a publically traded Canadian company that is involved in the exploration, mining, milling, refining and conversion of uranium containing materials as well as CANDU reactor fuel and components manufacturing. Cameco's headquarters are in Saskatoon, Saskatchewan. Cameco's uranium refining and conversion operations are located in Blind River and Port Hope, Ontario, respectively. The fuel and reactor components manufacturing facilities are located in Port Hope and Cobourg, Ontario. Collectively all of these operations are referred to as the Fuel Services Division (FSD) of Cameco. The processed uranium is part of the supply chain used in the manufacture of reactor fuel for electric utilities in Canada and around the world.

PHCF was initially established by a private company, Eldorado Gold Mines Limited in 1932 to process ore from Port Radium, in the Northwest Territories, into refined radium. The radium refining operation ran until 1939 when operations were suspended for a short period for economic reasons. In 1942 the company, Eldorado Gold Mines limited was purchased by the Canadian government when the strategic importance of uranium was recognized. This action made this operation a Crown Corporation. The operation was then converted to a uranium processing plant, and has provided feed for both the Canadian CANDU and the light water reactor programmes and the uranium metals industry.

The Crown Corporation was renamed Eldorado Nuclear Limited (ENL) in 1968. It was in this time period that the Canadian government enacted legislation to ensure that uranium was processed in Canada to the fullest extent possible and so this resulted in the construction of a uranium hexafluoride (UF_6) conversion facility at the Port Hope site. The facility was constructed and initial operations began in 1970. In the late 1970's, due to increased demand for UF_6 production a decision was made to build a new facility. The new facility went through regulatory and public approval processes that lead to the selection of the Hope Township site, about 15 km west of Port Hope. This was later changed to Blind River, and ENL sought and obtained government approval to split the operation, constructing the refinery in Blind River and new UO_2 and UF_6 conversions plants in an expanded Port Hope site.

The new UF_6 plant was constructed and commissioned by mid-1984 and employed a new process that better utilized the consumption of anhydrous hydrogen fluoride (AHF), reducing emissions and associated by-products.

In October 1988, the Federal Crown Corporation ENL and the Provincial Crown Corporation Saskatchewan Mining Development Corporation were merged to form a new entity, a Canadian Mining and Energy Corporation. This organisation was subsequently privatized in the early 1990s and the name was shortened to Cameco.

At present, PHCF is involved with the production of uranium dioxide (UO_2) for fuel in the CANDU reactors and as blanket fuel for light water reactors as well as UF_6 used as an intermediate feed product for light water reactors.

2. THE JOURNEY BEGINS

One of the worst industrial chemical disasters occurred without warning early on the morning of December 3, 1984, at Union Carbide's pesticide plant in Bhopal, India. While most people slept, a leak, caused by a series of mechanical and human failures, released a cloud of lethal methyl isocyanides over the sleeping city. Some two thousand people died immediately and another eight thousand died later. The hearings and media attention led to USA enactment of the Emergency Planning and Community Right to Know Act of 1986 (EPCRA), requiring companies to provide information about their potentially toxic chemicals. At the same time, US states were required to establish emergency planning districts and local committees to prepare for any emergency—a fire, an explosion, a flood that might result in the release of chemicals into the environment. Cameco itself had experienced environmental incidents at various operations and had initiated Environmental Risk Assessments throughout the organisation.

The Nuclear Industry in Canada and abroad has a long history of providing safety cases for the design and operation of nuclear power plants. The Port Hope facility is primarily a chemical/uranium process plant, pre-dated the establishment of the regulatory framework and was therefore not dealt with in the same manner. However, in the aftermath of the Bhopal disaster, other chemical and nuclear disasters since, and pending updates to the Canadian regulations, Cameco was required to produce a Safety Report.

In 1995, as part of the regulatory relicensing process, the PHCF was required to produce a safety analysis for the operation of the facility. The primary focus of attention was on potential spills or accidents and loss of containment of chemicals that could impact on our employees, the community in which Cameco operates as well as the environment. It should be noted that because the PHCF handles hydrofluoric acid (HF), hydrogen gas (H_2), uranium hexafluoride (UF_6), and fluorine (F_2), our operations were designed and built to the accepted safety standards and codes of the day and to industry best practices which were developed based on lessons learned and expertise of the chemical industry. These practices evolved with experience, internal and external, and are also a part of the company’s commitment to continual improvement. Some of these practices include:

- improvements have been made to the UF_6 storage and handling systems since the original construction. Additional containment has been supplied around the cylinder filling area, remotely actuated cylinder valve closures have been installed, installation of CO_2 system in the event of a major release and interlocks have been modified to incorporate cylinder type, fill capacities and motion detectors;

- improvements have also been made to the AHF system. The AHF suppliers have modified their railcars to provide for either remotely operated valve closures or remote shutdown of the main valve to isolate the rail car in the event of an accidental release. Additional liquid level instrumentation has been installed on the AHF storage tanks to prevent overfilling of the tanks and a de-railer device was installed on the incoming line;
- conversion of hydrogen supply from dissociated ammonia to bulk hydrogen and the use of diluted aqueous ammonia in the UO_2 plant has resulted in elimination of the liquid anhydrous ammonia inventory and simplified operations;
- additional detection systems have been installed in the bag house dust collector outlets to monitor for leaking/broken bags or increased loadings;
- consolidation of the fluorine production from the old UF_6 plant to the new UF_6 plant was adopted as a means of reducing the environmental risk and impact of an event. This also resulted in reduced chemical inventories;
- continuous improvement in the area of fluoride monitoring has resulted in additional installations in ducting systems to serve as troubleshooting tools for production personnel. They have also been utilized to automate responses, ensuring rapid response to in-plant leaks or loss of containment events.

The specific requirements were defined in section 21(e) of the PHCF license which required that by December 31, 1997, a formal site wide Safety Report was to be developed that met the following objectives:

- to provide an assessment of risk to the public and to the environment arising from the possibility of accidental release of hazardous chemicals stored, processed and transported;
- to identify the dominant contributors to risk and to ensure that appropriate safety measures are provided for the in Facility design, operating conditions or emergency procedures; and;
- to provide an information base to assist in optimizing safe plant operation and in safety-related decision making.

At the time, Cameco worked with the Canadian regulator, then referred to as the Atomic Energy Control Board and since changed to the Canadian Nuclear Safety Commission (CNSC) to best utilize the information and resources available to develop the Safety Report.

Most of the available regulatory documentation was focused on the nuclear reactors and there was less information available for uranium refineries or conversion plants. This analysis also heavily favoured probabilistic risk assessments. Further complicating this situation is the fact that a lot of the safety case assessment work being under taken at the time by the industry was being directed to new builds and their initial design. There was not a great deal of information on dealing with existing operating facilities, especially ones that had been in operation for decades.

Cameco conducted some research of academic papers and it became apparent that that there was a substantial body of work being done in the United Kingdom. This work embodied the entire nuclear fuel cycle. At the 1990, Safety and Reliability Society (United Kingdom) symposium held in Altrincham, England a specific paper was presented by British Nuclear Fuels Limited (BNFL). The paper abstract stated that "BNFL is preparing fully developed Safety cases, using qualitative and quantitative techniques for all its plants at the Sellafield irradiated nuclear fuel site in Cumbria. The Safety cases include hazard analysis for the fault quantification. Lessons learned from the first three years of the programme presented". This paper and discussions with BNFL representatives assisted in the development of Cameco's path forward to meet the requirements of the license condition 21(e).

It should be noted that in discussions between Cameco and the regulator, it became apparent that the value of quantitative risk assessment for our type of operation was limited. Some independent studies found that results could vary from 10^{-4} to 10^{-7} for the same evaluation of the same operation. Qualitative risk assessment approaches were therefore favoured.

3. PROCESS

The 1995 license conditions did not stipulate a specific methodology to be used. Since the entire site had to be reviewed in a very short time frame (2 years) and Cameco lacked the in-house expertise and resources, it was elected to seek outside expertise and assistance to identify the best technique for performing this work and to meet the license requirements, in the specified time frame. Again, in all the stages of this project Cameco was very careful to involve the regulator to ensure that the process and path forward was acceptable. The regulator did not give any formal approvals for steps along the way, however the feedback provided was helpful in assisting Cameco to focus and direct the work being conducted.

After an extensive and exhaustive process to identify the tool that would meet the intent of the license conditions, Cameco selected Imperial Chemical Industries (ICI)-Eutech, based in part to their work with BNFL. The technique used was referred to as Process Hazard Review (PHR). This technique was an evolution of the hazard identification and assessment process developed by ICI, and had been used at both nuclear and chemical facilities. ICI was an international chemical company that has a long association with hazard identification, assessment and control. It was one of the first chemical companies to develop and utilize process hazard identification of existing facilities. Their knowledge in the area of health, safety, and environment had been successfully used by other organisations, such as Eutech engineering Solutions, a wholly owned subsidiary of ICI. ICI was one of the originating organisations of what is now broadly referred to as HAZOP.

As a result, ICI-Eutech and their PHR methodology was selected because of it's leadership in the area of hazard assessment, successful application of the technique at a similar (BNFL) site and use of the technique with other chemical process industries.

To ensure that Cameco was heading down the correct path, in mid-1996 a consortium of Cameco, ICI-Eutech and BNFL individuals made a formal presentation to the regulator. Cameco provided an overview of the technique and plan forward using this tool as well as utilizing the expertise of ICI-Eutech to accomplish this license condition. The CNSC confirmed verbally that the approach was progressing in the appropriate direction. Cameco entered into a contract with ICI-Eutech and BNFL personnel to initiate and conduct this body of work.

To better confirm that this approach was appropriate, Cameco decided to perform a couple of test cases and then evaluate the process and the final product. Several key Cameco individuals were chosen to obtain the training from ICI-Eutech and become a core group of leaders to carry on this work for the entire site. These leader trainees were then qualified by completing the two pilot projects (liquid UF_6 and ammonia) under the guidance of the qualified personnel. Again the results were shared with the CNSC and found to be acceptable. This confirmed Cameco's plan and the work to complete the task was fully implemented.

The hazard identification is the most fundamental step in the process on identifying, assessing and controlling hazards. This step is often referred to as a Process Hazard Assessment or Analysis (PHA). It is an organised and systematic approach to identify and analyze the hazards associated with the processing of chemicals. The PHA focuses on equipment, instrumentation, human actions, and other factors in the process. Selection of the PHA methodology is influenced by many factors, including the existing knowledge of the process, the operating experience, size, complexity, etc. Some PHA methodologies available are listed below:

- What-If;
- Checklists;
- Hazard and Operability Study (HAZOP);
- Failure Mode and Effects Analysis (FMEA);
- Fault Tree Analysis;
- Change Analysis; and;
- Historical or literature reviews.

The above was based on the US OSHA requirements that were developed in response to the Bhopal event and is described in the *Code of Federal Regulation 29CFR1910 Occupational Health and Safety Standards.*

The selection of any one or combination of methods is based on the factors identified above and they all have their own limitations. In seeking third-party expertise, Cameco sought to find expertise with PHA at chemical operating facilities, and thus those more familiar with the problems associated with operating sites and the best method of dealing with them.

The PHR methodology was developed by ICI because of their experience with retroactive PHA's on their operating plants. In general they found that when dealing with older plants the conventional techniques had problems with the lack of technical information, outdated Process and Instrumentation Diagrams (P&IDs), ignored operating experience and actual operating practices and interactions, lacked involvement and operations commitment and ended up being intellectual exercises. As a result they developed the PHR methodology and have successfully implemented it in over 100 applications since 1990. One of the applications was a BNFL operation similar to PHCF as it is also a conversion facility. The age of the Port Hope site, dating back several decades made it a good candidate for their methodology.

PHR is a structured approach that involves an in–depth examination of an area by a team. The team utilizes available technical expertise and experience to bring to the review:

- operating experience and concerns;
- maintenance experience and concerns;
- service history, performance and safety;
- modifications, updates and creeping change;
- non-hardware changes; incidents and excursions;
- compliance with systems/procedures;
- emergency preparedness;
- surroundings;
- best practices.

The result of the PHR is an assessment that identifies the significant hazards and provides a mechanism for the following up on those hazards which require additional work. As stated earlier, all PHA methodologies have their limitations. In case of the PHR, Cameco was aware that the ICI-Eutech system did not provide a data base that could be readily updated. Therefore, the information obtained from the PHR had to be translated into a computer data base that would facilitate updating, monitoring, and assist in meeting the PHCF license condition. The software used was Dyadem International Ltd.'s PHA-Pro. Cameco is still using an updated version of this software to maintain it's Safety Report today.

A typical PHR team would consist of a Team Leader, selected from a core group, unit operation Process engineer, Area Operator, Maintenance engineer, and a Recording Secretary. The Team Leader assembled all of the relevant information, previous safety assessments, design manual, P&IDs, Material Safety Data Sheets (MSDS), operation history and other relevant information. An initial meeting was initiated with a physical inspection/tour of the unit operation. This would be followed up with several ½ days sessions during which time the unit operation was reviewed. If necessary, specialist options (e.g. Metallurgical engineer) would be sought and requested to attend the required session. The unit operation would be reviewed, utilizing P&ID if available, for possible mechanisms for Loss of Containment using their PHR user guide. An assessment of the risk is provided based on the team's knowledge and experience. This information is used to assign severity and likelihood value to each hazard.

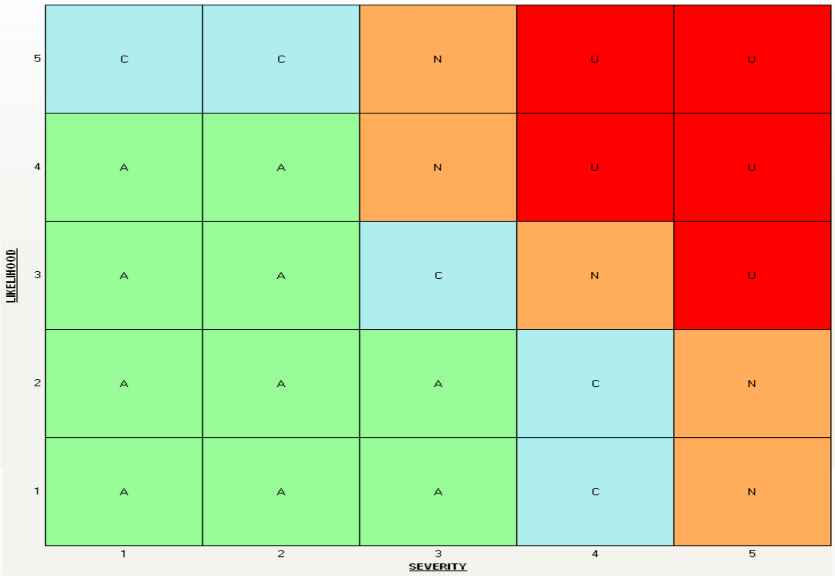
The assessment of risk to the public was provided in the following manner. Each of the events identified by the PHR was ranked in terms of likelihood and severity and placed on a grid. Likelihood and severity were defined as indicated on the following Table 1:

TABLE 1

Likelihood	**Severity**
1. Extremely unlikely to occur.	1. No injury or health effects to the general public or no environmental effects.
2. Not expected to occur during the lifetime of the facility.	2. Minor injury or minor health effects to the general public or minor environmental effects.
3. Expected to occur not more than once during the lifetime of the facility.	3. Injury or moderate health effects to the general public or moderate environmental effects.
4. Expected to occur several times during the lifetime of the facility.	4. Serious injury or health effects to the general public or serious environmental effects.
5. Expected to occur more than once per year.	5. Death or severe health effects to the general public or severe environmental effects.

The rankings, of likelihood and severity, for each event were then transcribed on to the following Safety Report Risk Matrix shown as figure 2.

Figure 2. **Safety Report Risk Matrix**



Risk for the hazards assessed in the PHR studies were ranked using this 5x5 risk matrix developed by Cameco. Risk is defined as a measure of both incident likelihood and the magnitude of the injury, damage or loss. The risk rankings are the combination of likelihood (L) and severity (S) and subsequently slotted into four categories as shown on Table 2.

TABLE 2

ACCEPTABLE	These risks are low either inherently or with controls in place. If controls are in place, periodic monitoring of the controls is recommended.
(C) CONTROLLED	These risks are generally acceptable provided all controls are effective. These risks should be periodically reviewed to ensure controls are in place and effective, but also to examine options for lowering overall risk.
(N) NOT DESIRABLE	These higher risks may have controls in place, require more controls, or may be controlled to the maximum extent practicable. Controls must be effective in this region as well as being frequently reviewed. Every effort should be made to reduce risks in this area through avoidance or elimination of the risk event, or by the addition of the controls to reduce or share the risk.
(U) UNACCEPTABLE	These risks are unacceptable from a corporate risk tolerance perspective due to their high consequence and/or frequency. Their immediate identification and control is imperative. The goal is to have no risks at Cameco in this region when controls are in place and effective.

The facility was broken up into production units (UF_6) and into major processes within the main production units (AHF storage and handling) to make the process more manageable. This resulted in 20 PHRs for the initial Safety Report. The outcome of the PHRs was ultimately reflected in tables that are reproduced in Figure 3 below.

Table 3. **Generic Sample of a PHR Output**

Area

Node

Deviation

Causes	Consequences	Safeguards	Risk with Safeguards			Remarks
			S	L	RR	
How the deviation occurs	What the results are	The barriers that are in place to prevent or mitigate the event	The Severity from Table 1	The Likelihood from Table 1	Risk Ranking from Figure 2	Any comments related to this, including follow-up activities.

Once all of this work was completed a report was issued which identified the hazards, cause, consequence, and level of risk. Where the safeguards/barriers were not deemed to be adequate, recommendations were provided for follow up activity, identifying the concern as a loss of containment, safety critical, non-conformance with standards or need for further study. This generated a substantial list of work that had to be completed to ensure that there were no events rated as U and that N events were moved to the C category, with appropriate corrective actions and process/equipment installed to minimize the risk to a lower level. All of this information is converted into a computer database, which is used to generate the Safety Report.

4. MODELLING AND ACCIDENT SCENARIOS

Based on the work from the PHRs there were 17 process upsets that were selected as having the most significant potential to impact the offsite receptors. These scenarios included: AHF unloading hose rupture, F2 discharge drum burn out, and UF_6 pigtail leak. In order to evaluate these scenarios Cameco used computer modeling and air dispersion techniques. A systematic assessment of various air dispersion models and regulatory requirements, culminated with the choice of AERMOD.

The AERMOD dispersion model is specifically designed for analysis of emission from complex industrial settings and takes into account multiple stacks, fugitive emissions, building wake effects, etc. The AERMOD model uses emission source characteristics, building and boundary layout, surrounding terrain, meteorological data and a receptor grid. The AERMOD dispersion model was used for Point of Impingement (POI) assessment as required by Ontario Regulation 419/05. Figure 3 below shows the basic configuration for the PHCF used in the modeling work.

Figure 3**. Computer Model of the Facility Used for Dispersion Modeling.**

All the process upsets identified have been used to assist Cameco in prioritizing its' efforts to further reduce the impact of a release. This can be achieved by means of shortening of the detection and response time or automating responses. Additional bag house equipment has been installed to handle UF_6 process upsets.

In all accident scenarios, all of the modeling results were and continue to be below the accidental release assessment criteria of 10 mg/m3 uranium (soluble), 25 mg/m3 hydrogen fluoride and 39.5 mg/m3 fluorine for a 30-minute exposure.

5. SAFETY REPORT HISTORY

Prior to the development of the Safety Report, safety analysis for the operating plants had been performed and captured in design documentation, licensing requirements and summarized in the Facility Licensing Manual (FLM).

In 1997, process hazard reviews were conducted for all active areas of the plant and where appropriate, off site facilities. A total of 20 PHR's were conducted over the span of approximately 1 year. Upon completion of the PHR report, the report was converted to a standard format that originally appeared in of Issue 1 of the Safety Report.

In Issue 2 (1999) of the Safety Report was converted to text format largely from the original table format. The process area descriptions were enhanced, protective measures associated with potential hazards documented and related area safety equipment and systems were summarized.

In Issue 3 (2002) of the Safety Report was reverted back to table format. A number of PHR's was also increased to 23. This increase reflects new installations in various areas that have taken place since 1997.

In Issue 4 (2006) of the Safety Report it was again presented in table format. A total of 25 PHR's now appeared, reflecting the addition of a PHR for the Clean-Up Program (CUP), which has grown in significance since 2002 and the separation of HF Recovery and Gaseous Effluent into two PHRs. Several additions have been made to the PHRs for Issue 4: a node to assess fire risk in the area has been added for each PHR; safeguards have been updated throughout the document; new causes have been identified and assessed for existing deviations; new deviations have been identified and assessed; and incidents that have occurred onsite between January 2003 and January 2006 have all been indicated in the remarks column of the PHRs.

In Issue 5 (2010) of the Safety Report was converted to a text and flowchart format. The table format still remains in place however and provides the details for the text version. Both the tables and text will be kept updated going forward. The process area descriptions, protective measures associated with potential hazards and related area safety equipment and systems were updated. Accident scenarios and process hazards were added based on credible scenarios from the PHR process. Several additions have been made to the Safety Report including the addition of the depleted uranium dissolution circuit, a section on radiation protection, emergency response and community notification. Incidents that have occurred since January 2006 and October 2010 have been reviewed and captured in either the environmental aspects or the Safety Report.

This present safety report for this facility is the culmination of significant effort conducted from 1995-2010. In fact even today this report is being revisited and updated to reflect operational changes, safety system changes as well as operational changes and the addition of abatement equipment. This paper will give an overview of the trials, tribulations and successes of this journey. This paper is intended to provide a "lessons learned" for future facilities that maybe undergoing a similar journey.

6. LESSONS LEARNED

The most significant lesson learned was that the PHR process focused on the P&IDs and so missed areas that would not be identified in these drawings. The concrete floors and trenches are part of the building containment but were not adequately addressed in the initial and subsequent Safety Report.

Lacking specific directions, the on-going discussions with the regulator were very beneficial. Cameco ensured that the CNSC were appraised at each significant step of the process. Presentations were given, when ICI-Eutech and their methodology were chosen and when the products from the two test risk assessments were completed. This action provided some assurance that the final product would be acceptable.

In an effort to ensure that Cameco met the licensing conditions of this initial Safety Report Cameco focused on the specific task at hand and did not look at the much bigger picture with respect to risk reduction and assessment. Therefore, the original work in the assessment had a substantial gap with respect to the safety of the employees and the potential impact to the environment with releases of minor amounts of hazardous chemicals. As Cameco committed to the principals of the ISO 14001 (environmental management system) and OHSAS 18001 (occupational health and safety management system), it was soon realized that the present safety report had to be revised and expanded to ensure coverage of potential risks for these important areas. For example, the ISO 14001 programme would expect consideration of consumption of raw materials which was not part of the initial Safety Report. The recently issued ISO 3100: Risk management – Principles and guidelines will expand this to business objectives.

Another lesson learned during this process is that there is a strong desire to keep all the risk assessments and information in a single report for the facility. This ensures an all encompassing approach that would be used for risk assessment in all applicable areas of the PHCF. Originally the software did not permit this but this has since been updated to permit this. The document should be written so that is can be shared in part or in whole depending on the information being requested. This aspect is still not fully resolved.

The Safety Report work involved an assessment of all the operating plants at the PHCF. The detail of the assessment gives rise to significant operational information on process equipment and design. As this can include proprietary commercial information, there was a concern that sharing all this detail of information to the public may/would compromise our competitive advantage. In order to mitigate these concerns a document was produced that would give sufficient information to alleviate stakeholders concerns but does not provide proprietary information that could jeopardize Cameco's process and design information. These multiple documents can adversely affect the management of changes.

For Cameco the assessment of risk to the employees, the public and environment arising from an accidental release of hazardous chemicals did identify one significant deficiency from previous assessments. The assessment of the release of chemicals through the area heating and ventilation (H&V's) was not assessed accurately because the releases were estimated to be of a short duration. The present re-assessment determined that these releases have more of an impact than those that are treated through the emissions control systems. This has resulted in an effort to further limit the environmental release by improving response time to detect and shutdown the normal H&V and revert to the emergency ventilation. Presently all of the modeled predicted releases are below the maximum recommended criteria for off site air concentrations for uranium, hydrogen fluoride and fluorine.

The linkage of the Safety Report to the design change process could have been better defined. Although design change modifications would undergo PHAs, the ensuing documentation tended to be a stand alone that would not be integrated into the Safety Report until a revision was performed.

7. GOING FORWARD

One of the challenges with this scope of work is to keep it up to date and relevant. Cameco has put in place several triggers in the operational processes that would bring this document into play and ensure that Cameco keep this document remains current and relevant. Within the incident/accident process there is a requirement in the investigation stage that the Safety Report be used to assess whether or not this present incident was already identified in the safety report and was it documented properly with the appropriate risk ranking and impact. Did the safety system perform as expected and documented? If not covered in the report there is a requirement to ensure that it is captured in the updating of the safety report. Any deficiencies identified in the recent investigation will be identified, corrected and incorporated in to the next revision of the Safety Report. To this end a single point of contact has been designated for this task within the PHCF. That individual is responsible for keeping the Safety Report relevant and a "living document".

Another trigger point is in the potential process change analysis management process. All significant process changes will be assessed against the Safety Report to ensure that they do not violate or compromise any of the conditions already identified in the Safety Report.

One of the issues identified early in this process is that there is not a lot of specific relevant external information available to assist in the process of preventing significant incidents. This is due to the small number of similar fuel cycle facilities in the world, currently there are 4 such operations. Presently Cameco is involved in two programmes that assist us in this area. Cameco is actively involved in the CANDU Owners Group (COG) as well as an international UF_6 safety producers group. COG does provide some basic information on a nuclear basis; however, sharing is somewhat limited due to the main focus of this group is nuclear power plants and electrical generation. The UF_6 safety working/producers group meets on an annual basis to present and discuss issues around safety and regulatory issues. Both of these groups are useful in sharing information and have been beneficial for Cameco, however maybe a specific group with similar operational issues could be formed to share risk assessment and reduction strategies. This could also be accomplished through interaction of the various regulatory bodies that govern similar operations.

Developing a Safety Report for an Existing Conversion Facility

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September 27-29, 2011

Overview

- Introduction
- Background
- Process
- Lessons learned
- Safety Report today
- Sharing experience

Cameco

Introduction

- Safety analysis is defined as:

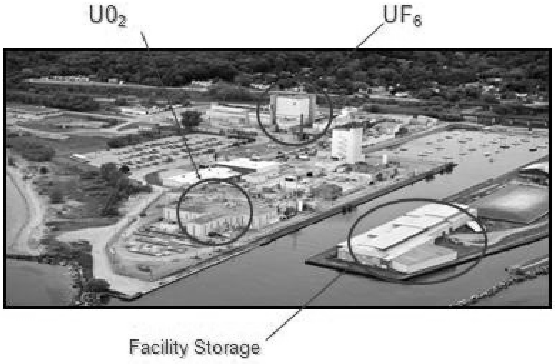
"a systematic examination of the structure and functions of a system aiming at identifying accident contributors, modeling potential accidents, assessing risk and finding risk reduction measures."[1]

[1] The Role of Safety Analysis in Accident Prevention, Jouko Suokas, Technical Research Centre of Finland Occupational Safety Engineering Laboratory, Accident Analysis & Prevention Volume 20 No. 1 pp 67-85 1988

Cameco

Background

- Ariel view of the Port Hope Conversion Facility



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Background

- 1932 - Site established for radium processing
- 1944 - Crown corporation formed
- 1946 - AECB established
- 1970 - Expanded to UF_6 production
- 1970 - First regulatory operating license for UF_6
- 1978 - New UO_2 facility
- 1983 – QA program for UF_6
- 1984 – New UF_6 facility
- 1986 – AECB begins updating regulations
- 1988 – Cameco created
- 1995 – Working to include license conditions in licenses
- 2000 – CNSC created with the new Act and Regulations

Cameco

Background

- Condition 21(e) of the PHCF license required a Safety Report with the following objectives:
  - provide an assessment of risk to the public and to the environment arising from the possibility of accidental release of hazardous chemicals
  - identify the dominant contributors to risk and to ensure that appropriate safety measures are provided; and
  - provide an information base to assist in optimizing safe plant operation

Cameco

Process

- Various safety or risk analysis completed
- Probabilistic risk assessment not done or available
- Plants built to codes but changed with time
- Change management inconsistent resulting in drawings may not be current
- Record of events limited to major events
- Stable and experienced work force
- Challenge – what process works best to meet requirements

Process

- Reviewed available resources and selected ICI-Eutech
- ICI-Eutech leadership in hazard assessment and experience at similar operations
- ICI-Eutech experience with older facilities was that they lacked technical information, out-dated P&IDs, people did not utilize operating experience and actual practices, did little involvement of people
- Process Hazard Review process developed to deal with these issues

Process

- Teams established to utilize available site technical expertise to review:
  - operating experience
  - maintenance experience
  - service history, performance and safety
  - modifications, updates and creeping change
  - non-hardware changes, incidents and excursions
  - compliance with system procedures
  - emergency preparedness
  - surroundings (i.e. railway)
  - best practices

Process

- Facility broken down into manageable sections
- Main plant areas such as UF_6, UO_2
- Each main plant into main operations - reduction, hydrofluorination, etc.
- General – materials handling, storage, etc.
- Total of 20 PHRs

Cameco

Process

- Safety analysis is defined as:

Likelihood	Severity
1. Extremely unlikely to occur.	1. No injury or health effects to the general public or no environmental effects.
2. Not expected to occur during the lifetime of the facility.	2. Minor injury or minor health effects to the general public or minor environmental effects.
3. Expected to occur not more than once during the lifetime of the facility.	3. Injury or moderate health effects to the general public or moderate environmental effects.
4. Expected to occur several times during the lifetime of the facility.	4. Serious injury or health effects to the general public or serious environmental effects.
5. Expected to occur more than once per year.	5. Death or severe health effects to the general public or severe environmental effects.

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Process

Risk Matrix

Likelihood

C	C	N	U	U
A	A	N	U	U
A	A	C	N	U
A	A	A	C	N
A	A	A	C	N

Severity

Cameco

Process

- Risk Ranking

(A) ACCEPTABLE	These risks are low either inherently or with controls in place. If controls are in place, periodic monitoring of the controls is recommended.
(C) CONTROLLED	These risks are generally acceptable provided all controls are effective. These risks should be periodically reviewed to ensure controls are in place and effective, but also to examine options for lowering overall risk.
(N) NOT DESIRABLE	These higher risks may have controls in place, require more controls, or may be controlled to the maximum extent practicable. Controls must be effective in this region as well as being frequently reviewed. Every effort should be made to reduce risks in this area through avoidance or elimination of the risk event, or by the addition of the controls to reduce or share the risk.
(U) UNACCEPTABLE	These risks are unacceptable from a corporate risk tolerance perspective due to their high consequence and/or frequency. Their immediate identification and control is imperative. The goal is to have no risks at Cameco in this region when controls are in place and effective.

Cameco

Process

- Results of analysis

Area Node Deviation						
Causes	Consequences	Safeguards	Risk with Safeguards			Remarks
			S	L	RR	
How the deviation occurs	What the results are	The barriers that are in place to prevent or mitigate the event	The Severity from Table 1	The Likelihood from Table 1	Risk Ranking from Figure 2	Any comments related to this, including follow-up activities.

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Process

- Outcome of results
  - All U or Unacceptable results were dispositioned before the report was finalized
  - All N or Not Desirable results identified further actions which were acted upon over the next several years
  - Some C or Controlled and A or Acceptable results identified further actions which were acted upon over the next several years

Cameco

Lessons Learned

- Focus on P&IDs missed containment structures (floors, sumps and trenches)
- Communications with regulator was critical to success
- Focus on environment and public safety biases risk low relative to workers
- Not apparent that you can capture all risk types (environment, public, safety, product quality, etc.) in one approach
- Report included too much technical info considered commercially confidential

Cameco

Safety Report Today

- Initial report has evolved and continues to evolve on the basis of continual improvement

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Sharing Experience

- Few converters and issues with competition so limited sharing
- 1986 - UF_6 safety working group
- 2006 - tried to set up something between the Class 1B facilities
- 2009 - Member of the CANDU Owners Group
- 2011 – FINAS - IAEA

Cameco

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.

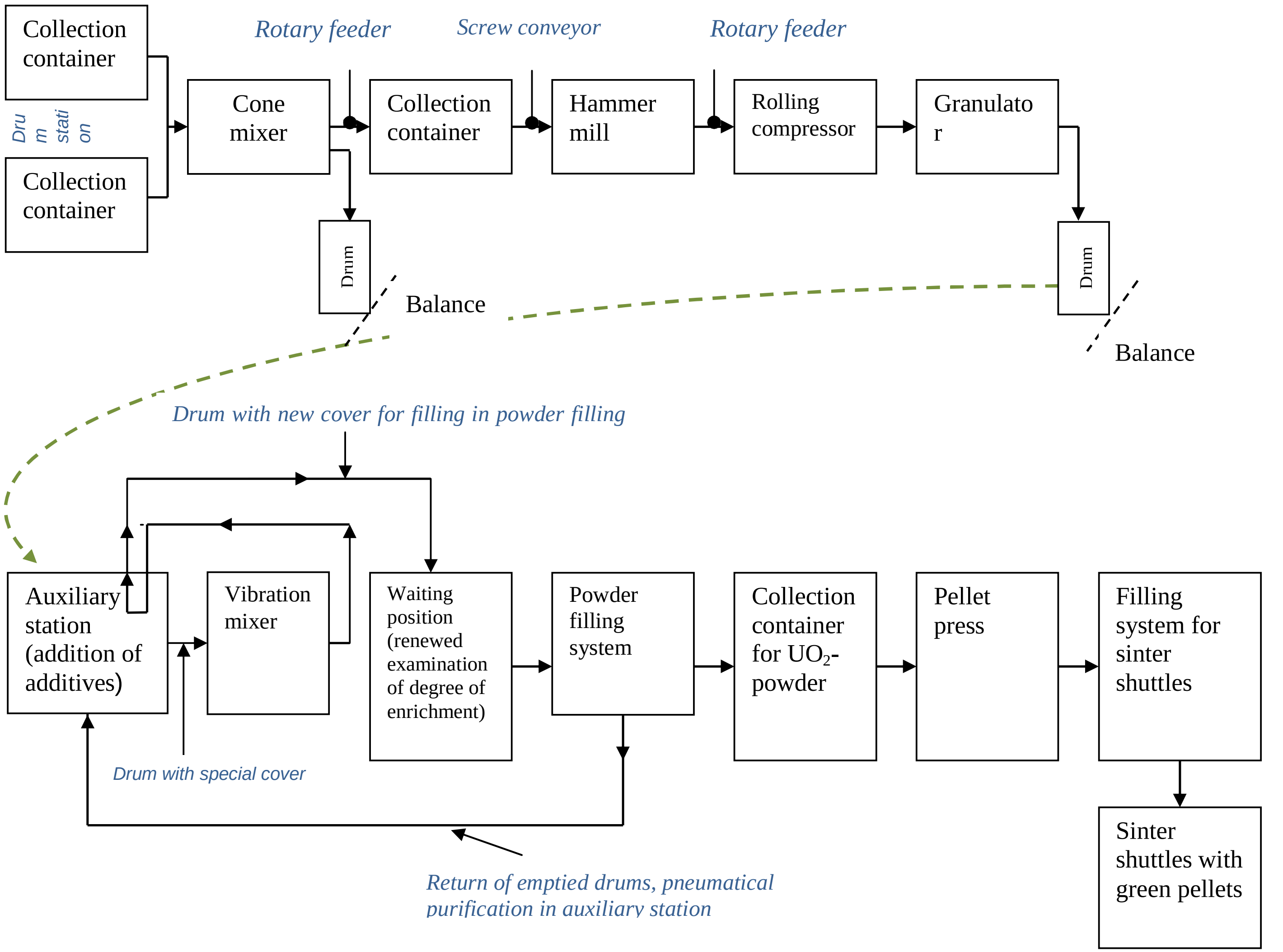
- Powder filling station: Feeding the powder via a collection container to the powder press, pressing of the green pellets, transfer of the green pellets to the filling equipment for sintering boats and carriage of the loaded sintering boats to the sintering furnace.

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.

References

[1] F. E. Moser, G. Gänssmantel, G. Mayer, Probabilistische Sicherheitsanalysen für Anlagen der Brennelementfertigung und der Anreicherung, GRS-A-3531, Oktober 2009

[2] IAEA-TECDOC-1267: Procedures for conducting probabilistic safety assessment for non-reactor facilities; January 2002

[3] L. Weil:Charakterisierung der Risiken der Kernenergienutzung; Habilitationsschrift; Mai 2002

[4] Atomrechtliche Sicherheitsbeauftragten- und Meldeverordnung Fassung: 8. 2002

[5] Datenbank VIBS: Meldepflichtige Ereignisse in Anlagen des Kernbrennstoffkreislaufs, Bundesamt für Strahlenschutz (BfS)

[6] BMU Leitfaden zur Durchführung der „Sicherheitsüberprüfung gemäß §19a des Atomgesetzes – Leitfaden probabilistische Sicherheitsanalyse –„ für Kernkraftwerke in der Bundesrepublik Deutschland, Bundesanzeiger Nr. 207a vom 3. 11. 2005

[7] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke Methoden zur probabilistischen Sicherheitsanalyse für Kernkraftwerke BfS-SCHR-37/05, August 2005

[8] Facharbeitskreis (FAK) Probabilistische Sicherheitsanalyse für Kernkraftwerke Daten zur probabilistischen Sicherheitsanalyse für Kernkraftwerke, BfS-SCHR-38/05, August 2005

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

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Toronto, Canada, 27-29 September 2011

Content

- Introduction
- Differences between a PSA compiled for nuclear power plants (NPPs) and for nuclear fuel cycle facilities (NFCFs)
- Basic concept of a PSA for nuclear fuel cycle facilities
- Example of an application

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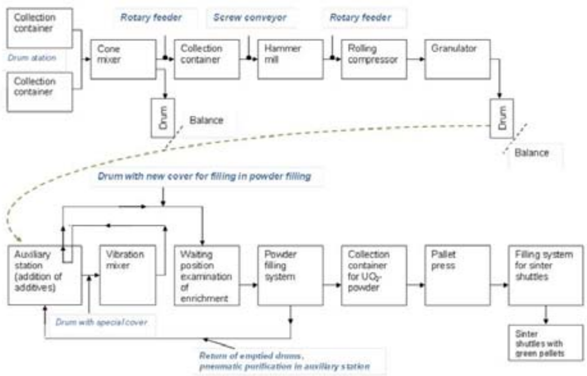
Example of application: Schematic view of the analyzed sub-process



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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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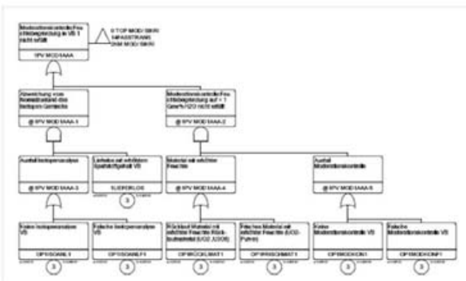
Example of a fault tree



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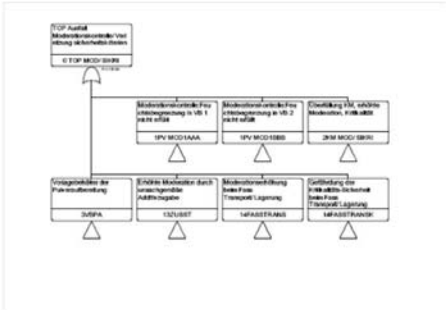
Example of a fault tree (cont.)



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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	$4{,}1 \cdot 10^{-8}$

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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

DEVELOPMENT OF ISA PROCEDURE FOR URANIUM FUEL FABRICATION AND ENRICHMENT FACILITIES: OVERVIEW OF ISA PROCEDURE AND ITS APPLICATION

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Abstract - Integrated Safety Analysis (ISA) procedure for uranium fuel fabrication and enrichment facilities has been developed for aiming at applying risk-informed regulation to these uranium facilities. The development has carried out referring to the ISA (NUREG-1520) by the Nuclear Regulatory Commission (NRC).

The paper presents purpose, principles and activities for the development of the ISA procedure, including Risk Level (RL) matrix and grading evaluation method of IROFS (Items Relied on for Safety), as well as general description and features of the procedure. Also described in the paper is current status in application of risk information from the ISA.

Japanese four licensees of the uranium facilities have been conducting ISA for their representative processes using the developed procedure as their voluntary safety activities. They have been accumulating experiences and knowledge on the ISA procedure and risk information through the field activities. NISA (Nuclear and Industrial Safety Agency) and JNES (Japan Nuclear Energy Safety Organization) are studying how to use such risk information for the safety regulation of the uranium facilities, taking into account the licensees' experiences and knowledge.

1. Introduction

NISA sets a goal of enhancing scientific justification in the safety regulations as well as realizing effective and efficient safety regulations as one of their major goals. To achieve the goal, NISA envisages using risk information as an effective tool[1].

JNES has been developing its own ISA procedures since 2004 intending to apply it to the uranium facilities in operation by four different licensees in Japan, by referring to the ISA used by the NRC[2-5] and other relevant documents[6-9]. JNES finalized the ISA procedure for the facilities in March 2010[10].

In 2010, the four licensees implemented ISA for their representative processes according to the ISA procedure prepared by JNES as part of their voluntary safety activities. Subsequently in 2011, the licensees plan to use the risk information including IROFS in addition to conventional safety activities as a test case, to accumulate knowledge on the effectiveness of the risk information obtained as a result of the ISA. NISA will reflect the results of the voluntary safety activities by the licensees to the promotion of application of the risk information in the safety regulation with a technical support from JNES.

2. Purpose, principles and activities for the development of the ISA procedure

2.1 Purpose for the development of the ISA procedure

Purpose of the JNES's development of the ISA procedure is to establish an appropriate risk analysis method to promote risk-informed regulation for the uranium facilities.

Main risk information to get is as follows;

(1) Exhaustive identification of accident sequences that could lead to exposure

(2) Determining IROFS[42], and

(3) Determining importance of the IROFS (i.e. Grading of the IROFS).

2.2 Principles for the development of the ISA procedure

Principles set for the development of the ISA procedure include;

(1) Use of risk information, for the time being, in the ex-post regulations (those after granting license) such as inspection of the facilities; accordingly, internal events[43] that are much relevant to the ex-post regulations, rather than external events, should be analyzed in the ISA ,

(2) Both radiological and chemical exposure should be analyzed for general public at the peripheral supervised area boundary (offsite public), and workers in the facilities,

(3) Analysis should be conducted for the period of the maintenance/inspection in addition to those of normal operation, startup and shutdown operations,

(4) Consequences and likelihood of occurrence of accident sequences should be categorized depending on their severity and frequency.

(5) The Risk Level (RL) of accident sequences should be expressed by the products of the integers assigned to the respective Consequence Category (CC) and Likelihood Category (LC); that is, RL = CC×LC. These products are compiled in a table to form a RL matrix, where acceptable RL bound should be defined. These RL should be used as an index to decide sufficiency of the safety measures.

(6) Methods should be developed to select IROFS required to meet the acceptable RL, and to evaluate importance of the selected IROFS.

2.3 Activities for developing the ISA procedure

In order to develop a practical standard ISA procedure, following activities were made;

(1) Provisional application to representative processes of the uranium facilities in operation,

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[42] IROFS are structures, systems, components and operators' actions to prevent potential accident (i.e. to prevent initiating event and progress of the sequences) and to mitigate its consequences.

[43] Internal events are a type of initiating events that occur inside the facilities. These include, for example, events occurred by causes inherent to facilities or systems, such as random failure of components or erroneous operation by workers. However, loss of external power due to events such as thunderbolt is considered as an internal event.

(2) Interviewing four licensees who made a voluntary ISA according to the draft ISA procedure about their implementation processes and ISA results,

(3) Survey of additional published information by NRC[6, 7] and discussion with NRC staff,

(4) Incorporation of the PSA procedure for MOX fuel fabrication facility by the Japan Atomic Energy Agency (JAEA)[11], and,

(5) Consultation with experts at the "The expert panel for Use of Risk Information at Fuel Fabrication and Enrichment Facilities" established within JNES. It was useful in establishing a policy for determining LC based on the tentative safety goals for nuclear facilities[12].

2.4 Risk Level (RL) matrix

(1) Setting Consequence Category (CC)

The Consequence Category (CC) for radiological and chemical exposures is shown in **Table 1**.

Radiological consequences were grouped into four categories. Hazards requiring IROFS were defined as CC $\geq$ 3 (> 5 mSv) for the offsite public, and the CC $\geq$ 2 (> 50 mSv) for the workers.

Chemical consequences were grouped into three categories referring to the NRC categories[3]. Note that criteria for the chemical exposures are based on the Acute Exposure Guideline Level (AEGL)[13] established by USEPA, because there are no such criteria in Japan.

(2) Setting Likelihood Category (LC)

The Likelihood Category (LC) for accident sequence consists of four as shown in **Table 2**.

(a) Upper occurrence limit of the LC1 "Extremely unlikely"

The upper limit of the LC1 is set as 10^{-6}/year, based the reasons described below, that it meets, with a margin, the quantitative tentative target values of the average fatality risk by cancer for public member within a certain distance from the concerned facility[44], as defined in the safety goal[12] by NSC of Japan to be applied to all nuclear facilities;

i. For hypothetical criticality accident of radiological exposure analysis, dose to individual in the outdoor at 100 m from the uranium facility is estimated to be 1 Sv at maximum[45],

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[44] The interim report[12] includes two types of quantitative tentative target values; average acute fatality risk for public member around the site boundary of a concerned facility and average fatality risk by cancer for public member within a certain distance from the concerned facility. As described later, effective dose to the offsite member will be less than 1 Sv even in the hypothetical criticality accident for the uranium facilities. The acute fatality will occur at the dose level of 1 Gy (1 Sv for γ-radiation) or more[14, 15], therefore no acute fatality risk is considered and only fatality risk by cancer need to be considered.

[45] In the JCO criticality accident where a total nuclear fission was 2.5×10^{18}, the dose that a hypothetical man who remained outdoor at 100 meters away from the site might have received is estimated to be 53 mSv[17]. Meanwhile, the total nuclear fissions in the worst criticality accident in the past in nuclear fuel facilities (Idaho, USA in 1959)[18] was 4×10^{19}. Based on these data, the dose at

ii. Fatality rate for the radiological dose of 1 Sv is 0.1 death/Sv [16],

iii. Likelihood of occurrence corresponding to the fatality risk by cancer of 10^{-6}/year is 10^{-5}/year (=10^{-6} death/year / (1 Sv × 0.1 death/Sv)),

iv. Meanwhile, because risk level of the uranium facility could be reduced to sufficiently low; such as one order smaller than this value, therefore, the upper limit is set as 10^{-6}/year.

(b) Upper occurrence limit of the LC2 "Highly unlikely"

Similarly to the case of **(a)**, the likelihood of occurrence corresponding to the fatality rate by cancer of 10^{-6}/year is 4×10^{-5}/year (=10^{-6} death/year / (0.25 Sv × 0.1 death/Sv)).

Meanwhile, for the offsite public and the workers, the lower occurrence limit of the radiological exposure for the CC3 is one order lower than that for the CC4.

Consequently, the upper occurrence limit for the LC2 is set as 10^{-5}/year, one order of magnitude higher than that for the LC1 of 10^{-6}/year.

(c) Upper occurrence limit of the LC3 "Unlikely"

Similarly to the case of **(a)** and **(b)**, the likelihood of occurrence corresponding to the fatality rate by cancer of 10^{-6}/year is 4×10^{-3}/year (=10^{-6} death/year / (0.005 Sv × 0.05 death/Sv)), using cancer fatality rate of 0.05 death/Sv [16] for the radiological exposure.

Meanwhile, the likelihood that an accident would occur once during the facility life time, i.e. approximately 10^{-2}/year, will fall within the LC4 "Not unlikely".

Consequently, the upper occurrence limit for the LC3 "Unlikely" (i.e. lower limit for the LC4 "Not unlikely") is set as 10^{-3}/year.

(3) Risk Level (RL) matrix

The Risk Level (RL) matrix defined as above is shown in **Figure 1**. In the matrix, the RL in white box is considered as acceptable level. Safety measures with the RL in shaded box need to be revised or required additional IROFS.

Furthermore, the exposure risk in LC 4 should be confirmed to be low enough by the IROFS and candidates of other accident sequences so that it does not exceed the cancer fatality rate.

2.5 Grading evaluation method of IROFS

IROFS should be selected based on both RL matrix and the following rules;

(i) Priority in selection of IROFS

(a) Should select preferentially engineered controls and enhanced administrative controls (e.g. combined an alarm with a personnel activity) over administrative controls.

(ii) From the viewpoint of accident prevention

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100 meters away can be calculated to be 848 mSv (= 53 mSv × $4\times10^{19}/2.5\times10^{18}$), suggesting that dose by the worst criticality accident at the uranium facilities will not be greater than 1 Sv.

(a) For hazards in CC 3, should select one preventive IROFS, at least.

(b) For hazards in CC 4, should select two or more preventive IROFS.

The grading of the selected IROFS is evaluated by two different methods; risk-based and scenario-based.

(1) Risk-based grading evaluation method

For the selected IROFS, this method evaluates contribution of the IROFS to exposure risks on offsite public and workers separately, using an importance index for accident sequences of the whole facility in principle. Basic procedure includes;

(i) Calculate exposure risks of each accident sequence, which should be summed for the whole facility to obtain total exposure risk of the facility: R,

(ii) The following two indices should be used for the grading evaluation of IROFS in the facility.

(a) FV (Fussell-Vesely) importance

This index indicates what extent the failure of the concerned IROFS will contribute to the assumed accident sequence, as defined in the formula [1].

$$FV = (R - R(X_i=0)) / R \qquad [1]$$

Where, $R(X_i=0)$ is sum of exposure risks of the accident sequence assumed to occur in the facility when the failure probability X_i of $IROFS_i$ is zero.

(b) RAW (Risk Achievement Worth)

This index indicates the degree of increase in the exposure risk when the concerned IROFS always fail, as defined in the formula [2].

$$RAW = R(X_i=1) / R \qquad [2]$$

Where, $R(X_i=1)$ is sum of exposure risks of accident sequence assumed to occur in the facility when the failure probability X_i of $IROFS_i$ is 1.

(iii) Select IROFS with the importance “High” based on the calculated FV and RAW values for the IROFS, and criteria established depending on features of the facility to be evaluated and evaluation purposes. The criteria could be set referring to $FV \geq 0.005$ or $RAW \geq 2$ which are extensively used for nuclear power plants[19, 20].

(2) Scenario-based grading evaluation method

This method weighs IROFS depending on relevant of IROFS to the deviation from constraints, placing emphasis on such constraints as nuclear criticality and thermal limits as well as limiting conditions on radioactive release during the power outage; IROFS that corresponds to the followings should be considered that with importance “High”;

i. IROFS, the loss of whose function directly results in deviation of the constraints,

ii. IROFS right after the initiating event, when occurrence of the initiating event could become deviation from the constraint, and

iii. Engineered IROFS or enhanced administrative IROFS that is relevant to the safety of the offsite public after the deviation from the constraints.

(3) Determination of importance of IROFS

Based on the risk-based importance and scenario-based importance of IROFS as described above **(1)** and **(2)**, the importance of IROFS should be determined considering the following (i) through (iv);

i. When the both importance are "High", the importance of the IROFS should be determined as "Rank A", and when they are "Low", it should be determined as "Rank B".

ii. If one of them is "High" and the other "Low", the reasons and causes of the difference should be examined.

iii. If it proves that the reason of "High" scenario-based importance is due to extreme conservatism, or "High" risk-based importance is due to excessive failure rate, the importance of the IROFS should be determined as "Rank B". On the other hand, if the rational reason to define it "Rank B" could not be identified, the importance of the said IROFS should be remained as "Rank A".

iv. For IROFS that involves several accident sequences, when the IROFS should be determined as "Rank A" based on any accident sequence that results in radiological or chemical exposure on the offsite public or workers, then the IROFS should be determined as "Rank A".

3. General description and features of the ISA procedure

3.1 General description of the ISA procedure

General description of the developed ISA procedure is illustrated in **Figure 2**. The points are;

In (4), for the identified hazards, calculate the release of radioactive material and chemicals by the Five Factor Formula[5] under the condition without safety measures[46] to evaluate radiological and chemical exposure on the offsite public and workers. Based on the evaluation results, a CC value should be assigned. Note that in the case of criticality accident in the uranium facilities, the CC value of 4 should be assigned without evaluation in this stage, because of apparent severe radiological exposure to workers. Hazards, which could result in larger radiological exposure than the specified value (see **2.4 (1)**), should be selected as hazards requiring IROFS. The maximum value of 4 should be conservatively assigned to the LC without evaluation in this stage.

In (5), for the selected hazards requiring IROFS, accident sequences should be determined considering safety measures provided to the current facilities based on event tree or others. Note that safety measures relevant to the accident sequences should be IROFS candidate at this stage. Evaluate consequence and likelihood of occurrence for each accident sequence considering the IROFS candidates. The CC should be determined based on the evaluation of exposure according to **Table 1**.

[46] The confinement capability of static components (other than connections and seals), buildings and structures shall be maintained, because of their extremely unlikely to suffer from failures by aging.

The LC should be determined according to **Table 2** based on the evaluation using parameters[47] including likelihood of occurrence of initiating events, components failure rate, human error rate and failure duration. Product of the category numbers for CC and LC should become a RL of the accident sequence.

In (6), determine whether the RL of the accident sequence is acceptable (i.e. falls in white box in the RL matrix in **Figure 1)**. Select IROFS from among IROFS candidates that satisfy the acceptable RL. If any IROFS candidates selected would be unacceptable, additional IROFS candidate should be evaluated to identify IROFS that satisfy the acceptable RL. Conduct grading evaluation of the selected IROFS.

3.2 Features of the ISA procedure

Features of the ISA procedure include;

(1) Two steps of HAZOP analysis to identify initiating events and hazards systematically and exhaustively; preparation of an initiating event review sheet to compile initiating events in the first step and identification of hazards in the second step.

(2) Selection of hazards requiring IROFS based only on CC value assuming the hazard could occur, without evaluating LC.

(3) For the radiological exposure, four CC and their boundaries were set based on laws and regulations in Japan. The four LC and their boundaries were defined with a margin to meet safety goals of the nuclear facilities[12]. These allowed formulation of a detailed RL matrix.

(4) IROFS were selected based on RL value as well as from the viewpoint of accident prevention, and from preferring engineered and enhanced administrative controls over administrative controls.

(5) Importance of IROFS was determined by reviewing and analyzing the results of both risk-based and scenario-based grading evaluation of IROFS.

The ISA procedure developed by JNES with features described above may contain possible responses to the general observation identified during the review of ISA Summary in USA described in the reference 6. The features are outlined in **Table 3** for further information.

4. Current use of risk information in uranium facilities

The four licensees have revised the ISA results obtained until December 2010 for representative processes based on NISA and JNES review, and reflect the revise to their voluntary safety activities.

Taking into account knowledge obtained from information on the licensees' activities, JNES will prepare an evaluation manual to ISA results and review the licensees' ISA results; revise the ISA procedure; prepare a draft basic guidance for use of risk information; prepare a draft manual of technical bases specified in the design and construction authorisation.

NISA, licensees and JNES will exchange information periodically, and will revise the approach, if necessary.

[47] Including index tables, Table A-9 and A-10, shown in Appendix A, Section 3, NUREG-1520 Rev.1[3].

5. Conclusion

The ISA procedure developed by JNES is considered to be a practical standard procedure that would allow identification of IROFS requiring safety measures and appropriate grading evaluation of the IROFS in the uranium facilities; which could also allow use of risk information obtained by the ISA in, for example, revising inspection items, method and frequency to achieve more effective and efficient inspection system. In addition, combination with information on the type of hazards, accident sequences, severity of the consequences and component failure rates would allow achieving more effective and transparent safety regulation.

In March 11, 2011, severe accidents involving loss of all power supplies occurred at the Fukushima Daiichi Nuclear Power Station of TEPCO due to the Great East Japan Earthquake. Such accidents will definitely add a momentum to discussion on reviewing the regulations. In the development of ISA procedure, external events such as earthquake and tsunami will also be addressed.

Table 1**. Consequence Category (CC) for Accident Sequences** (Radiological and Chemical Exposure)

Consequence Category	**Offsite public (At peripheral supervised area boundary)**	**Workers**
4	RD[a)] > 250 mSv[c)] (Deterministic effects)	RD > 1 Sv[g)]
	CD[b)] > AEGL[d)]-2 (Irreversible or other serious long-lasting health effects)	CD > AEGL-3 (Endanger life)
3	5 mSv[e)] < RD ≤ 250 mSv (Exceeding health risk that is considered small)	250 mSv[c)] < RD ≤ 1 Sv (Deterministic effects)
	AEGL-1 < CD ≤ AEGL-2 (Effects such as significant discomfort or stimulation)	AEGL-2 < CD ≤ AEGL-3 (Irreversible or other serious long-lasting health effects)
2	1 mSv[f)] < RD ≤ 5 mSv	50 mSv[h)] < RD ≤ 250 mSv
1	Accident with less radiological/ chemical exposure than those shown above	Accident with less radiological/ chemical exposure than those shown above

a) Radiological Dose (in effective dose)
b) Chemical Dose
c) Guide in Rreactor Site Evaluation (May 27, 1964)
d) Acute Exposure Guideline Level[13]
e) Guide for Safety Review of Reprocessing Plants (April 2001)
f) Article 3, Notice No. 13 Regarding to dose limits in accordance with the provisions in the Rule regarding Nuclear Fuel Fabrication Business (December 2000)
g) ISA standard in USA (64 FR 41338)
h) Article 6, Notice No. 13 Regarding to dose limits in accordance with the provisions in the Rule regarding Nuclear Fuel Fabrication Business (December 2000)

Table 2. **Likelihood Category (LC) of Accident Sequences**

Likelihood Category (LC)	**Likelihood of occurrence of accident sequence**	**Definition of Likelihood of occurrence (L per event)**
1	Extremely unlikely	$L \leq 10^{-6}$/year
2	Highly unlikely	10^{-6}/year < $L \leq 10^{-5}$/year
3	Unlikely	10^{-5}/year < $L \leq 10^{-3}$/year
4	Not unlikely	10^{-3}/year < L

Table 3**. Outline of features in JNES's ISA procedure**

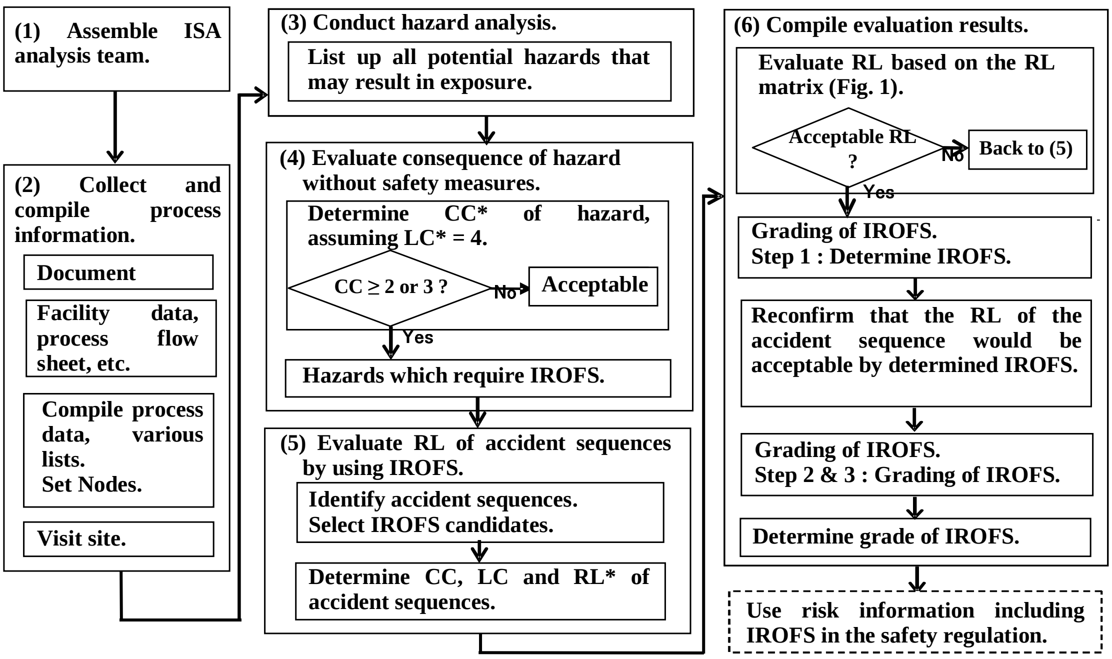
Items	Features in JNES[10]	Notes: As in NRC
Purpose	Grading evaluation of IROFS and their utilisation in inspection, etc.	Reviewing the validity of design and utilisation of risk information in the core inspection.
(1) Risk Level (RL) matrix	For the radiological exposure, four consequence categories and their boundaries were set based on laws and regulations in Japan. Also four likelihood categories were set and their boundaries were defined with a margin to meet safety goals of the facilities[12]. These allowed formulation of a detailed RL matrix.	Three categories for both consequence and likelihood of occurrence[3]; No regulatory definition on bounding value for the likelihood categorisation. (Highly unlikely, unlikely, etc are defined by licensees and authorized by NRC[3])
(2) Analysis scope	Subjects of the ISA should be internal events that are much relevant to safety regulation such as inspection, and therefore external events are postponed.	Internal events and external events (including natural phenomena such as earthquake, flooding, tornado, tsunami and hurricane[3])
(3) Hazard analysis method	Two steps of HAZOP analysis was employed to identify initiating events and hazards systematically and exhaustively; preparation of an initiating event review sheet to compile initiating events in the first step and identification of hazards by HAZOP or FMEA technique in the second step.	What-If technique, HAZOP technique, FMEA. (A variation of the What-if technique was in practice used which was less time consuming compared to other techniques[7])
(4) Hazard screening method	Selection of hazards requiring IROFS based only on Consequence Category (CC) value assuming the hazard could occur, without evaluating Likelihood Category (LC).	Evaluate consequence and likelihood of occurrence of hazards, and whether or not IROFS is required is determined based on its RL.
(5) Likelihood evaluation method	The LC was determined based on the evaluation using parameters including likelihood of occurrence of initiating events, components failure rate, human error rate (including index table in the reference 3) and failure duration.	Index method, reliability data-based method or qualitative method. (Most licensees employ the index method[6])
(6) Identification of IROFS and its approach	IROFS were selected based on RL value as well as from the viewpoint of accident prevention, and from preferring engineered and enhanced administrative controls over administrative controls.	Compliance to performance requirements. Nuclear criticality need to be prevented[2]. (IROFS is basically selected at licensee's judgment)
(7) IROFS grading evaluation	Importance of IROFS was determined by reviewing and analyzing the results of both risk-based and scenario-based grading evaluation of IROFS in order to reflect in the study for improving safety regulations such as inspection.	None.
Procedure guide	Draft ISA procedure guide was prepared in March 2010.	Different guidance documents have been issued[3-5].

Figure 1. **Risk Level (RL) matrix**
(Left: For offsite public; Right: For workers; RL = CC×LC; RL in white box is acceptable.)

For Offsite Public		Likelihood Category (LC)			
		1	2	3	4
Consequence Category (CC)	4	4	8	12	16
	3	3	6	9	12
	2	2	4	6	8
	1	1	2	3	4

For Workers		Likelihood Category (LC)			
		1	2	3	4
Consequence Category (CC)	4	4	8	12	16
	3	3	6	9	12
	2	2	4	6	8
	1	1	2	3	4

Figure 2**. Overview of ISA procedure**



Note* : CC = Consequence Category.
LC = Likelihood Category.
RL = Risk Level (= CC×LC)

References

[1] Near-term implementation plans for risk-informed regulation, NISA and JNES, May（2005）(in Japanese)

[2] 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material," 72 Federal Register 63974, Nov.14（2007）

[3] NUREG-1520 Rev.1, Integrated Safety Analysis Standard Review Plan, May（2010）

[4] NUREG-1513, Integrated Safety Analysis Guidance Document, May（2001）

[5] NUREG/CR-6410, Nuclear Fuel Cycle Facility Accident Analysis Handbook, March（1998）

[6] K. J. Morrissey, C. S. Tripp, et al., Transition from a traditional deterministic nuclear criticality safety framework to a risk-informed Integrated Safety Analysis approach: ISA summary, The 8th International Conference on Nuclear Criticality Safety, St Petersburg, May（2007）

[7] C. S. Tripp, F. H. Burrows, et al., Transition from a traditional deterministic nuclear criticality safety framework to a risk-informed Integrated Safety Analysis approach: methodology, The 8th International Conference on Nuclear Criticality Safety, St Petersburg, May（2007）

[8] AIChE, Guidelines for Hazard Evaluation Procedures, Second Edition with Worked Examples, Center for Chemical Process Safety, New York（1992）

[9] AIChE, Layer of Protection Analysis, Center for Chemical Process Safety, New York（2001）

[10] K. Yamate, T. Arakawa, et al., "Development of ISA Procedure for Uranium Fuel Fabrication and Enrichment Facilities," Transactions of the atomic energy society of Japan, Vol.10, No.2, p.105-118 (2011) (in Japanese)

[11] H. Tamaki, K. Yoshida, "PSA Procedure Guide for MOX Fuel Fabrication Facility," JAEA-Technology 2010-004 (2010) (in Japanese)

[12] NSC, Interim Report on Examination of Safety Goals for Nuclear Facilities, Committee on safety goal, Dec. (2003) (in Japanese)

[13] USEPA, Acute Exposure Guideline Levels, http//www.epa.gov/opt/aegl/process.htm, April（2004）

[14] NUREG/CR-4214 Rev.2, Health Effects Models for Nuclear Power Plant Accident Consequence Analysis, Part Ⅰ ITRI-141, Oct.（1993）

[15] ICRP Publication 103（2007）

[16] ICRP Publication 60（1990）

[17] NSC, "The Final Report of Investigation Subcommittee on the Criticality Accident at the Conversion Facility of the JCO," (1999) (in Japanese)

[18] JNES, "Survey and Analysis of Accident and Troubles in Reprocessing Plants and Other Facilities," JNES/SAE04-015 (2004）(in Japanese)

[19] Nuclear Energy Institute, 10 CFR 50.69 SSC Categorization Guideline, NEI00-04 (Rev0), July（2005）

[20] Nuclear Energy Institute, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, NUMARC 93-01 Revision 3, July（2000）

Development of ISA Procedure for Uranium Fuel Fabrication and Enrichment Facilities
－ ISA procedure and its application －

NEA/CSNI Workshop on
Safety Assessment of Fuel Cycle Facilities
Regulatory Approaches and Industry Perspectives

Toronto, Canada
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Japan Nuclear Energy Safety Organization (JNES)
Takashi Yamada

Contents

1. Purpose, Principles and Framework of the development

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Purpose of the development

Purpose :

Establishing an appropriate risk analysis method to promote risk-informed regulation for uranium facilities.

Main risk information to get :

1. Exhaustive accident sequences leading to exposure
2. IROFS (Items Relied on for Safety)*
3. Graded IROFS, etc.

*IROFS : Structures, systems, equipment, components, and activities of personnel that are relied on to prevent potential accidents at a facility or to mitigate their potential consequences.

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Principles for the development

1. Use of risk information in the ex-post regulations such as inspection of the facilities
2. First priority to internal events that are much relevant to the ex-post regulations
3. Both radiological and chemical exposure analyses for both offsite public and workers
4. Risk analyses for the duration of maintenance/inspection in addition to normal and startup/shutdown operations
5. Categorization of consequences and likelihood of occurrence of accident sequences, depending on their severity and frequency
6. Development of the ways for selection of IROFS to meet the acceptable risk level and for grading of the selected IROFS

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Uranium facilities in Japan

Licensee	Japan Atomic Energy Agency (JAEA) Ningyo-Toge Environmental Engineering Center	Japan Nuclear Fuel Limited	Global Nuclear Fuel Japan Co., Ltd.	Mitsubishi Nuclear Fuel Co., Ltd.	Nuclear Fuel Industries, Ltd. Kumatori Works	Nuclear Fuel Industries, Ltd. Tokai Works
Licensed Date	Oct 18, 1985	1st phase : Aug 10, 1988 2nd phase : July 12, 1993	Aug 30, 1968	Jan 11, 1972	Sep 1, 1972	Sep 29, 1978
Facility	Enrichment (centrifugal method)	Enrichment (centrifugal method)	Fuel fabrication (for BWR)	Re-conv. & Fuel fab. (for PWR)	Fuel fabrication (for PWR)	Fuel fabrication (for BWR)
Date of Operation start	Mar 10, 1988 (Shutdown in 2001)	Sep 27, 1991	Aug 29, 1970	July 28, 1972	Sep 1, 1972	Jan 4, 1980
Max. enrichment	5% UF_6	5% UF_6	5% UO_2	5% UO_2	5% UO_2	5% UO_2
Max. capacity	200t-SWU/y	1050t-SWU/y	750t-U/y	Re-conv. : 450t-U/y Fuel fab. : 440t-U/y	284t-U/y	200t-U/y
Current Status	Terminated	In operation	In operation	In operation	In operation	In operation

Note : JAEA's facility was established for research and development.

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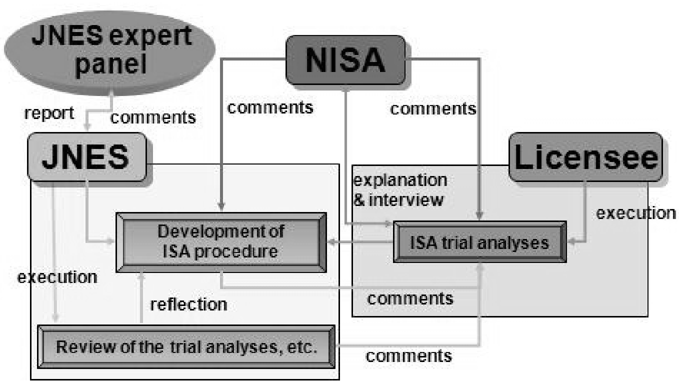
Framework of the Development



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2. ISA procedure developed

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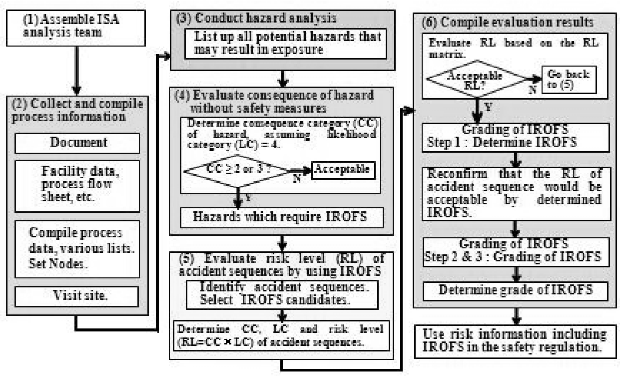
ISA procedure developed



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3. Setting of Risk Level matrix

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Consequence Category (CC) for radiological exposure

In order to reflect the ISA results to improvement of inspections and maintenances, low-level radiological exposures are needed to be closely evaluated.

Consequences of radiological exposure are classified into four categories. Each category is assigned a number 1, 2, 3 or 4.

Category	Offsite public (At peripheral supervised area boundary)	Facility workers
4	* RD > 250 mSv (Deterministic effects)	RD > 1 Sv (Life-threatening)
3	5 mSv < RD ≤ 250 mSv (Exceeding health risk that is considered small)	250 mSv < RD ≤ 1 Sv (Deterministic effects)
2	1 mSv < RD ≤ 5 mSv	50 mSv < RD ≤ 250 mSv
1	RD ≤ 1 mSv	RD ≤ 50 mSv

* Radiological Dose (in effective dose)

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Exposure level in regulations / standards

:Fuel cycle facilities in Japan :Nuclear reactors in Japan :Criterion used in USA

	Exposure level	Regulations and standards	Remarks
Facility workers	1 Sv	Not stipulated in Japan; a criterion for the ISA in the USA	Based on 64 FR 41338 in the USA
	250 mSv	"Guideline for siting nuclear reactors" (May 27, 1964)	For whole body
	50 mSv	Article 6, "Notification No. 13 specifying radiation dose limit in accordance with the provision in the rules regulating nuclear fuel fabrication businesses" (December 26, 2000)	Dose limit for radiation workers (per year) in ICRP Pub. 60 (1990)
Offsite public	250 mSv	"Guideline for siting nuclear reactors" (May 27, 1964)	For whole body
	5 mSv	"Examination guide for safety design of light water nuclear power reactor facilities" (March 2001), "Examination guide for the safety of reprocessing facilities" (April 2001)	Risk may be considered "low" if the estimated effective dose for the public is below 5 mSv per accident.
	1 mSv	Article 3, "Notification No. 13"	Concentration limit in the area other than supervised area (for one year), ICRP Pub. 60 (1990)

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Consequence Category (CC) for chemical exposure

Consequences of chemical exposure are classified into three categories by referring to the NRC categorization. Each category is numbered 1, 3 or 4, considering consistency with CC for radiological exposure.

Category	Offsite public	Facility workers
4	* CD > ** AEGL-2 (Long-lasting or serious health effects)	CD > AEGL-3 (Life-threatening)
3	AEGL-1 < CD ≤ AEGL-2 (Effects such as extreme discomfort, or strong irritation)	AEGL-2 < CD ≤ AEGL-3 (Long-lasting or serious health effects)
1	CD ≤ AEGL-1	CD ≤ AEGL-2

*CD = Chemical dose (exposure)
**AEGL: Acute Exposure Guideline Level by USEPA

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Likelihood Category (LC)

Likelihood category		Likelihood of accident occurrence	
1	Extremely unlikely	$\leq 10^{-6}$/year	(Frequency index ≤ -6)
2	Highly unlikely	$> 10^{-6}$/year, $\leq 10^{-5}$/year	(Frequency index ≤ -5)
3	Unlikely	$> 10^{-5}$/year, $\leq 10^{-3}$/year	(Frequency index ≤ -3)
4	Not unlikely	$> 10^{-3}$/year	(Frequency index > -3)

Note : Likelihood category is common to both offsite public and facility workers.

Definition of boundary of likelihood category :

(1) Frequency index -6: Defined within the range of the probability of cancer fatality (10^{-6}death/y) shown in the safety goal (draft) in Japan

(2) Frequency index -5: Defined within the range of the probability of cancer fatality, considering radiation dose for the consequence categories 4 & 3

(3) Frequency index -3: Defined within the range of the probability of cancer fatality, considering likelihood of accident occurrence during the facility lifetime

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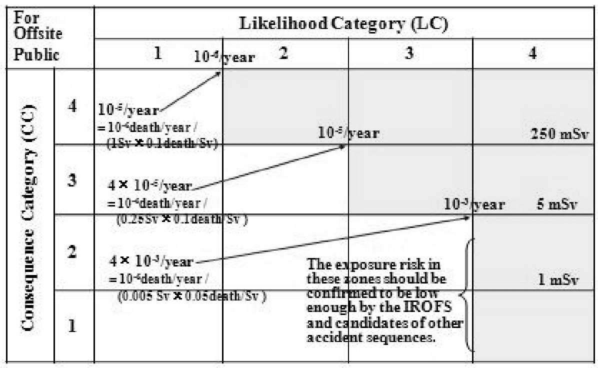
Definition of boundary of likelihood category



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Risk Level (RL) matrix

The Risk Level (RL) is defined as the product of CC and LC.
Check RL of individual accident sequence. If it is in yellow box, measures should be taken by providing additional IROFS.

For Offsite Public		Likelihood Category (LC) 1	2	3	4
Consequence Category (CC)	4	4	8	12	16
	3	3	6	9	12
	2	2	4	6	8
	1	1	2	3	4

For Facility Workers		Likelihood Category (LC) 1	2	3	4
Consequence Category (CC)	4	4	8	12	16
	3	3	6	9	12
	2	2	4	6	8
	1	1	2	3	4

(for radiological and chemical exposure)

Distribution of RL of accident sequences found from trial analyses for representative processes of uranium facilities

		Likelihood Category (LC) 1	2	3	4
Consequence Category (CC)	4	Mostly workers and partly public			
	3				
	2				
	1	Workers and public		Workers and public	

4. Grading method of IROFS

Selection rules of IROFS

I. Priority in selection of IROFS

(a) Should select preferentially engineered controls and enhanced administrative controls (e.g. combined an alarm with a personnel activity) over administrative controls.

II. From the viewpoint of accident prevention

(a) For hazards in CC 3, should select one preventive IROFS, at least.

(b) For hazards in CC 4, should select two or more preventive IROFS.

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Grading method of IROFS

Risk-based grading method

Evaluating the contribution of the IROFS to exposure risks on offsite public and workers separately.

By using FV (Fussell-Vesely) and RAW (Risk Achievement Worth) indexes for accident sequences of the whole facility in principle.

Scenario-based grading method

Considering IROFS, which take part in deviating from constraints, to be important for safety.

Constraints are such as nuclear criticality and thermal limits.

RAW		
2	III	I
	IV	II
	FV 0.005	

Risk-based importance:
Zone I, II, III → High
Zone IV → Low

Determination of importance of IROFS

IROFS	Risk-based	Scenario-based	Importance
Case 1	High	High	Rank A
Case 2	High	Low	Rank A or B[1)]
Case 3	Low	High	
Case 4	Low	Low	Rank B

1) For determining the importance of IROFS, the reasons and causes of the difference should be examined.

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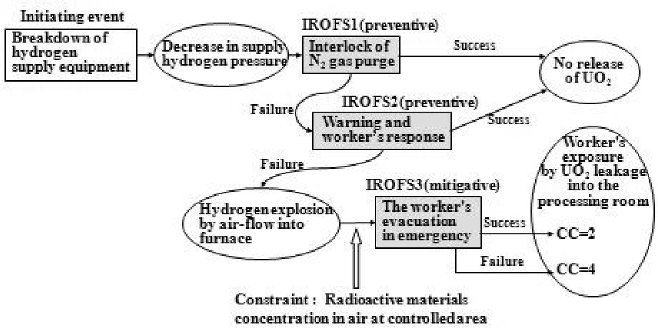
An example of analysis of worker's exposure caused by hydrogen explosion in a sintering furnace



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An example of grading result of IROFS

IROFS	Risk-based	Scenario-based	Importance
Interlock of N_2 gas purge	Low	Low	Rank B
Warning and worker's response	Low	High	Rank A
Worker's evacuation in emergency	High[1)]	Low	Rank A

1) Effective in the exposure reduction. And there are a number of sequences that the worker's evacuation becomes effective.

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5. Features of ISA procedure developed

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Outline of features of ISA procedure (1/3)

Items	Features of ISA procedure	Notes : As in NRC
Purpose	Grading of IROFS and the use of the results for inspection, etc.	Review of the validity of design and use of risk information for the core inspection.
(1) Risk Level (RL) matrix	For the radiological exposure, four CCs and their boundaries were set based on laws and regulations in Japan. Also four LCs were set and their boundaries were defined with margin to meet safety goals (draft) of nuclear facilities.	Three categories for both CC and LC ; No regulatory definition of boundary for LC (Highly unlikely, unlikely, etc are defined by licensees and authorized by NRC)
(2) Analysis scope	Internal events are the first target. External events are the next.	Internal events and external events (including natural phenomena such as earthquake, flooding, tornado, tsunami and hurricane)

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Outline of features of ISA procedure (2/3)

Items	Features of ISA procedure	Notes : As in NRC
(3) Hazard analysis method	Two steps to identify initiating events and hazards systematically and exhaustively. In the first step : Preparation of an initiating event review sheet to compile initiating events In the second step : Identification of hazards by HAZOP or FMEA technique	What-If technique, HAZOP technique, FMEA. (A variation of the What-if technique was in practice used which was less time consuming compared to other techniques)
(4) Hazard screening method	Screening of hazards which require IROFS based only on CC without evaluating LC	Evaluate CC and LC of hazards, and IROFS is determined based on RL.
(5) Likelihood evaluation method	Use of reliable quantitative values for parameters such as initiating events, components failure rate, human error rate and failure duration. Use of index table in NUREG-1520 rev.1	Index method, reliability data-based method or qualitative method. (Most licensees employ the index method)

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Outline of features of ISA procedure (3/3)

Items	Features of ISA procedure	Notes : As in NRC
(6) Identifi-cation of IROFS	Identification of IROFS based on RL matrix and selection rules.	Compliance to performance requirements. Nuclear criticality need to be prevented. (IROFS is basically selected at licensee's judgment)
(7) Grading of IROFS	Grading of IROFS from both risk-based and scenario-based evaluation	None.
Procedure guide	Draft ISA procedure guide was prepared in March 2010.	Different guidance documents have been issued, such as NUREG-1520.

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6. Current use of risk information

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Current use of risk information for uranium facilities

1. Being independent from regulation, licensees have voluntarily started the use of risk information obtained from the ISA, aiming at data accumulation to understand the usefulness of the risk information to the safety.
2. Licensees will report the data to NISA.
3. NISA and JNES will review the data to know the effects of the use of risk information in the regulation.

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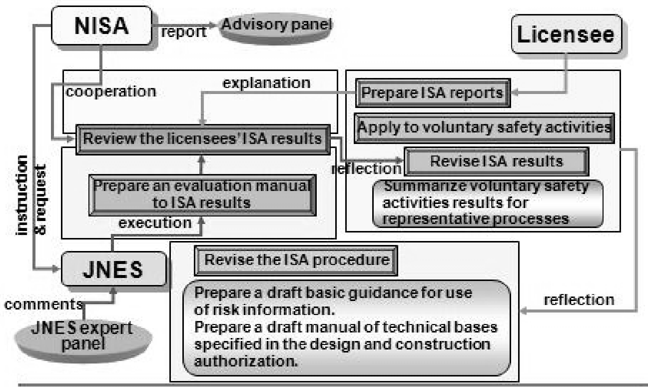
Framework of current use of risk information



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Conclusions

1. A practical ISA procedure (internal event oriented) for uranium facilities has well developed.
2. This ISA procedure enables ;
   - identification of IROFS, and
   - grading of the identified IROFS.
3. Licensees have started voluntary activity aiming at data accumulation to understand the usefulness of the risk information from the ISA to the safety.
4. NISA and JNES will review the data to know the effects of the use of risk information in the regulation.

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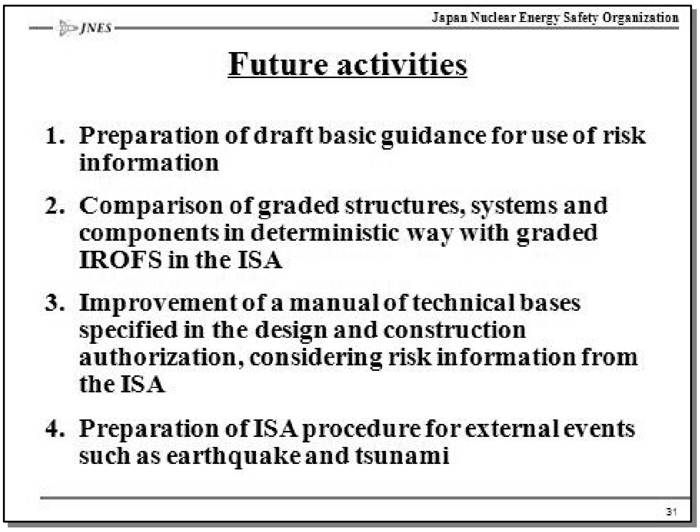
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Future activities
1. Preparation of draft basic guidance for use of risk information
2. Comparison of graded structures, systems and components in deterministic way with graded IROFS in the ISA
3. Improvement of a manual of technical bases specified in the design and construction authorization, considering risk information from the ISA
4. Preparation of ISA procedure for external events such as earthquake and tsunami
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INTEGRATED SAFETY ANALYSIS TO OPERATE WHILE CONSTRUCTING URENCO USA

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Abstract - The URENCO USA (UUSA) site in Lea County, New Mexico, USA is authorized by the U.S. Nuclear Regulatory Commission (NRC) for construction and operation of a uranium enrichment facility under 10 CFR 70 (Ref 1). The facility employs the gas centrifuge process to separate natural uranium hexafluoride (UF_6) feed material into a product stream enriched up to 5% U-235 and a depleted UF_6 stream containing approximately 0.2 to 0.34% U-235. Initial plant operations, with a limited number of cascades on line, commenced in the second half of 2010. Construction activities continue as each subsequent cascade is commissioned and placed into service.

UUSA performed an Integrated Safety Analysis (ISA) to allow the facility to operate while constructing the remainder of the facility. The ISA Team selected the What-If/Checklist method based on guidance in NUREG-1513 (Ref 2) and AIChE Guidelines (Ref 3). Of the three methods recommended for high risk events HAZOP, What-If/Checklist, or Failure Modes and Effects Analysis (FMEA), the What-If/Checklist lends itself best to construction activities. It combines the structure of a checklist with an unstructured "brainstorming" approach to create a list of specific accident events that could produce an undesirable consequence.

The What-If/Checklist for Operate While Constructing divides the UUSA site into seven areas and creates what-if questions for sixteen different construction activities, such as site preparation, external construction cranes, and internal construction lifts. The result is a total of 112 nodes, for which the Operate While Constructing ISA Team created hundreds of what-if questions. For each what-if question the team determined the likelihood, consequences, safeguards, and acceptability of risk. What-if questions with unacceptable risk are the accident sequences and their selected safeguards are the Items Relied on For Safety (IROFS).

The final ISA identified four (4) new accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61. The ISA Hazard Evaluation also identified eight (8) new IROFS to make the credible high consequence events highly unlikely and intermediate consequence events unlikely as appropriate. Details of how the ISA evolved into the final accident sequences and IROFS, including feedback from the regulator, are included in the paper. Lesson learned from months of operating and constructing with the IROFS in place will be shared.

Facility

The URENCO USA (UUSA) site in Lea County, New Mexico, USA is licensed (Ref 1) by the U.S. Nuclear Regulatory Commission (NRC) for construction and operation of a uranium enrichment facility under 10 CFR Part 70 (Ref 4). URENCO operates in a pivotal area of the nuclear fuel supply chain which ends with the sustainable generation of electricity for consumers around the world. The supply chain itself can be subdivided into four key processes: mining; conversion; enrichment; and fabrication. With its industry-leading centrifuge technology and around 27% global market share, URENCO is firmly positioned in the enrichment stage, which is the highest value added segment of the supply chain.

URENCO's principal activity is the provision of a service to enrich uranium to provide fuel for nuclear power utilities. The enrichment service is mostly provided on a toll basis using customers' uranium. URENCO fulfils its customer's requirements through its four operational enrichment plants in the UK, Germany, the Netherlands, and the US.

URENCO also owns a 50% interest in Enrichment Technology Company Limited (ETC), a joint venture company jointly owned with Areva. The UUSA facility employs industry leading gas centrifuges to separate natural uranium hexafluoride (UF_6) feed material into a product stream enriched up to 5% U-235 and a depleted UF_6 stream containing approximately 0.2 to 0.34% U-235. Initial plant operations, with a limited number of cascades on line, commenced in the second half of 2010. Construction activities continue as each subsequent cascade is commissioned and placed into service. Figure 1 shows the major buildings and areas of the UUSA facility.

Figure 1. **Arial Photo of URENCO USA Facility on 6-May-2011**

Workers at the facility are proud of their safety record with over 12 million man-hours of operation without a construction-related lost time accident. The record reflects the facility's focus on safe operations.

The facility features a modular design allowing operations in part of the facility while constructing adjacent parts. The separation process takes place in a Separations Building Module (SBM). An SBM houses the cascade halls and a UF_6 handling area with solid feed and take-off stations for supply and removal of UF_6 gas from the centrifuges. The centrifuges are arranged in a cascade and many cascades are grouped to form an assay. There are multiple assays in one SBM, each capable of producing product at a specific enrichment. In addition to the SBMs, the facility has a Centrifuge Assembly Building (CAB), Uranium By-Product Cylinder (UBC) Storage Pad, Cylinder Receipt and Dispatch Building (CRDB), Central Utilities Building (CUB), and Technical Support Building (TSB) which support operations in the SBMs.

Phased Approach

The original Safety Evaluation Report (SER) (Ref 5) and Integrated Safety Analysis (ISA) Summary meeting the requirements of 10 CFR 70.61(c) considered a complete facility as described in the original Safety Analysis Report (SAR). It did not consider construction activities or the impact of construction on operating process equipment. The SAR (Ref 6) and ISA Summary have since been revised to recognize startup of the facility is done in a phased approach.

Using the phased approach, Initial Plant Operation (IPO) may begin as soon as the required facilities, systems, processes, and Items Relied on for Safety (IROFS) are operational. As the construction of systems is completed, or nears completion, the systems are turned over from construction to operations responsibility. The facility operates in a series of phases determined by operational requirements and documented in the revised SAR.

Purpose of the ISA

Operating while constructing represents a change to the activities of personnel at the facility. The facility's configuration management system requires the ISA to be evaluated for any such change.

The purpose of the ISA is to identify accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61. The performance requirements are related to radiological exposure, chemical exposure to licensed material or hazardous chemicals produced from licensed material, and nuclear criticality accidents.

The likelihood of credible high-consequence accident sequences must be reduced to highly unlikely and the likelihood of credible intermediate-consequence accident sequences must be reduced to unlikely. Table 1 shows acceptable combinations of likelihood and consequence categories. Controls established to manage risk are called Items Relied On For Safety (IROFS).

IROFS are the structures, systems, equipment, components, and activities of personnel that are relied on to prevent potential accidents at a facility that could exceed the performance requirements in 10 CFR 70.61 or to mitigate their potential consequences (Ref 4). IROFS may be passive or active engineered features or administrative in nature.

Table 1. **Risk Matrix with Risk Index Values (Ref 6)**

	Likelihood of Occurrence		
Severity of Consequences	**Likelihood Category 1 Highly Unlikely (1)**	**Likelihood Category 2 Unlikely (2)**	**Likelihood Category 3 Not Unlikely (3)**
Conseq. Category 3 High (3)	Acceptable Risk 3	Unacceptable Risk 6	Unacceptable Risk 9
Conseq. Category 2 Intermediate (2)	Acceptable Risk 2	Acceptable Risk 4	Unacceptable Risk 6
Conseq. Category 1 Low (1)	Acceptable Risk 1	Acceptable Risk 2	Acceptable Risk 3

Construction activities adjacent to operating areas of concern introduce hazards not previously analyzed in the ISA. The ISA assesses the affects of on-going construction activities while operating the process systems to support production (i.e., cascades operating), identifies new accident sequences, and IROFS required to manage risk. UUSA performed an ISA to allow the facility to operate while constructing the remainder of the facility. Upon completion of the entire facility, the new accident sequences and controls associated with construction activities will no longer be applicable.

Method

The process hazard analysis (PHA) method used to identify any new accident sequences and IROFS is the What-if/Checklist method. The ISA Team selected the What-If/Checklist method based on guidance in NUREG-1513 (Ref 2) and American Institute of Chemical Engineers Guidelines (Ref 3). The ISA guidance document, NUREG-1513, recommends one of three methods for high risk events; Hazard and Operability Analysis (HAZOP), What-If/Checklist, or Failure Modes and Effects Analysis (FMEA). The What-If/Checklist lends itself best to construction activities because it combines the structure of a checklist with an unstructured "brainstorming" approach to create a list of specific accident events that could produce an undesirable consequence.

The usual HAZOP method using process guidewords did not lend itself well to construction activities. As the results show, loss of containment appeared to be the only deviation leading to consequences of interest. Systematically going through the remaining guidewords (less, more, part of, as well as, reverse, and other than) would not yield meaningful deviations because construction activities do not generally affect process parameters, such as pressure, flow, and temperature.

The ISA Team consisted of individuals with experience and knowledge specific to construction activities, operations/maintenance, nuclear criticality safety, radiation safety, fire safety, and chemical process safety. The ISA Team divided the facility into seven operating areas and considered sixteen different types of construction activities in each area. Table 2 lists the areas and construction activities considered. The result is 112 nodes, for which the Operate While Constructing ISA Team created hundreds of what-if questions during about 60 hours of meetings over ten weeks.

For each What-If question, the team determined the likelihood, consequences, safeguards, and acceptability of risk. What-if questions with unacceptable risk are the accident sequences and their selected safeguards are the IROFS. For example, Node 1A is for site preparation activities affecting operation of the UBC Storage Pad. The first What-If question in Node 1A considers construction equipment causing a fire with high consequences to the worker and public. Table 3 shows the proposed accident sequence OC1-1 requires IROFS to reduce risk to an acceptable level. The IROFS selected establish barriers to keep site preparation vehicles away from the UBC Storage Pad.

Table 4 shows the initiating event frequency, IROFS, likelihood and consequence categories, and risk index for accident sequence OC1-1. The initiating event frequency is a -1, which correspond to a few events expected during facility lifetime. The frequency is based on limited evidence from industry events involving fires caused by construction vehicles. Two preventive IROFS are applied to reduce the likelihood of the event to Category 1, or Highly Unlikely. With IROFS applied, the risk is managed to an acceptable level.

Table 5 provides a narrative of the event including the initiating event, uncontrolled consequences, controlled consequences, initiating event frequency and IROFS failure probability.

Table 2**. Node Designation**

Areas	Construction Activities
1. UBC Storage Pad	A. Site Preparation (Asphalt Paving, Excavation, Backfill, Earth Moving, Compaction, etc.)
2. SBM Mini-Halls (1001, 1003, 1005, 1007, and Extensions)	B. Concrete, Building Foundation/Slab
3. UF_6 Handling Area (1001, 1003, 1005, and 1007)	C. Building Shell (Concrete Panels, Steel Erection, secondary Steel Supports, Platforms, etc.)
4. Process Service Corridor (1001, 1003, 1005, 1007 and Extensions)	D. External Overhead Cranes (Moveable/Tower) /Heavy Loads
5. CRDB Shell/Bunker (Ventilated, GEVS, etc.)	E. Roofing (Composite, Concrete, etc.)
6. External to Uranic Material (TSB, CUB, etc.)	F. Electrical (Conduit, Cable Trays, Lighting, Cabinets,)
7. Miscellaneous Events	G. Compressed Gas Cylinder (CGC)
	H. Internal Construction Lifts (Scissor and Boom)
	I. Finishing (Man Doors, Roll up Doors, Internal Walls, Coatings, etc.)
	J. Utilities (Normal Power, Emergency Power, Chilled Water, Cooling Water, HVAC, Security, etc.)
	K. Machine/Equipment Installation
	L. Construction Vehicles
	M. Mechanical Piping (Header Pipe, Control Piping, Fire Protection, Vacuum, Hanger Systems, and Process
	N. Toxic and Hazardous Substances
	O. Welding and Cutting (Construction, Process, etc.)
	P. Testing (Vacuum, Electrical, Mechanical, System, etc.)

Table 3**. Example What-If Question for Site Preparation near the UBC Storage Pad**

Node	**What-If/Cause**	**Consequences**	**Likelihood**	**Safeguards**	**Risk**
1A1. – Site Preparation (Asphalt Paving, Excavation, Backfill, Earth Moving, Compaction, etc.)	Construction Equipment causes a fire.	Chemical Release - Rupture of UBC caused by fire. High Consequence to local/area worker and public.	Not Unlikely	UBC robust design IROFS36e Storage Pad Drain-off UBC Storage Pad fencing Ditch/trench Set back for construction vehicles.	Acceptable Accident Identifier - OC1-1 {Ref. IROFS50a and IROFS50h}

Rev 5

Table 4**. Example Event Sequence and Risk Index**

Accident Identifier	Initiating Event Index	Preventive Safety Parameter 1 or IROFS 1 Failure Index	Preventive Safety Parameter 2 or IROFS 2 Failure Index	Mitigation IROFS Failure Index*	Likelihood Index Uncontrolled (U)/ Controlled (C)	Likelihood Category	Consequence Category (Type of Accident)	Risk Index (h=f x g) Uncontrolled (U)/ Controlled (C)	Comments & Recommendations
	(a)	(b)	(c)	(d)	(e)	(f)	(g)	(h)	
OC1-1	-1	N/A	N/A	N/A	-1 (U)	3	3 (T)	9 (U)	IROFS Required
OC1-1	-1	IROFS50a -2	IROFS50h -2	N/A	-5 (C)	1	3 (T)	3 (C)	Acceptable Risk

Table 5. **Example Event Description**

Accident Identifier	Description
Accident Identifier: OC1-1 (UBC Storage Pad)	The initial failure (initiating event) involves a fire in a construction site preparations vehicle located near the UBC Storage Pad resulting from an impact or failure of an item in the construction vehicle (e.g., ruptured fuel line, electrical short). For the uncontrolled accident sequence, the construction site preparations vehicle is located such that the vehicle fire propagates to the operating UBC Storage Pad due to fuel run-off or presence of combustible material. Flame impingement on a UBC cylinder results in a ruptured cylinder. A release of UF_6 occurs, resulting in high consequences to the public. For the controlled accident sequence, the preventive measures are: (1) administratively control the proximity of construction vehicles to the UBC Storage Pad by a temporary barrier of sufficient strength to alert the vehicle operator upon impact with the barrier (IROFS50a) and (2) administratively control the proximity of construction vehicles to the UBC Storage Pad by a second and independent temporary barrier of sufficient strength to alert the vehicle operator upon impact with the barrier (IROFS50h). The frequency index number for the initiating event was determined to be (-1). The NUREG-1520 criteria – a few failures may occur during facility lifetime – apply. This failure frequency index was selected based on limited evidence from industry events involving fires caused by construction vehicles. The failure probability index of (-2) was selected for IROFS50a. IROFS50a is enhanced by the use of a physical device (barrier) corresponding to a failure probability index of (-3) for an enhanced administrative IROFS per NUREG-1520, but increased by an order of magnitude for conservatism. The IROFS justification for enhanced administrative control is discussed in Section 3.8.3. The failure probability index of (-2) was selected for IROFS50h. IROFS50h is enhanced by use of a physical device (barrier) corresponding to a failure probability index of (-3) for an enhanced administrative IROFS per NUREG-1520, but increased by an order of magnitude for potential dependent failures between IROFS50a and IROFS50h. The IROFS justification for enhanced administrative control is discussed in Section 3.8.3.

Result

The final ISA (Ref 7) identifies four (4) new accident sequences that, unless mitigated or prevented, would exceed the performance requirements of 10 CFR 70.61. All are associated with chemical consequences, not criticality. The initiating events for the other three accident sequences are;

- An external construction site preparations vehicle impacts an area of concern with subsequent UF_6 release and high consequences to the public and/or worker.

- An internal construction vehicle impacts process equipment resulting in a UF_6 release due to equipment damage and intermediate consequences for the worker.

- An external construction crane failure or human error results in an impact to an area of concern with subsequent UF_6 release and high consequences to the public and worker.

The phrase "area of concern" affords some flexibility in determining where and when an IROFS applies. An area of concern is a location where the What-If/Checklist method determined the accident sequence is possible. An IROFS only applies when the area is in operation.

The ISA also identified eight (8) new IROFS to make the credible high consequence events highly unlikely and intermediate consequence events unlikely as appropriate. Table 6 describes the eight IROFS required to manage the risk of the new accident sequences associated with construction activities.

Table 6. **Items Relied on for Safety Required to Operate While Constructing**

IROFS	Description
IROFS50a	Administratively control proximity of site preparations vehicles around the UBC Storage Pad to prevent a fire from an impact with UBCs resulting in a release of UF_6.
IROFS50b	Administratively control proximity of external site preparations vehicles around areas of concern to prevent an impact with areas of concern resulting in a release of UF_6.
IROFS50c	Administratively control proximity of external site preparations vehicles around areas of concern to prevent an impact with areas of concern resulting in a release of UF_6.
IROFS50d	Administratively control proximity of internal construction vehicles relative to operating process equipment of concern to prevent a release of UF_6 associated with an impact.
IROFS50e	Administratively control movement of internal construction vehicles to prevent impact with operating process equipment of concern resulting in a release of UF_6.
IROFS50f	Administratively control proximity of external construction cranes around areas of concern to prevent a release of UF_6.
IROFS50g	Administratively control movement of external construction cranes around areas of concern to prevent a release of UF_6.
IROFS50h	Administratively control proximity of site preparations vehicles around the UBC Storage Pad to prevent a fire from an impact with UBCs resulting in a release of UF_6.

Regulatory Review

How the ISA evolved into the final four accident sequences and eight IROFS provides some insight into the regulatory process. The process took almost nine (9) months with one Request for Additional Information (RAI). Table 7 provides a chronology of the evolution from start of ISA Team meetings to present.

The initial License Amendment Request (LAR) submitted in June, 2009 (Ref 8), combined all identified hazards into two accident sequences prevented by one existing IROFS for fire control and two new IROFS; one to control proximity and another to control movement of all construction vehicles. The generic nature of the proximity and movement IROFS required specific controls to be identified during implementation. UUSA proposed having qualified individuals use an approved procedure to perform a risk review. The risk review would select specific controls for the construction activity from a list of suggestions in the procedure. Table 8 lists the suggested controls for proximity and movement.

Table 7**. Chronology**

Date	Event
3 March, 2009	Start ISA Team Meetings
19 June, 2009	Submit LAR to Operate While Constructing the Facility
24 Sept, 2009	NRC Issues Request for Additional Information (RAI)
23 Oct, 2009	Response to NRC RAI
7 Jan, 2010	Submit Revised LAR
11 March, 2010	NRC Issues SER allowing UUSA to Operate While Constructing
30 June, 2010	The UUSA Facility Begins Initial Operations
Present	Construction Continues

Table 8**. Suggested controls for proximity and movement**

Proximity	Movement
Fence – permanent or temporary	Designated path for construction vehicles
Jersey barriers – temporary plastic/concrete	Establishing a load movement paths
Walls – permanent or temporary	Limiting the size of mobile equipment
Tape – construction or barrier	Limiting the height of lifted loads
Posts – temporary plastic with rope/chain	Use of additional spotters
Barrels – temporary plastic traffic sand-filled	Use of speed control devices
Removing UF_6 from the area	Use of proximity warning devices

The NRC's Request for Additional Information (RAI) (Ref 9) identified two concerns with using qualified individuals and the approved procedure. The first concern is the qualified individuals replacing an ISA Team to determine controls for a specific construction activity. The qualified individual may select an inappropriate control for the activity, such as barrier tape to control proximity of an earth mover relative to an area of concern. The ISA Team needs to determine in advance which barriers are substantial enough for each specific activity. In response to the RAI (Ref 10), UUSA created separate accident sequences for indoor and outdoor construction activities and one to cover use of cranes.

The second concern is for independence of proximity and movement controls. UUSA considered the control independent because the have different functional attributes, are not subject to common cause failures, and one IROFS does not cause the failure or increase the likelihood of failure of the other IROFS. The proximity control establishes a barrier that must not be breached. The second IROFS controls movement of vehicles relative to components in an operating area of concern. The argument to the contrary is that a barrier can only be breached by a moving vehicle, therefore barriers and control of movement cannot possibly be independent.

The regulator also requested additional information to support the initiating event frequency of -2 corresponding to no failure of this type in URENCO's 30-year operating history. The request stated operating history from gas centrifuge facilities alone is not sufficient. A search of industry operating experience found five vehicle accidents and one crane accident at nuclear-related facilities. None of the accidents involved a fire of sufficient magnitude to exceed 10 CFR 70.61 performance requirements if they had occurred at UUSA during operation. UUSA raised the initiating event frequency for all construction-related accident sequences to -1, corresponding to a few events expected during facility lifetime. The event frequency is based on limited evidence from industry events involving fires caused by construction vehicles.

Operating Experience

The IROFS have been effective. There have been no events involving chemical releases due to construction activities. Based on the number of condition reports written, IROFS50f has been the most challenging to implement.

IROFS50f controls the proximity of external construction cranes around areas of concern to prevent a release of UF_6. After three months of operation, operators identified a potential adverse trend in IROFS50f barrier placements. The five condition reports are summarized as follows;

- one for moving an IROFS50f barrier without operation's permission;
- one for a flagman not adequately trained to perform IROFS50 duties;
- two for openings in the IROFS50f barriers large enough to allow the passage of the crane they were designed to contain, and;
- one for a crane not found in the designated location on the permit.

The apparent cause is inadequate training of construction personnel on the requirements, purpose, expectations or implications of IROFS50f. URENCO Construction Management conducted training using a mock up of IROFS50f barrier requirements and expectations. All construction field management and supervision attended, from area foreman up to project manager.

Since implementing the corrective action, there have been five similar conditions over six months. For all other construction-related IROFS only two conditions have been identified. The difficulty with implementing IROFS50f stems from the frequent need to move barriers when re-positioning cranes as construction work changes.

Conclusion

The What-If/Checklist method effectively identified hazards resulting from construction activities while operating the UUSA facility. A regulatory review resulted in more specific accident sequences and IROFS along with raising the initiating event frequency. The resulting construction-related accident sequences and associated IROFS meet the performance requirements of 10 CFR 70.61. The UUSA facility is allowed to operate while constructing the remainder of the facility with these IROFS in place.

References

[1] Materials License, SNM-2010 through Amendment 45, Issued by the U.S. Nuclear Regulatory Commission, Dec 30, 2010.

[2] NUREG-1513, Integrated Safety Analysis Guidance Document, May 2001, U.S. Nuclear Regulatory Commission.

[3] Guidelines for Hazard Evaluation Procedures, American Institute of Chemical Engineers, Second Edition, A John Wiley & Sons, 1992.

[4] Title 10 of the U.S. Code of Federal Regulations, Part 70.

[5] NUREG-1827, Safety Evaluation Report for the National Enrichment Facility in Lea County, New Mexico, Docket No. 70-3103, June 2005.

[6] URENCO USA, Safety Analysis Report, Revision 30c, 2011.

[7] NRC Safety Evaluation Report, License Amendment Request for the National Enrichment Facility to Operate While Construction (LAR 09-14), March 11, 2010.

[8] Letter from Louisiana Energy Services to the U.S. Nuclear Regulatory Commission, NEF-09-00085-NRC, License Amendment Request for the National Enrichment Facility to Operate While Constructing, June 19, 2009.

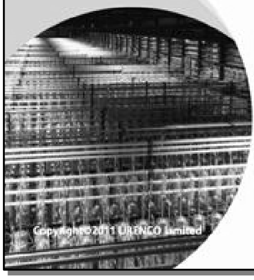
[9] Letter from the U.S. Nuclear Regulatory Commission to Louisiana Energy Services, Louisiana Energy Services Request for Additional Information on License Amendment Request to Operate while Constructing and Integrated Safety Analysis Revision 10, September 24, 2009

[10] Letter from Louisiana Energy Services to the U.S. Nuclear Regulatory Commission, NEF-09-00198-NRC, Response to Request for Additional Information on LAR-09-014, October 23, 2009.

Welcome

Integrated Safety Analysis to Operate While Constructing URENCO USA

R. Kohrt
R. Lehman
S. Su



Outline

- URENCO USA Facility
- Phased Approach
- ISA for Operate While Constructing
  - Purpose
  - Method
  - Result
- Regulatory Review
- Operating Experience
- Conclusion

URENCO USA Facility

- Uranium enrichment facility located in Lea County, New Mexico, USA
- Licensed for construction and operation under 10 CFR Part 70
- Uses gas centrifuges to separate natural uranium hexafluoride (UF_6) feed material into a product enriched up to 5% U-235
- Initial operations commenced in June of 2010
- Construction activities continue as each subsequent cascade is placed into service

URENCO USA Facility

- Modular design allows operations in part of the facility while constructing adjacent parts
- Separation takes place in a Separations Building Module (SBM) which houses;
  - a UF_6 handling area with feed and take-off stations for supply and removal of UF6 gas from the process
  - cascade halls with centrifuges arranged in cascades
  - cascades are grouped to form an assay
  - there are multiple assays in an SBM, each capable of producing product at a specific enrichment

URENCO USA Facility

- In addition to the SBMs, the facility has
  - a Centrifuge Assembly Building
  - Uranium By-Product Cylinder Storage Pad
  - Cylinder Receipt and Dispatch Building
  - Central Utilities Building
  - Technical Support Building

URENCO USA Facility

URENCO USA Facility

- URENCO operates four enrichment plants
- Others located in the UK, Germany, and the Netherlands
- URENCO also owns a 50% interest in Enrichment Technology Company (ETC)
- Service provided on a toll basis for nuclear power utilities using customers' uranium

URENCO USA Facility

Workers at the facility are proud of their safety record with almost 13 million man-hours of operation without a construction-related lost time accident. The record reflects the facility's focus on safe operations.

Phased Approach

- Original license documents considered a complete facility
- Construction activities or the impact of construction on operating process equipment were not considered
- The Safety Analysis Report (SAR) and ISA Summary have since been revised to recognize a phased approach to startup

Phased Approach

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- Operations may begin when required facilities, systems, processes, and Items Relied on for Safety (IROFS) are ready
- Systems are turned over from construction to operations
- SAR describes a series of phases determined by operational requirements

Purpose of the ISA

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- Operating while constructing represents a change to 'activities of personnel'
- Requires an ISA impact evaluation
- Impacts are accident sequences that, unless mitigated or prevented, would exceed performance requirements of 10 CFR 70.61;
  - radiological exposure
  - chemical exposure
  - nuclear criticality accidents

Purpose of the ISA

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- Credible high-consequence accident sequences must be highly unlikely
- Credible intermediate-consequence accident sequences must be unlikely
- Controls established to manage risk are called Items Relied On For Safety (IROFS)
- IROFS are the structures, systems, equipment, components, and activities of personnel relied on to prevent or mitigate accidents

Purpose of the ISA

URENCO

Table 1: Risk Matrix with Risk Index Values (Ref 6)

	Likelihood of Occurrence		
Severity of Consequences	Likelihood Category 1 Highly Unlikely (1)	Likelihood Category 2 Unlikely (2)	Likelihood Category 3 Not Unlikely (3)
Conseq. Category 3 High (3)	Acceptable Risk 3	Unacceptable Risk 6	Unacceptable Risk 9
Conseq. Category 2 Intermediate (2)	Acceptable Risk 2	Acceptable Risk 4	Unacceptable Risk 6
Conseq. Category 1 Low (1)	Acceptable Risk 1	Acceptable Risk 2	Acceptable Risk 3

ISA Method

URENCO

- Selected What-If/Checklist method
  - Consistent with NUREG-1513 and American Institute of Chemical Engineers Guidelines
  - HAZOP or FMEA could also have been used
    - HAZOP guidewords did not yield meaningful deviations because construction does not affect process parameters
  - What-If/Checklist lends itself to construction activities
    - combines the structure of a checklist with an unstructured "brainstorming" approach to create a list of accidents

ISA Method

URENCO

- ISA Team experience
  - construction activities
  - operations/maintenance
  - nuclear criticality safety
  - radiation safety
  - fire safety
  - chemical process safety
- Divided the facility into seven operating areas
- Considered sixteen types of construction activities in each area
- Result is 112 nodes

ISA Method

urenco

Areas

- UBC Storage Pad
- SBM Mini-halls
- UF_6 Handling area
- Process Services Corridor
- CRDB
- Other buildings
- Miscellaneous events

Construction Activities

- Site preparation
- Concrete foundation
- Building shell
- External cranes
- Roofing
- Electrical
- Compressed gas
- Internal lifts
- Finishing
- Utilities
- Equipment installation
- Construction vehicles
- Mechanical piping
- Hazardous materials
- Welding and cutting
- Testing

ISA Method

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- ISA Team created hundreds of What-If questions during about 60 hours of meetings
- For each What-If question the team determined
  - consequences
  - likelihood
  - safeguards
  - acceptability of risk
- What-if questions with unacceptable risk are the accident sequences and their selected safeguards are the IROFS

ISA Method

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- An example, is site preparation activities at the UBC Storage Pad
- What about construction equipment causing a fire?
- Consequences are high (Category 3) due to cylinder rupture
- Likelihood is not unlikely (Category 3)
- Risk is unacceptable (risk index = 3x3 = 9)
- The accident sequence requires IROFS
- The IROFS are barriers around the Pad

ISA Method

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Result

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- Four new accident sequences
- Initiating events for the other three are;
  - external construction vehicle impact with high consequences to the public and worker
  - internal construction vehicle impact with intermediate consequences for the worker
  - external construction crane failure or human error results in an impact with high consequences to the public and worker
- Eight new IROFS to manage the risk

Result

urenco

- The eight new IROFS are;
  - Proximity of vehicles around UBC Storage Pad – 2 IROFS
  - Proximity of vehicles around other areas of concern – 2 IROFS
  - Proximity and movement of internal construction vehicles – 2 IROFS
  - Proximity and movement of external cranes – 2 IROFS

Regulatory Review

URENCO

Date	Event
3 March, 2009	Start ISA Team Meetings
19 June, 2009	Submit LAR to NRC
24 Sept, 2009	NRC issues RAI
23 Oct, 2009	URENCO USA responds to RAI
7 Jan, 2010	Submit revised LAR
11 Mar, 2010	NRC issues SER
30 June, 2010	URENCO USA begins operations
Present	Construction continues

Regulatory Review

URENCO

- Initial submittal combined all hazards into two accident sequences prevented by
  - one existing IROFS for fire control
  - one new IROFS to control proximity and
  - one new IROFS to control movement
- Generic nature required specific controls to be identified during implementation
- Risk review by qualified individuals using an approved procedure
- Select specific controls for the activity from a list of suggestions in the procedure

Regulatory Review

URENCO

Proximity

- Jersey barrier
- Tape
- Fence
- Walls
- Posts
- Barrels
- Removing UF_6 from the area

Movement

- Use of spotters
- Load movement path
- Vehicle path
- Limit vehicle size
- Limit lift height
- Speed control device
- Proximity warning device

Regulatory Review

URENCO

The NRC's RAI identified three concerns

1. Qualified individuals cannot replace an ISA Team to determine controls for a specific construction activity
   – The qualified individual may select an inappropriate control for the activity
   – The ISA Team needs to determine in advance which barriers are substantial enough for each specific activity
2. Proximity and movement controls are not independent
   – a barrier can only be breached by a moving vehicle

Regulatory Review

URENCO

3. Initiating event frequency of -2 is too low
   – corresponds to no failure of this type in 30 years
   – operating history from gas centrifuge facilities alone is not sufficient
   – search of industry operating experience found six vehicle/crane accidents at nuclear facilities
   – UUSA raised the frequency to -1, corresponding to a few events expected during facility lifetime
   – higher event frequency is based on limited evidence from industry events

Operating Experience

URENCO

- The IROFS have been effective
  – no events involving chemical releases due to construction activities
  – proximity of cranes has been the most challenging to implement
- Adverse trend
  – moving a barrier without operation's permission
  – flagman not adequately trained to perform duties
  – opening in barrier large enough to allow the passage of the crane, and
  – crane not found in the designated location on the permit

Operating Experience

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- Apparent cause is inadequate training on requirements, purpose, expectations or implications of the IROFS
- URENCO Construction Management conducted training using a mock up of a barrier and crane
- Condition frequency reduced by about half
- Challenge stems from the frequent need to move barriers when re-positioning cranes as construction work changes
- Only two conditions identified for all other construction-related IROFS

Conclusion

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- What-If/Checklist method effectively identified hazards resulting from construction activities
- Regulatory review resulted in more specific accident sequences and IROFS along with raising the initiating event frequency
- Construction-related accident sequences and associated IROFS meet the performance requirements of 10 CFR 70.61
- The UUSA facility is allowed to operate while constructing the remainder of the facility with eight new IROFS in place

SESSION THREE

CHEMICAL HAZARDS – RELEASE LIMITS

INDUSTRY PERSPECTIVE ON SETTING EMISSIONS LIMITS

Paul Desiri
GE Hitachi Nuclear Energy Canada Inc., Canada

Abstract - GE Hitachi Nuclear Energy Canada Inc. (GEH-C) has operated a uranium dioxide fuel pelletizing plant and fuel bundle assembly facility in Toronto and Peterborough respectively, since the mid- 1960's. A primary operational objective is to ensure a consistently high degree of environmental protection, and to accomplish this requires properly established emission control limits.

This paper will summarize the system of environmental and public protection constraints currently in place at GEH-C including derived emission limits, operational limits and action levels. It will discuss the basis of any values chosen and describe the methods used to calculate their magnitude. Perspectives on how to ensure operational effectiveness of these constraints will also be addressed.

1. Introduction

GE Hitachi Nuclear Energy Canada Inc. (GEH-C) Class 1B facilities in Peterborough and Toronto are covered under a single operating licence and share a common organization and common environment health and safety programs. A primary operational objective is to ensure a consistently high degree of environmental protection, and to accomplish this requires properly established emission control limits.

Properly set emission limits are essential for effective control of environmental emissions. A sufficient margin of safety must be built in to ensure the routine emission limit is never approached. Corresponding administrative levels are also necessary to facilitate early response to potential process upsets and thus eliminate the possibility of any major unplanned releases occurring.

This paper will use the example of the system of constraints for air emissions from the pelleting operation. However the approach described in this paper applies equally to all emissions.

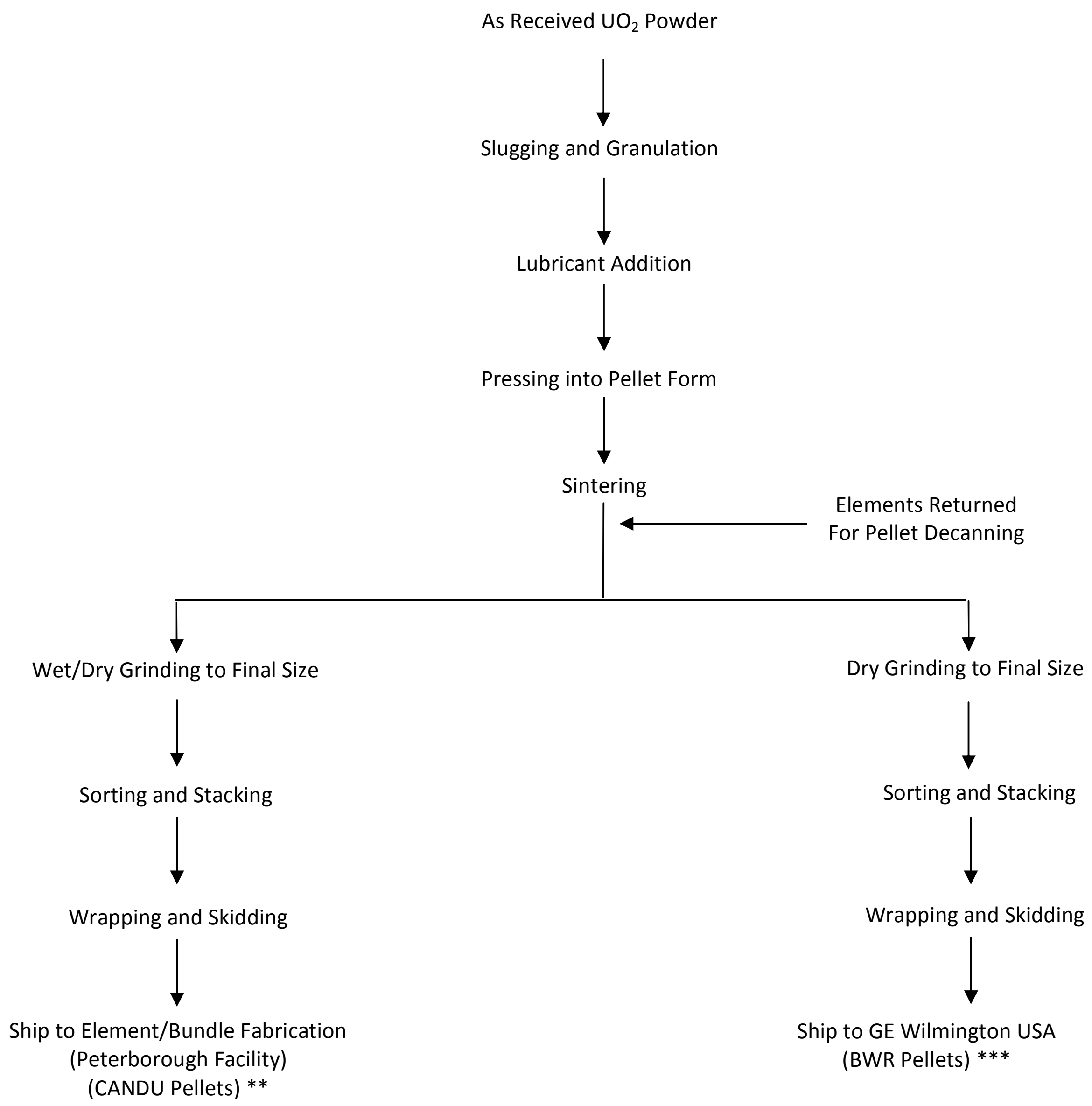
2. Facility background

The GEH-C pelleting facility is located in an area consisting of industrial, light industrial, and mixed commercial/residential occupancy. An aerial photo showing the facility in relation to its surroundings is provided in Figure 1 below.

The facility is licensed to produce natural uranium pellets for use in the manufacture of fuel bundles. It has produced fuel pellets since 1965. A summary of the basic process flow is provided in Figure 2. Below.

Figure 1. **Facility Aerial Photo**

Figure 2**. Process Flow**[48]



[48] GEH-C Radiation Protection Manual Volume 1, June 8, 2011

Local ventilation applied to production work stations and room ventilation throughout the plant, both ensure that airborne uranium is kept to a minimum for workers in accordance with the ALARA principle. Plant air is collected by independent systems that protect each different plant area. These systems provide filtration through multiple stages of HEPA filter banks before emission.

Stacks are sampled continuously using isokinetic sampling. The quantity measured is the average daily uranium in air concentration for each stack. Air concentration is the key parameter for short term operational control. However, the parameter used to gauge long term impact and overall compliance is the mass discharged which is calculated from the measured concentration and known stack characteristics, ie emission rate. For this paper, mass discharge is considered to define air emissions.

3. Design objectives

The following characteristics of emission limits are considered essential:

- Derivation of overall air emission limit is directly tied to the regulatory limit through a recognized methodology
- Sufficient conservatism is built in to each particular constraint, ie, derived emission limit, operational limit, etc.
- Administrative limits must provide additional safety margin to the emission limit and be sensitive enough to trigger corrective action when appropriate without triggering false alarms
- Program design must include effective operational reviews and facilitate independent verification measurements to achieve high level of confidence in accuracy of data.

4. Site specific limits

The 1 mSv public dose limit is the starting point for emission limit derivation. It represents the upper ceiling as it is the legal limit for routine emissions. In actuality though, once the derived emission limit is determined, it is not of significant operational importance, other than for reporting purposes. This is because discharges at this facility are (and by necessity must be) a very small fraction of the derived emission limit.

The current derived emission limit of 15.2 kg was calculated as part of the application for the 2001 CNSC facility operating licence[49]. The actual detailed calculation is outside the scope of the paper. However, it is important for this paper to discuss the strategy for selecting a derivation method and to illustrate the relationships between corresponding regulatory limits.

Air emission stacks must be approved by the Ministry of Environment, Ontario through its certificate of approval process. The dispersion model[50] that was tied to provincial regulations at the time was used as the original methodology to calculate the derived emission limit. It was also the most conservative model available.

Using default data from recognized standards (ICRP etc.) and conservative assumptions for quantities such as occupancy, e.g., 24 hours and 365 days, a derived air concentration of 0.5 $\mu g/m^3$ was determined as being equivalent to a public dose limit of 1 mSv. When this concentration is input into

[49] Radiation Protection Manual Instruction – 2.2, P. Desiri, Nov 14, 2000

[50] Regulation 346 of the Environmental Protection Act of Ontario, 1990

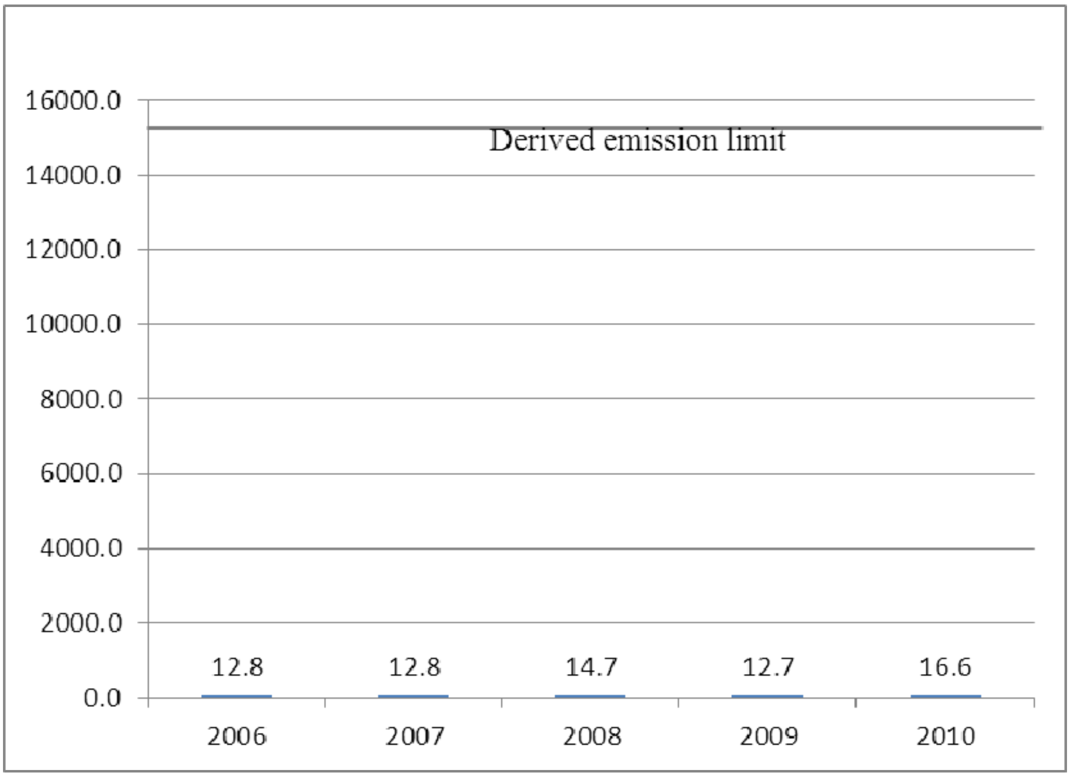
the standard dispersion formula, an annual discharge of 15.2 kg is calculated. The dispersion model is considered conservative as it would tend to overestimate the predicted concentration for a given discharge.

This was later confirmed when the derivation was recently updated using AEROMOD[51]. The AEROMOD model showed the concentrations predicted by the previous calculation were conservative by a factor of about 3.

For the most recent relicensing in 2010, a new constraint, the operating limit, was proposed by GEH-C. The rationale for GEH-C introducing this constraint was that emissions were consistently three orders of magnitude below, or one thousandth of, the derived emission limit. This is illustrated in figure 3.

The operating limit was therefore intended to be a more meaningful limit from an operational perspective. It could be a more representative compliance measure and introduce a significant safety factor in relation to the public dose limit.

Figure 3. **Emissions performance compared to derived emission limit – linear scale**[52]



[51] Section 26 of O Reg. 419/05 and Ontario Ministry of the Environment (MOE) “Procedure for Preparing an Emission Summary and Dispersion Modelling Report

[52] GEH-C Annual Compliance Reports

It was difficult at first to select an appropriate magnitude for the proposed operating limit. The public dose limit has a clear basis and is written into the regulations. Rather than select some fraction that was arbitrary, GEH-C decided to base the operational limit on the concept of trivial dose. The *deminimus* dose of 50 µSv, which has had application in other regulatory standards as a trivial dose[53], was selected.

The ratio between the public limit and *deminimus* dose is 1/20. Applying this ratio is to the derived emission limit of 15.2 kg, yields an operating limit of 760 g. This limit was written in the 2011 CNSC Class 1B licence.

Table 1**. Discharge limits for air effluent**

Parameter	Magnitude
Derived Emission Limit	15.2 kg/yr.
Facility Licence Operating Limit	760 g/yr.

5. Action level determination

Like the derived emission limit, the current action level for airborne emissions was first calculated for the 2001 CNSC licence. An action level[54] is by definition a significant break down that merits a full, formal investigation. The challenge for an operator is to select one that truly reflects such a breakdown in control so that operational resources are not unnecessarily consumed by investigation of insignificant events and also avoid the potential to bring undue negative attention to the facility. As Action Levels are not easily changed, it is critical that they be set properly.

One possible approach to determining an action level is to statistically analyze past results in order to determine the upper range of normal operation. If the standard deviation is sigma, then a specified multiple of sigma would represent an emission that is outside the range of what is considered normal. The problem with this approach is that for fairly stable results, such as this facility has seen, the sigma will be relatively small. And unless the time period selected is long enough in duration to capture all possible variables (production cycles, seasonal variations etc.), the constraint could be unnecessarily restrictive. This is particularly an issue for those variables that are unrelated to any break down, e.g., production level variations.

Another approach, which is considered more effective, is to review past events at the facility where breakdowns actually occurred and use the magnitude of the measured release as an action level. A review of past data yielded a 1995 incident where a stack was operated without all of the required filters in place. The measured concentration on this stack for a 24 hour period was about 1 µg/m^3.

Once a proposed action level magnitude has been identified, there needs to be verification against historical operating performance and other standards. Because of the importance of setting the action level correctly, all available data and standards should be consulted to confirm the value is comparable

[53] Regulatory Document R-85, Atomic Energy Control Board, Radiation Protection Requisites for the Exemption of Certain Radioactive Materials From Licensing Upon Transferral For Disposal, Aug 1, 1989

[54] CNSC Radiation Protection Regulations, May 31, 2000

so that it can be easily supported. In the case of this facility this value was about 2-3 times the maximum concentration for normal operation so it was concluded it was most likely in the appropriate range. This value was also in the range of other air limit standards in place at the time (MOE).

This concentration 1 $\mu g/m^3$ can be compared to the other constraints by converting it to an equivalent mass discharge. This is calculated by taking the particular stack emission characteristics, ie emission rates and then assuming the stacks are all operating at the action level. The sum of all products is then the action level equivalent emission. This quantity (330 g) is shown as one of several constraints in Figure 4.

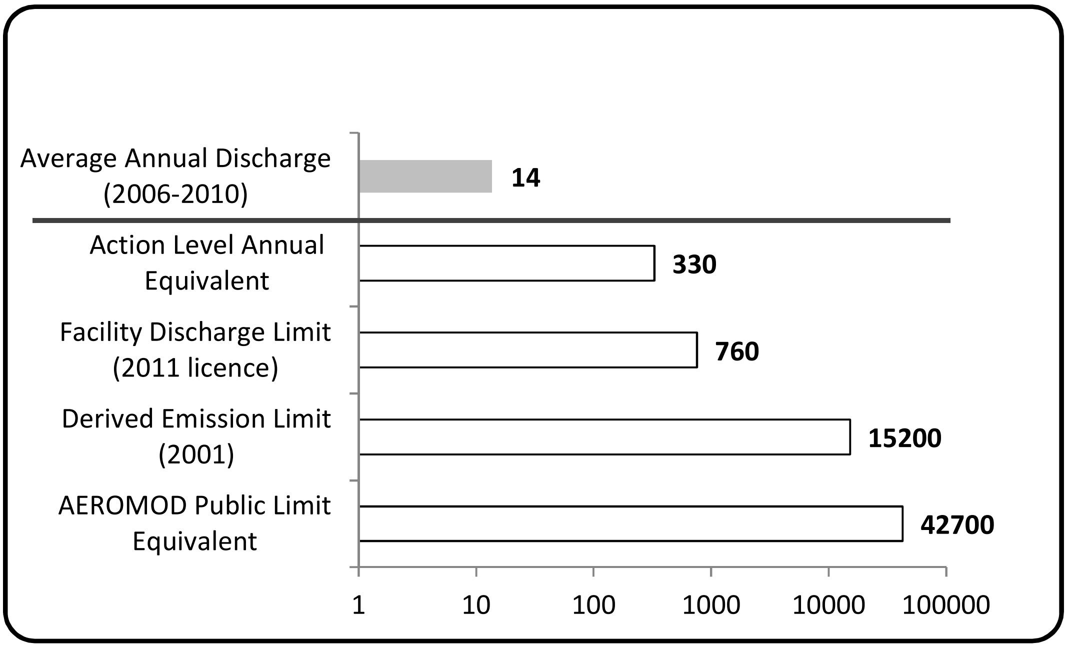
Action levels should be review periodically to ensure they continue to be set at the proper magnitude. The review should consider not only performance data trends, but also a review of any new incidents during the review period.

Based on 10 years of operating performance since this action level was introduced, the magnitude is still considered to be appropriate. It has served the facility well by giving advance warning of possible issues. Although there have been occurrences where results have approached the action level in the past 10 years, it has not yet been exceeded.

The fact that the action level has not been exceeded does not mean it is set too high. As tempting as it may be to push for a lower action level for ALARA purposes, it is not enough of a reason to reduce an action level simply because it has not been exceeded. If this the sole reason for lowering an action level, it is equivalent to penalizing an operator for good performance. A stable process, particularly for a parameter as important as air emissions, is highly desirable and should be acknowledged as a success when it is achieved. Internal control levels could be used as an ALARA measure in this case instead.

Figure 4**. Illustration of safety margin – routine emissions in grams versus constraints**

(Logarithmic scale)



6. Compliance monitoring program

After the Action Level has been approved, it must then be fully implemented in the sites environmental management system. It is essential that all employees involved in controlling and measuring air emissions are aware of the action level's magnitude and also its importance to the continued safe operation of the facility. Staff that are responsible for emission measurements, must also be fully knowledgeable of reporting obligations including time limits which for this site are 24 hours as per the CNSC regulations[55]

Even when emissions are below the action level, it is important that adverse trends be identified promptly so that they can be investigated and corrective actions taken as soon as possible. This can be a challenge in a large company but it is absolutely essential. To do this effectively, a well- established review rhythm must be a fundamental part of the operations. In this facility, there are several different levels of review, each with its own frequency and varying involvement of different levels of company management. Generally, the reviews conducted at the lowest levels in the organization, occur the most frequently.

Table 2**. Operational reviews**

Organizational level	Scope of Review - Frequency
Environmental Technicians	100% of raw results, immediately after they are determined - Daily
Environmental Health and Safety Management, Facility management	Any unusual results, as soon as possible after determination – As required
Operations Management	Unusual results, adverse trends, corresponding action plans - Monthly
Corporate Health and Safety Management	Any results approaching Action Level, as soon as possible – As required
Canada Business management	Emission results - Monthly
Company Board of Directors	Emission results - Quarterly
Top Safety Executives of Company	Emission results - Annual

In addition to operational reviews, periodic reviews of measurement accuracy must be conducted. In this facility, these are carried out by an independent third party on a five year cycle. Inter-comparisons were completed in 2002 and 2007. In both studies, favorable comparisons resulted, ie the results were well within performance standards.

[55] CNSC Radiation Protection Regulations, May 31, 2000

7. Conclusions

The following characteristics are considered essential and should be employed in the setting of emission limits:

Derivation of overall air emission limit is directly tied to the regulatory limit through a recognized methodology

Sufficient conservatism is built in to each particular constraint, ie, derived emission limit, operational limit, etc.

Administrative limits must provide additional safety margin to the emission limit and be sensitive enough to trigger corrective action when appropriate without triggering false alarms

Program design must include effective operational reviews and facilitate independent verification measurements to achieve high level of confidence in accuracy of data.

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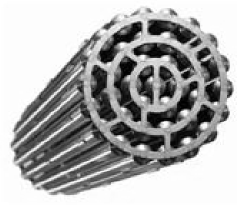
Industry Perspective on Setting Emissions Limits



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Presentation Outline

- Background on operation
- Design objectives
- Site specific limits
- Action level determination
- Compliance monitoring program

Background on Operation

Arnprior Tubing Operation

Toronto Pelleting Operation

Peterborough Bundle Assembly

- Fuel bundle assembly in Peterborough using zirconium tubes from Arnprior and natural uranium dioxide (UO_2) pellets manufactured from UO_2 powder in Toronto
- Also a nuclear services business based at the Peterborough facility
- GEH-C has manufactured uranium oxide fuel for over 50 years.

Facility Pictures courtesy Google Earth®

Design objectives

- Emission limit tied to regulatory limit through recognized methodology
- Sufficient conservatism is built in to each particular constraint
- Administrative limits provide additional safety margin and sensitive enough to trigger action when appropriate without false alarms
- Program design must include effective operational reviews and facilitate independent verification to achieve high level of confidence in data.

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Site specific limits – Example: Toronto air emissions

- 1 mSv public dose limit is the starting point for emission limit derivation.
- Derived emission limit of 15.2 kg part of application for 2001 CNSC licence
- Ministry of Environment, Ontario dispersion model tied to provincial regulations at the time was used
- Using default data from recognized standards, conservative assumptions for other quantities, derived air concentration of 0.5 µg/m3 determined.
- AEROMOD model update (2010) showed concentrations predicted by the previous calculation were conservative by a factor of about 3.
- Operating limit of 760 g in 2010 relicencing a more meaningful limit from an operational perspective.

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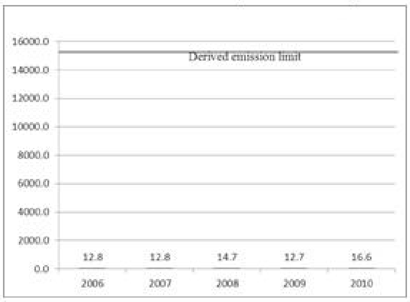
Site specific limits (cont'd)

Air emissions performance compared to derived emission limit – linear scale, all values in grams



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Action level determination – Example: air emissions

- Action level a significant break down that merits full, formal investigation.
- One approach - statistically analyze past results to determine the upper range of normal operation - several problems with this
- Better to review events at facility where breakdowns actually occurred and use the magnitude of the measured release as an action level.
- 1995 incident where stack was operated without all the required filters. The measured concentration was about 1 µg/m3.
- Verification against operating performance, other standards necessary
- Action levels should be reviewed periodically and consider not only performance data but also any new incidents during the review period.
- Fact that action level not exceeded does not mean it is set too high.

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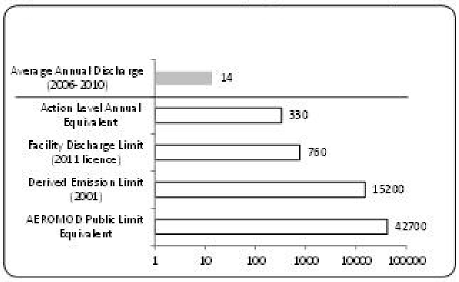
Action level determination (cont'd)

Illustration of safety margin – routine air emissions in grams versus constraints (Logarithmic scale)



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Compliance monitoring program

Organizational level	Scope of Review - Frequency
Environmental Technicians	100% of raw results, immediately after they are determined - Daily
Environmental Health and Safety Management, Facility management	Any unusual results, as soon as possible after determination – As required
Operations Management	Unusual results, adverse trends, corresponding action plans - Monthly
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Canada Business management	Emission results - Monthly
Company Board of Directors	Emission results - Quarterly
Top Safety Executives of Company	Emission results - Annual

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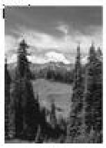
Conclusions

The following characteristics are considered essential and should be employed in the setting of emission limits:

- Emission limit tied to regulatory limit through recognized methodology
- Sufficient conservatism is built in to each particular constraint
- Administrative limits provide additional safety margin and sensitive enough to trigger action when appropriate without false alarms
- Program design must include effective operational reviews, facilitate independent verification to achieve high level of confidence i



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November 16, 2011

TAKING INTO ACCOUNT CHEMICAL SAFETY FOR FRENCH BASIC NUCLEAR INSTALLATIONS

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Abstract - Among nuclear installations, some fuel cycle facilities present a high level of chemical hazards. In France, the TSN law of the 13 June 2006 requires taking into account all the risks generated by a basic nuclear installation (BNI). But, as most of the implementing regulatory texts are under development at this time, part of the previous regulation settled down in the 1990s is still applying: the order of the 31 December 1999 concerning technical regulation in order to prevent and to limit hazards generated by nuclear facilities; the decree of the 4 May 1995 and the order of the 26 November 1999 that deal with BNI discharges. Moreover, some parts of BNI or of nuclear sites can be submitted to the general regulation concerning chemical hazards, which is part of the environment code. As a result, even if the TSN law and its implementing decree Nr 2007-1557 of the 2 November 2007 settle clearly that safety of BNI is not only radiological, but must take into account chemical hazards, the latter aspects are still under development. Moreover the application of the existing regulation, even if complex, has helped to assess chemical risks inside BNI and nuclear sites.

Introduction

A lot of fuel cycle facilities (FCF) have the specificity, among nuclear installations, to use chemical substances in great amounts, with al the risks and hazards linked to: toxicity, corrosive aspect, explosion, fire, etc. Consequently chemical risks must be assessed for FCF as well as radiological ones. In France the law Nr 2006-686 of the 13 June 2006 [5] concerning transparency and nuclear safety (known as the “TSN law”) requires to take into account all the risks generated by a basic nuclear installation (BNI). However most of the regulation requiring to take into account chemical risks was settled down in the 1990s on the basis of the previous general regulation concerning BNI, the decree Nr 63-1228 of the 11 December 1963 [1].

1. French regulation concerning chemical hazards – the order of the 31 December 1999

The French regulation in force dealing with chemical hazards is mostly based on the order of the 31 December 1999 [2] which applies to civil basic nuclear installations (BNI). This order lays down technical regulation in order to prevent and to limit hazards generated by nuclear facilities and that

[56] ASN : Autorité de sûreté nucléaire = French nuclear safety authority
DRC : ASN directorate in charge of nuclear waste, research facilities (including research reactors) and fuel cycle facilities

could have an impact on people living or working around, public security, public health, agriculture, environment or monuments. The order of the 31 December 1999 [2] deals with practical measures to take concerning:

- Noise and vibrations,
- Gaseous pollution,
- Water pollution,
- Waste management,
- Traffic inside nuclear sites,
- Load handling,
- Management of radiological, chemical, toxic, flammable, corrosive or explosive substances,
- Electrical devices,
- Lightning,
- Fire,
- Nuclear hazards.

The order parts actually dealing with chemical hazards are:

- Title III – prevention of air pollution (articles 10 and 11), that refers to the regulation concerning BNI releases (see next part) ;
- Title IV – prevention of water pollution (articles 12 to 19), that :
- Also refers to the regulation concerning BNI releases (article 12) ;
- Deals with toxic, radioactive, flammable, corrosive or explosive liquids ;

Puts constrains on liquid storages (article 14), as for example minimum volumes of retentions under tanks, or no presence of tanks of incompatible substances above the same retention;

Puts constrains on transfer areas (article 15) and on pipes (article 16);

- Requires that liquid waste do not be able to damage sewer networks (article 18) ;
- Requires that the operator takes all measures to prevent accidental leaks or releases (article 19).
- Title VI – prevention of other risks (articles 28 to 47), that :
- Deals with handling and transports inside nuclear sites (articles 28 and 29) of all dangerous substances ;
- Requires a continuous confinement of nuclear substances (article 30) ;
- Requires written operating instructions especially for installations containing toxic, radioactive, flammable, corrosive or explosive substances (article 31) ;
- Requires periodical assessments for these installations (article 40) ;
- Deals with fire prevention and measures to put out fire (articles 41 to 44).

This regulatory text is still on application. However the TSN law [5] has now put in force to take into account all the risks generated by BNI, radiological and non-radiological. This will be developed in an order called the "BNI order", and more particularly in ASN decisions as the one concerning the environment protection that will be issued before the end of the year.

2. French regulation concerning BNI discharges – the decree of the 4 May 1995 and the order of the 26 November 1999

The decree of the 4 May 1995 [3] settles down the general regulation concerning discharges of civil BNI. Its article 1 requires specific authorizations for gaseous and liquid radioactive releases.

Its article 2 settles down specific rules for parts of BNI or installations classified for the environment protection (ICPE)[57] operating inside nuclear sites, that are submitted to the ICPE regulation which is part of the environment code. This article 2 deals with some chemical risks.

The article 7 of the decree of the 4 May 1995 [3] says that radioactive and non radioactive liquid discharges of some substances into underground waters are forbidden.

The rest of the decree deals with administrative procedures to follow to deliver a discharge authorization. Concerning installations submitted to what-is-called "authorization procedure" to be allowed to operate, which is the case of BNI, the article 11 says that the discharge authorization is an inter-ministerial order containing specific limitations that must take into account regional water-uses planning, that deals with chemical pollution of waters, and radioprotection. With the TSN law [5], the administrative form of the discharge authorization has now changed into ASN prescriptions that have to be countersigned by ministers in charge of nuclear safety

The TSN law [5] requires also to take into account all the risks generated by a BNI, radiological and non-radiological. But as there is still no new regulation replacing the decree of the 4 May 1995 [3], the content of the authorization is described by the article 11 of the decree of the 4 May 1995 and its application order, the order of the 26 November 1999 [4].

The order of the 26 November 1999 [4] settles downs the content of limitations concerning water supplies and liquid and gaseous discharges of BNI. It deals with radiological and non-radiological releases. Chemical releases are dealt with:

In article 10 for gaseous releases;

In article 17 for liquid releases.

The order of the 26 November 1999 [4] also settles down basic technical measures that operators have to put in place, as for example:

Gaseous discharges must come out a facility through chimneys, except very specific justifications (article 11-I);

All radioactive gaseous discharges must be treated or filtrated before release (article 11-III);

[57] installations using or storing chemical substances. These installations are divided in different types according to their potentially dangerous aspects : the less dangerous installations are "declared installations" ; the other ones are "authorized installations", inside which the higher possible classifications are the "SEVESO type" and the "SEVESO II type".

There must be separated networks to prevent mixing of different types of liquid waste (articles 18 and 19).

3. Other regulatory documents taking into account chemical hazards: internal urgency plan (PUI), and intervention particular plan (PPI)

Similarly to requirements of the decree of the 11 December 1963 [1], the article 20 of the "Procedures decree" [6] requires that every BNI has an internal urgency plan (PUI). The PUI describes organization, means and methods of intervention that are planned by the operator to manage accidental scenarios in order to protect its staff, the public and the environment, and in order to preserve or to restore the BNI safety.

Moreover the decree Nr 2005-1158 of the 13 September 2005 [7], that is not specific to the nuclear industry, requires that each industrial site containing at least one BNI has an intervention particular plan (PPI). The PPI is developed by the local administrative authorities, under the control of the Prefect that is the local State representative. The PPI describes measures and means needed to manage specific risks during the occurence of incidents or accidents whose consequences can get out of the physical limits of an industrial site. The PPI contains:

A general description of the industrial site;

The geographical application area of the PPI, listing all the concerned municipalities;

The information and protection measures planned for populations and, if necessary, description of evacuation plans with possible accommodation locations;

The measures that the operator has to take to immediately alert the local authorities and to inform them concerning the situation evolution;

The measures that the operator has to take concerning the protection of the population living or working around its site;

The diffusion of warning for the populations living or working around the site;

If necessary, the measures for warning and informing the authorities of a neighbor State;

General planned measures concerning the long-term environment cleaning after the occurrence of a major accident.

Both PUI and PPI are developed on the basis of the design assessment of the BNI, that must take into account all the risks inherent to the BNI and accidental scenarios. Consequently, PUI and PPI take into account, when it is appropriate, the chemical risks that can generate incidental and accidental conditions that need specific management.

4. Some other French regulation concerning chemical hazards – the environment code

The regulation listed above is specific to BNI, except the decree of the 13 September 2005 [7]. But some parts of BNI or of nuclear sites can be submitted to a more general regulation concerning environment protection: the environment code [8]. This code settles down a classification of uses of chemical substances, that takes into account possible dangerous aspects (toxic, corrosive, etc.) of the substances related to quantities used or stored.

Linked to this classification, the environment code settles down regulation concerning installations using or storing chemical substances, classifying these installations in different types of installations classified for the environment protection (ICPE) according to their potentially dangerous aspects: the less dangerous installations are "declared installations" ; the other ones are "authorized installations", inside which the higher possible classifications are the "SEVESO type" and the "SEVESO II type".

The main types of ICPE that operate inside nuclear sites are:

Fuel storages for electricity generators,

Uses or storages of great amounts of chemical products that can be corrosive or toxic.

Because of amounts of chemical substances, some parts of BNI or of nuclear sites can also be submitted to the SEVESO II European directive, even if there is no or no more radioactive materials, as for example the Phenix and the Superphenix reactors because of sodium that was used as coolant.

Even though taking into account of the different French regulations that can apply to a BNI may reveal to be sometimes quite difficult, the use of the environment code classification of ICPE and of uses of chemical substances helps to find out which chemical parameters must be taken into account in addition to nuclear risks, in order to assess major risks that can be generated by BNI or nuclear sites. Examples are given here after.

5. Example Nr 1 of chemical risks taken into account for French nuclear sites : the Tricastin site

The Tricastin civil AREVA site hosts several BNI:

The Comurhex facility that processes chemical transformation of natural uranium;

The TU5/W facility that processes chemical transformation of uranium coming from retreatment process;

The Eurodif plant that enriches uranium up to 5% of uranium 235, and that will be replaced by the GB II facility;

The Socatri plant that treats uranium liquid waste coming from the Eurodif and GB II facilities.

All the operators are parts of the AREVA group.

The processes used inside the facilities are chemical ones, with no great radiological or critical issues. In particular, the Comurhex, TU5/W, Eurodif and GB II plants use fluorine components to create UF_6.

The amounts of fluorine substances present in the Eurodif plant are enough to make the facility also classified as ICPE (as "SEVESO II type" installation, in fact).

Assessing the major accident that could occur in the Tricastin site has required to examine:

Criticality accidents that could occur in the Eurodif and GB II plants, more precisely inside the storages of enriched uranium;

Massive leaks of uranium substances, from any BNI;

Massive leaks of fluorine substances stored or used in the Comurhex, TU5/W, Eurodif and GB II facilities, and that could also generate HF gas;

Uncontrolled hydrolysis of UF6, that could also generate HF gas.

At the end of the assessing process, it has appeared that the major accident that could occur in the Tricastin site, that is also called “design basis accident”, is a plane crash on the UF_6 storage of the Eurodif plant (or the GB II plant, when the Eurodif plant will be under decommissioning), that would generate massive leaks of fluorine substances. These uncontrolled releases would then generate enough HF gas to corrode many equipments or materials inside the site, and to touch seriously many people inside the site. In fact, in such a case, the evacuation zone would be a circle of about 6.6-km radius around the Eurodif plant, which includes the whole Tricastin site. The uncontrolled releases would also generate releases of great amounts of uranium that could have significant impacts (mostly chemical, because uranium is chemically toxic more than radiological) on environment or on health

The consequences of a criticality accident are very weak compared to the chemical risk.

Because of the consequences it could have, a leak of fluorine substances is a major safety issue. That is why the major safety measures put in place in the Tricastin site deal with containment of chemical products. And the release authorizations concerning the site have specifications dealing with HF releases in air.

6. Example Nr 2 of chemical risks taken into account for French nuclear sites: the La Hague site

The La Hague site contains the nuclear fuel reprocessing plants. Reprocessing is a chemical process of liquid-liquid extraction leading to separate uranium and plutonium form the fission products, and to recover the nuclear matter. As the raw materials of its process are spent nuclear fuels, criticality and radioprotection are two major issues on the site.

But the process also uses very strong acids (HNO_3 at pH < 1), highly concentrated sodium hydroxyde, hydrazine, hydroxylamine, tributylphosphate, dodecane and other solvents. All these products can generate chemical issues that can become safety matters.

First, the storage of these chemical products must be safe, because they can corrode equipments or materials, and/or because they can become explosive if they are not safely controlled, and/or because their impact on environment cannot be neglected. As a result any uncontrolled release of these products must be prevented. As a consequence, some chemical storages in La Hague are classified as ICPE, even as “SEVESO II type” installations.

Second, the major accidental scenarios of chemical matters, that can be defined taking into account compositions of spent fuels and presence of the chemical products listed above are as follow:

Unwanted mixing of incompatible products, that can explode;

Solvent fire;

Explosion because of unwanted generation of explosive components like nitride or azide;

Uncontrolled redox reaction outside the process scope, that can generate plutonium precipitation in one part of the process, possibly leading to a criticality reaction;

Fire of some fuels components containing graphite or magnesium.

Third, existence of potentially dangerous chemical products in one part of an installation makes it not easy to operate in incidental or accidental conditions, or can create worst situations to manage.

As a result chemical hazards are taken into account:
While working out incidental and accidental scenarios;

In the general operating rules of the site, for the need of maintenance of equipments, or for the management of incidental or accidental conditions;

In discharges authorization.

7. Conclusion

The French regulation in force concerning chemical hazards inside BNI or nuclear sites is at the moment a mix between previous regulation settled down in the 1990s, and the new regulation based on the TSN law. This TSN law clearly requires taking into account all the risks generated by a BNI, which is a progress compared to the previous regulation that was not that precise.

The general regulation dealing with chemical hazards, settled down in the environment code, can also have to be taken into account in some cases, mostly for FCF because some of them use dangerous chemical substances in great amounts.

As a result, the regulation in force concerning chemical hazards inside nuclear sites can look quite complex, because it refers to several texts not necessarily linked to each other. But it still has helped taking into account chemical hazards in the BNI that had really some. However it must be said that chemical hazards were plainly taken into account mostly for buildings classified as part of BNI or when they were linked to radiological hazards. Other buildings existing on nuclear sites, but not part of BNI (classified as ICPE, for example) were often not designed to resist events like earthquake, even if they contain chemical products.

The regulation still under development, mostly the BNI order, is expected to help to take this kind of risks into account more accurately and more simply. The feedback from the Fukushima nuclear accident of March 2011 is also expected to help on this subject, because it requires to consider accidental situations that are beyond design, or that were not considered in the design because involving installations or buildings considered as of “minor interest” as they are not active part of the nuclear process.

References

[1] Decree Nr 63-1228 of the 11 December 1963 dealing with general regulation concerning basic nuclear installations

[2] Order of the 31 December 1999 concerning technical regulation in order to prevent and limit hazards generated by basic nuclear installations and that could have an impact on people living or working around, public security, public health, agriculture, environment and monuments

[3] Decree Nr 95-540 of 4 May 1995 concerning releases of basic nuclear installations

[4] Order of the 26 November 1999 concerning water supply and liquid and gaseous releases of basic nuclear installations

[5] Law Nr 2006-686 of the 13 June 2006 concerning transparency and nuclear safety, known as the “TSN law”

[6] Decree Nr 2007-1557 of the 2 November 2007 concerning basic nuclear installations, known as the “Procedures decree”

[7] Decree Nr 2005-1158 of the 13 September 2005 concerning intervention particular plans required for some types of installations

[8] Environment code – articles L.521-1 to L.521.24 concerning chemical risks

Bibliography

- Draft of the order concerning general regulation for BNI (known as the “BNI order”)

Taking into account chemical safety for French nuclear installations

ASN/DRC
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French regulation concerning chemical risks in nuclear facilities

The French regulation in force is a mix between "new" regulation (in force since 2006) and the previous regulation (written in the 1990s).

The "new" regulation :

The "TSN law" of the 13 June 2006 that gives a clear regulatory body to the nuclear safety. It deals with safety, radioprotection, and with protection of the following interests : pubic security, public health, public safety, nature protection, environment protection.

⇒ It requires to have an integrated approach of all risks that could be generated by a facility.

⇒ Same requirement in the implementing regulatory text called the "Procedures Decree" of the 2 November 2007

French regulation concerning chemical risks in nuclear facilities

But some implementing texts of the TSN law are still under development.

=> The new regulation has to cohabit with the previous one.

French regulation concerning chemical risks in nuclear facilities

The "previous" regulation, still in force :

- Specific to nuclear installations : the order of the 31 December 1999 and some texts regulating releases
- General regulation concerning chemical risks : the parts of the environment code dedicated to chemical industries

The regulation specific to nuclear facilities is largely inspired by the regulation concerning chemical industries (first written in the 1970s).

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French regulation concerning chemical risks in nuclear facilities

The order of the 31 December 1999

= technical regulation in order to prevent and to limit hazards generated by nuclear facilities and that could have an impact on people living or working around, public security, public health, agriculture, environment or monuments.

It deals with practical measures to take concerning :

- ❑Noise and vibrations
- ❑Gaseous pollution
- ❑Water pollution
- ❑Waste management
- ❑Traffic inside nuclear sites
- ❑Load handling
- ❑Management of radiological, chemical, toxic, flammable, corrosive or explosive substances
- ❑Electrical devices
- ❑Lightning
- ❑Fire
- ❑Nuclear hazards (criticality, radiation protection)

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French regulation concerning chemical risks in nuclear facilities

Complementary to the order of the 31 December 1999 : the regulation concerning releases of nuclear installations

General text : the decree of the 4 May 1995 :

General principles and administrative procedures

Implementing text : the order of the 26 November 1999 :

Settles downs the content of limitations concerning water supplies and liquid and gaseous discharges

Deals with radiological and chemical releases

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French regulation concerning chemical risks in nuclear facilities

Complementary to the order of the 31 December 1999 : the regulation concerning incidental or accidental procedures for nuclear installations

The "Procedures Decree" requires that every nuclear installation has an internal emergency plan **(IEP – *PUI in French*)** developed by the licensee.

The decree of the 13 September 2005, which deals with every type of industry, requires that an particular intervention plan **(PIP – *PPI in French*)** is developed by the local authorities for each industrial site containing at least one nuclear facility.

PUI and PPI deal with incidental and accidental situations, with procedures to follow in these cases, and with areas that could have to be evacuated.

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French regulation concerning chemical risks in nuclear facilities

PUI and PPI are developed on the basis of the design assessment of the nuclear facility (risks inherent of the facility ; accidental scenarios).

=> PUI and PPI have to take into account chemical risks, when it is appropriate

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French regulation concerning chemical risks in nuclear facilities

Complementary to the order of the 31 December 1999 : the regulation concerning chemical industries – the environment code

The environment code can apply to parts of a nuclear site that are classified as "classified installations for the environment protection" (CIEP – *ICPE in French* = major chemical industries)

=> Mostly fuel storages for electricity generators, or storages of chemical substances

But parts of a nuclear facility can have a double classification because of chemical risks :

- ❑ as nuclear facility
- ❑ and as ICPE

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EXAMPLES

Example of the Tricastin site

Several nuclear facilities dealing with uranium chemistry and uranium enrichment

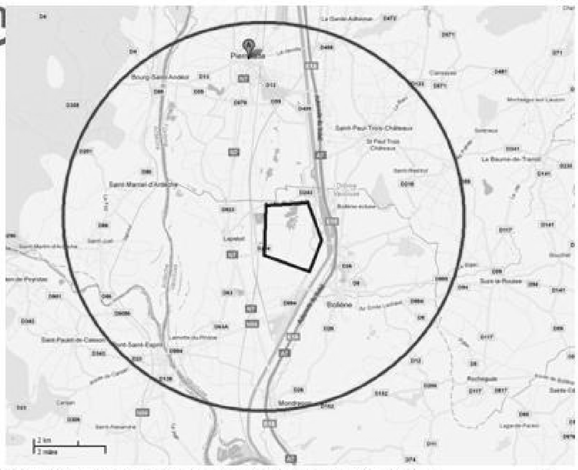
Possible accidental situations :

❑ criticality accident because of enriched uranium

❑ chemical accident because of the use of fluorine substances to create UF_6

The review of these possible accidents has concluded that the most serious accident on this site would be an air crash on the Eurodif UF_6 storage : it would generate a massive uncontrolled leak of UF_6 that would combine with the air to create great amounts of HF (very corrosive).

=> Evacuation zone = circle of 6.6-km radius



Example of the La Hague site

Nuclear fuel reprocessing facilities (liquid-liquid extraction process)

Raw materials : spent nuclear fuels => criticality risks + highly radioactive materials

+ Use of chemical products : very strong acids (HNO_3 at pH<1), highly concentrated NaOH, solvents…

=> Major accidental scenarios :

- ❑ Unwanted mixing of incompatible products, that can explode
- ❑ Solvent fire
- ❑ Explosion because of unwanted generation of explosive components like nitride or azide
- ❑ Uncontrolled red-ox reaction outside the process scope, that can generate plutonium precipitation in one part of the process, possibly leading to a criticality reaction
- ❑ Fire of some fuels components containing graphite or magnesium

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Example of the La Hague site

In addition, existence of potentially dangerous chemical products in one part of an installation makes it not easy to operate in incidental or accidental conditions, or can create worst situations to manage.

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CONCLUSION

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The French regulation in force concerning chemical hazards inside nuclear sites or installations is at the moment **a mix** between previous regulation settled down in the 1990s, and the new regulation based on the TSN law (2006).

It looks quite complex. But…

- ❑ it already requires taking into account all the risks (radiological and non-radiological ones)
- ❑ it still has helped taking into account chemical hazards in the nuclear installations that had really some

The regulation under development is expected to be more complete, more accurate and more simple.

Thank you for your attention

DETERMINATION OF DISCHARGE AUTHORIZATIONS FOR FRENCH BASIC NUCLEAR INSTALLATIONS[58] AND PUBLIC INFORMATION

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Abstract - The determination of discharge authorized limits for a French nuclear site is initiated by the request of the operator, based on the maximum nuclear and chemical inventory that could be released during normal operating conditions, accompanied with justifications. Request and justifications are analyzed and discussed by the ASN and the IRSN, taking into account nuclear and chemical inventories expected inside BNI, current regulations (BNI specific regulation, environment code, public health code), operating feedback (release feedback for an operating BNI, feedback coming from other nuclear sites or installations, etc.) and best available technologies that can be used to treat liquid or gaseous waste before release. After taking into account potential suggestions coming from public information or public enquiry concerning the operator request, the discharge authorized limits are settled down in specific ASN prescriptions that have to be ratified by the State secretaries in charge of nuclear safety. The whole process runs through 2 or 3 years to be achieved.

Communication has revealed to be quite an uneasy task, even for administrative procedures. This aspect is mostly tested while communicating about events. Consequences of this communication can hardly be foreseen because of multiple external parameters like : news on the front pages at the same moment ; historic communication difficulties still in the public mind ; technical vocabulary not easily understood ; public fear of things being hidden ; power of ecologist or nongovernmental associations.

Introduction

In application of the first French regulation concerning basic nuclear installations (BNI), the decree Nr 63-1228 of the 11 December 1963 [1], nuclear installations operators had to ask for discharge authorizations. These demands were based on an envelope calculation of the potential maximum impact the BNI or the nuclear site could have on the environment. In the 1980s it appeared that the release authorized limits settled down on the basis of this approach were much too higher than the real releases: in fact, BNI and nuclear sites only used a few percents of these limits (less than 10%), mostly because these limits were determined taking into account some incidental conditions in addition to

[58] Basic nuclear installation = facility containing amounts of radioactive substances or of fissile materials greater than the thresholds defined in the decree Nr 2007-830 of the 11 May 2007

[59] ASN : Autorité de sûreté nucléaire = French nuclear safety authority
DRC : ASN directorate in charge of nuclear waste, research facilities (including research reactors) and fuel cycle facilities

normal operating conditions. Moreover it appeared that authorized limits omitted some chemical parameters that could have an impact on water quality (drinkable or underground waters) and that were already regulated for chemical industries through the environment code.

In order to deliver more realistic and more complete authorized limits, a new regulation was settled down in the 1990s: the decree Nr 95-540 of the 4 May 1995 [2] and its application order of the 26 November 1999 [3]. This regulation is still applying and will be abrogated when the new regulation (mostly the BNI order – see Bibliography), implementing the TSN law [7], will be emitted, probably at the end of 2011.

1. Basic data for determining discharge limits

Discharge limits are suggested by an operator on the basis of the maximum potential inventory, radiological and chemical, that can be released in the environment by its nuclear site during normal operating conditions (including some foreseen and usual degraded operating conditions, like maintenance operations). As a consequence discharge limits are close to routine discharges.

The maximum inventory is used to calculate the impact on environment and the sanitary impact (chemical and radiological) of the considered nuclear site. Because this maximum inventory is based on the normal operating conditions, the calculated sanitary impact on the reference groups for any French nuclear site is very weak: usually less than 10 µSv/year for radiological issues, and under all prescribed limits for chemical substances.

This impact is controlled through numerous measures in the environment by operators, which results are sent to the ASN. Some of these measures are part of the release authorization. And the ASN controls the operators on this subject through specific inspections, eventually by sampling on the ground.

2. French regulation concerning BNI discharges – the TSN law, the Procedures decree, the decree of the 4 May 1995 and the order of the 26 November 1999

The TSN law [7] requires to take into account all the risks generated by a BNI, radiological and non-radiological. It also requires an integrated approach for all aspects related to the creation and the operating or dismantling of a BNI, in order to protect public security, public health, public safety and environment protection.

But the implementing regulation of the TSN law [7] is still not complete: in fact quite only the Procedures decree [8] is in force, which describes procedures to follow. As there is not yet new regulatory text describing the content of discharge authorizations, this content is settled by an old regulation, still in force: the decree of the 4 May 1995 [2] and the order of the 26 November 1999 [3].

As a consequence the current regulation concerning BNI discharges can be described as follows:

The Procedure decree [8] gives minimum requirements concerning the content of the operator request for discharges:

- For liquid discharges : types of waste to be treated, origins of these waste, quantities, physical characteristics, chemical and radiological composition, planned treatment processes, conditions of discharge in the receiving environment, composition of treated liquid waste before discharge, incidence of the discharges on the water resources, on the aquatic environment, on the water flows, quality and level;

- For gaseous discharges : all gaseous releases, including aerosols, dusts and their ground deposition ; incidence on air quality and soils quality;
- Public exposure due to all the discharges and the radionuclides transfers, including food chain;
- Compliance with the local water uses plans (see § 3 below);
- Means planned by the operator to prevent, limit or compensate the facility nuisances, with the corresponding estimated costs;
- The administrative procedure to follow is settled down by the Procedures decree [8];
- The administrative form of the discharge authorization is settled by the TSN law [7] and the Procedures decree [8] : ASN prescriptions countersigned by ministers in charge of nuclear safety;
- The content of the discharge authorization is described by the article 11 of the decree of the 4 May 1995 [2] and its application order, the order of the 26 November 1999 [3].

The decree of the 4 May 1995 [2] used to settle down the general regulation concerning discharges of civil BNI:

- Its article 1 requires specific authorizations for gaseous and liquid radioactive discharges;
- Its article 2 settles down specific rules for parts of BNI or installations classified for the environment protection (ICPE)[60] operating inside nuclear sites, that are submitted to the ICPE regulation which is part of the environment code [4]. This article 2 deals in fact with some chemical risks (the most important ones);
- The article 5 indicates that there is only one demand of discharge authorization for one nuclear site with one operator, including discharges of all the installations existing on the site (BNI and ICPE) and run by the same operator. As a result, a discharge authorization is global for one nuclear site, in order to take into account all its possible impacts on environment;
- The article 7 of the decree of the 4 May 1995 says that radioactive and non radioactive liquid releases of some substances into underground waters are forbidden;
- The article 11 requires that the discharge authorization deals with : limits for discharges and for water supplies ; means of analysis, measures and controls ; operator reports concerning discharges and water supplies ; and public information;
- The rest of the decree describes administrative procedures that had to be followed to deliver a discharge authorization, and that were abrogated by the TSN law [7].

The order of the 26 November 1999 [3] settles down the content of limitations concerning water supplies and liquid and gaseous releases of nuclear sites:

- Article 1 clearly says that limits concerning ICPE existing inside nuclear sites must be in accordance with the prescriptions of the environment code ;
- Title II – articles 3 to 7 – deals with water supplies;

[60] installations using or storing chemical substances. These installations are divided in different types according to their potentially dangerous aspects: the less dangerous installations are “declared installations” ; the other ones are “authorized installations”, inside which the higher possible classifications are the “SEVESO type” and the “SEVESO II type”.

- Title III – articles 8 to 14 – deals with gaseous releases, radiological and non-radiological ones:
  - Article 8: uncontrolled gaseous releases are not allowed;
  - Article 8 also: installations must be designed, run and maintained in order to limit releases. Gaseous waste must be captured at their production sources as much as possible, canalized and, if possible, treated before release, and the releases must be as low as possible. The release authorization settles down limits on the basis of the best available practices, taking into account costs of these practices and specificities of the environment around the considered nuclear site;
  - Article 9: limits must be put, when it is accurate, on the following radiological parameters : tritium, gaseous iodine, rare gas, carbon 14, other beta-and-gamma-emitting radionuclides, and alpha emitting radionuclides;
  - Article 10: limits must be put on chemical releases when these releases are submitted to authorization in accordance with the decree of the 4 May, that is to say releases coming from an activity classified as ICPE inside the nuclear site;
  - Article 11: gaseous releases must come out a facility through chimneys, except very specific justifications ; and all radioactive gaseous releases must be treated or filtrated before release;
- Title IV – articles 15 to 23 – deals with liquid releases, radiological and non-radiological ones :
  - Article 15: installations must be designed, run and maintained in order to limit releases. Liquid waste must be captured at their production sources as much as possible, canalized and, if possible, treated before release, and the releases must be as low as possible. The release authorization settles down limits on the basis of the best available practices, taking into account costs of these practices and specificities of the environment around the considered nuclear site. Limits must be in accordance with the quality objectives of the receiving waters, with their fish vocation and with the water-uses management planning;
  - Article 16: limits must be put, when it is accurate, on the following radiological parameters : tritium, radioactive iodine, carbon 14, other beta-and-gamma-emitting radionuclides, and alpha emitting radionuclides;
  - Article 17: limits must be put on the acceptable pH for receiving waters. Choice of parameters to limit must be made on the basis of fluxes and toxicity of released substances. The releases should generate neither smell nor color in receiving waters. The releases must not generate any harm to fish populations ; they must not contain enough hydrocarbons to generate an oil film on the water surface ; and their temperature must be acceptable for the receiving waters;
- Title V – articles 24 to 26 – deals with global means that the operator has to have on its site (sampling and analysis means) and records that the operator has to put in place and to update;
- Title VI – articles 27 to 30 – deals with the information that the operator must provide to authorities, and with the controls that the authorities can make;

The regulation in force concerning BNI discharges settles down the radiological parameters to examine while determining the discharge authorization (tritium, radioactive iodine, rare gas, carbon

14, other beta-and-gamma-emitting radioelements, and alpha emitting radioelements). It requires also to examine chemical parameters, but mostly refers to other regulations on this subject: the environment code and the water-uses planning.

The regulation under development is expected to replace the previous still-in-force regulation, and to require that every discharge prescriptions list precisely all substances, chemical and radiological, that have to be surveyed, and not only groups of substances.

3. Others regulation that have to be taken into account: the environment code, the water-uses planning and the public health code

The environment code [4] regulates the ICPE, that is to say the chemical industries, and chemical pollutions. As it was said before, releases coming from parts of nuclear sites submitted to the ICPE regulation must comply with the environment code.
Moreover, as it determines maximum acceptable release limits for each chemical substance classified as possibly dangerous for health or environment, the environment code gives reference values that can be compared with the chemical limits suggested by the operators.

The environment code introduces also ***the water-uses plans***. These plans were created in the 1990s in order to limit and prevent water pollution and to coordinate water uses. A plan concerns a geographic water-side basin, and determines water-quality objectives for each river inside the considered water-side basin. Every installation that releases substances in waters has to be in accordance with the concerned water-uses plan.

Moreover, as most BNI release substances in waters that can be used for human activities (potable waters, leisure activities), they have also to comply with ***the public health code*** [5] that determines acceptable limits of concentrations of chemical substances in waters that can be used as potable waters or for leisure activities.

The use of limits contained in the environment code [4] and in the public health code [5] has lead in the 2000s to precise parts of discharge orders concerning chemical releases : compared to discharge orders enacted before 2000, the last orders list precisely the chemical elements and/or substances to survey, and not only groups of substances.

At last, some nuclear sites can be submitted to the specific regulation concerning greenhouse effect, which is composed by orders limiting releases of greenhouse-effect gases [6]. The concerned French nuclear sites are the La Hague site and the Grenoble CEA site, because of their boilers.

4. Methodology for determining regulatory limits on routine releases

As it was said before, the operator proposes discharge limits based on the maximum potential inventory, radiological and chemical, that can be released in the environment by its nuclear site during normal operating conditions (including some foreseen and usual degraded operating conditions, like maintenance operations). This request can be of different types:

- for a new facility : a global request for discharge limits, proposed with the request of the BNI authorization creation in application of the Procedures decree [8];
- for an operating BNI or nuclear site : a global revision of previous limits, or a change for some limits.

According of the type of the request, it can include all or parts of the following parameters:

- Radiological releases, liquid and gaseous, with precise list of concerned radionuclides, or accurately chosen inside the following list : tritium, radioactive iodine, rare gas, carbon 14, other beta-and-gamma-emitting radionuclides, and alpha emitting radionuclides;
- Chemical releases, on the basis of the activities run inside the nuclear site, and in accordance with the environment code and, when necessary, with the public health code;
- Biological releases when it is accurate, mostly when the BNI cooling system includes cooling-towers because of the legionella risk. This risk is regulated by the public health code.

The operator request and its justifications are then analyzed by the ASN and the IRSN, and discussed between the operator, the ASN and the IRSN.

For operating BNI or nuclear sites, the discussion is based on:

- The experience feedback concerning the actual releases measured during the 10 last years at least;
- The best available technologies allowing better treatments before release or allowing less releases;
- The comparison with the limits granted to similar BNI and with actual releases of similar BNI;
- The comparison with the limits for chemical releases settled down by the environment code and the public health code.

For new facilities, that sometimes have no equivalent in France and sometimes none elsewhere, the discussion is based on:

- The nuclear and chemical inventories expected in the new facility;
- The feedback concerning mostly French facilities that can have some similarities with the planned plant, concerning the nuclear or chemical inventories, the process or the releases;
- The best available technologies that can be used to treat liquid or gaseous waste before release;
- The limits for chemical releases settled down by the environment code and the public health code.

The process of analysis and discussion can be long, up to 2 or 3 years.

In parallel with these analysis and discussion, the administrative procedure requires:

- For new BNI, or for BNI requesting modifications of their discharge authorizations in parallel with modifications of their decrees : a public enquiry covering all the requested modifications;
- For operating BNI requesting modifications of their discharge authorizations only: a public enquiry or a public information, if the modifications are limited.

The law of the 12 July 2010 known as the “Grenelle 2 law” [9] now requires public information for any small increment in discharge limits.

Public enquiries or public information should allow to present to the public the operator request, and should allow the public to ask questions and to tell concerns. The results of these procedures can lead to modify or lower limits concerning radiological or chemical substances.

The draft of discharge limits is submitted to the examination of the Departmental Council for environment and for technical and sanitary risks.

At the end, the new discharge limits are settled down into ASN prescriptions that have to be countersigned by ministers in charge of nuclear safety. These limits can only be equal or lower than the limits suggested by the operator, and cannot regulate releases that were not in the operator request.

As the process for officially settling down discharge limits is quite long, discharge limits are not often changed. That is why operators have to integrate the foreseen modifications of their facilities when they request a modification in the discharge limits.

The BNI order, which is still under writing, is expected to concatenate and update the regulation above.

5. Example of discharge authorized limits: the La Hague site

The discharge authorization for the La Hague site is the order of the 10 January 2003, modified by the order of the 8 January 2007 [10]. This order is representative of the improvement process that has been at work since the beginning of the 2000s, in order to settle down discharge authorization closer to routine releases, particularly in terms of listed substances and of level of authorized limits.

5.1.Limits settled down for gaseous discharges

The radiological substances are the same as those listed in the order of the 26 November 1999 [3]:

- Tritium;
- Iodine;
- Radioactive rare gas;
- Carbon 14;
- Other artificial beta-and-gamma emitting radionuclides;
- Artificial alpha-emitting radionuclides.

But, concerning chemical gaseous discharges, all chemical components or elements are clearly listed.

The other improvement is the detailed list of points of surveillance or of analysis, that indicates which type of radionuclides is measured at each point and whether the measures are continuous or offline.

5.2.Limits settled down for liquid discharges

Radiological liquid waste must be stored in specific tanks before release. The authorized limits concern liquid waste radiological activity measured at the tank exit at the time of release.

The radiological substances are more detailed than for the gaseous discharges: in addition of the 6 groups listed in order of the 26 November 1999 [3], and because of the feedback of the routine releases of the site, there are:

- Strontium 90;
- Cesium 137;
- Cesium 134;
- Ruthenium 106;
- Cobalt 60.

Chemical liquid waste are released at several outlets. Consequently limits are settled down for each outlet.

Chemical substances are also detailed, in order to take into account all the chemical products used on the site and all their derivatives that can be traced in liquid waste.

Moreover the parameters listed below have to be surveyed:

- pH ;
- Color;
- Odor;
- Temperature;
- Toxic substances able to kill fishes or ruin flora;
- Dissolved oxygen;
- Detergents;
- Hydrocarbons;
- Bacteriological quality, especially Escherichia coli and enterococcus.

These parameters are surveyed in application of the article 17 of the order of the 26 November 1999 [3]. The list is inspired by discharge authorizations for ICPE and parameters figuring in the environment code [4] or in the public health code [5].

5.3.Environment surveillance plan

The discharge authorization settles down minimum measures in the site environment (article 14 for gaseous discharges; article 27 for liquid discharges):

- A minimum number of surveillance points is clearly defined, eventually grouped by types;
- List of parameters to survey, for each surveillance point or group of points ;
- Sampling frequency.

Article 30 indicates what minimum means the operator has to have to perform the discharges surveillance.

5.4.Information concerning discharges

The discharge authorization requires that the operator sends an annual report concerning routine discharges. This report can be communicated to the public. It contains elements concerning at least :

- Recall of the discharge authorization ;

- Balance sheet of the water supplies ;
- Balance sheet of discharges ;
- Balance sheet of environment surveillance measures ;
- When it is pertinent, the exceptional works that made the operator used chemical substances that were traced in the discharges ;
- Balance sheet of the discharges effects, chronicle or accidental, on air, water and soils ;
- Estimation, the more realistic as possible, of doses to people due to discharges ;
- Description of maintenance works done on equipments part of water-supplies or discharges circuits ;
- Description of events that occurred during the past year, and concerning water supplies or discharges ;
- Evolution of water supplies and of discharges over several years ;
- Description of actions done by the operator in favor of environment protection.

6. Communication / public information

Public information, whatever form it takes, is required for some administrative procedures concerning BNI: creation of a BNI, significant modifications, determination or modification of discharge limits, decommissioning authorization. Most of the time in the case of BNI, this public information is made with a public enquiry that is regulated by the environment code.

But public information does not only deal with administrative procedures. It usually has to be done after an incident and has revealed to be not an easy task. Communication was previously the prerogative of the French government. But since the TSN law (2006), the ASN is in charge of the communication to the public on nuclear activities.

First, the focus of the public on events is difficult to anticipate, because it partly depends on the media background and on the other news that will place at the same moment. Example: at the end of December 1999, two big storms touched France, one in the north part, the other in the south part. They seriously damaged significant parts of the electricity network, provoking electricity breakdowns in hundreds of thousands homes. At the same time, the Blayais nuclear site was flooded because of the combination of the storm in the south part of France and of high tide. It was the first time since years that the national nuclear crisis system was activated. But news did not talk about the Blayais crisis; they focused on electricity breakdowns.

Second, public still keeps in mind historic communication difficulties of the French government, and often fears that the nuclear industry and the government are hiding or minimizing the events. The origin of this suspicion is clearly the Tchernobyl accident and the rather poor management of the sanitary aspects by the French government at the time.

Third, this aspect is strengthened by the almost technical aspects of the nuclear industry. As a result, the technical terms used to describe a facility or a process are understood with difficulty by the public and must be clearly explained in order to prevent any reaction implying that something is hidden behind the technical vocabulary.

And four, there are in France some powerful ecologist associations and nongovernmental organizations (NGO), some of them requiring France to abandon nuclear industry, as for example Greenpeace or Sortir du Nucléaire.

Example: the event that occurred on the 7 July 2008 in the SOCATRI facility:

The SOCATRI facility is located in the Tricastin site, and treats uranium liquid waste for the Eurodif and the GB II plants.

In 2008 the SOCATRI facility was under big modifications, in order to modernize its installations. In particular important works were in progress in its tanks rooms. On the 7 July 2008, a liquid waste transfer was ordered between 2 old tanks, not modernized. But, because of a switch error, the waste were sent to another tank, still quite full, that was quickly overflowed. The liquid waste felt then in the tank retention that was unconfined and close to an open wall because of modification works. As a consequence the liquid waste came out of the building polluting both the soil and the river “La Gaffière” that received about 30 m^3 of liquid waste containing about 75 kg of natural uranium. No significant consequences on the environment or on the public health occured.

As the operator and the ASN gave information on this incident classified on level 1 of the INES scale, the event became of public concern and was subject to many press releases, mainly because it took place during holidays at a time without any other significant topic.

This event revealed the gaps of the media culture concerning nuclear industry: media persisted in showing the Eurodif cooling-towers that are near the SOCATRI facility, and were talking of the Tricastin NPP instead of the waste treatment facility of Socatri. As a consequence, the operator and the ASN had to explain, during several days, what was the SOCATRI facility. But their explanations were not necessarily taken into account, because the Eurodif cooling-towers were more photogenic than the SOCATRI building.

Moreover the event occurred only a few days after that an existing uranium pollution, located in the south of the site, with no identified cause, was revealed. This pollution was announced to the local stakeholders. Precise public information was given in a meeting on the 4 July 2008, but did not get significant media coverage at this time. The SOCATRI event of the 7 July 2008 suddenly increased the media coverage.

In these circumstances all the events that occurred in the Tricastin site facilities during July and August 2008, whoever the operator was, were highlighted by the media, even those rated at level 0 of the INES scale. The operators were suspected of hiding things since years.

Moreover the Government representative, the ASN and the IRSN had to explain and justify how they had managed the historic pollution, how they had informed the public and how they had taken decisions in time concerning protection measures for the public after the releases of the 7 July 2008.

The media pressure became lesser on August 2008, because of the Olympic games, went up at the beginning of September 2008 because of the occurrence of a new event in the EDF Tricastin site (a fuel assembly that kept hanged on the vessel cap during the opening of the vessel), and disappeared after the treatment of this last event (November 2008). The Tricastin nuclear site has not been of media interest since this time.

7. Conclusion

The determination of release authorized limits for a French nuclear site is initiated by the request of the operator, based on the maximum nuclear and chemical inventory that could be released during normal operating conditions, accompanied with justifications. Request and justifications are analyzed and discussed by the ASN and the IRSN, taking into account nuclear and chemical inventories expected inside BNI, different regulations (BNI specific regulation, environment code, public health code), operating feedback (release feedback for an existing BNI, feedback coming from other nuclear sites or installations, etc.) and best available technologies that can be used to treat liquid or gaseous waste before release. After taking into account potential suggestions coming from public information or public enquiry concerning the operator request, the release authorized limits are settled down in specific ASN prescriptions that have to be ratified by the State secretaries in charge of nuclear safety. The whole process can need 2 or 3 years to be achieved.

The BNI order is expected to be issued at the end of the year. It will update the French regulation concerning release authorized limits, in the following aspects:

- Settling down proceedings to follow to determine new release limits for a facility or a nuclear site;
- Regulating water supplies;
- Requiring the collect and the treatment of all effluents before release;
- Regulating gaseous and liquid releases;
- Settling down minimum survey of the environment;
- Requiring nuisances prevention.

Communication has revealed to be quite an uneasy task, even for administrative procedures. This aspect is mostly tested while communicating about events. Results of this communication can hardly be foreseen because of multiple external parameters as news on the front pages at the same moment ; historic communication difficulties still in the public mind ; technical vocabulary not easily understood ; public fear of things being hidden ; power of ecologist or nongovernmental associations. Lessons have already been learned, mostly in France from the Tchernobyl accident and the SOCATRI event of the 7 July 2008. But as communication is quite a new prerogative of the ASN in France, there are still many lessons to learn, as for example from the 2011 Fukushima accident.

References

[1] Decree Nr 63-1228 of the 11 December 1963 dealing with general regulation concerning basic nuclear installations

[2] Decree Nr 95-540 of 4 May 1995 concerning releases of basic nuclear installations

[3] Order of the 26 November 1999 concerning water supply and liquid and gaseous releases of basic nuclear installations

[4] Environment code:

- articles L.212-1 to L.212-1 concerning water-uses planning
- article L.213-10 concerning license fee for water pollution or for some water uses
- articles L.218-73 to L.218-80 concerning harmful releases in seas
- articles L.219-1 to L.219-18 concerning protection of sea environment
- articles L.222-1 to L222-8 concerning air protection
- articles L.229-1 to L.229-54 concerning greenhouse effect
- articles L.300 to L.350 concerning specific protected sites because of their environmental interest
- articles concerning installations classified for environment protection : especially articles L.511, L.512 (general regulation), L.515-8 to L.515-12 (installations requiring constraints of public utility, L.515-15 to L.515-26 (installations submitted to a plan for technological risks prevention)
- articles L.123-1 to L.123-16 concerning public enquiry
- articles L..124-1 to L.124-8 concerning public right of information concerning environment
- articles L.125-1 to L.125-9 concerning of other means of public information

[5] Public health code:

- articles R.1321-1 to R.1322-4 concerning quality of drinkable waters, quality of underground waters
- articles L.1321-1, L.1321-4, L.1335-2, R.1321-1, R.1321-2 and R.1321-46 concerning the legionella risk
- articles L.1333-1 to L.1333-20 concerning radioactive radiations

[6] Orders concerning quotas for gaseous releases having an impact on greenhouse effect, and concerning some nuclear sites:

- order of the 22 March 2007 concerning the Grenoble CEA site
- order of the 17 March 2008 concerning the La Hague site

[7] Law Nr 2006-686 of the 13 June 2006 concerning transparency and nuclear safety, known as the “TSN law”

[8] Decree Nr 2007-1557 of the 2 November 2007 concerning basic nuclear installations, known as the “Procedures decree”

[9] Law of the 12 July 2010, known as the “Grenelle 2 law”

[10] Order of the 10 January 2003, modified by the order of the 8 January 2007, concerning water supplies and discharges of the La Hague site

Bibliography

– Draft of the order concerning general regulation for BNI, known as the “BNI order”

Determination of discharge authorizations for French nuclear installations

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Brief history

The decree Nr 63-1228 of the 11 December 1963 required that nuclear installations licensees asked for discharge authorizations.

In the 1980s it appeared that discharge limits :

- were too higher than the real releases, because the limits were calculated taking into account some incidental situations in addition to normal operating conditions
- often omitted some chemical parameters that could have an impact on water quality, and that were already regulated for chemical industries (through the environment code)

=> A new regulation was settled down in the 1990s concerning discharge of nuclear facilities

Basic data for determining discharge limits

Discharge limits are suggested by the operator on the basis of the maximum potential inventory, radiological and chemical, that can be released in the environment by its nuclear site during **normal operating conditions** (including some foreseen and usual degraded operating conditions, like maintenance operations)

⇒ Discharge authorization deals with the releases of the one nuclear facility, and can deal with the releases of the whole nuclear site

⇒ Discharge limits are close to routine releases

French regulation concerning discharges of nuclear installations

The "TSN law" of the 13 June 2006 requires to take into account all the risks, radiological and non-radiological, generated by a nuclear facility.

As its implementation texts are mostly still under development, the regulation in force concerning discharges of nuclear installations is a mix between "previous regulation" (written in the 1990s) and the "new" one (after 2006), and can be summarized as follows.

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French regulation concerning discharges of nuclear installations

Procedures to follow : "Procedures decree" of the 2 November 2007

Minimum requirements for the content of the operator request : "Procedures decree" of the 2 November 2007 :

- ❑ description of liquid discharges
- ❑ description of gaseous discharges
- ❑ Public exposure due to all the discharges and the radionuclides transfers, including food chain
- ❑ Compliance with the local water-uses plans
- ❑ Means planned by the operator to prevent, limit or compensate the facility nuisances, with the corresponding estimated costs

Administrative form of the discharge authorization : "TSN law" of the 13 June 2006 and "Procedures decree" of the 2 November 2007

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French regulation concerning discharges of nuclear installations

Content of the discharge authorization : decree of the 4 May 1995 and order of the 26 November 1999 :

- ❑ water supplies
- ❑ gaseous releases, radiological and non-radiological ones ; *list for radiological ones : tritium, gaseous iodine, rare gas, carbon 14, other beta-and-gamma-emitting radionuclides, and alpha emitting radionuclides ; for chemical parameters : refers to the environment code and to water-uses plans*
- ❑ liquid releases, radiological and non-radiological ones ; *list for radiological ones : tritium, radioactive iodine, carbon 14, other beta-and-gamma-emitting radionuclides, and alpha emitting radionuclides ; for chemical parameters : refers to the environment code and to water-uses plans*
- ❑ global means that the operator has to have on its site (sampling and analysis means) and records that the operator has to put in place and to update
- ❑ information that the operator must provide to authorities, and the controls that the authorities can realize

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French regulation concerning discharges of industrial installations

The environment code :

- ❑ regulates discharges of chemical industries
- ❑ determines maximum acceptable release limits for each chemical substance classified as possibly dangerous for health or environment

The water-uses plans :

- ❑ introduced by the environment code
- ❑ determine water-quality objectives for each river inside the considered water-side basin

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French regulation concerning discharges

The public health code :

- ❑ determines acceptable limits of concentrations of chemical substances in waters that can be used as drinkable waters or for leisure activities

Specific regulation concerning greenhouse effect :

- ❑ orders limiting releases of greenhouse-effect gases
- ❑ the La Hague site and the CEA Grenoble site are submitted to.

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Methodology for determining regulatory limits for routine discharges

1. The operator suggests discharges limits based on the maximum potential inventory, radiological and chemical, that can be released in the environment by its nuclear site during normal operating conditions

- ❑ global request or change of some limits (for operating nuclear installation)
- ❑ can deal with radiological, chemical or biological releases
- ❑ suggested limits must be justified and accompanied by impact calculations on people

WGFCS – 27-29 September 2011 – Determination of discharge authorizations for French nuclear installations - Communication 9

Methodology for determining regulatory limits for routine discharges

2. The suggested discharge limits are analyzed and discussed between the operator, the ASN and the IRSN (technical support of ASN) based on :

- ❑ for operating nuclear installation : experience feedback over the 10 last years
- ❑ for new facility : nuclear and chemical inventories expected
- ❑ for new facility : feedback concerning mostly French facilities that can have some similarities with the planned facility
- ❑ best available technologies allowing best/better treatments before release or allowing less releases
- ❑ comparison with the limits granted to similar facilities and with actual releases of similar facilities
- ❑ comparison with the limits for chemical releases settled down by the environment code and the public health code

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Methodology for determining regulatory limits for routine discharges

3. Parallel to point 2, the operator request is submitted to an administrative procedure :

public enquiry, dealing also with the eventual modifications or creation of the facility decree

The results of the public enquiry can lead to lower discharge limits.

Points 2 and 3 can be long, up to 2 or 3 years long.

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Methodology for determining regulatory limits for routine discharges

4. The draft of discharge limits is submitted to the examination of the Departmental Council for environment and for technical and sanitary risks.

5. The new discharge limits are settled down into ASN prescriptions that have to be countersigned by ministers in charge of nuclear safety.

These limits can only be equal or lower than the limits suggested by the operator, and cannot regulate releases that were not in the operator request.

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Perspective

The order concerning general regulation for nuclear installations will update the French regulation concerning release authorized limits, in the following aspects :

❑The settling down proceedings to be followed to determine new release limits for a facility or a nuclear site

❑The regulation of water supplies

❑The requirements concerning the collect and the treatment of all effluents before release

❑The regulation of gaseous and liquid releases

❑The minimum survey of the environment

❑The nuisances prevention

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EXAMPLE : discharge limits for the La Hague site

Discharge authorization = order of the 10 January 2003, modified by the order of the 8 January 2007

List of installations on the site only submitted to the environment code regulation (chemical installations - ICPE)

Water supplies :

❑ List of supply points : sea, groundwater, brook "Les Moulinets", brook "Froide Fontaine"

❑ maximum quantities

❑ supply conditions

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EXAMPLE : discharge limits for the La Hague site

Gaseous discharges :

❑ Place and height of all chimneys ; origins of releases for each chimney

❑ limits for a list of radionuclides = list existing in the order of the 26 November 1999 : tritium, iodine, radioactive rare gas, carbon 14, beta and gamma emitters, alpha emitters

❑ limits for a large list of chemical substances : SO2, dusts, NOx, CO, cadmium, mercury, thallium, arsenic, lead, cobalt, copper…

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EXAMPLE : discharge limits for the La Hague site

Liquid discharges :

- ❑ radioactive waste must be stored before release
- ❑ points of discharges
- ❑ limits for a list of radionuclides = list existing in the order of the 26 November 1999 (tritium, iodine, carbon 14, beta and gamma emitters, alpha emitters) + strontium 90, cesium 137, cesium 134, ruthenium 106, cobalt 60 (*feedback of the routine releases of the site*)
- ❑ limits for a large list of chemical substances : SO_2, dusts, NOx, CO, cadmium, mercury, thallium, arsenic, lead, cobalt, copper…
- ❑ other regulated parameters : pH, color, odor, temperature, toxic substances, dissolved oxygen, detergents, hydrocarbons, bacteriological quality

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EXAMPLE : discharge limits for the La Hague site

Environment surveillance plan :

- ❑ Minimum number of surveillance points is clearly defined, eventually grouped by types
- ❑ List of parameters to survey, for each surveillance point or group of points
- ❑ Sampling frequency

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EXAMPLE : discharge limits for the La Hague site

Annual report :

- ❑Recall of the discharge authorization
- ❑Balance sheet of the water supplies
- ❑Balance sheet of discharges
- ❑Balance sheet of environment surveillance measures
- ❑When it is pertinent, the exceptional works that made the operator used chemical substances that were traced in the discharges
- ❑Balance sheet of the discharges effects, chronicle or accidental, on air, water and soils
- ❑Estimation, the more realistic as possible, of doses to people due to discharges
- ❑Description of maintenance works done on equipments part of water-supplies or discharges circuits
- ❑Description of events that occurred during the past year, and concerning water supplies or discharges
- ❑Evolution of water supplies and of discharges over several years
- ❑Description of actions done by the operator in favor of environment protection

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COMMUNICATION

PUBLIC INFORMATION

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Regulatory context

Before 2006 : communication was the prerogative of the French government.

The ASN was a directorate of the ministry of industry.

Since the effectiveness of the “TSN law” of the 13 June 2006, the ASN is an independent administrative authority = it is no more a directorate of a ministry

=> The ASN is free to communicate independently of ministries, and has to.

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ASN observations

Public information : through public enquiry for creation or big modifications of a facility, and for discharge limits

But it is an “administrative” and formal way of communication.

Actual issue : to communicate out of administrative procedures, mostly on nuclear incidents or accidents and/or on public / journalists requests

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ASN observations

Communication is a hard task because :

- very difficult to anticipate the focus of the public and of the journalists
- suspicion of the public because of the poor management of the sanitary consequences of the Tchernobyl accident
- suspicion of the public due to the very specific language => fear that something is hidden behind this language
- powerful environmental associations or non-governmental organizations in France

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EXAMPLES

Flooding of the EDF Blayais site (December 1999)

- **What happened ?**

On the 26 and 27 December 1999 France was crossed by 2 major storms

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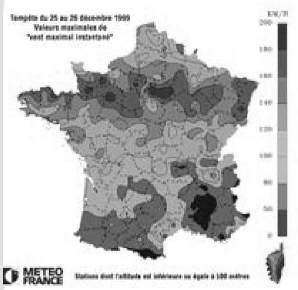
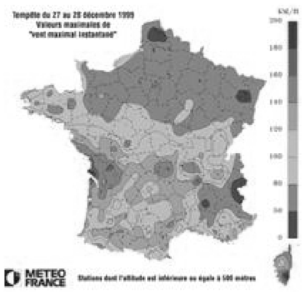
EXAMPLES

Storms of December 1999





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EXAMPLES

Flooding of the EDF Blayais site (December 1999)

❑Consequences :

- significant parts of the French electricity network seriously damaged
- The Blayais site flooded => emergency shutdown of the reactors + basements flooded => activation of emergency plans and of crisis centers

❑ **News and public focused on the electricity breakdowns**

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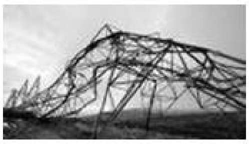
EXAMPLES

Consequences of the storms of December 1999



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EXAMPLES

Consequences of the storms of December 1999 :

the Blayais site

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EXAMPLES

Uncontrolled uranium liquid releases from the SOCATRI facility (7 July 2008)

❑ ***What happened ?*** A liquid radioactive waste transfer was ordered but unfortunately sent to the wrong tank (switch error). This tank was already quite full => it quickly overflowed in a retention that was unconfined and close to an open wall because of modification works => the liquid waste came out of the building polluting both the soil and the river "La Gaffière"

❑ **Consequences :**

- the river "La Gaffière" received about 30 m3 of liquid waste containing about 75 kg of natural uranium
- No significant consequences on the environment or on the public health

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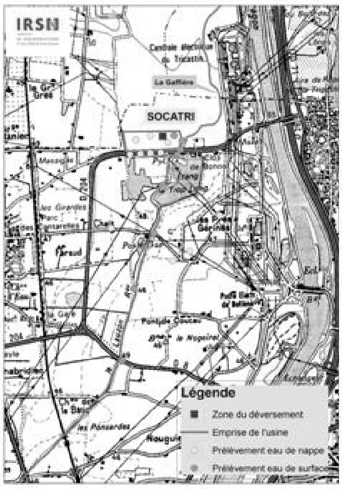

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EXAMPLES

Uncontrolled uranium liquid releases from the SOCATRI facility (7 July 2008)

❑ there were no other news of importance at the moment => journalists focused on the event

❑ By ignorance, journalists first talked about the Tricastin NPP, because EDF is more well-known as the Socatri society.

❑ The journalists kept on showing pictures of the Eurodif cooling towers, because they were more photogenic than the SOCATRI square buildings.

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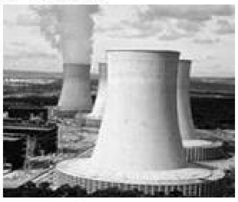
EXAMPLES

Photos in the media in 2008



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EXAMPLES

Photos in the media in 2010,

at the moment of the trial court concerning the water pollution offense and delay in notification of the event

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EXAMPLES

Lessons learned :

- for journalists, nuclear facilities = NPP ; and it is very hard to disabuse them
- journalists are looking for photogenic or symbolic shots => they prefer cooling towers or reactor buildings
- ASN press releases have to be very clear and understandable (no technical vocabulary, or the less as possible), to prevent journalists from giving wrong information.

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EXAMPLES

Furnace explosion at the CENTRACO facility (12 September 2011)

CENTRACO facility = incinerator of slightly contaminated waste

=> 2 furnaces : one for waste containing liquid elements (including oils) ; one for metallic items

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EXAMPLES : the CENTRACO event : *The Marcoule site*

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EXAMPLES : the CENTRACO event : *The Marcoule site*

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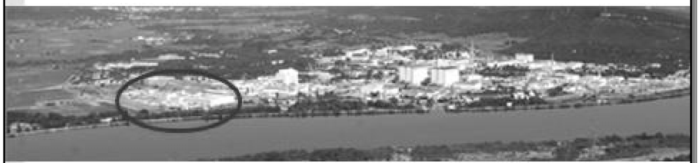
EXAMPLES : the CENTRACO event :
The Marcoule site



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EXAMPLES

The CENTRACO facility

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EXAMPLES :

The CENTRACO facility

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EXAMPLES

Furnace explosion at the CENTRACO facility (12 September 2011)

❑ ***What happened ?*** Known elements (investigations still in progress) : metallic items were under melting inside the melting furnace. But the melting could not be done. Two workers came inside the furnace room, opened the furnace, and one of them tried to mix the metallic items with a perch. A very few seconds later, liquid melted metal spread out of the furnace.

EXAMPLES

Furnace explosion at the CENTRACO facility (12 September 2011)

Consequences :

- One dead worker (the one who was using the perch), the other seriously injured (burned up to 80%)
- The pressure wave blown up the furnace room door and injured 3 more workers (that were in the next room)
- There were only 67 kBq inside the furnace + ventilation was still operating + no damage on the building

⇒ no contamination of the installation or of the workers + no measurable release in the environment

Provisional INES rating : level 1

EXAMPLES

Furnace explosion at the CENTRACO facility (12 September 2011)

Communication :

- ASN, IRSN and CEA provided press releases in the 12 September 2011 afternoon => it could not prevent journalists from talking of "an explosion at the Marcoule reactor"
- Journalists still prefer photogenic or symbolic images => 2 kinds of images in the news : emergency evacuation of the seriously-injured worker // various photos of the Marcoule site, not linked to the CENTRACO facility
- Content of the press articles were quite good : they resume the ASN, IRSN and CEA press releases


EXAMPLES
Gard Exclusif : le grand brûlé dans l'explosion de Marcoule transféré à Paris
13/09/2011, 15 h 39 | Mis à jour le 13/09/2011, 16 h 03
23 réactions
Facebook
Twitter
Envoyer par mail
Imprimer
A + grand A + petit
PARTENARIATS
Midi Libre
Présente les 100 métiers qui recrutent en Languedoc-Roussillon vidéos et fiches métiers
Gravement brûlé dans l'explosion du four de Centraco sur le site de Marcoule, le jeune homme a été héliporté au CHU Lapeyronie à Montpellier et vient d'être transféré en région parisienne. (Midi Libre Bagnols)
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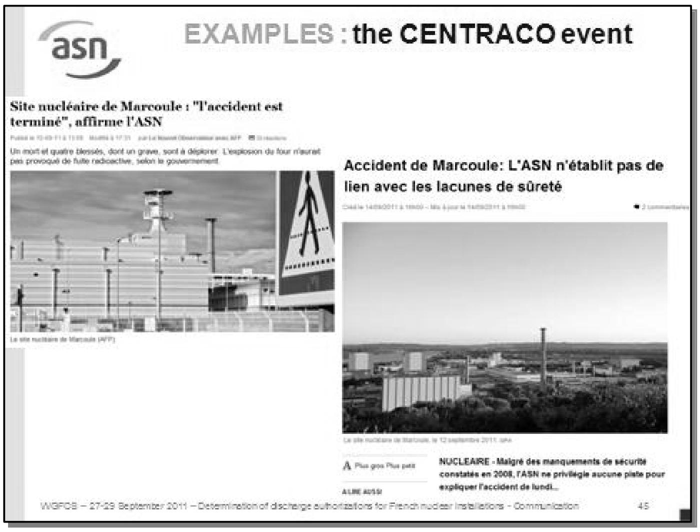
MARCOULE (GARD)
Explosion sur un site nucléaire: un mort mais pas de fuite radioactive
Partagez
Outils
It is a photo of the Phenix reactor…
L'usine de retraitement des déchets nucléaires Centraco (centre nucléaire de traitement et de conditionnement) à Marcoule. Photo Archives DL
WGFCS – 27-29 September 2011 –


EXAMPLES : the CENTRACO event
Site nucléaire de Marcoule : "l'accident est terminé", affirme l'ASN
Un mort et quatre blessés, dont un grave, sont à déplorer. L'explosion du four n'aurait pas provoqué de fuite radioactive, selon le gouvernement
Le site nucléaire de Marcoule (AFP)
Accident de Marcoule: L'ASN n'établit pas de lien avec les lacunes de sûreté
A Plus gros Plus petit
NUCLEAIRE - Malgré des manquements de sécurité constatés en 2008, l'ASN ne privilégie aucune piste pour expliquer l'accident de lundi...
WGFCS – 27-29 September 2011 – Determination of discharge authorizations for French nuclear installations - Communication
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IRSN GLOBAL PROCESS FOR LEADING A COMPREHENSIVE FIRE SAFETY ANALYSIS FOR NUCLEAR INSTALLATIONS

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Abstract - A fire safety analysis (FSA) is requested to justify the adequacy of fire protection measures set by the operator. A recent document written by IRSN outlines a global process for such a comprehensive fire safety analysis.

Thanks to the French nuclear fire safety regulation evolutions, from prescriptive requirements to objective requirements, the proposed fire safety justification process focuses on compliance with performance criteria for fire protection measures. These performance criteria are related to the vulnerability of targets to effects of fire, and not only based upon radiological consequences out side the installation caused by a fire.

In his FSA, the operator has to define the safety functions that should continue to ensure its mission even in the case of fire in order to be in compliance with nuclear safety objectives. Then, in order to maintain these safety functions, the operator has to justify the adequacy of fire protection measures, defined according to defence in depth principles.

To reach the objective, the analysis process is based on the identification of targets to be protected in order to maintain safety functions, taken into account facility characteristics. These targets include structures, systems, components and personal important to safety. Facility characteristics include, for all operating conditions, potential ignition sources and fire protections systems.

One of the key points of the fire analysis is the assessment of possible fire scenarios in the facility. Given the large number of possible fire scenarios, it is then necessary to evaluate "reference fires" which are the worst case scenarios of all possible fire scenarios and which are used by the operator for the design of fire protection measures.

Introduction

This document presents the IRSN[61]'s process for a comprehensive analysis of fire hazards in nuclear installations.

The purpose of the analysis process described in this document is to prove that the fire protection measures in the facility are acceptable and sufficient. The justification of the measures retained should be based on the fulfilment of technical criteria considering the fire hazard analysis. On the basis of the safety functional requirements in combination with the protection requirements for safety targets[62], the operator shall therefore determine the performance criteria that shall be met by the fire protection measures. Compliance with these criteria ensures that the safety objectives will be met.

This guide is a tool is to explain the analysis process of the fire hazards in nuclear installations. Thanks to its general nature, very specific configurations may require to be adapted to the process described in this document.

Fire and nuclear safety considerations

In the room where fire breaks out, the fire will cause a temperature increase, a change in pressure, turbulence as well as the production of hot gases and combustion aerosols toxic, corrosive and possibly flammable. Fire is often the source of opaque and explosive atmospheres.

It is important for all of the effects of the fire to be considered in the analysis, on the one hand to assess the vulnerability of the targets that need to be protected against the effects of fire and, on the other, to establish the fire protection measures.

Only considering fire characteristics is not enough to determine and design the fire protection measures needed for a satisfactory level of safety. It is also essential to consider the unfavourable effects of the fire extinguishing systems selected (excess pressure due to the release of an extinguishing agent, malfunction of safety equipments due to the extinguishing agent used, criticality accident caused by a mechanical or moderating effect, etc.) and the mechanical effects induced (behaviour of confinement barriers, etc.).

Fire in the presence of radioactive materials

In the presence of radioactive materials, a fire can scatter the materials, thereby producing a situation in which the workers' exposure can't be controlled and even a release of radioactive materials into the environment. A fire can also trigger a criticality accident by damaging the measures and systems used to control the criticality units. Furthermore, specific measures are necessary to deal with the effluents produced by the extinguishing systems or the fire fighting, which could constitute the source of contamination (dispersion of radioactive materials) or the source of criticality accidents.

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[61] Institut de Radioprotection et de Sûreté Nucléaire – French TSO

[62] The elements needed to perform a safety function or to protect against the effects of a fire. Targets can be diverse in nature: radioactive materials, radioactive materials containment systems, criticality units, material and human resources that play a role in the safety functions, escape routes and access routes to equipments which have to be operated to make and keep the installation safe.

Structures that accommodate or support safety targets are to be protected against the fire. Equipment and structures, different from those mentioned above, which due to deterioration in the form of a domino effect caused by a fire could affect the safety of the installation are also to be protected against the fire and its effects.

Fire in the presence of safety equipment

The fire and the extinguishing systems used can significantly damage safety equipments. Even at a distance from the fire and in different ways, the corrosive material and the soot contained in the smoke may produce equipment malfunctions which differ from those caused by the simple thermal effects of fire. Because of toxicity and visibility effects for employees smoke may jeopardize actions to tackle the fire and to put the installation in safe conditions.

Structural stability in case of fire

In nuclear installations, the structural stability of buildings containing targets shall be guaranteed to lead and maintain the facility in safe conditions. This stability is requested for a fire occurring inside or outside of these structures. Consideration also has to be given to the consequences of any interactions between buildings caused by a fire that grows up in a contiguous building.

To demonstrate that the structural stability in case of fire is sufficient, the actual stress (temperature and, pressure fields) that these structures are likely to experience, including the cooling phase after extinction, and the structures' behavioural requirements during a fire have to be known.

The analysis process defines the fire scenarios, leading to the worst effects on the structures.

Hazard analysis process for a fire in a nuclear installation

According to the defence in depth, the safety of a nuclear installation in case of fire shall be demonstrated for all of the operating states, including shutdown states, with the operating ranges being retained for each operating state. The measures retained to meet the safety objectives have to be described. However, it is the measures' adequacy that has to be justified in particular.

The analysis process shall be based on the verification of the fulfilment of the technical performance levels which are justified through a fire and safety hazard analysis. The operator has to define the safety functions to be protected, the associated functional requirements, the technical performance levels of fire protection measures (FPMs) retained and demonstrate the adequacy of these performance levels in relation to the needs and how they are assured by the design adopted. Justification of the measures therefore concerns compliance with technical performance criteria. Calculation of the potential radiological consequences of a fire will only be carried out in a verification step of the safety demonstration.

Principles to be met

Aims of the fire protection measures

The control of hazards linked to an event such as a fire in a nuclear installation requires the examination of both plausible fires and the targets to be protected as part of nuclear safety.

The aim of the protection measures implemented on the basis of this examination is to:

prevent fires and limit their number, spread and duration,

maintain functional safety requirements,

limit the radiological consequences of the fire.

Defence in depth applied to fire protection

To fulfil the aforementioned aims, the fire protection is defined and designed according to the defence in depth principle. These measures are therefore implemented and organized in successive levels that are as independent as possible. Each level of defence against the fire shall prevent the situation from deteriorating and moving to the next level and limit the consequences of the failure of the previous level.

Applied to fire hazards, the levels of defence can be defined as:

- Preventing fires from starting;
- Detecting and extinguishing quickly those fires which do start, thus limiting the damage;
- Preventing the spread of those fires which have not been extinguished, thus minimizing their effects on the installation's safety and their consequences.

It may also be necessary to have other levels of defence including last measures inside the installation and the protection of the population in the event of a radioactive materials release. These last levels are generally specified in the installation's internal emergency plan and the corresponding external emergency plan. These levels are not covered in this document because fire is not the only initiating event to be considered when defining the corresponding measures.

Fire protection means shall be designed and dimensioned to meet the aims of the aforementioned defence in depth levels as well as possible.

Consideration of a combination of events

A combination of events is the occurrence that several events are able to affect the same installation in the same period of time. If there is no link between these events, they are independent events. Otherwise, depending on the strength of the correlation, the dependence of the events is potentially proven.

This section looks at the combination of fire with other events, to be considered when designing and dimensioning fire protection measures (FPMs). These events may be internal events caused by the failure of equipment involved in a safety function or caused by internal or external hazards.

As a general rule, the combinations shall be explicitly considered whenever there is a proven dependency and no design solution can rule out such a dependency. Any absence of dependency shall be justified. Combinations of events not selected shall be specified and their exclusion shall be justified with regard to their frequency and consequences.

Combinations with a potential dependency shall be examined to determine whether they should be considered. The following situations are to be examined in particular: lightning and fire, airplane crash and fire, explosion and fire, earthquake and fire.

Furthermore, an independent fire is to be considered:

- in conjunction with each event with a high frequency that is likely to affect the fire protection measures (freezing, loss of external power supply, etc.),
- after an event that undermines the safety of the installation over a sustained period without any - compensatory measures.

For all of the combinations of events considered, all of the direct and indirect effects brought about by the initial event are to be studied. Therefore, the effects of these events on the fire protection measures and the associated back-up elements as well as the possible intervention of the external emergency services shall be assessed. If necessary, these fire protection measures will have to be protected against the associated hazards and qualified on the basis of the specific conditions induced.

Margins and uncertainties

The hazard analysis process needs to assess the different effects of fire and to compare them with the performance criteria while taking into account the failure conditions of the targets to be protected.

The modelling of the fire scenarios and the evaluation of the scenarios' effects comprise uncertainties that are linked to the input data used to model the scenarios, the values associated with the data, the tools used to determine the effects, the models implemented, etc. These uncertainties shall be assessed and taken into consideration. Furthermore, the margins considered when defining the performance criteria of the fire protection measures shall cover the variability of the situations examined in the installation's safety analysis, which are finally represented by a small number of studied scenarios (“reference” fire scenarios).

The margins selected and the parameters used to determine them shall be presented and justified in the analysis.

Fire hazard analysis

The fire hazard analysis shall prove that the performance level of the FPMs meets the safety objectives and prove the robustness of the operator's safety demonstration. For the "reference" fire scenarios, this means that the operator will have to show that:

- the fire protection measures (defined according to the defence in depth process) are adapted to the fire hazards,
- their global design ensure, despite the failure of one of these measures, that the consequences for the installation's safety are controlled and the consequences for individuals and the environment remain acceptable.

The performance level of the FPMs is defined according to specific conditions. The hazard analysis shall be updated in case of modification of the installation or safety reassessments required by regulations.

The analysis shall reveal the key parameters of the safety demonstration. Then, minor modifications and those likely to significantly undermine the safety demonstration's conclusions can be easily pointed out.

As part of the final verification of the design, the operator shall show that consequences would remain acceptable, even in case of fire in a given room despite the FPMs taken.

Demonstration of the sufficiency of the fire protection measures

Elements required for the analysis

This chapter lists the elements which shall be known in addition to the description of the installation's characteristics (dimensions, organization of rooms, lay-out, procedures, etc.) for the fire hazard analysis.

Functions to be safeguarded and associated functional requirements

The operator identifies the safety functions and the associated support functions to be maintained or to be re-established within relatively short period in case of fire. The operator associates the functional requirements needed to ensure that the corresponding systems and components work correctly during the various operating states.

Targets to be protected against the fire and its effects

The operator identifies the targets to be protected so that the functions defined previously can be safeguarded in case of fire. These targets include, in particular: radioactive materials, material confinement systems, criticality units, SSCs (structures, systems and components) important for safety, material and human resources that play a role in the safety functions to be safeguarded in the event of a fire, escape routes and access routes to equipments which have to be operated to make and keep the installation safe.

Structures that accommodate or support safety targets are to be protected against the fire. Equipment and structures, different from those mentioned above, which could affect the safety of the installation due to domino effects are also to be protected against the fire and its effects.

Performance criteria to be met

A nuclear installation operator shall demonstrate by a fire safety analysis that the functional safety requirements are met thanks to the fire protection measures retained. Considering the process, the effectiveness of these measures has to be compared with quantitative performance criteria. These technical criteria may be threshold values based on, for example, data on the failure of equipment (temperature and thermal flux values that trigger a malfunction, soot particle or toxic particle concentration levels, etc.); whenever possible, they shall include a margin in relation to the experimental or theoretical data.

These performance criteria vary according to the goal of the operator's demonstration. For example, if the operator seeks to demonstrate the resistance of the last stage filter, the performance criteria can be defined in relation to the failure criteria of the filtering system during a fire (i.e. the maximum temperature values and the difference in pressure at the filter terminals).

Furthermore, specific attention shall be given to the fire stability requirement for structures that accommodate or support safety targets. Attention shall also be given to the associated performance criteria, because this functional requirement generally conditions compliance with all of the others.

Fire hazards

The fire hazards within the installation which are likely to impact on the targets have to be identified for all of the installation's operating states (normal, maintenance, shutdown states…). These fire hazards are linked to the products and materials used and, also to the installation's equipment and operating conditions.

The fire hazards outside of the installation, which form part of the industrial, human activities or natural environment, such as lightning, external road or railroad hazards, etc. shall also be identified in order to define the fire protection measures needed to control external fire hazards.

Fire protection measures retained

The operator indicates the reliable FPMs for its safety demonstration. The operator justifies the fire protection measures' ability to fulfil their functions (fire detection, heat insulation, smoke-tightness, access and extinction of the fire in a room, etc.). The justification of these measures can be based on the fact that they comply with different reference documents or standards, as long as the associated conditions and qualification criteria are adapted to the situation.

With regard to the response to a fire, whether it relies on human actions or technical instruments, the response time to be used for the safety demonstration is the sum of all of the amounts of time needed for the effective implementation of the intervention means.

"Reference" fire scenarios

The selection of "reference" fire scenarios is an important part of the safety demonstration. These scenarios justify the suitability and the sufficiency of the fire protection measures retained, considering the fire hazards, by comparing the performance levels of these fire protection measures with the performance criteria.

In practice, the number of fire scenarios that could arise in an installation can be high. Nevertheless, it is usually necessary to reduce the possible fire scenarios to a manageable number of credible “reference fire scenarios”

Two stages are therefore necessary:

identification of the fire scenarios,

selection of "reference" fire scenarios for the design of the fire protection measures.

Definition of fire scenarios

The definition of deterministic fire scenarios is carried out by room or group of rooms. Conservative assumptions are to be retained with regard to the parameters used in the scenarios' development (ventilation flow rates, diagnosis and response times, etc.). As part of a deterministic approach, the outbreak of a fire will always be considered.

Selection of "reference" fire scenarios

The fire scenarios thereby identified may be grouped in accordance with their specific characteristics and similarities as long as the fire hazards are of the same nature. The rooms or groups of rooms concerned are covered by the same type of fire protection measures (similar nature and performance levels) and their fire effects are similar.

For each fire scenario group, one or more representative scenarios are retained if they are likely to have the most harmful direct or indirect effects on the targets. These scenarios will be used to check the design of the fire protection measures. They are known as "reference" fire scenarios. Each reference scenario shall be chosen to ensure that the fire protection measures can also ensure compliance with these objectives for all of the other scenarios of the group.

The conclusions produced following the examination of each reference scenario apply to all of the rooms or groups of rooms covered by a same reference scenario.

Quantification of the effects of "reference" fire scenarios

For each "reference" fire scenario, a quantitative assessment of the characteristic factors of the fire is necessary to assess the effects the fire can have on targets and the effectiveness of the fire protection measures.

The methods and tools used for this quantification process shall be adapted to the scenarios and the parameters studied. The parameters and groups of hypotheses retained shall be reasonably inclusive.

Selection of methods and tools

When the assessment of the characteristic factors under investigation is based on a numerical tool or an analytical calculation method, it is important to show that the tool selected matches the degree of complexity of the phenomenon studied. The accuracy, the physical factors to be characterised and the performance criteria are also to consider. It is important to demonstrate that these tools have been validated and used in their relevant area.

If quantification is carried out on the basis of experimental results, it is important to make sure that the experimental results were obtained under conditions that are sufficiently representative of the scenarios. The test results shall be analysed to make sure that the conclusions drawn apply to the cases considered.

In certain specific cases (for example, if there is no adapted calculation method or experimental data), the opinion of experts can be sought. However, a prudent approach should be retained. In any case, resorting to an expert's opinion shall be clearly mentioned and justified.

Characteristic factors investigated

The characteristic factors to be quantified vary with the "reference" fire scenario(s) retained and the requirements. Aside from the temperature reached in the room in which the fire started and the duration of the fire, factors such as pressure, thermal flux values received by the targets, the quantities of soot and unburned materials produced, the toxicity of the smoke, etc. and their associated uncertainties (inherent to the input data, the tool, etc.) may be factors to characterise.

Input data and groups of hypotheses

Regardless of the quantity of input data needed to forecast the characteristic factors of the fire under investigation, input data that may significantly affect the results. They shall be identified and their values justified (physical parameters, values of thresholds for automatic actions, criteria for manual actions, time required for manual action, etc.). The uncertainties associated with these values shall be appraised.

Verification of the performance level of the fire protection measures

The characteristic factors of fire effects in the "reference" scenarios shall therefore be compared with the criteria retained. Then:

- one or more criteria are not met: corrective measures shall be taken (new design of fire protection measures, additional measures, modification of initial project, etc.) and the demonstration shall be reconsidered,

- all of the criteria are met: justification of the performance level of the fire protection measures is provided for the reference scenarios; the robustness of the demonstration shall now be proven.

Verification of the robustness of the safety demonstration

The robustness of the safety demonstration, and therefore the sufficiency of the fire protection measures and their design, is proven on the basis of the fire scenario study whose effects could prove to be more harmful than the "reference" scenarios retained at the dimensioning stage; this involves:

- "aggravated" scenarios comprising a fire protection measure failure;

- one or more “maximum possible fire” scenarios.

Within the scope of checking whether the consequences for safety, people and the environment remain acceptable.

Aggravated scenarios comprising a fire protection measure failure

The failure of a fire protection measure can result in fire scenarios that are more harmful than those retained during the dimensioning stage. Consequently, the performance criteria of some of these measures may no longer be respected. This stage therefore consists of checking the robustness of the safety demonstration while making sure that the consequences remain acceptable despite the hypothetical failure of a fire protection measure.

The acceptability of the demonstration is assessed on a case by case basis, while taking the installation, its specific characteristics and its environment into consideration. To test the robustness of the demonstration, two approaches are possible:

- a deterministic approach,

- a probabilistic approach.

Deterministic approach

Using this approach and on the basis of the "reference" fire scenarios, it is important to determine the plausible failures of the fire protection measures. Then these failures, considered separately, shall not allow the development of a fire whose effects would result in unacceptable consequences. It is important to recall that in certain cases, the lack of effectiveness of one fire protection measure can impact on the effectiveness of one or more of the other fire protection measures. These cases shall be clearly identified and considered in the demonstration.

The failure can concern material that may belong to an active system, such as the automatic fire detection (failure of a sensor, for example) or a extinguishing system (failure of a valve, for example), or a passive system, such as fire area elements (doors, fire dampers, in particular).

Failure may be the result of human actions, such as the failure of an action or diagnosis or a delayed intervention (incorrect diagnosis by an operator, slow response and slow implementation of equipment by the emergency team).

If the operator provides proof of the robustness of certain fire protection measures, their failure may be ruled out. The operator shall, nevertheless, provide proof that the level of performance of the measures concerned and the measures' functional character are maintained during the fire scenario conditions retained and for the period required.

Probabilistic approach

This approach allows situations comprising complex events and an accumulation of events to be studied. In particular, situations in which redundant systems are lost and situations involving the occurrence of an external or internal hazard such as a fire. The hazard in terms of the probability of occurrence of the undesirable event is assessed. It includes both failures of a material and of a human or organizational action.

The event tree method is commonly used to represent fire scenarios. It shows how each scenario will develop, determine the events to be studied, assess the influence of measures (fire protection measures, systems and support systems, behavioural procedures, etc.) and consider time-related and functional dependencies between events.

The addition of the values of the frequencies calculated for each "branch" of the event tree that leads to the undesirable event gives then the total frequency of the undesirable event for the "reference" scenario. The frequency associated with the reference scenario is then applied to all of the scenarios covered by the reference scenario in order to exhaustively assess the hazard associated with this "group" of scenarios.

To assess the robustness of the demonstration, the following factors are cross-referenced:

the total frequency associated with the group of scenarios considered,

the contribution of the failure of each fire protection measure to the total frequency,

the corresponding level of consequences.

The approach adopted to create the trees and the input data retained shall be presented and justified as the robustness of the results of the probabilistic approach chiefly depends on the quality of the input data.

Worst case fire scenarios

In addition to check that a fully developed fire in one room or a group of rooms cannot result in unacceptable consequences for safety, people and the environment, this stage assesses the robustness of fire compartmentation.

The rooms or groups of rooms concerned by this stage are those that accommodate mobile radioactive material which contains - or is likely to contain for a transient period - permanent combustible loads.

The spreading of a fire to all of the loads is to be considered separately from any consideration of the quantity of air available (air tightness or possible ventilation) and fire extinguishing systems that may be present.

The boundaries of the rooms or groups of rooms to be retained for the maximum possible fire study are those for which a fire resistance and a radioactive material confinement capacity are justified. Where a facility is not subdivided by these areas, the maximum area should be defined by the exterior walls and roof of the facility.

Assessment of consequences for safety, people and the environment

The consequences of a fire are to be assessed by considering:

the functional damage brought about by equipment failures,

the radiological impact.

The failure of equipment important for safety or the loss of back-up systems needed for the functioning of such equipment due to the effects of a fire shall induce the operator to carry out a functional analysis in order to check that the safety functions required in case of fire remain intact.

If the time needed to re-establish a function is below the lead times needed to recover and maintain the installation in a safe state and if there is a possibility of a functional redundancy, the safety demonstration is acceptable.

The assessment of the radiological consequences of a fire combined with the dissemination of radioactive materials or irradiation exposure concerns both workers, members of the public, rescue team and the environment.

The quantification of the effects of the fire scenarios retained shows the quantities of radioactive materials that could be involved. The fractions of these materials which are airborne are estimated while taking into account the nature of the radionuclides present, as well as their physicochemical form and volatile character. For each radionuclide or each group of radionuclides, the retained airborne release fraction shall be justified. If results from experiments are used, it shall be guaranteed that the experiment conditions sufficiently represent the case considered.

The various modes of transfer and deposition mechanisms in the buildings and ventilation systems are to be considered with specific attention to leakages into the environment.

If the radiological consequences thereby assessed are considered to be tolerable, the safety demonstration in the event of a fire is acceptable.

IRSN
INSTITUT DE RADIOPROTECTION ET DE SÛRETÉ NUCLÉAIRE

CSNI Workshop

Toronto, Canada, 27-29 september 2011

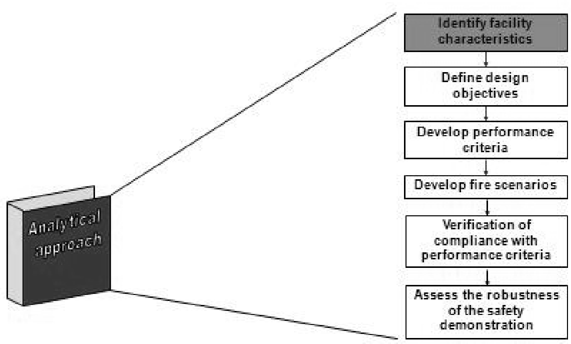
The IRSN approach on fire safety analysis of nuclear facilities

Yannick ORMIERES, Jocelyne LACOUE

Principles to include in a safety demonstration

- Defense in depth principles
- Use appropriate tools
- Take into account margins and uncertainties…

Fire safety analysis for nuclear facilities IRSN 2



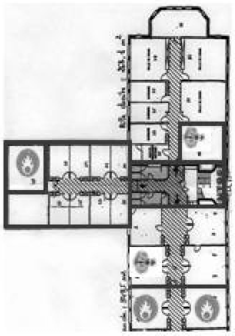
Fire safety analysis for nuclear facilities IRSN 3

Identify facility characteristics

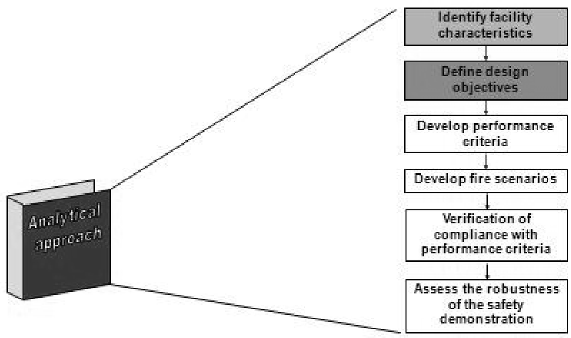
Basic characteristics
- Type of construction
- Number of floors
- Processes
- etc.

Identify fire loads
- Electrical rooms
- Flammable liquids
- Transient fire loads
- etc.

Fire protection measures
- Fire detection systems
- Sprinklers
- Fire compartments/walls
- etc.



Fire safety analysis for nuclear facilities IRSN 4



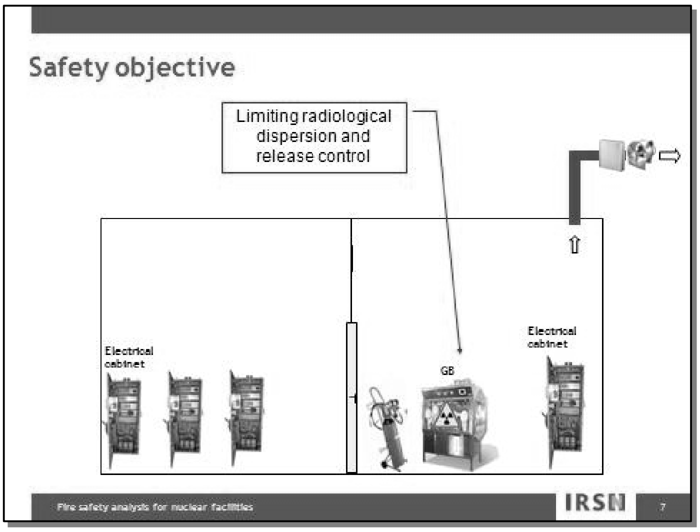
Fire safety analysis for nuclear facilities IRSN 5

Define safety targets in case of fire

Identify objectives
- Nuclear safety
- Radiation release
- Life safety
- Interruption

Determine « targets » to be protected from the fire to achieve these objectives:
- Radiological materials
  - nature and physical status (gas, liquid, solid)
  - quantity
- Structures, systems and components important for safety (SSCs)
  - identification of equipment taking part in safety functions
  - redundancy identification
- Personnel
- etc.

Fire safety analysis for nuclear facilities IRSN 6


Safety objective
Limiting radiological dispersion and release control
Electrical cabinet
GB
Electrical cabinet
Fire safety analysis for nuclear facilities
IRSN
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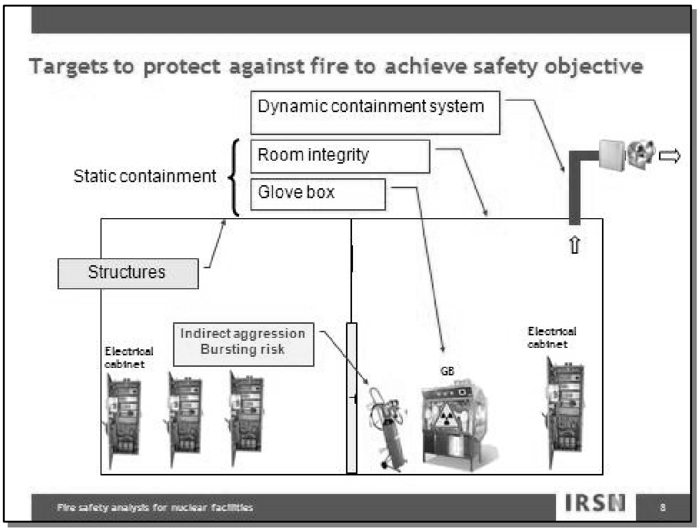
Targets to protect against fire to achieve safety objective
Dynamic containment system
Room integrity
Static containment
Glove box
Structures
Indirect aggression
Bursting risk
Electrical cabinet
GB
Electrical cabinet
Fire safety analysis for nuclear facilities
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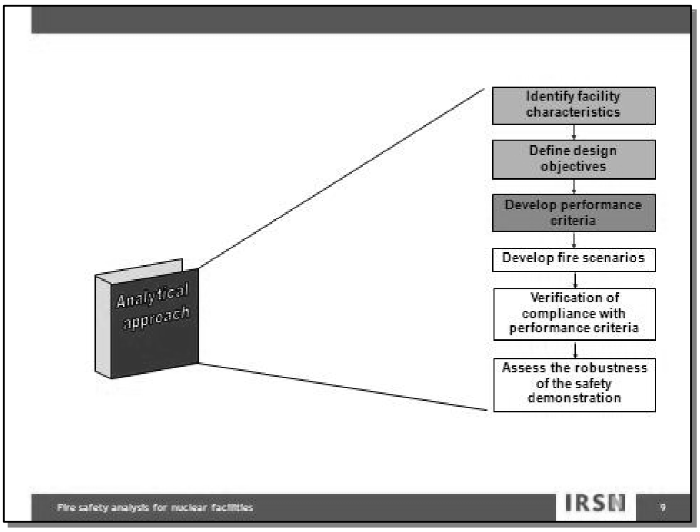
Identify facility characteristics
Define design objectives
Develop performance criteria
Develop fire scenarios
Verification of compliance with performance criteria
Assess the robustness of the safety demonstration
Analytical approach
Fire safety analysis for nuclear facilities
IRSN
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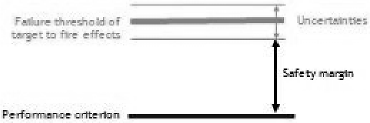
Performance criteria

- **To adapt fire protection measures to the vulnerability of the safety targets**

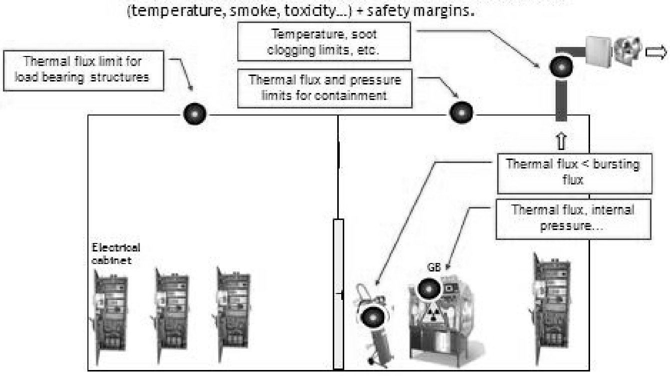
Performance criteria

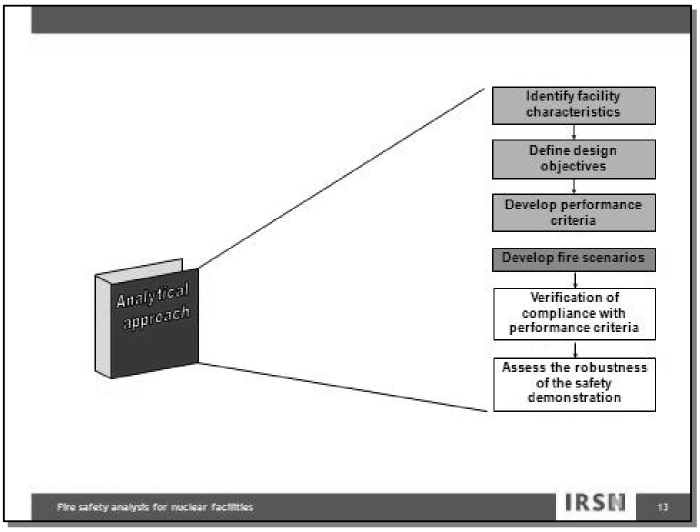
- These criteria will be used to evaluate the designs proposed by the operator.
- Definition of performance criteria is based on the vulnerability of targets to the effects of fire.
  For example, performance criteria might include values for thermal radiation exposure (kW/m²) or air temperature.
- These criteria are determined with margins.



Performance criteria

- Performance criteria depend on the sensitivity of targets to fire (temperature, smoke, toxicity...) + safety margins.





Identification of possible fire scenarios

Ignition is postulated

- A fire is by definition an accidental event that occurs only in abnormal operating conditions. It makes no sense to say that nothing can burn because it refers only to normal operation.
- The feedback shows that the risk of a fire is significant

Consider operating conditions of the facility

- Normal operating conditions
- Reduced power operation
- Scheduled maintenance or shutdown
- etc.

Take into account combined events with fire

Fire safety analysis for nuclear facilities IRSN 14

Identification of possible fire scenarios

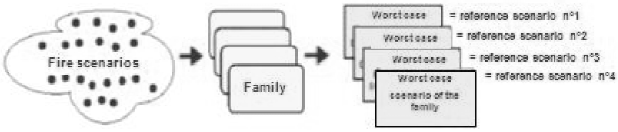
Combined events

- Occurrence of events that affect an installation in the same time interval. If there are no links between these events, they are considered as independent :
  - Fire and dependent events
    - » Earthquake and fire,
    - » Explosion and fire...
  - Fire and INdependent events is postulated
    - in conjunction with each event with a high frequency rate that is likely to affect fire protection measures:
      - » winter conditions (freeze, snow...)
      - » Loss of offsite power...
    - after an event downgrading the long term safety installation without compensatory provisions

Fire safety analysis for nuclear facilities IRSN 15

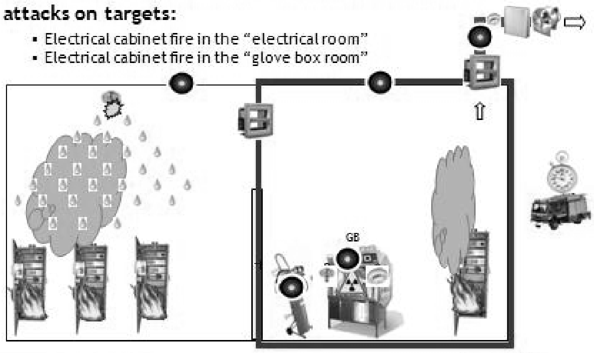
Selection of "reference fire scenarios"

- It is usally necessary to reduce the possible fire scenarios to a manageable number of credible "reference fire scenarios".
  - "Reference fire scenarios" are a subset of the possible fire scenarios



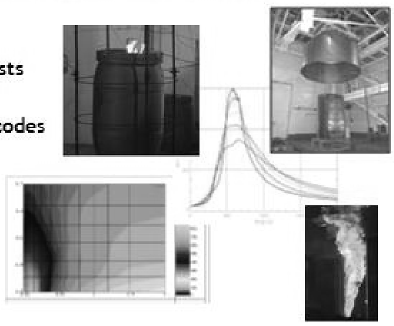
Reference fire scenarios

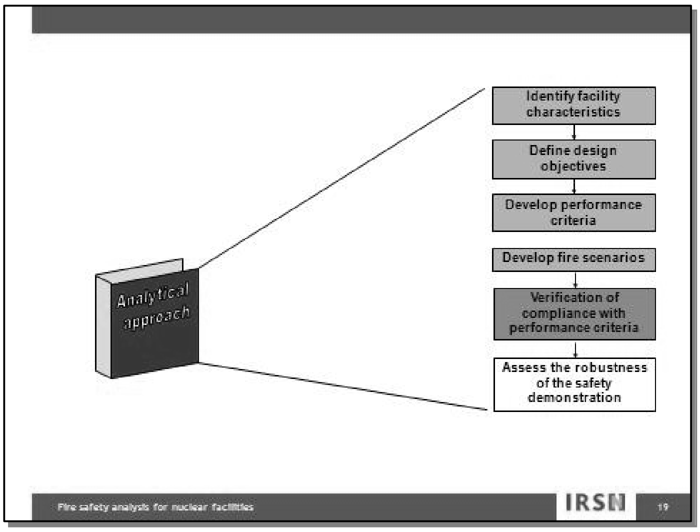
- these scenarios were selected because they are representative of all scenarios identified in terms of attacks on targets:
  - Electrical cabinet fire in the "electrical room"
  - Electrical cabinet fire in the "glove box room"



Estimate "reference fire scenario" effects

- Result of real fire tests
- Result of computer codes
  - Zone models
  - CFD models
  - ...
- Expert advice
- Feedback




Analytical approach
Identify facility characteristics
Define design objectives
Develop performance criteria
Develop fire scenarios
Verification of compliance with performance criteria
Assess the robustness of the safety demonstration
Fire safety analysis for nuclear facilities
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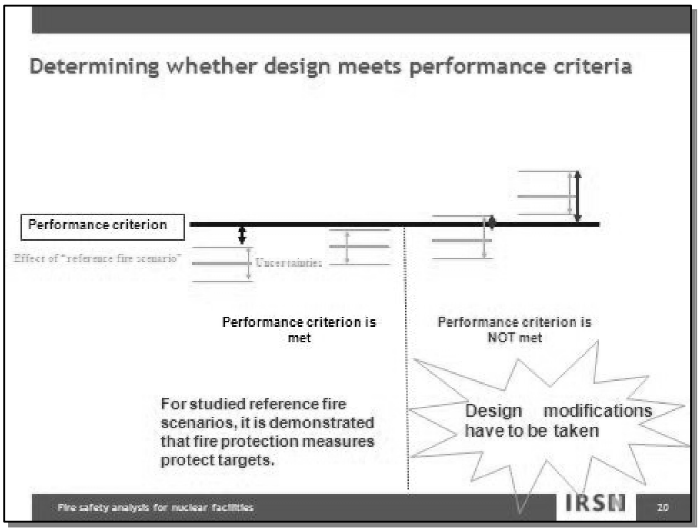
Determining whether design meets performance criteria
Performance criterion
Effect of "reference fire scenario"
Uncertainties
Performance criterion is met
Performance criterion is NOT met
For studied reference fire scenarios, it is demonstrated that fire protection measures protect targets.
Design modifications have to be taken
Fire safety analysis for nuclear facilities
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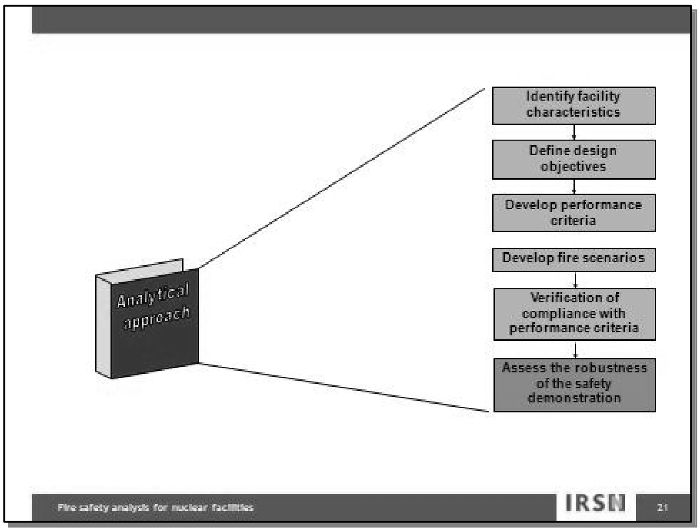
Analytical approach
Identify facility characteristics
Define design objectives
Develop performance criteria
Develop fire scenarios
Verification of compliance with performance criteria
Assess the robustness of the safety demonstration
Fire safety analysis for nuclear facilities
IRSN
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Assess the robustness of the safety demonstration

- The failure of fire protection measures can lead to fire scenarios more severe than design fire scenarios.
  - So, performance criteria can not be respected.

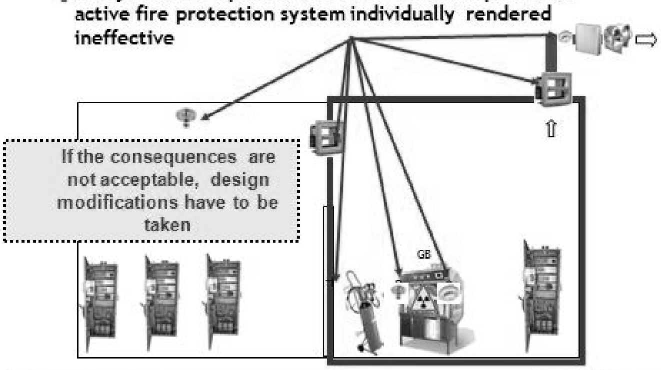
    "When properly implemented, defence in depth ensures that no single [...] failure could lead to harmful effects, and that the combinations of failures that could give rise to significant harmful effects are of very low probability" [IAEA SF-1]

- This stage consists in checking the robustness of the safety demonstration by making sure that the consequences for safety remain acceptable in spite of:
  - 1) Failure of a fire protection measure
    - Deterministic approach
    - Probabilistic approach
  - 2) "Worst case fire scenarios" = The spreading of a fire to all of the loads is to be considered separately from any consideration of the quantity of air available and fire extinguishing systems that may be present.

Fire safety analysis for nuclear facilities IRSN 22

1) Assess the robustness of the safety demonstration

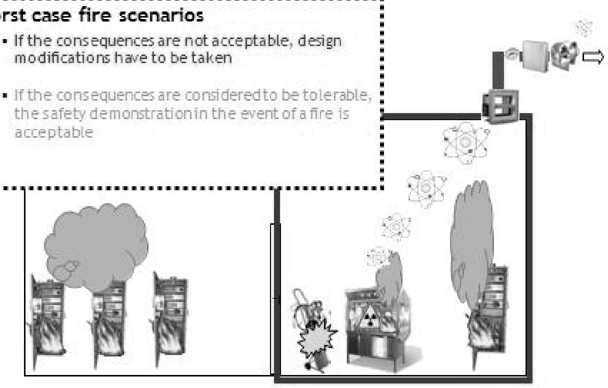
- Study the consequences of a fire with each passive or active fire protection system individually rendered ineffective



Fire safety analysis for nuclear facilities IRSN 23

2) Assess the robustness of the safety demonstration

- Worst case fire scenarios
  - If the consequences are not acceptable, design modifications have to be taken
  - If the consequences are considered to be tolerable, the safety demonstration in the event of a fire is acceptable



Fire safety analysis for nuclear facilities IRSN 24

RISK-INFORMING SAFETY REVIEWS FOR NON-REACTOR NUCLEAR FACILITIES: AN EXAMPLE APPLICATION

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Abstract - This paper describes a methodology used to model potential accidents in fuel cycle facilities that employ chemical processes to separate and purify nuclear materials. The methodology is illustrated with an example that uses event and fault trees to estimate the frequency of a specific energetic reaction that can occur in nuclear material processing facilities. The methodology used probabilistic risk assessment (PRA)-related tools as well as information about the chemical reaction characteristics, information on plant design and operational features, and generic data about component failure rates and human error rates. The accident frequency estimates for the specific reaction can be useful to help to risk-inform a safety review process and assess compliance with regulatory requirements.

1. Introduction

U.S. Department of Energy (DOE) has recently been carrying out research on chemical separation technologies related to closed nuclear fuel cycle architectures[63]. Part of this research is related to solvent extraction processes, e.g., PUREX or modified PUREX processes[64] used in fuel reprocessing plants to remove impurities from the feed that may consist of reactor spent fuel or other fissile material. The extraction operation generally employs an organic solvent, usually tributyl phosphate (TBP), diluted in an organic matrix, as an extractant along with concentrated nitric acid in various processes. In this process, one or more components that are present in the solution, e.g., uranium and/or plutonium as well as metal impurities, are transferred between two immiscible liquid phases,

[63] U.S. Department of Energy, Nuclear Energy Research and Development Roadmap, Report to Congress, April 2010, DOE/NE.

[64] M. Benedict, T.H. Pigford, and H.W. Levi, "Nuclear Chemical Engineering," Second Edition, McGraw-Hill Book Company, New York, 1981.

typically an organic phase and an acidic aqueous phase. One concern in the facilities that utilize this process is the occurrence of an explosive, runaway nitration oxidation reaction (NOR) that can occur when the organic solvent TBP, and its degradation products, comes in contact with concentrated nitric acid at elevated temperatures. Such events have occurred before, in the U.S. and other countries, in facilities that employ solvent extraction. These reactions occur continuously over a wide temperature range but the reaction rates and the heat and gases generated at temperatures below about 60 ^{0}C are low and passive heat removal and normal venting are adequate. At higher temperatures (about 80 ^{0}C and higher), facility-specific heat removal measures are needed along with actions to ensure that the amount of TBP that can enter heated acid-bearing vessels is limited. A recent report issued by the Defense Nuclear Facilities Safety Board (DNFSB)[65] summarizes the events that have occurred in the extraction operations at Savannah River and Hanford in the U.S. and the accident at the Tomsk plant in Russia and states that “maintaining a temperature of less than 130°C is generally accepted as a means to prevent” any explosive reactions.

2. Safety Strategy

Considering that the undesired reaction occurs if the organic solvent comes into contact with concentrated nitric acid at an elevated temperature, the safety strategy and approach for coping with the possibility of NORs is as follows:

1. Segregation of separate phase solvent (TBP) from acid bearing and heated process equipment such as evaporators; this is meant to ensure that a separate phase of TBP or TBP in excess of its solubility limit that could be entrained with the aqueous phase does not come into prolonged contact with highly concentrated nitric acid at elevated temperature. This strategy is usually implemented through process sampling and density monitoring and control, and may include a passive engineered system to allow for the separation of organic and aqueous phases based on their density difference. The equipment and procedures that would credited for this strategy may include sampling points and procedures, process density control loops and monitors, and a passive system to separate organic and aqueous phases.

2. Heat transfer strategy; this relies on passive convective and radiative heat transfer mechanisms to the surrounding environment. The strategy should demonstrate an adequate heat transfer to the room environment of heat that may be generated from all possible sources including the exothermic reactions such as the solvent nitric acid reaction (at relatively low temperatures). The temperature of the surrounding environment needs to be controlled to ensure adequate heat transfer during routine and pre-defined upset conditions. The equipment and procedures credited may include: the geometry of process vessels, temperature sensors and control loops to detect and limit self-heating, off-gas venting to relieve pressure from any gases evolved in the reactions, and reagent sampling controls to ensure that the proper diluent is used.

3. Evaporative cooling strategy; this provides for heat removal via evaporation of water in the aqueous phase in heated process vessels where some (limited) amount of TBP is expected to be present, and where the possibility of the exothermic nitration oxidation reaction exists. This strategy depends on the large latent heat of vaporization associated with the aqueous phase, and it also requires the fulfillment of certain criteria, such as maintaining a minimum aqueous to TBP ratio, a maximum TBP layer depth, a maximum process solution temperature and an open, vented system. The equipment and procedures credited for this strategy could be: process sampling and administrative flushing controls to limit the amount of TBP accumulation in undesired vessels or locations, level controls to maintain the minimum aqueous to TBP mass ratio, temperature

[65] U.S. Defense Nuclear Facilities Safety Board, DNFSB/TECH-33, November 2003.

controls to limit solution temperatures, and an offgas venting system of sufficient capacity to relieve pressure from any gases released in the reactions.

3. Risk Assessment Considerations and Limitations

A limited-scope probabilistic risk assessment (PRA) model can be used to evaluate the failure of some of the safety strategies due to internally initiated process deviations and assess their contribution to the facility risk posed by the nitration oxidation reaction (NOR). Such a model was recently used to study the risk posed by a NOR for a facility that would be based on a modified PUREX process. In particular, the PRA model focused on (1) the failure of evaporative cooling in selected process vessels, and (2) the failure of the TBP segregation strategy, through such events as emulsification, and the formation of a third phase or a rag layer, leading, eventually, to a violation of the success criteria for evaporative cooling. The PRA can be considered a limited-scope risk assessment for several reasons:

1. The generic risks due to external hazards, such as seismic events or loss of offsite power events, including loss of alternate source of alternating current, were excluded from the analysis. These initiating events can potentially lead to other high consequence outcomes, similar to NORs, and would have greatly enlarged the scope of the study.

2. Failures of the heat transfer strategy were not considered in the analysis. This strategy applies to the adequacy of passive heat transfer to the room environment from process vessels containing solutions at lower temperatures (about 55^{o}C and below) and depends for its success on the availability of room cooling, i.e., the proper operation of the facility's HVAC system. Consideration of the failures of the HVAC system, however, would have greatly enlarged the scope of the analysis.

3. A semi-empirical model for the TBP-nitrate reactions based on thermal decomposition alone, used to set the success criteria for the evaporative cooling safety strategy, was accepted as the basis for further evaluation of the phenomenon.

The end state of the quantitative analysis was considered to be the annual probability of transport of a separate organic phase into process equipment with conditions that could promote NOR. Consequence of NOR event was assumed to be high for the facility worker where safety controls should be in place to reduce the likelihood of occurrence of the event to highly unlikely. A consequence analysis of the events was not performed.

4. Qualitative Risk Assessment

A qualitative assessment was performed to screen and prioritize resources to processes of highest risk. An evaluation of the factors that may contribute to the possibility of NOR in the various process units was first carried out to determine in which process units organics and nitric acid either contact each other during normal operation or have the potential to come into contact and where there is somewhat higher risk of a NOR occurring. Each of the process units was evaluated for the possibility of a NOR in terms of the equipment employed, the sequence of operations, and the conditions (temperature, pressure, etc.) under which the operations occur. Based on this assessment and taking into account the heat sources present, the heat balance and the potential for TBP transfer, several vessels in two process units were selected for more detailed evaluation. For each of the vessels selected, a qualitative safety review was performed followed by a quantitative risk assessment of NOR. The qualitative review is summarized first followed by a summary of the quantitative risk assessment.

Two of the vessels selected are evaporators: the first one is a natural recirculation thermosiphon type boiler which utilizes pressurized super heated water as a heating fluid. The first evaporator operates under vacuum. The normal process temperature is below 66^{o}C and the normal super heated hot water temperature is 105 ^{0}C. The hot water system (HWS) temperature is equipped with controls to ensure a maximum temperature of 122^{o}C is not violated. The mitigation strategy applied to the first evaporator is evaporative cooling. Two conditions are necessary for a viable NOR scenario to occur in this vessel: (1) a rising process temperature above 80^{o}C; this can be due to an inability to maintain the hot water system temperature below 122^{o}C or the occurrence of a heat exchanger tube rupture, and (2) failure of evaporative cooling to successfully mitigate the event. The success criteria for evaporative cooling involve maintenance of a minimum aqueous phase to TBP ratio, a maximum TBP layer depth, a maximum process solution temperature, and an open adequately vented system. The conditions under which these criteria could be violated include equipment failures (loss of temperature control, heat exchanger tube ruptures, venting system failure), human failures (operator failure to flush the system on schedule as required), and process failures (formation of emulsions, or a third phase or rag layer).

Another vessel selected for evaluation is a collection tank for concentrates drawn off from the evaporator. The tank is cooled by a cooling water loop, and is maintained in a well-mixed condition by an air sparger to prevent the formation of any hot spots within the tank, which operates normally at a temperature around 40 ^{0}C. If the temperature reaches a set point of 80 °C, steam jets will be shut off, and the solution volume is verified and maintained at a level to ensure that the evaporative cooling would be successful. The safety strategy for the concentrates collection tank is also evaporative cooling. Flushing of the tank contents is performed once every six months to ensure that any accumulation of TBP is limited to an amount that is within the criteria for successful evaporative cooling. Semi-annual flushing ensures that the amount of TBP is limited in the tank. Two conditions are necessary for a NOR scenario to occur: (1) a rising tank temperature above 80 ^{0}C due to failure or degradation of the tank cooling/mixing system and (2) failure of evaporative cooling. An assessment of the conditions under which the success criteria for evaporative cooling in the tank could be violated include equipment failures of the tank cooler and/or sparger, human failures (flush tank contents every six months), and venting system failures.

The second evaporator selected for more detailed evaluation is also a natural circulation thermo-siphon evaporator which concentrates liquors, supplied from a feeding tank. The evaporator includes a boiler used for evaporation of the feed solution and reflux from a rectification column. It has a tubular heat exchanger. The heating fluid (steam) occupies the shell side and the mother liquor to be evaporated circulates in the tubes. The conditions for a NOR is this vessel readily exist only if sufficient TBP is present. Hence, TBP prevention is the main safety strategy applied to this evaporator. The amount of TBP that enters the evaporator from the feeding tank is controlled below its solution detection limit. This small amount of TBP will be fully and safely reacted in the aggressive environment that exists in this evaporator. The study conservatively assumed that the undesired reaction could occur if the soluble TBP amount is not controlled or if a separated phase of TBP is transferred to the evaporator. These could happen either through a slow accumulation of mechanically entrained droplets that could eventually create a separate phase of TBP or a severe process malfunction leading to a transfer of a relatively large amount of solvent from other process units. Both ways of TBP transfer involve the circumvention of multiple barriers, including the passive organic-aqueous phase separation unit and process sampling controls that ensure that the amount of soluble TBP passing through the unit downstream to the evaporator remains sufficiently low. Operational failures in the units that could circumvent the barriers and allow TBP transfer to the evaporator were analyzed in the study.

Table1. **Summary of Qualitative Risk Assessment (above). Process vessels in the table were identified as higher risks and selected for further evaluation in the Quantitative Risk Assessment**

Process Vessel	Safety Strategy	Success Criteria
Evaporator: natural recirculation thermo-siphon type boiler	Evaporative Cooling	- Maintain hot water system temperature below 122°C - Maintain minimum aqueous phase to TBP ratio - Maintain maximum TBP layer depth - Maintain maximum process solution temperature - Adequately vented system
Collection Tank	Evaporative Cooling	- Maintain tank temperature below 80°C - Maintain minimum aqueous phase to TBP ratio - Maintain maximum TBP layer depth - Maintain maximum process solution temperature - Adequately vented system - Tank flush once every six months
Evaporator: natural circulation thermo-siphon evaporator	TBP Prevention	- Prevent transfer of TBP to evaporator

5. Quantitative Risk Assessment

Quantitative evaluation, using accident sequence delineation presented in the form of event trees and fault trees, was carried out to gain further insights into possible combinations of failures that could lead to NOR in the process vessels selected after the qualitative assessment. Quantification was carried out using the SAPHIRE code[66] to obtain the point frequency of a NOR and a 5th percentile and 95th percentile frequency to show the range of uncertainty.

The NOR scenario in the first evaporator is modeled under two conditions of TBP accumulation:

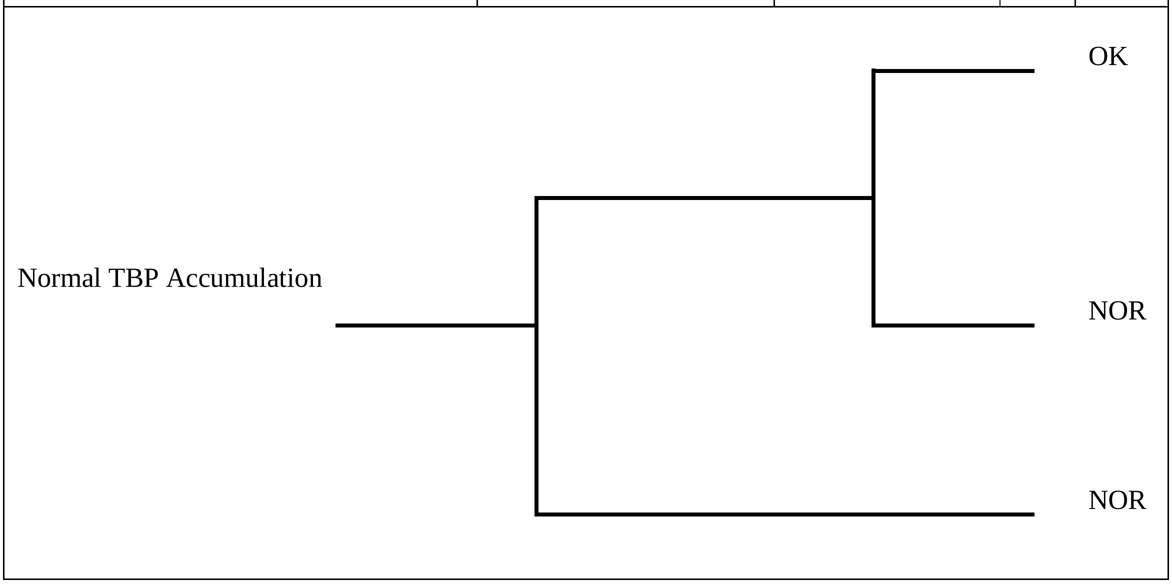
(1) normal accumulation of TBP, which refers to an accumulation of a small amount by mechanical entrainment with the aqueous phase, and (2) upset accumulation of TBP, which can occur due to a severe process malfunction such as formation of an emulsion that can transfer large quantities of solvent. Under the first condition, high solution temperature and failure of the evaporative cooling strategy is necessary for a NOR to occur. The initiating event for this scenario is the increase in solution temperature if the evaporative cooling strategy fails. This initiating event can happen due to a loss of temperature control or a heat exchanger tube rupture. The former is modeled via a standard fault tree model and the latter via generic data. The next top event in the event tree models the different ways by which the various success criteria for evaporative cooling, viz., maintaining the aqueous to TBP mass ratio and the TBP layer thickness, can be violated. The first can happen due to

[66] U.S. Nuclear Regulatory Commission, SAPHIRE Code Reference Manual, NUREG/CR-6952, September 2008.

operator failure to flush the vessel at the end of a six month period, which is conservatively assumed to cause an unavailability of evaporative cooling for six months until the next flushing action is required. This failure is modeled via a fault tree based on human error probability to carry out an action. The failure probability for the second criterion, maintaining TBP level, is estimated through standard fault tree methodology. The last top event in the tree represents the success of venting to ensure that the solution temperature is maintained below the azeotropic limit for the nitric acid/water solution. Venting is provided by a two-train system consisting of fans and HEPA filters with an additional fan as standby. Failure of venting is modeled via a fault tree to evaluate the venting failure probability. There are two NOR sequences for this scenario; in the first the level control is successful but venting fails, while in the second, the amount of TBP accumulated is sufficient to violate the criteria for evaporative cooling. The dominant cutset in the first sequence is common cause failure of plugging of two sets of HEPA filters. In the second sequence, the dominant cutset is the failure of the operator to carry out the six-month flush out of the vessel. Under the second condition, multiple failures of the barriers that prevent TBP transfer from upstream process units to the evaporator have to occur. The failure probabilities for these were assigned based on very limited data. Further barriers to the transfer of organics to the evaporator are provided by sampling and density controls. Failure of these controls was modeled via standard fault tree modeling. The initiating event for this scenario is again a loss of temperature control or a heat exchanger tube rupture that leads to a rise in solution temperature. The top events in the event trees relate to the success/failure of the various pulse columns in breaking up entrained organic material followed by the success/failure of the sampling and density controls. Venting is not modeled as the amount of TBP assumed to be transferred in the upset accumulation condition would violate the criteria for the success of evaporative cooling. The dominant cutsets in one sequence are the ineffectiveness of density controls, common cause failure of the density transmitter, failure of sampling analysis, failure of diluent wash column, and malfunction of the pulse extraction column. In the other sequence, the dominant cutsets are ineffectiveness of density controls, fraction of the time sampling failed between successive sampling intervals, failure of diluent wash column, and malfunction of the pulse extraction column.

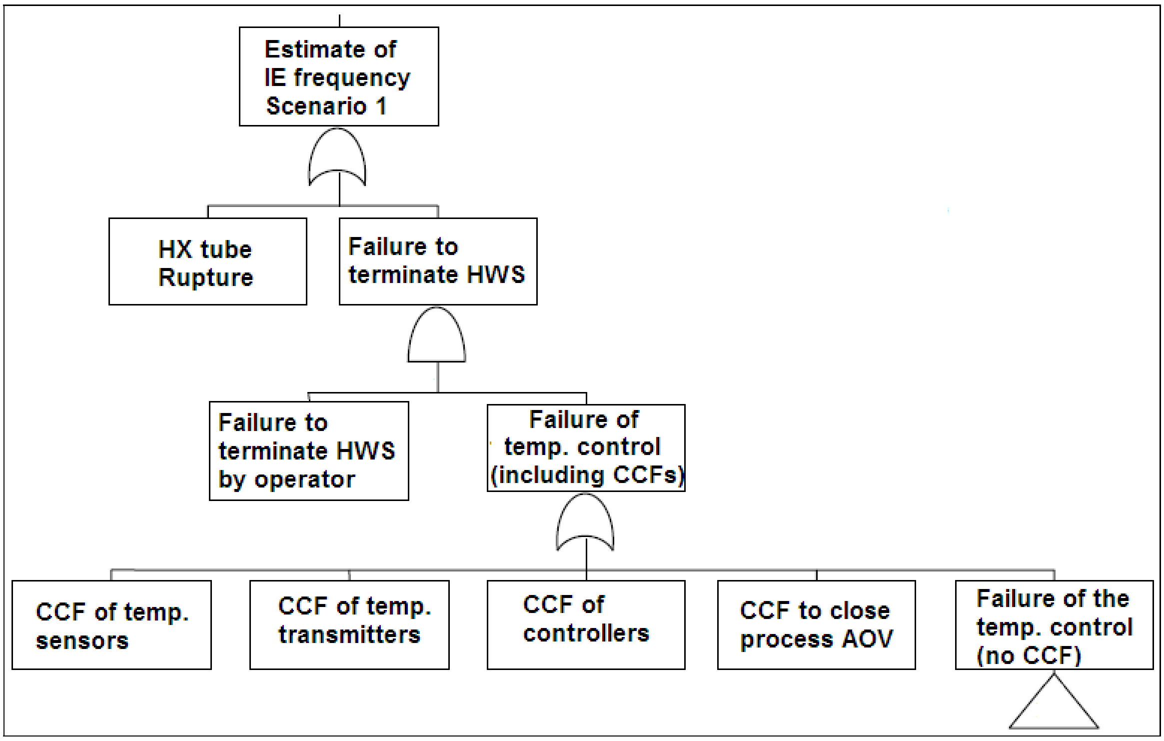
Figure 1**. Event tree for condition 1 (below): normal accumulation of TBP.**

Loss of temperature control or heat exchanger tube rupture	Maintain TBP Level Control	Venting system of the evaporator		End States



The PRA model for NOR in the concentrate collection tank assumes: (1) failure to provide cooling flow to the tank heat exchanger could result in tank heat up and initiation of evaporative cooling (HVAC system failures that could also lead to tank heat up were not modeled as it was assumed that facility response to HVAC failure would be shutdown of the unit), (2) failure of spray mixing inside the tank could create hot spots leading eventually to initiation of evaporative cooling, and (3) if there was an increased amount of TBP in the tank due to inadvertent transfer, then loss of cooling or mixing would lead to NOR as the criteria for evaporative cooling would have been violated. The initiating event is the loss of cooling or mixing; its frequency was estimated from fault tree evaluations of the systems involved. The next top event is “no transfer of separate organics”, which was estimated using the models developed earlier for the evaporator, due to the common pathways for transport of separate phase TBP to the process vessels in the unit, including the evaporator and the concentrates tank. The next top event labeled “level control or No excessive TBP” addresses the operator actions needed to provide aqueous make up to maintain the criteria for success of evaporative cooling on the appropriate branches under conditions (1) and (2) above. The last top event in the tree, “venting”, represents the success of venting to maintain the solution temperature at a safe level to prevent a NOR. There are four NOR sequences. Two of them involve the transfer of large amounts of TBP to the tank due to malfunctions in the pulsed extraction columns and subsequent failures of the sampling and density controls; they are very similar to the scenarios under upset accumulation in the first evaporator accident scenario and the dominant cutsets are also similar. The dominant cutset in the venting failure sequence is common cause failure of plugging of HEPA filters. In the remaining sequence it is the failure of the operator to recognize the level alarm and take proper action.

Figure 2**. Simplified Fault Tree of Initiating Event frequency of Scenario 1 (above)**



The PRA model for NOR in the second evaporator is based on the evaluation of the various pathways by which organics can be transferred to this evaporator. Two scenarios with their respective event trees are modeled; in the first scenario, the initiating event is solvent transfer by mechanical entrainment, in the second by a severe process malfunction leading to the transfer of a relatively large amount of solvent. Both event trees consider the failures and successes of various barriers to the transfer of TBP, including success of wash columns to break up and separate the entrained organics, the effectiveness of passive systems in preventing transfer of any separate phase organics in excess of their solubility limit, and failures of sampling for organics in batch tanks. These failures were modeled by a combination of fault trees and corrosion rate data for failure of a baffle in the passive system. Three NOR sequences resulted from the analysis. The dominant cutsets in all of them include operational failures of the passive, failure of diluent wash columns and failure of an air lift to stop process solution transfer to the unit where the evaporator is located.

6. Conclusion

The example analyzed in this paper has demonstrated that PRA methods and tools can be successfully applied to model accident sequences in fuel cycle facilities that chemically process nuclear materials and to identify major vulnerabilities that may arise from combinations of equipment failures and human errors to cause undesired outcomes. However, while the results of the quantitative assessments show that the point estimate frequencies of the nitration-oxidation reaction in various process units are low, they must be considered preliminary for several reasons. The failure rate database for equipment failures and human reliability in fuel cycle facilities, especially for equipment that may be exposed to harsh chemical environments, is very sparse and uncertain. Moreover, the PRA carried out was a limited-scope one for several reasons as stated above. However, the analysis performed using PRA techniques can be considered as risk-informing the qualitative analyses. In particular, the identification of dominant cutsets in the various sequences helps to focus attention on the more important systems that impact the safety of the design with regard to reducing the frequency of nitration oxidation reactions.

NEA/CSNI Workshop on Safety Assessment of Fuel Cycle Facilities – Regulatory Approaches and Industry Perspectives
Toronto, Canada, 27-29 September 2011

Risk-informing Safety Reviews of Fuel Cycle Facilities: An Example Application

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U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Agenda

- Introduction and Purpose of Study
- Risk Assessment Considerations and Limitations
- Qualitative Risk Assessment
- Quantitative Risk Assessment
- Conclusion

Introduction

- Focus of paper is in the analysis methodology
- Hazard of concern: explosions due to nitration oxidation reactions (NOR) in solvent extraction technology

Purpose of Review

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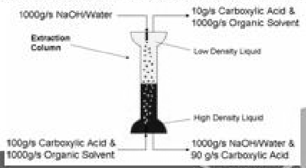
- PUREX process uses solvent extraction
- Solvent or Liquid-liquid extraction consist in removing compounds from two immiscible liquids (aqueous and organic phases)

Figure: Solvent Extraction Column Example



Risk Assessment Considerations and Limitations

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- Limited scope PRA:
  - Generic risk due to external hazards were excluded from the analysis
  - Failures of heat transfer strategy were not considered in the analysis
  - Semi-empirical model for the tributyl phosphate (TBP)-nitrate reactions based on thermal decomposition alone

Risk Assessment Considerations and Limitations

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- Quantitative analysis end state: annual probability of transport of TBP into process equipment with conditions that could promote NOR
- Consequence of NOR was assumed to be high for the facility worker
- Safety controls should be in place to reduce the likelihood of occurrence of the event to highly unlikely
- Consequence analysis was not performed

Qualitative Assessment

U.S.NRC

- Screen and prioritize resources to processes of highest risk.
- Factors that may contribute to possibility of NOR:
  - Contact of organic and aqueous phase
  - Potential of TBP transfer
  - Heat sources
  - Heat balance

Process vessels in the table were identified as higher risks and selected for further evaluation in the Quantitative Risk Assessment

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Process Vessel	Safety Strategy	Success Criteria
Evaporator: natural recirculation thermo-siphon type boiler	Evaporative Cooling	- Maintain hot water system temperature below 122°C - Maintain minimum aqueous phase to TBP ratio - Maintain maximum TBP layer depth - Maintain maximum process solution temperature - Adequately vented system
Collection Tank	Evaporative Cooling	- Maintain tank temperature below 80°C - Maintain minimum aqueous phase to TBP ratio - Maintain maximum TBP layer depth - Maintain maximum process solution temperature - Adequately vented system - Tank flush once every six months
Evaporator: natural circulation thermo-siphon evaporator	TBP Prevention	- Prevent transfer of TBP to evaporator

Quantitative Assessment

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- Quantitative evaluation using accident sequence delineation (event trees and fault trees)
- Quantification using SAPHIRE code.
- Point Frequency of conditions that could promote NOR was calculated
- Uncertainty (5th and 95th percentile)

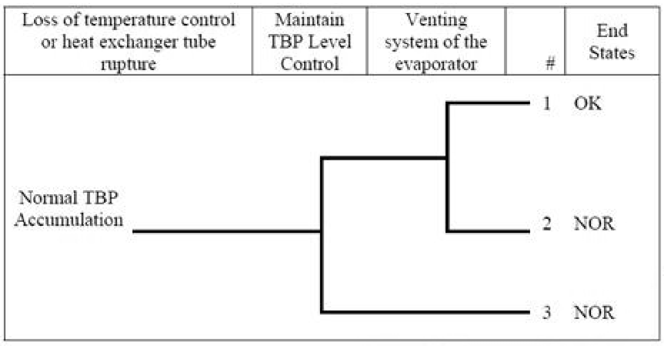
Evaporator 1

- Upset conditions:
  - Normal TBP accumulation
  - Upset accumulation of TBP (Failure of process upstream barriers)
- Initiating Events:
  - Loss of temperature control
  - Heat exchanger tube rupture

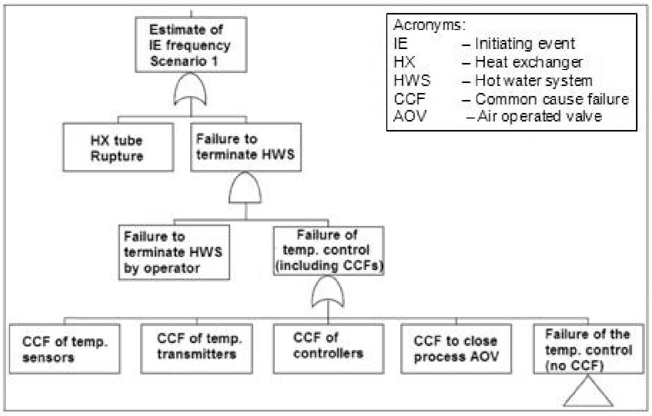
Evaporator 1

- Dominant cut-sets:
  - Failure or ineffectiveness of density controls
  - Failure of sampling
  - failure diluents wash column
  - Malfunction of pulse extraction column

Simplified Event tree for condition 1: normal accumulation of TBP



Simplified Fault Tree of Initiating Event frequency of Scenario 1

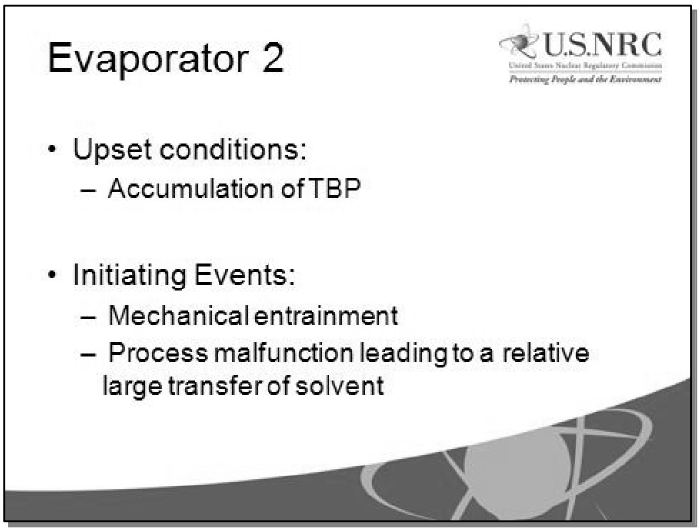


Concentrate collection tank

- Upset conditions:
  - Excessive accumulation of TBP
  - Large transfer of TBP
- Initiating Events:
  - Loss of cooling or mixing
  - Failure of spray mixing
  - Loss of mixing

Concentrate Collection Tank

- Dominant cut-sets:
  - Plugging of HEPA filters
  - Operator failure to recognize level alarm and take proper action


Evaporator 2
U.S.NRC
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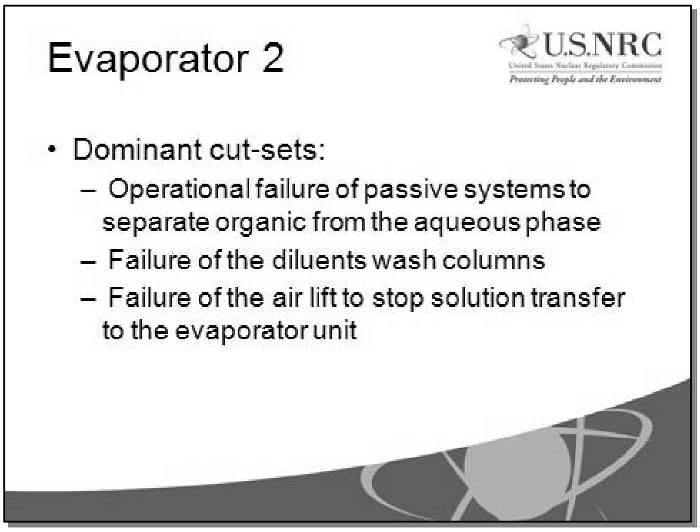
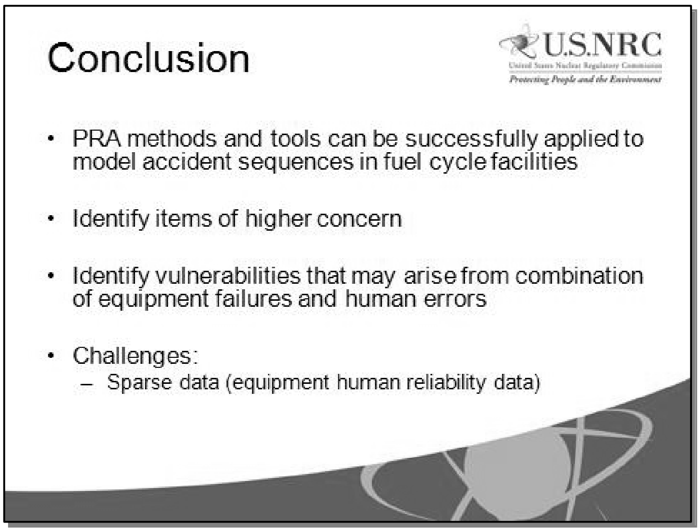
• Upset conditions:
– Accumulation of TBP
• Initiating Events:
– Mechanical entrainment
– Process malfunction leading to a relative large transfer of solvent


Evaporator 2
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment
• Dominant cut-sets:
– Operational failure of passive systems to separate organic from the aqueous phase
– Failure of the diluents wash columns
– Failure of the air lift to stop solution transfer to the evaporator unit


Conclusion
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• PRA methods and tools can be successfully applied to model accident sequences in fuel cycle facilities
• Identify items of higher concern
• Identify vulnerabilities that may arise from combination of equipment failures and human errors
• Challenges:
– Sparse data (equipment human reliability data)

REGULATION OF CHEMICAL SAFETY AT FUEL CYCLE FACILITIES BY THE UNITED STATES NUCLEAR REGULATORY COMMISSION

Kevin M. Ramsey

U.S. Nuclear Regulatory Commission, U.S.A.

Abstract - When the U.S. Nuclear Regulatory Commission (NRC) was established in 1975, its regulations were based on radiation dose limits. Chemical hazards rarely influenced NRC regulations. After the Three Mile Island reactor accident in 1979, the NRC staff was directed to address emergency planning at non-reactor facilities. Several fuel cycle facilities were ordered to submit emergency plans consistent with reactor emergency plans because no other guidance was available. NRC published a notice that it was writing regulations to codify the requirements in the Orders and upgrade the emergency plans to address all hazards, including chemical hazards. The legal authority of NRC to regulate chemical safety was questioned. In 1986, an overfilled uranium hexafluoride cylinder ruptured and killed a worker. The NRC staff was directed to address emergency planning for hazardous chemicals in its regulations. The final rule included a requirement for fuel cycle facilities to certify compliance with legislation requiring local authorities to establish emergency plans for hazardous chemicals.

As with emergency planning, NRC's authority to regulate chemical safety during routine operations was limited. NRC established memoranda of understanding (MOUs) with other regulatory agencies to encourage exchange of information between the agencies regarding occupational hazards. In 2000, NRC published new, performance-based, regulations for fuel cycle facilities. The new regulations required an integrated safety analysis (ISA) which used quantitative standards to assess chemical exposures. Some unique chemical exposure cases were addressed while implementing the new regulations. In addition, some gaps remain in the regulation of hazardous chemicals at fuel cycle facilities. The status of ongoing efforts to improve regulation of chemical safety at fuel cycle facilities is discussed.

I. Introduction

The U.S. Nuclear Regulatory Commission (NRC) was created in 1975 when the former Atomic Energy Commission was re-organized. NRC authority was limited to radioactive materials associated with nuclear reactor operations. Specifically, NRC regulated source material for manufacturing reactor fuel, the reactor fuel itself, and byproduct materials from reactor operations. The safety of other radioactive materials was regulated by other agencies. NRC regulations were based primarily on compliance with radiation dose limits. Radioactive materials with hazardous chemical properties were noted, but radiation was the primary hazard associated with the materials under NRC's jurisdiction. Chemical hazards rarely influenced NRC's regulations.

II. Emergency preparedness

In 1979, there was a severe accident at the Three Mile Island nuclear reactor. At that time, no formal emergency plan was required for any fuel cycle facility. One of the follow-up actions from this accident was to evaluate the need for emergency plans at non-reactor facilities. The NRC staff developed criteria for selecting licensees that should have a formal emergency plan. The criteria were based on quantities of radioactive material that could result in a serious radiation exposure during an accident. In February 1981, Orders were issued to 61 non-reactor licensees to submit a formal emergency plan for approval or request a license amendment to reduce the amount of radioactive material authorized. Half of the licensees submitted emergency plans. The licensees were instructed to use reactor guidance when preparing their emergency plans because no other guidance was available.

In June 1981, NRC published a Notice of Proposed Rulemaking to codify the emergency plan requirements in the Orders and upgrade the plans to address all hazards to the public, including chemical hazards. In response to the notice, the legal authority of NRC to require emergency plans to address chemical releases was questioned because other Federal agencies regulated chemical safety. In addition, NRC received several comments that the rule was not cost effective and there was no need for the rule. Despite the concerns, the NRC staff believed the rule was in the public interest and proceeded with writing a proposed rule.

In January 1986, an overfilled, uranium hexafluoride cylinder ruptured at the Sequoyah Fuels facility in Gore, Oklahoma. One worker died from pulmonary edema caused by exposure to hydrofluoric acid vapors. A follow-up action was to review the authority of various Federal agencies to regulate chemical hazards at fuel cycle facilities.

In March 1986, the NRC staff requested Commission approval to publish a proposed emergency plan rule for public comment. Legal Counsel concluded that the chemical toxicity of uranium compounds may be considered in NRC's regulations. However, the authority of NRC to regulate other chemical hazards continued to be questioned. The Commission disapproved the proposed rule and directed the staff to address chemical hazards and lessons learned from the Sequoyah Fuels accident.

In October 1986, the Emergency Planning and Community Right-To-Know Act[67] was signed into law. The Act established a program under the U.S. Environmental Protection Agency. The Act required each State to establish local emergency planning committees (LEPCs) in each area with a facility possessing hazardous chemicals in excess of threshold quantities. Local public safety officials served on the committees. Each LEPC was required to establish an emergency response plan for chemical releases from facilities in its area. The Act required each facility operator to provide its LEPC with the information needed to prepare and implement its emergency plan. The NRC staff revised its proposed rule to require each licensee submitting an emergency plan to certify that it was in compliance with the Emergency Planning and Community Right-To-Know Act. This approach resolved concerns about NRC's legal authority because it utilized a program implemented by another Federal agency. Another advantage of this approach was that decisions concerning emergency response to hazardous chemicals releases were left to local officials. In 1989, the Commission approved the revised rule because it

[67] Emergency Planning and Community Right-To-Know Act of 1986 (http://www.epa.gov/oecaagct/lcra.html).

believed that any NRC obligation to ensure adequate emergency planning for releases of hazardous chemicals could be met by requiring licensees to certify compliance with the new Act.

III. Licensing and inspection

As noted above, NRC's authority has been viewed historically as limited to hazards posed by the radioactive materials NRC regulates. In October 1988, the NRC established a Memoranda of Understanding (MOU) with the U.S. Occupational Safety and Health Administration (OSHA)[68]. The MOU provides guidelines for exchange of information between NRC staff and OSHA staff on occupational hazards at NRC-licensed facilities. Concerns regarding radiation safety and occupational safety often overlap. Generally, NRC has jurisdiction to regulate the following issues at NRC-licensed facilities:

- Radiation risk produced by radioactive materials licensed by NRC.
- Chemical risk produced by radioactive materials licensed by NRC.
- Facility conditions that affect the safety of radioactive materials licensed by NRC.

Generally, OSHA has jurisdiction to regulate facility conditions that result in an occupational risk at almost all U.S. facilities (many more than those licensed by NRC). Under the MOU, NRC agrees to have its inspectors share information with OSHA concerning non-radiological hazards brought to their attention or observed during inspections.

In 2000, NRC published new regulations for fuel facilities licensed to use enriched uranium and plutonium[69]. The new regulations used a risk-informed and performance-based approach that included the following:

- Establishing performance requirements for preventing accidents or mitigating their consequences.
- Requiring a safety program that demonstrates compliance with the performance requirements.
- Requiring an Integrated Safety Analysis (ISA) that identifies:
- Radiological hazards related to licensed material,
- Chemical hazards of licensed material and hazardous chemicals produced from licensed material, and
- Facility hazards that could affect the safety of licensed material and thus present an increased radiological risk.
- Requiring a description of the quantitative standards used to assess the consequences of acute chemical exposure to licensed material and chemicals produced from licensed material.

In March 2002, NRC published NUREG-1520, Standard Review Plan (SRP) for a License Application for a Fuel Cycle Facility[70]. The SRP described acceptable ways to demonstrate compliance with the

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[68] NRC Inspection Manual Chapter 1007, Interfacing Activities Between Regional Offices of NRC and OSHA (http://www.nrc.gov/reading-rm/doc-collections/insp-manual/manual-chapter/mc1007.pdf).

[69] Final Rule: Title 10, Code of Federal Regulations, Part 70, Domestic Licensing of Special Nuclear Material, Federal Register, Volume 65, Page 56211, September 18, 2000 (http://www.gpoaccess.gov/fr/index.html).

[70] NUREG-1520, Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility (http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1520/).

new requirements. The SRP stated that acceptable standards for assessing acute chemical exposures included, but were not limited to, the following:

- Emergency Response Planning Guidelines (ERPGs)[71],
- Acute Exposure Guideline Levels (AEGLs)[72], and
- exposure limits in International Organization for Standardization (ISO) standards.

All the standards listed in the SRP defined airborne limits for inhalation exposures. The industry interpreted the guidance as requiring analysis of airborne releases only.

IV. Unique issues during implementation

Bulk chemical storage tanks

When licensees began performing ISAs, the NRC staff noted that releases from bulk chemical storage tanks posed some of the greatest hazards to workers and the public. Concerns were raised that NRC should require bulk chemical storage tanks to be addressed in the ISAs. After reviewing the concerns, NRC management concluded that no changes were warranted because NRC does not regulate hazardous chemicals before process addition to licensed material, or after process separation from licensed materials, unless the chemicals could potentially contact licensed material or cause an increased radiological risk.

Dermal exposures

NRC recognizes that inhalation is the primary exposure pathway for members of the public off-site. However, worker exposures may involve direct contact of hazardous chemicals with the skin. When accident scenarios involving dermal exposure have been identified, NRC has required licensees to establish a quantitative standard for assessing the risk. Licensees have noted that few industry standards exist for dermal exposures. NRC acknowledges a lack of formal standards for dermal exposures; however, accepted industry practices provide a basis for establishing a site-specific standard, which NRC can accept as reasonable for assessing the consequences of acute chemical exposures. The Honeywell manual[73] for medical treatment of hydrofluoric acid exposures is an example of accepted industry practice, which NRC has accepted as a basis for a site-specific standard.

V. Physical security of hazardous chemicals

As noted above, NRC does not regulate the safety (or security) of hazardous chemicals before process addition to licensed material, or after process separation from licensed material. In 2007, the U.S. Department of Homeland Security (DHS) published the Chemical Facility Anti-Terrorism Standards (CFATS)[74,75]. The CFATS regulation imposed physical security requirements for hazardous

[71] U.S. Department of Energy, Office of Emergency Management and Policy, Emergency Response Planning Guidelines (http://orise.orau.gov/emi/scapa/chem-pacs-teels/erpg-definitions.htm).

[72] U.S. Environmental Protection Agency, National Advisory Committee for the Development of Acute Exposure Guideline Levels for Hazardous Substances, Acute Exposure Guideline Levels (http://www.epa.gov/opptintr/aegl/pubs/define.htm).

[73].Honeywell, Recommended Medical Treatment for Hydrofluoric Acid Exposure, 2006 (http://www.honeywell.com/sites/servlet/com.merx.npoint.servlets.DocumentServlet?docid=DB49512F4-4D4F-7ADE-D038-44F6DF1771F9).

[74] Interim Final Rule: Title 6, Code of Federal Regulations, Part 27, Chemical Facility Anti-Terrorism Standards, Volume 72, Page 17688, April 9, 2007 (http://www.gpoaccess.gov/fr/index.html).

chemicals above specified threshold quantities. However, the CFATS regulation exempted facilities licensed by NRC from the security requirements. NRC recognized that the exemption may create a significant gap in the regulation of security for hazardous chemicals. An MOU is being negotiated between the NRC and DHS to define clear lines of responsibility between the agencies for the security of high-risk chemicals.

VI. Conclusion

NRC's regulation of chemical safety at fuel cycle facilities has improved over the years. Limits in NRC authority continue to pose challenges when regulating chemical safety. Coordination and cooperation with other regulatory agencies is essential to maintaining effective oversight of the industry.

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[75] Final Rule: Title 6, Code of Federal Regulations, Part 27, Appendix to Chemical Facility Anti-Terrorism Standards, Volume 72, Page 65396, November 20, 2007 (http://www.gpoaccess.gov/fr/index.html).

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Regulation of Chemical Safety at Fuel Cycle Facilities in the US

NEA/CSNI Workshop on

Safety Assessment of Fuel Cycle Facilities – Regulatory Approaches and Industry Perspectives

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Outline

Regulation of Chemical Safety at Fuel Cycle Facilities

- Initial Regulatory Focus of the US NRC
- Emergency Preparedness: Historical Review
- Licensing and Inspection
- Unique Issues During Implementation
- Physical Security of Hazardous Chemicals
- Conclusion

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Initial Regulatory Focus of the U.S. NRC

- US Nuclear Regulatory Commission was created in 1975
  - Authority limited to radioactive materials associated with reactor operations
  - Regulations specified compliance with radiation dose limits
  - Chemical hazards rarely influenced NRC regulations

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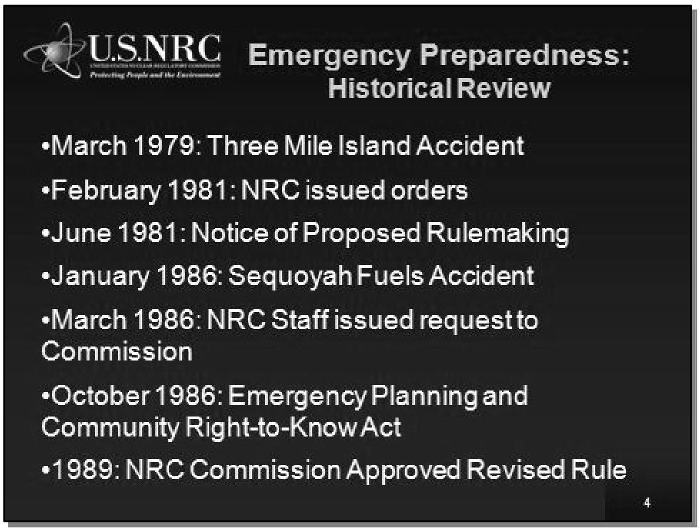
U.S.NRC
Emergency Preparedness: Historical Review
•March 1979: Three Mile Island Accident
•February 1981: NRC issued orders
•June 1981: Notice of Proposed Rulemaking
•January 1986: Sequoyah Fuels Accident
•March 1986: NRC Staff issued request to Commission
•October 1986: Emergency Planning and Community Right-to-Know Act
•1989: NRC Commission Approved Revised Rule
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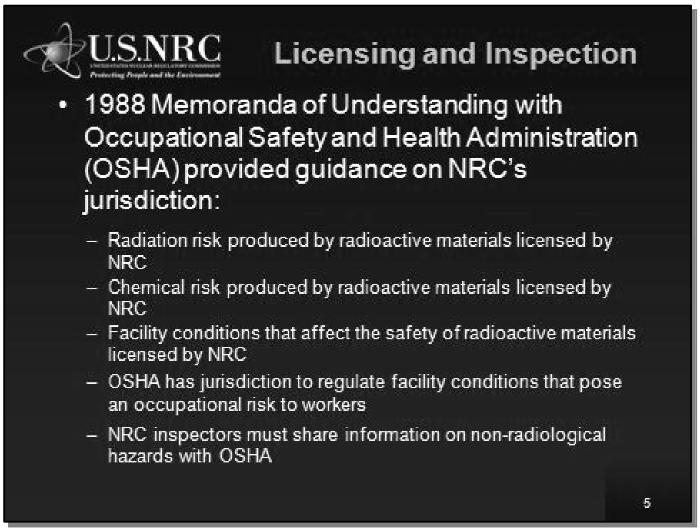
U.S.NRC
Licensing and Inspection
• 1988 Memoranda of Understanding with Occupational Safety and Health Administration (OSHA) provided guidance on NRC's jurisdiction:
– Radiation risk produced by radioactive materials licensed by NRC
– Chemical risk produced by radioactive materials licensed by NRC
– Facility conditions that affect the safety of radioactive materials licensed by NRC
– OSHA has jurisdiction to regulate facility conditions that pose an occupational risk to workers
– NRC inspectors must share information on non-radiological hazards with OSHA
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U.S.NRC
Licensing and Inspection - continued
In 2000, NRC published new regulations using risk-informed and performance-based approach:
• Established performance requirements for preventing accidents or mitigating their consequences
• Required a safety program to demonstrate compliance
• Required an Integrated Safety Analysis (ISA) with:
– Radiological hazards
– Chemical hazards
– Facility hazards
• Required description of quantitative standards for assessing consequences of acute exposure
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U.S.NRC

Licensing and Inspection - continued

In 2002, NRC published NUREG-1520, Standard Review Plan for a License Application for a Fuel Cycle Facility.

- Acceptable standards for assessing acute chemical exposures include:
  - Emergency Response Planning Guidelines,
  - Acute Exposure Guideline Levels,
  - Exposure limits in ISO standards
- The industry interpreted these standards as requiring analysis of airborne hazards only

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U.S.NRC

Unique Issues During Implementation

- Bulk Chemical Storage Tanks
  - Posed greatest hazards to workers and public
  - NRC does not regulate hazardous chemicals not mixed with licensed material, unless chemicals could cause an increased radiological risk
  - Not initially required to be addressed in the ISA
- Dermal Exposures
  - Inhalation is primary pathway for public off-site, but workers may have direct contact with skin
  - Licensees must establish a quantitative standard for assessing risk of dermal exposure
  - Few industry standards exist – but industry practices provide basis for site-specific standards

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U.S.NRC

Physical Security of Hazardous Chemicals

- NRC does not regulate the safety (or security) of hazardous chemicals before process addition or after process separation
- U.S. Department of Homeland Security (DHS) published the Chemical Facility Anti-Terrorism Standards (CFATS) in 2007
  - Imposed physical security requirements for threshold levels of hazardous chemicals
  - NRC licensed facilities were exempted (regulatory gap)
  - NRC negotiating MOU with DHS to address regulatory gap and clarify responsibilities

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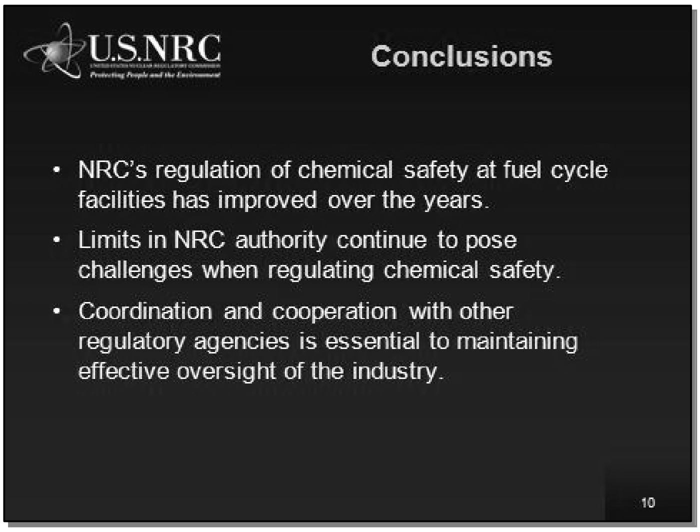
U.S.NRC
Conclusions
• NRC's regulation of chemical safety at fuel cycle facilities has improved over the years.
• Limits in NRC authority continue to pose challenges when regulating chemical safety.
• Coordination and cooperation with other regulatory agencies is essential to maintaining effective oversight of the industry.
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U.S.NRC
Questions ?
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SESSION FOUR

BACK END FACILITIES

Dry Interim Storage of Radioactive Material in Germany
C. Drobniewski, J. Palmes *(BfS, Germany)*

Overview of the Paks Modular Vault Dry Storage Probabilistic Safety Assessment
D. Hollo (Nubiki, Hungary); K. Toth (SOM System Ltd., Hungary)

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P. Nocture (Areva, France)

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S. Segawa, K. Fujita *(JNFL, Japan)*

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Why Risk Assessment in Long-Term Storage of Spent Nuclear Fuel?
T. Ahn, J. Guttmann, J. Davis *(NRC, USA)*

DRY INTERIM STORAGE OF RADIOACTIVE MATERIAL IN GERMANY

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Abstract - In accordance with the waste management concept in Germany, spent fuel is stored in interim storage facilities for a period of up to 40 years until deposition in a geological repository. In twelve on-site interim storages in the vicinity or directly on the sites of the nuclear power plants, spent fuel elements from reactor operation are stored after the necessary period of decay in wet storage basins inside the reactors. Additionally, three central interim storage facilities for storage of spent fuel of different origin are in operation.

The German facilities realize the concept of dry interim storage in metallic transport and storage casks. The confinement of the radioactive material is ensured by the double lid system of the casks, of which the leak tightness is monitored constantly. The casks are constructed to provide adequate heat removal and shielding of gamma and neutron radiation.

Usually the storage facilities are halls of thick concrete structures, which ensure the removal of the decay heat by natural convection.

The main safety goal of the storage concept is to prevent unnecessary exposure of persons, material goods and environment to ionizing radiation. Moreover any exposure should be kept as low as reasonable achievable. To reach this goal the containment of the radioactive materials, the disposal of decay heat, the sub criticality and the shielding of ionizing radiation has to be demonstrated by the applicant and verified by the licensing authority.

In particular accidents, incidents and disasters have to be considered in the facility and cask design.

This includes mechanical impacts onto the cask, internal and external fire, and environmental effects like wind, rain, snowfall, flood, earthquakes and landslides. In addition civilizatoric influences like plane crashes and explosions have to be taken into account.

In all mentioned cases the secure confinement of the radioactive materials has to be ensured.

On-site storage facilities have to consider the interplay with the nearby facilities too.

The facilities have to be monitored upon aging effects. This includes recurrently checks of the casks to ensure the manageability and the save confinement of the materials.

1. Introduction

The goal of the article is to provide a comprehensive overview over the safety assessment structure in Germany for storage facilities of spent fuel. This also includes depicture of necessary tasks performed to meet the requirements set by the legal authorities based on national law.

By the term "storage facility for spent fuel" the article refers to the dry interim storage facilities build for the storage of spent fuel casks like the CASTOR® series. Those facilities are located either on-site with the nuclear power plants or as a centralized storage facility on a separate site.

The on-site facilities are solely storing said spent fuel casks, while the central storage facilities also store additional types of radioactive waste. The presented article will only deal with the spent fuel casks, although the safety requirements depicted are also valid for casks storing highly active conditioned radioactive waste or other radioactive waste in general.

Main safety element of the storage facilities are the storage casks themselves, they provide the necessary confinement of the radioactive waste. The inventory integrity is provided by the structural stability of the steel cask, the special holding structure for the fuel inside and the rod claddings. Leak tightness is usually secured by a double lid closing. Proper shielding of radiation is achieved by the casks and inbuilt neutron moderator material.

For the licensed radioactive inventory the cask provides thermal stability of the waste, the radiation shielding, the sub criticality and the leak tightness. The storage facilities in Germany are buildings of concrete with wall thicknesses between 80cm and 1.2 m. They are sectioned to provide separate rooms for reception, repair and maintenance and storage respectively.

The licensing procedure is defined by the German law for nuclear power (AtG [**76**]) supplemented by the requirements stated in the regulations StrSchV[**77**] and the RSK guidelines [**78**].

The applicant, private or governmental company, has to apply for a licence (in this case) at the Federal Bureau for Radiation Protection (BfS). Compliance to the above mentioned requirements has to be demonstrated by the applicant and is then reviewed / checked and validated by the BfS with the support of external expert organisations. The demonstration of compliance has to follow the state of scientific knowledge and can be performed by experiments and/or simulations.

The licences for the storage facilities are given for a specific period of time (40 years).

Furthermore the licences for the casks are divided into one for transport and one for storage only.

[76] AtG – Gesetz über die friedliche Verwendung der Kernenergie und den Schutz gegen ihre Gefahren (Atomgesetz) – in the present valid version

[77] StrSchV – Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung) – in the present valid version

[78] RSK Guides – Leitlinien der Reaktor-Sicherheitskommision

Enclosed in the licensing process is the check of the performed actions inside the facility in normal operation. In addition a set of so called design based accidents is identified which set the impacts / events the facility and casks have to withstand without violating the limits set in the regulation rules.

The following section 2 will therefore deal with the normal operation and section 3 will focus on the design based accidents. Due to the recent events, an inquiry into the safety margins provided by the layout of the facility and properties of the casks was indicated. Section 4 describes a qualitative examination of extreme events by extrapolation on the design based accidents.

2. Assessment of normal operation

When the facility runs in normal operation the confinement of the radioactive material and the prevention of unnecessary exposition to ionizing radiation have to be provided.

Furthermore the decay heat has to be safely transported outside of the facility.

To provide continuously confinement, the leak tightness of the cask has to be monitored regularly.

As for the CASTOR-Series casks there are pressure sensors installed to the double-lid system that report automatically all irregularities to the working staff. The double-lid system is constructed in a way that it will induce an error long before actual release of radioactive material will happen. This way a maintenance and repair schedule can take action to restore the leak tightness.

Since commissioning of the storage facilities and installing the automated pressure monitoring of the casks there has been no reported leakage of the casks due to malfunctioning lids, only malfunctioning pressure sensors were detected so far [79].

Unnecessary exposition to radiation for the public is avoided due to the shielding of the cask and the concrete facility building. At the nearest reachable point for the public persons, the radiation is shielded down to a sub natural radiation level.

For the working personal, prescript working schedules as well as the sectioning of the facility building reduces the exposition as much as reasonable achievable. Keeping track of the actual exposition of the working personal through calibrated dosimeters (calibrated by the responsible legal authority) ensures abiding the legal dose limits for the staff.

The decay heat removal is an integral part of the safety of the casks and the facility. The decay heat of the radioactive inventory is transported by heat conduction to the cask surface and removed via natural convection out of the facility. The facility is designed to provide effective passive cooling for the maximum licensed decay heat at any outside conditions. Due to these regulations the building integrity as well as the working temperatures for facility components is secured.

In addition the lid-temperatures inside the casks stay in a secure level to provide the leak tightness for the storage time. The stability of the inventory (like fuel rod claddings) and the moderation material is also secured.

[79] D. Wolff, U. Probst: Beurteilung bisher aufgetretener Ausfälle des Druckschalters DPS 220 hinsichtlich systematischer Versagensursachen., BAM III.4/10 299-DS Dezember 2010

Keeping track of the leak tightness is only a part of the undertaken periodic safety measures.

The casks and the facility are repeatedly checked for ageing effects. This includes corrosion checks on the casks, on the facility components, the facility structure.

3. Design based Accidents

Design based accidents are either natural events that can strike the storage facility or man made accidents in or outside of the facility. All such events shall not have any impact on the safety of the facility that can violate the radiation release limits as in StrSchV §49 [80**]**.

All design based accidents and their respective results have to be assessed in order to get a license for the facility. The events are assessed by a combination of deterministic and probabilistic inquiries.

The following design based accidents are included:

a) Dropping of Casks / dropping onto casks / collisions

Casks have to withstand collisions of the cask with other casks or facility structures while handled with the facility crane. In addition the computer driven crane is programmed to avoid collision on its own.

Furthermore droppings of casks inside the facility are considered for all possible accidents connected with handling, for instance the storage process or maintenance handling. The facility specific topology (steps), layout (sectioning, doors ...) is taken into consideration while identifying possible events.

Dropping of facility parts onto the casks are also considered but conservatively assessed by the design based accident for an accidental plane crash as described below.

b) Earthquakes

Earthquakes are by nature events hard to assess, as there is no absolute way to access the real impact strength due to the incomplete knowledge of the internal processes and status of the geology at the facility site. An interplay of deterministic (assessment based on historic events, geological structure …) and probabilistic (rate of occurrence for quakes with certain strength in given time …) methodologies have to be used to derive the so called design earthquake. In Germany each facility site has to be earthquake risk assessed for earthquakes based on the regulations stated in KTA 2201 [81]. The facility itself has to sustain the derived “design earthquake” to ensure the safety. Furthermore it has to be proved, that casks won’t topple in case of the said earthquake and therefore no mechanical impact can change the condition of the casks.

c) Fires

The design based accident fire for casks is either 600°C for 1h or 800°C with 30min in duration.

The cask has to withstand those fire conditions without substantial release of radioactive material.

Due to the high mass and specific heat of the cask material, the thermal impact is mostly limited to near surface areas within the cask. The possibility of such a fire is greatly reduced by the facility itself,

[80] StrSchV §49 - Release for a design based accident shall not exceed 50mSv.
[81] KTA 2201 - Design of Nuclear Power Plants against Seismic Events

as the amount of burnable substances in the storage facility is kept as low as possible and there are no ignition sources inside.

d) Flood

The casks have to be designed in a way that they withstand submerging into water at a depth of 200m. Furthermore the sub criticality has to be proven even for a cask filled with water (which is highly unlikely as the cask is leak tight under the submerging mentioned.) The risk of such a submersion depth is practical impossible in the case of storage in the facility.

The facilities themselves are designed against the flood height which has an occurrence probability of

10^{-4} per year. This is achieved by a heightened building site and/or temporary deployable flood protection systems.

e) Plane crashes (accidental from military vessel) (FLAB)

In safety assessment the accidental crash of a small military aircraft hitting the storage facility is considered. It is assumed that a turbine shaft hits the casks right on top. It has to be proven that the impact onto the cask does not lead to a substantial release even including the assumed fire cause by the accident. The protecting properties of the concrete building are usually neglected for this scenario.

f) Explosions

Explosions of for instance burnable gases have to be considered at the closest reasonable vicinity of the facility in the assumed available amount. This includes especially the transport by ships as the on-site storage facilities are often close to rivers with active transport routes like the NPP themselves. The strength of the explosion is derived from the maximum transport amount and ideal conditions for the explosion. The facility has to withstand the explosion without structural damage and therefore without impact onto the casks.

4. Extreme Scenarios

By the Term “Extreme Scenarios” we summarize a set of events that exceed the design based accidents. We will use those to assess safety margins of the storage facilities. Therefore the scenarios are not on a quantitative but qualitative level to identify the scaling of accident consequences in dependence of the accident/disaster extent. Wherever possible the consequences of the scenarios are assessed by extrapolating the consequences starting from the most alike design based accident.

The following procedure is used to pin down the scenarios:

First we identify events which cause radiation consequences near or above the radiation limits. Based on those limits we try to identify the types of impacts that can lead to such consequences. The possible disasters or accidents that can lead to those impacts are then considered. By choosing an extreme (but feasible) extent of such disasters we estimate the consequences based on extrapolations starting from the alike base design accident. The consequences (radioactive) of such disasters have to be compared to the limits again. Following the above depicted procedure we end up with the following scenarios:

a) Flooding

Even extreme floods that exceed the design based flood disaster can't create any substantial release as the casks are leak tight up to 200m submerge. Furthermore the submerging in water won't have any negative thermal impact on the casks. Assuming that floods are no permanent conditions, a corrosive damage to the cask can be excluded, especially as the casks can be checked in the following cleaning procedure. Besides very fast rising flood events (Tsunami etc.) it is also very unlikely that the storage facility itself will be damaged, and therefore the casks are also safely enclosed at the storage area.

In an event damaging the facility structure, one could argue about consequences comparable to the ones of an extreme Earthquake as described below.

b) Earthquakes

Extreme earthquakes can lead to two possible results, firstly tipping of the casks which are a mechanical impact, and secondly burying of the casks under debris of the facility which will lead to a long term thermal impact. It is secure to assume that the case of tipping casks is the one more likely to happen instead of the facility collapse. Therefore we will take the toppled casks scenario as given when considering the facility collapse.

1) Toppling of the casks in the storage facility:
The mechanical impact of a toppling is much lower than the impact of the design based accident FLAB. Therefore it is conservative to assess the releases of this case by this design base accident (FLAB). When extrapolating the releases to the amount of casks possibly affected at the storage facility it leads to no unsustainable consequences.

2) Burying the cask below debris due to collapse of the facility building:

In an event of overwhelmingly strong earthquakes, the possibility of partly or full collapse of the facility building is feasible. The casks will then be toppled and buried below debris.

However the mechanical consequences for the casks of such a collapse are not stronger than the ones coming from the FLAB. It is feasible to assume the releases to be in the same dimension. Therefore main focus lies on the thermal impact due to reduced air cooling of the casks.

Under realistic assumptions the heat up of the cask will take weeks until the final temperature is reached. Furthermore the reached temperatures are below temperatures harming the integrity of the casks. Nevertheless, special attention is advised for the working personal in the recovery / retrieval of the casks, as the neutron moderator material will be lost in the long run, imposing special attention to radiation protecting of the working personal.

c) Fire

As seen in the former scenario the thermal condition of the cask is an important issue. While due to an earthquake the thermal impact is on a daily/weekly scale, a fire scenario is of much shorter duration.

In the design based accidents the duration is 1 h at maximum.

Fires with longer durations than 1h are very unlikely to happen from natural sources.

Even including man made burnable substances it is highly unlikely to sustain a fire for a longer period due to the sheer amount of needed material. In addition the response time of fire-fighters is not to be expected to exceed hours.

Anyhow, assuming fires with varied duration shows that fire of several hours can lead to a cask temperature inside, so that the lid seal maximum temperature is exceeded.

In such a case the radiation release would be gradually higher than the releases calculated for the design based accidents, depending on fire duration. A temperature endangering the overall integrity of the cask is not feasible with substances viable in an accidental/disaster situation.

d) Explosions

Including explosions of sources closer and/or stronger lead basically to the same impacts as earthquakes. Moreover an explosion has a much shorter "burning" time, so that the assumptions made in the extreme fire scenario cannot be reached. And therefore the thermal/mechanical consequences are more of theoretical value than actual feasible.

In summary, Table 1 shows a collection of the results. Given the extreme scenario one can clearly see that the feasible scenarios are not leading to radiological consequences above the legal limits.

The row "feasible as accident/disaster" describes whether the accident/disaster is imaginable as natural event or accident or the overall probability of the disaster extent. Finally row "not radiological impact to the surrounding area" is assessing the probable damage to the surroundings, as for instance civilian buildings. By earthquake 1 we address the earthquake that topples the casks but does not damage the facility, while earthquake 2 is assessing an earthquake strong enough to damage and collapse the facility itself.

The disasters fall mainly in 2 categories. One is the feasible disasters, which have a very low probability but are imaginable, the other are the scenarios which are not feasible as a pure accidental situation. The first category leads to no radiological consequences exceeding the legal limits, but have disastrous effects on the surroundings. The second category is only affecting the close vicinity without damage to the civilian buildings in greater distance. The possible radiological consequences are possible in the dimension of the radiological limits. It must be kept in mind however, that due to the not feasible nature they are not relevant for a pure safety assessment situation.

Table 1**. Extreme Scenarios - Overview**

Accident/ Disaster	Radiologic consequences compared to limits	Feasible as accident/disaster	Not radiological impact to surrounding area
Flood	Very limited to none	Unlikely in the extreme extent	Severe flood damage in a large surrounding area
Earthquake 1	Safely below limits	Yes	Depending on the structural integrity, high damage to buildings in the area
Earthquake 2	Safely below limits for a timeframe of several weeks[82]	Yes	Devastating effects on even strong concrete buildings and therefore for most civilian buildings
Fire	Below limits, limits are reachable for long durations (several hours)	No[83]	Only close vicinity will be affected
Explosion	Safely below limits	No[83]	Only close vicinity will be affected

5. Conclusion

The licensing process for radioactive waste interim storage facilities in Germany as defined by law secures a traceable and reliable way for storage of radioactive waste. The safety of the facilities has to be demonstrated by the applicant and is checked by the regulatory body. This includes the normal operation as well as so called design based accidents for which has to be proven that the consequences stay safely below the regulatory limits. As presented in the article, a wide range of nature based events and man-made accidents are covered. Extrapolations of those events to extreme scenarios show that the storage facilities in Germany include a high level of safety margins. It is therefore safe to assume that the interim storage facilities and stored casks are safe for the investigated events.

[82] Depending on the time buried the release scales.
[83] The amount of burnable/explosive material is not feasible originating from an accidental situation.

Dry Interim Storage of Radioactive Material in Germany

Christian Drobniewski

Workshop on Safety Assessment of Fuel Cycle Facilities –

Regulator Approaches and Industry Perspectives

27-29.9.2011

Toronto

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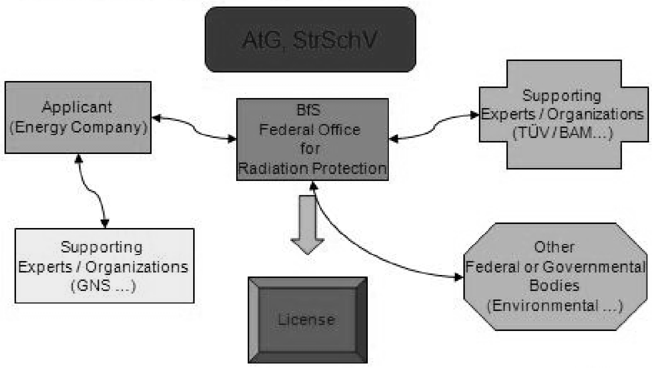
Licensing Process (Overview)



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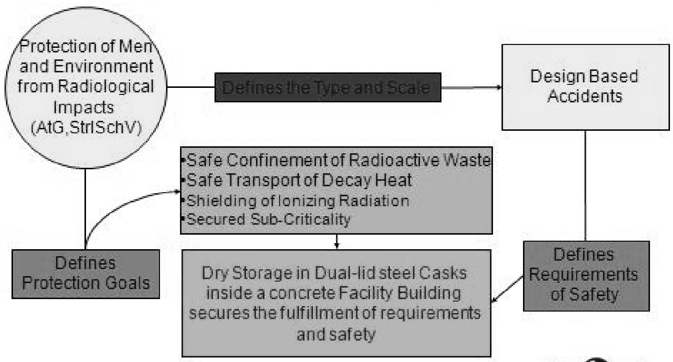
Principles for Dry Intermediate Storage in Germany



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Aspects Covered in the Licensing process

- Protection / Safety Goals
- Normal Operation (Cask handling, Surveillance, Maintenance ...)
- Design Based Accidents
- Protection of Environment
- Protection of Public (Dose limits for public etc.)
- Protection of Workers (Dose limits for working personal)

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Design Based Accidents

- Wind / Rain / Snow / Lightning ...
- Flood
- Earthquake
- Handling Accidents (Cask Dropping, Collisions etc.)
- Fire (inside / outside)

- Crash of an military Aircraft
- Explosion of combustible Gases (from outside)

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„Assessment" of Extreme Events

- Events exceeding the prior mentioned design based accidents are NOT covered in the licensing process.
- The following inquiry is based on extrapolation and good knowledge.
- Aim of the inquiry is to indentify safety margins and the grade of aggravation of event consequences.

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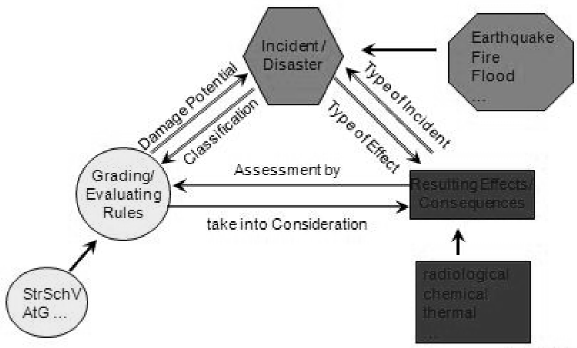
Identifying Incidents / Extreme Events (1)



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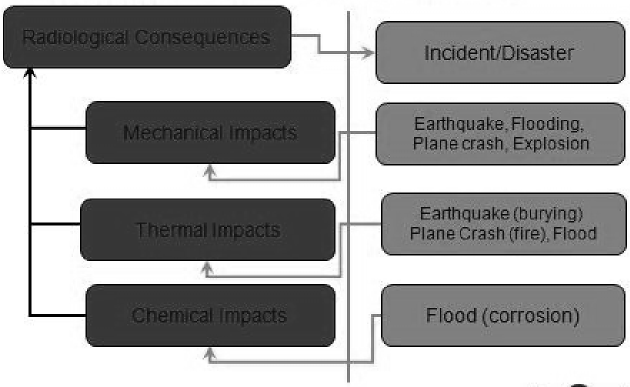
Identifying Incidents / Extreme Events (2)



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Extreme Event: Flood

Assessing the Impacts:

— **Chemical Impact:**
  - **Very little, if any, as casks are protected against corrosion and the metal thickness is way too large for corrosion in a short timeframe.**

— **Mechanical Impact:**
  - **Little to none, as casks (toppling due to flooding nearly impossible)**

— **Thermal Impact:**
  - **None as water is effectively cooling the casks.**

Resulting Consequences:

— **Very little to none**

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Extreme Event: Earthquake

Assessing the Impacts:

— Chemical Impact: none

— Mechanical Impact:
- Toppling of casks – already considered in handling accidents and conservative below effects of a airplane crash onto a cask.
- Burying of casks with debris – dropping of Building parts onto the cask is conservatively below the effects of an airplane crash.

— Thermal Impacts:
- Burying of casks with debris – reduced thermal conductivity and convectional cooling of the casks.
- Timeframe for reaching a final heat up temperature is several weeks but safely below critical values.
- Special arrangements have to be made when recovering the casks due to increased neutron radiation (loss of moderation material)

Resulting Consequences:

— Save below dose limits for several weeks of buried casks.

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Extreme Event: Fire

Assessing the Impacts:

— Chemical Impact:
- Little, at worst oxidation damages at the surface of the casks

— Mechanical Impact:
- Little to none as the cask materials have comparable heat expansion rates (except the moderation material)

— Thermal Impact:
- Depending on the fire duration the thermal impact can lead to a temperature exceeding the design temperature of the lid system
- Necessary for this is a fire duration of several hours.

Resulting Consequences:

— Gradually increasing consequences with fire duration.

— However there is no reasonable way to accumulate the amount of necessary burnable material by a natural event or accident.

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Extreme Scenarios Summary

Accident/ Disaster	Radiologic consequences compared to limits	Feasible as accident/disaster	Not radiological impact to surrounding area
Flood	Very limited to none	Unlikely in the extreme extent	Severe flood damage in a large surrounding area
Earthquake (toppling)	Safely below limits	Yes	Depending on the structural integrity, high damage to buildings in the area
Earthquake (burying)	Safely below limits for a timeframe of several weeks[1]	Yes	Devastating effects on even strong concrete buildings and therefore for most civilian buildings
Fire	Below limits, limits are reachable for long durations (several hours)	No[2]	Only close vicinity will be affected
Explosion	Safely below limits	No [2]	Only close vicinity will be affected

[1] Depending on the time buried the release scales.
[2] The amount of burnable/explosive material is not feasible originating from an accidental situation.

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OVERVIEW OF THE PAKS MODULAR VAULT DRY STORAGE PROBABILISTIC

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Abstract - Currently in Paks Nuclear Power Plant site four VVER-440/213 type PWR units are operating. The spent fuel assemblies are taken to an onsite interim storage called Modular Vault Dry Storage (MVDS) after 3 years minimum decay. The construction of the Paks MVDS has been going on in phases. The Transfer Cask Reception Building (TCRB) and a vault module including the first three vaults were constructed in 1997. Additional storage models were erected in the following years. A new vault module including the vaults 17-20 is under construction. Further enlargements of the storage facility are licensed to accommodate up to a total of 16,159 spent fuel assemblies in a total of 33 vaults.

As a regulatory requirement the Hungarian Atomic Energy Authority prescribed a PSA for the MVDS as a general design requirement. The PSA is used to justify the nuclear safety of the facility and to support the safety assessment in certain license obtaining procedures.

In 1996 the designer company (GEC ALSTHOM later Alstec) performed the first PSA for the TCRB and the 1-11 vaults as part of their Safety Assessment Report. The PSA was part of the MVDS Pre-construction Safety Report that served as a basis of the implementation licence and construction permit. It has included the usual elements, namely initiating event analysis, event tree and fault tree construction, input data analysis and quantification. The analysis covered the internal and external initiating events, operator errors, common cause/common mode failures, control system reliability issues. The presented results included point estimates of risk, the results of sensitivity and importance analyses as well as comparisons with pre-defined acceptance criteria. Because of the continuous expansion of the storage facility, and of the change of the Hungarian regulation, the PSA has been updated a number of times.

On the occasion of the last Periodic Safety Review the licensee of the MVDS decided to review the PSA considering the latest advancements in analysis techniques as well as feedback from available operating experience and data. A feasibility study has been performed which assessed how to update the whole PSA in regard to changes in safety regulation, operating experiences and to the effect of the expansion. The aim is to create an updated, living PSA model and documentation that considers the actual state of the facility state and the future expansion plans.

In the paper the main characteristics of the current PSA of the Paks MVDS, and its planned update are reviewed.

1. Description of the facility

In Hungary four VVER-440/213 type PWR units are operating at Paks site. The spent fuel assemblies from the reactors are placed in a spent fuel pool within the reactor hall for minimum 3 years. When the decay period has expired, the spent fuel assemblies are taken into the Modular Vault Dry Storage (MVDS) which is located at the adjacent site of the Paks NPP. The Paks MVDS provides for at least 50 years of interim storage for the fuel assemblies of the Paks NPP in a contained and shielded arrangement. For the scheme of the facility see Figure 1[84].

The bare fuel assemblies are stored vertically in individual fuel storage tubes, the matrix of storage tubes being housed within a concrete vault module that provides shielding. To prevent the development of corrosion and degradation processes, the fuel assemblies are housed in an inert nitrogen environment inside the storage tubes. Decay heat is removed by a once-through buoyancy driven ambient air flow across the exterior of the fuel storage tubes, through the vault and the outlet stack. There is no direct contact between the fuel assemblies and the cooling airflow.

The Paks storage facility can be divided functionally into three major structural units.

The first one is the Vault Module where the spent fuel assemblies are stored in the vertical tubes. These vault modules include a minimum of three or maximum five vaults depending on the geometrical arrangement. Each vault includes 450 storage positions. The number of the storage tubes in vaults is increased to 527 from the vault 17.

The second major structural unit is known as the Charge Hall where the fuel handling machine travels during the fuel handling operations. The charge hall is bordered by the reinforced concrete wall of the ventilation stack on the one side and by a steel structure with steel sheeting on the other side.

The third major unit is the Transfer Cask Reception Building in which the reception, preparation, unloading and loading of the transfer cask take place. The fuel handling system and other auxiliary systems are installed in the TCRB.

The spent fuel assemblies are transported to the MVDS from the spent fuel pool using the C-30 transfer cask and its railway wagon. The fuel assemblies are under water in the cask during the transportation. The transfer cask is received in the TCRB where it is removed from the railway wagon and prepared for fuel assembly unloading. The fuel is raised into a drying tube directly above the cask where the fuel assembly is dried prior to being lifted into the fuel handling machine. Each fuel assembly is transferred individually within the fuel handling machine to the vertical fuel storage tubes in the vaults. Once the fuel handling machine has moved away from the storage tube the air is evacuated from the tube and replaced with nitrogen gas. Then the storage tube is connected to the built-in nitrogen system that monitors the storage environment of the spent fuel assemblies.

The Paks MVDS is being increased in storage capacity to match the spent fuel output from the nuclear power plant. The storage capacity of the exiting 16 vaults is 7,200 spent fuel assemblies. Construction work to further extend the storage capacity by an additional 2,108 storage positions in a new storage vault module were started in 2010. Further enlargements to the storage facility are licensed to accommodate up to a total of 16,159 spent fuel assemblies in a total of 33 vaults.

[84] Paks Modular Vault Dry Storage Final Safety Analysis Report Rev 6/c, 2011

2. Current state of the PSA

The first Probabilistic Safety Assessment (PSA) documentation was part of the Final Safety Analysis report made by the facility designer company. The analysis was made for 33 vaults. As the facility expansion advanced and some changes were made in its technology and operation the PSA was updated a number of times. The most notable changes in the PSA documentation were made during the vault extension of the facility that affected most of the chapters in the PSA documentation, however the result of the analysis did not changed significantly. The most important features of the PSA for the storage facility are presented in the following.

The main objective of the safety analysis is to demonstrate that the risk of radiation from the storage facility is acceptable both for the operators and the public. To accomplish this objective probabilistic (event tree, fault tree analyses) and deterministic (radiation dose calculations) analyses are required.

Probabilistic safety assessment was performed by using the most commonly applied analysis steps including: initiating event analysis, event tree and fault tree construction, input data analysis and quantification.

The first task was to identify the possible initiating events. The final initiating event list was derived from a larger set of potential initiating events by the application of screening criteria. Two approaches were used to identify the initiating events: HAZOP (Hazards and Operability) study and assessment of the functional sub-system response to check list derived from hazards and failure types. The first method was based upon a systematic assessment of the basic operating sequence using keywords and deviations. In the second method each sub-system was assessed in turn against a list of fault and hazard types derived from check lists. The results of the analysis were several plant specific internal and external initiating events. A unique and consistent coding system was used during the whole analysis, in order to easily identify initiating events, event trees, fault trees and its elements.

Fault sequences were developed for the identified initiating events. This was done by constructing event trees using appropriate computer software (see an example in Figure 2[85]). The event tree analysis was prepared for three parts of the facility: the Transfer Cask Reception Building, the Fuel Handling Machine in the Charge Hall and the Vault modules. The aim of the analysis was to identify all fault sequences that can lead to radiological consequence and to assign a frequency values to all sequence. All sequences that had a frequency less than 10^{-7}/year were considered beyond design basis accidents and they were excluded from further analysis. The dose consequence analyses were made on a deterministic basis. Where appropriate fault trees were used to assess initiating and coincidental fault frequencies (see an example on Figure 3). The results of the analyses were fault sequence frequencies and where appropriate dose values both for operators and public. These results were summarised in fault schedule tables for every fault sequence.

In the fault tree models the operator errors were modelled using the Technique for Human Error Rate Prediction (THERP) method. During the study errors of commission type human failures were considered. The common mode failures were taken into account, rates were calculated using Beta factor method. In the calculations an average Beta value was used. The I&C system of the facility includes programmable logic controllers (PLCs). PLCs are used to provide control and single channel interlocking to prevent minor plant damage and minor injury to personnel. Where additional protection was required to prevent more severe plant damage, serious operator injury or a radiological hazard, further interlocking was incorporated. Because computer systems reliability could not be assessed

[85] Paks Modular Vault Dry Storage Final Safety Analysis Report Rev 4, 2006

adequately, the most pessimistic failure rate has been assigned to all control systems containing programmable electronic systems.

The component reliability data were collected from nuclear industry. Where it was not possible non-nuclear data were used. Since in the facility there are several non-nuclear commercial components it was allowable to use such data. Since PSA was made in the design phase no operational data were used. Data collection was performed by considering component boundaries defined for each component type. The individual component failure rate data used in the analysis were collected in tables arranged in mechanical, electrical, instrumental equipment groups (see an example in Table 1[86]). In order to perform the uncertainty analysis, distribution functions and associated parameters were required for each component failure.

Quantification of the PSA model was done by the use of computer code. It consisted of generating point estimates of risk, uncertainty calculations as well as importance and sensitivity analysis. The latter was performed mainly to determine the sensitivity of each sequence to selected parameters. The parameters were control failure, operator error and CMF. Single failure analysis was also performed. The analysis identified some single failures but the associated doses were low for these sequences.

In the summary of the analysis the results were compared with acceptance criteria for on-site and off-site consequences (allowed frequency for the different dose bands) derived from nuclear safety regulation in the UK (Safety Assessment Principles for Nuclear Facilities Rev 1, 2006). Where the calculated sequence frequencies exceeded the limit, additional dose calculations were made in order to check whether there was any unnecessary conservatism in the probabilistic analysis that could be removed. The final result of the whole analysis showed that the risk associated with the operation of the facility is acceptable both for the facility operators and the members of the public.

3. Planned update of the PSA

On the occasion of the Periodic Safety Review the licensee of the MVDS decided to review the PSA considering the latest advancements in analysis techniques as well as feedback from available operating experience and data. A feasibility study has been performed which assessed how to update the whole PSA in regard to changes in safety regulation, operating experiences and to the effect of the expansion. The aim is to create an updated, living PSA model and documentation that considers the actual state of the facility and the future expansion plans.

More specifically the planned PSA update will include:

review of the initiating events from the point of view of completeness and development of new event trees and fault trees as necessary,

review of the reliability data used in the analysis including data update by the use of operational data,

more detailed modelling of the power supply system,

extension of the analysis to internal and external hazards,

review of the risk criteria, modifications as necessary,

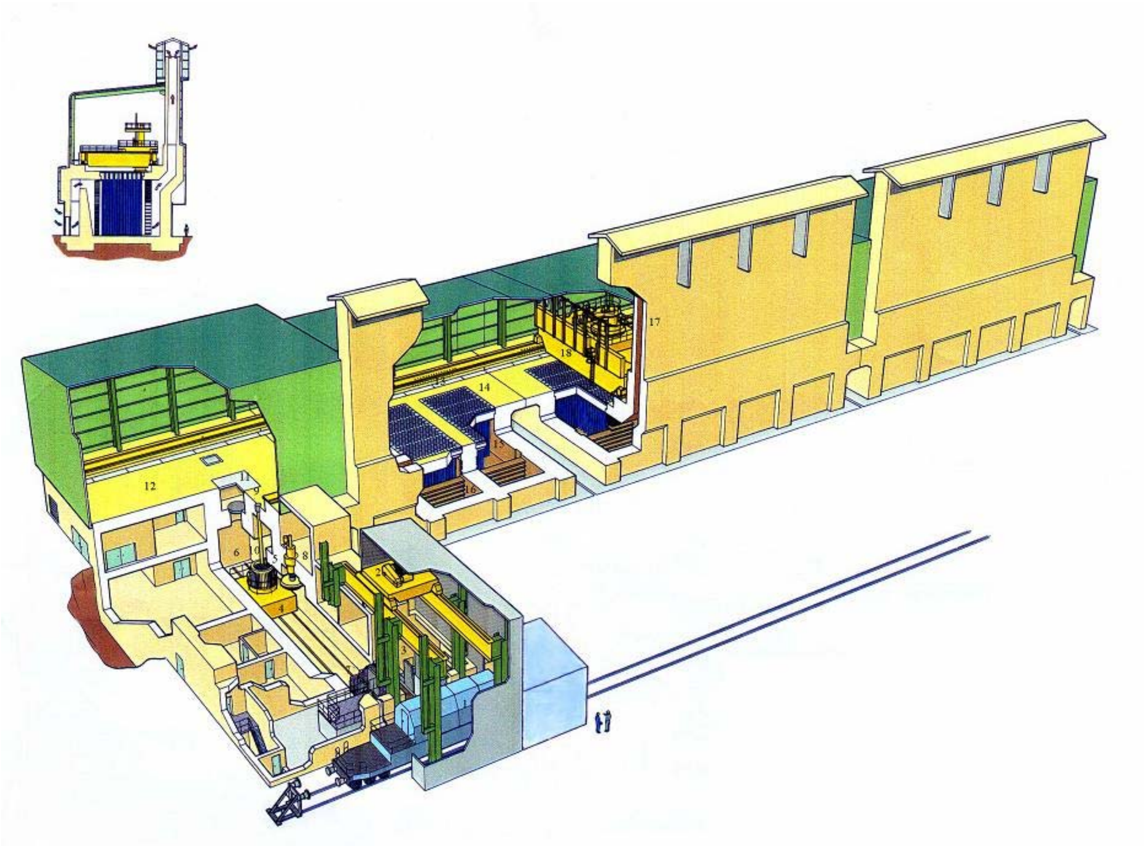
[86] Paks Modular Vault Dry Storage Stage II/III Pre-Construction Safety Report, 2004

modification of the PSA model in accordance with current operational practice, and current operating controls.

4. Conclusions

The current version of PSA and its documentation for Paks MVDS accomplishes the main objective for safety assessments, it demonstrates that the risk of radiation from the storage facility is acceptable both for the operators and the public. In order to prove that the assessment is as realistic and as complete as possible regular updates are needed. In the future our aim is to update the PSA model and documentation in several areas of the analysis. As a result of this process a living PSA programme to be initiated that considers the actual state of the facility and the future expansion plans.

Figure 1. **Scheme of the MVDS**



1 Rail Wagon
2 Cask Handling Crane
3 Cask Preparation Area
4 Cask Transfer Trolley
5 Transfer Cask
6 Fuel Drying And Unloading Cave
7 Roller Shutter Door
8 Lid Lifting Station
9 Load/Unload Port
10 Fuel drying tube
11 Maintenance Hatch
12 Charge Hall
13 Fuel Handling Machine Rail
14 Vault Modules
15 Storage Tubes
16 Collimators
17 Stack
18 Fuel Handling Machine

Figure 2. **Example of an Event Tree**

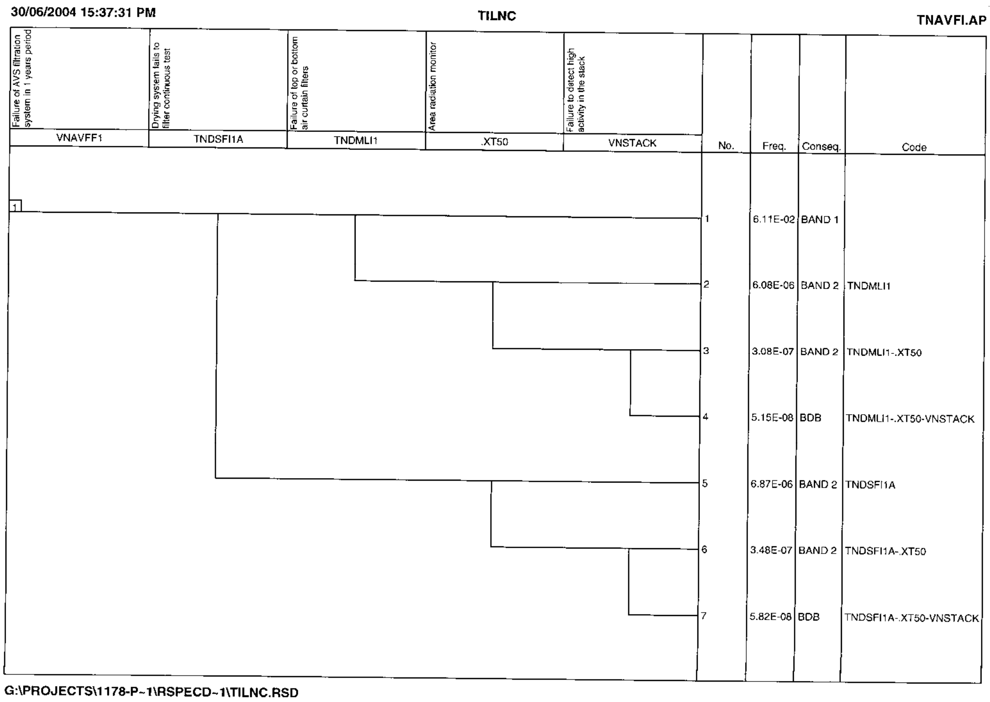


Figure 3. **Example of a Fault Tree**

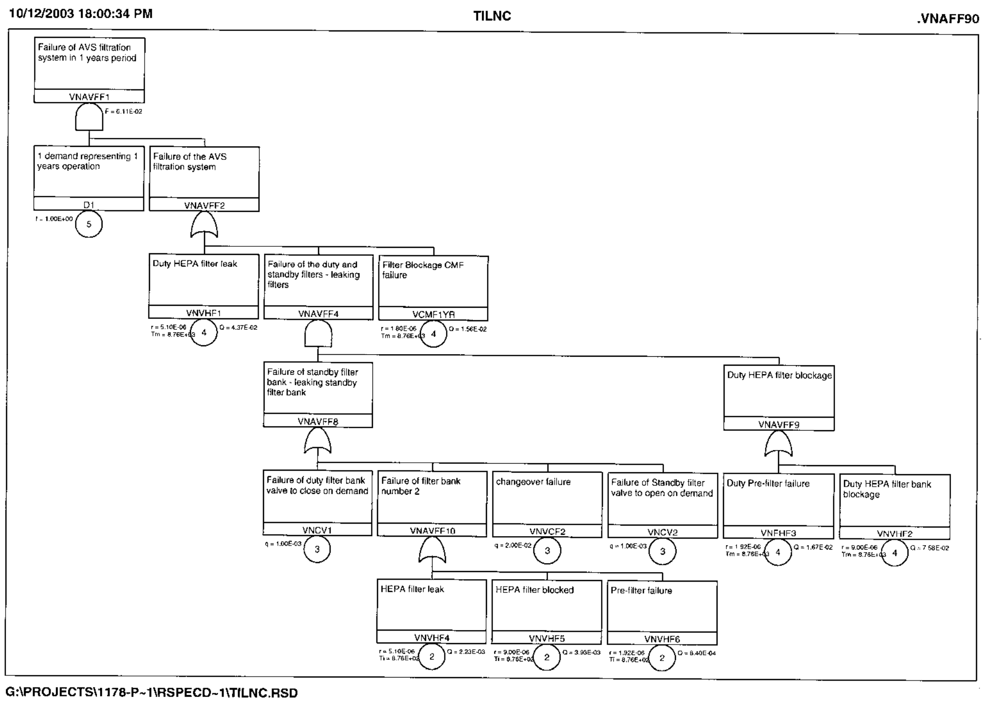


Table 1**. Example of component failure data**

Component	Failure Mode	Failure Rate or Probability (Mean)	Notes	Error Factor EF	Source (EF)	Component Type
HEPA Filter	Leak Blockage	1.7 x 10-6/hr 3 x 10-6/hr		3	Alstec	Filter
Pre-Filter	Fails to Function	6.4 x 10-7/hr		3	Alstec	Filter
Metal-Filter	Fails to Function	3.97 x 10-6/hr	All modes	10	Alstec	Filter
Ventilation Fans (Centrifugal)	Failure to start Failure to run	4.4 x 10-5/hr (includes starter) 4.2 x 10-5/hr		10 10	Alstec	Fans
Ventilation Fan (Axial)	Failure to Start Failure to Run	2.6 x 10-5/hr (includes starter) 5.7 x 10-5/hr		10	Alstec	Fans
Isolation Dampers (Manual)	Jam in position	3 x 10-3/dem or 1.0 x 10-6/hr		3 3	Alstec Alstec	Valves
Non Return Dampers	Jam in position either open or closed	1.2 x 10-6/hr		10	Alstec	Valves
Motorised Damper	Fails to Function	1.6 x 10-5/hr		3	Alstec	

NUBIKI

SOM SYSTEM mérnöki iroda

Overview of the Paks Modular Vault Dry Storage Probabilistic Safety Assessment

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NEA/CSNI Workshop
Toronto, Canada, 27-29 September, 2011

Outline

- Overview of the Paks MVDS facility
- Current State of PSA
- Planned update of the PSA

Paks MVDS PSA 27-29 September, 2011, NEA/CSNI Workshop 2

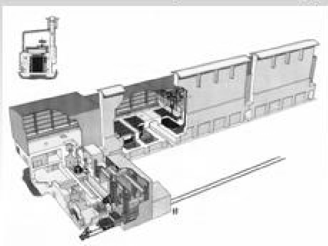
Overview of Paks MVDS facility

- Four VVER-440/213 PWR units at Paks site
- Spent fuel are placed in spent fuel pool within the plant for min. 3 years
- After decay period, assemblies are taken into the Modular Vault Dry Storage (MVDS)
- MVDS provides at least 50 years of interim storage
- MVDS is located at Paks site, near the plant

Paks MVDS PSA 27-29 September, 2011, NEA/CSNI Workshop 3

Overview of Paks MVDS facility

- The facility consists three main areas:
  - Vault Module
  - Charge Hall
  - Transfer Cask Reception Building



Paks MVDS PSA 27-29 September, 2011, NEA/CSNI Workshop 4

Overview of Paks MVDS facility

- Fuel assemblies are stored vertically in individual fuel storage tubes in inert nitrogen environment
- Decay heat is removed by ambient air flow, cooling the exterior of the fuel tube
- Capacity increase of the storage is continuous to house the spent fuel output from Paks NPP units
- Current capacity: 16 vaults, 7200 spent fuel assemblies
- Design capacity: 33 vaults, approx. 16,000 fuel assemblies

Paks MVDS PSA 27-29 September, 2011, NEA/CSNI Workshop 5

Current state of PSA

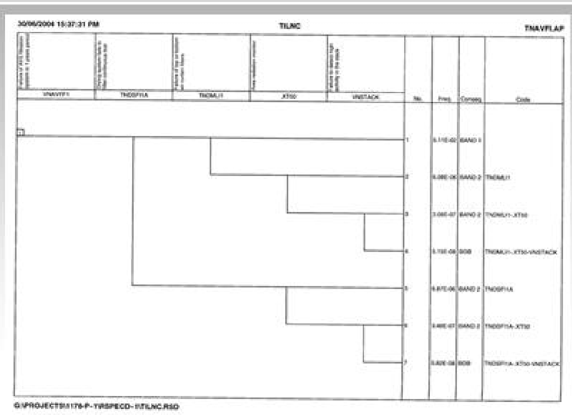
- Main objective of MVDS safety analysis is to demonstrate that risk of radiation from the facility is acceptable both for operators and the public
- To accomplish the objective deterministic and probabilistic safety assessments are required
- PSA documentation is part of the MVDS Final Safety Analysis Report
- First PSA report was produced by the facility designer
- As facility expansion advanced, PSA has been updated a number of times

Paks MVDS PSA 27-29 September, 2011, NEA/CSNI Workshop 6

Current state of PSA

- PSA was performed by using the most commonly applied analysis steps:
  - Initiating event identification
    - HAZOP study and functional sub-system analysis
  - Fault sequence development
    - Event tree construction to identify all fault sequences that can lead to radiological consequences
    - Screen beyond design basis sequences
    - Dose consequence analyses were made on deterministic basis

Paks MVDS PSA 27-29 September, 2011, NEA/CSNI Workshop 7

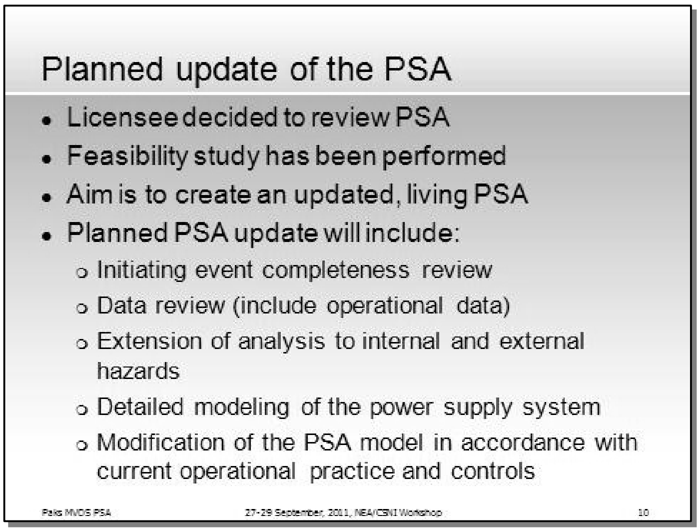


Paks MVDS PSA 27-29 September, 2011, NEA/CSNI Workshop 8

Current state of PSA

- Fault tree construction
  - Where appropriate, fault trees were used to assess initiating and coincidental fault frequencies
- Data compilation
  - Human errors (HRA using THERP method)
  - Common mode failures (Beta-factor method)
  - Component reliability data collection (including distribution functions and associated parameters)
- Quantification
  - Point estimates of risk, uncertainty calculations, importance and sensitivity analysis
  - Comparison of results with acceptance criteria

Paks MVDS PSA 27-29 September, 2011, NEA/CSNI Workshop 9


Planned update of the PSA
Licensee decided to review PSA
Feasibility study has been performed
Aim is to create an updated, living PSA
Planned PSA update will include:
Initiating event completeness review
Data review (include operational data)
Extension of analysis to internal and external hazards
Detailed modeling of the power supply system
Modification of the PSA model in accordance with current operational practice and controls
Paks MVDS PSA
27-29 September, 2011, NEA/CSNI Workshop
10

SAFETY ASSESSMENT DURING DESIGN

Pierre Nocture (AREVA)

NO PAPER AVAILABLE

AREVA

BACK-END

Safety Assessment during Design

Pierre NOCTURE
2001 WGFCS Workshop
Toronto

AREVA

CONTENT

- 1 - Back- end characteristics
- 2 – Back-end hazards
- 3 – Safety Assessment during design
- 4 – Heat Release Risk Analysis

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1- BACK- END CHARACTERISTICS (1)

	Back- end facilities	*PWR*
Distribution of radioactive / fissile materials and risks	Spread in very numerous units, equipments and rooms	In the core
Processes implementation	Chemical processes, no pressure, low temperatures	Mechanical processes, high pressure, high temperature
Energy concentration in equipment and piping	Much lower	
Physical-chemical phenomena that can lead to accidental situation	Slow kinetics	Fast kinetics
Time limits for operation in case of incidents or accidents	Higher (10 to > 100 hrs)	1 to 10 hrs
Energy released during a loss of reactivity control	much lower	

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1- BACK- END CHARACTERISTICS (2)

- **Specificities** :
  - Dispersible materials (oxide powders, gases, solutions)
  - Variation of the volumes, masses and physical-chemical nature of fissile materials during the process operations → Criticality hazard
  - Physical-chemical changes involving chemical agents → reactions, corrosion, fire, explosion hazards
  - Large diversity of processes and equipment → more diversified maintenance operations that in a reactor
  - Some processes require frequent work in glove boxes (operations involving U/Pu oxide)
  - Minor events with limited consequences

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2- BACK- END HAZARDS

- **Nuclear risks:**
  - Radioactive materials dispersion
  - External exposure
  - Criticality
  - Explosion of radiolysis gases
  - Decay heat release
- **Internal non- nuclear risks:**
  - Load handling
  - Fire
  - Internal explosion
  - Use of chemical products
  - Use of electrical equipment
  - Heating and cooling fluids
  - Pressure equipment
  - Internal flooding
  - Human factor
- **External non- nuclear risk:**
  - Earthquake /tsunami
  - External flooding
  - Meteorological conditions
  - Loss of electrical power
  - Fluids supply failure
  - Aircraft crash,
  - Industrial and transport infrastructure

» Diversity of risks and origins → safety approach of the fuel cycle back-end facilities requires being adapted to the risks and to their extent.

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2 – SAFETY ASSESSMENT (1)

- In th 80's
  - Internal risks:
    - safety assessments based on a deterministic approach generally
    - use of acceptability diagram (frequency/consequences) for particular analyses relating to cooling systems, H_2 radiolysis scavenging air systems, load handling scenarios, plant electrical power supply systems
  - Some external risks based on a probabilistic approach
- Currently, for new design :
  - Internal risks:
    - no approach based on a pre- determined probability/radiological consequences graph is retained any more for the design of a new installation
    - Practical elimination of accidents that needs emergency countermeasures to protect the public
    - the assessment of consequences is an element used to check that design has been carried out correctly
  - Some external risks based on a probabilistic approach

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2 – SAFETY ASSESSMENT (2)

- Risk by risk approach
- Essentially deterministic
- Defense in depth : succession of barriers intervening between radioactive substances, staff and environment :
  - physical barriers:
    - equipments,
    - glove- boxes,
    - rooms
  - and/ or organizational barriers:
    - monitoring devices,
    - operating procedures.

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2 – SAFETY ASSESSMENT (3)

- Risk-by-risk approach ≠ accidental situations based on approach
  - Fuel cycle back-end facilities involve diverse chemical and mechanical processes
  - The importance of the source terms depends on the technological choices and on the characteristics of equipment
  - The distribution of the source terms depends on the location of equipment inside the facilities

→ the events and accidents to be considered for design depend on the design itself › no "standard" accidental scenario › DBA can not be pre-defined (≠ reactors approach)

» The accidents to be considered for design are identified and analyzed through the safety demonstration

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2 – SAFETY ASSESSMENT (4)

About using PRA for the safety demonstration

- Probabilistic Risk Analysis (PRA) may not necessarily be appropriate for fuel back-end cycle facilities:
  - no standardized facilities
  - diversity of processes (mainly chemical processes)
  - non-Boolean failure modes associated with chemical processes
  - diversity and specificity of equipment › difficulties for data collection
  - location of radioactive material (distributed in the whole process)

  › large number of radioactive sources (rather low energy level)
  › large number of sequence of events that depend on the design itself and can not be pre-defined

- Probabilistic Risk Analysis (PRA) may not necessarily be feasible for fuel back-end cycle facilities:
  - lack of applicable failure data for equipment specially designed for reprocessing
  - difficulty for obtaining reliable values of failure probability (large uncertainties)

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2 – SAFETY ASSESSMENT (5)

About safety criteria

- No approach based on a pre- determined probability / radiological consequences graph is retained any more for the design of a new installation

› the acceptability is based on an individual basis, by taking into account of
  - the installation
  - its environment
  - the quality of the associated lines of defense…

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2 – SAFETY ASSESSMENT (6)

- Safety demonstration steps:
  - Inventory of the sources of danger:
    - where are located radioactive materials ?
    - which quantity ?
    - which physical and chemical form ?
  - Identification of the safety functions
  - Identification of the risks:
    - in normal situation
    - in incidental / accidental situations : search for the initiating events (internal or external aggression) and for scenarios which can lead to the risk
  - Risk analysis:
    - application of the defense in-depth concept,
    - to representative penalizing scenarios,
    - › definition of design (and operating) devices for:
      - prevention
      - surveillance
      - mitigation of consequences
    - assessment of radiological consequences
    - › checking of their acceptability, i.e. checking that the design is correct
    - inventory of the safety requirements (= the implementation of the above defined devices)

» The safety methodology is adapted to the risks and to their extent

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2 – SAFETY ASSESSMENT (7)

- **Safety design rules:**

The design principles of systems which have to be reliable are, as far as possible :

- redundancy : double or even triple systems (electrical power supply, cooling systems..)
- limitation of the common modes (to enhance reliability of the redundant systems)
- physical separation of the redundant systems
- functional separation : design of different systems to insure a same function as far as possible

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2 – SAFETY ASSESSMENT (8)

Safety functions (La Hague):

- containment functions for :
  - radioactive liquids
  - radioactive gases and aerosols
  - radioactive solids
  - chemical reagents
- functions ensured by structures :
  - civil engineering
  - equipment support
  - fixed shields against the external radiations
- auxiliary functions (for preservation of the containment) :
  - temperatures control
  - transfer of radioactive materials
  - maintenance and operation - lifting and handling - mobile additional shields
  - energy power and utilities supply
  - buildings ventilation
- control functions :
  - operation control
  - Health Physics

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3– EXAMPLE: HEAT RELEASE RISK ANALYSIS

- **Deterministic approach (in most of the cases)**
  - Definition of maximum acceptable temperatures for safety related materials and equipment (under normal and accidental situations)
  - Calculation of the time delay before reaching unacceptable temperatures (boiling point of a fission products tank content, temperature of in-depth damage of concrete of structures…)
  - Definition of the design safety requirements (depend on the time delay):
    - totally redundant cooling systems for short time delay
    - partially redundant cooling systems for intermediate time delay
    - no redundancy for cooling systems for long time delay

AREVA

THE NEED TO STUDY OF BOUNDING ACCIDENT IN REPROCESSING PLANT

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Abstract - There is a clear consensus that the severe accident corresponds to the core damage accident for power reactors. On the other hand, for FCFs, there is no clear consensus on what is the accident to assess the safety in the region of beyond design basis, or what is the accident which has very low probability but large consequence.

The need to examine a bounding consequence of each type of accident is explained to advance the rationality of safety management and regulation and, as a result, to reinforce the safety of a reprocessing plant. The likelihood of occurrence of an accident causing a bounding consequence should correspond to that of a severe accident at a nuclear power plant. The bounding consequence will be derived using the deterministic method and sound engineering judgment supplemented by the probabilistic method. Once an agreement on such a concept is reached among regulators, operators and related experts it will help to provide a solid basis to ensure the safety of a reprocessing plant independent of that of a nuclear power plant.

In this paper, we show a preliminary risk profile of RRP calculated by QSA (Quantitative Safety Assessment) which JNFL developed. The profile shows that bounding consequences of various accidents in a range of occurrence frequency corresponding to a severe accident at a nuclear power plant. And we find that the bounding consequence of high-level liquid waste boiling is the largest among all in this range. Therefore, the risk of this event is shown in this paper as an example.

To build a common consensus about bounding accidents among concerned parties will encourage regulatory body to introduce such an idea for more effective regulation with scientific rationality. Additionally the study of bounding accidents can contribute to substantial development for accident management strategy as reprocessing operators.

I. PURPOSE

In a nuclear power plant (NPP), safety-related structure, system, components, and management measures, providing defense in depth, are incorporated to prevent a severe accident, that is a core damage accident and the subsequent large release of radioactive material. The definition of the severe accident is clear, and its make-up has been thoroughly studied and then well understood among the concerned professionals.

In a reprocessing plant, various types of accident such as boiling due to decay heat, hydrogen explosion, solvent fire, nuclear criticality, leakage, and like are postulated. Numerous safety measures have been identified for preventing and mitigating such accidents, and so many obligations, such as strict quality assurance, configuration management and audits are added to ensure continuing their high reliability.

However, the consequences of individual accidents with the same level of likelihood of occurrence as that of the severe accident do not appear to have been sufficiently examined. (Hereinafter such a consequence is referred to as a “bounding consequence”.) If such an examination is not sufficient, the same level of requirements as those for NPPs is apt to be applied. Those requirements will be sufficient, even if not deemed necessary, since the bounding consequences should be inherently much smaller than those of severe accident.

In order to rationalize and, as a result, to reinforce the safety management and regulation, it is necessary to clarify the difference of the safety features between an NPP and a reprocessing plant. An examination of the bounding consequences of individual accidents and to identify bounding accidents should be helpful for this purpose.

The definition of the bounding consequence and the bounding accident will be given below with examples.

II. EXPLANATION OF THE BOUNDING CONSEQUENCE

(1) General safety features

(a) Modern commercial plant

The safety of a modern commercial plant has been improved greatly over the 50-years of experience with PULEX plants around the world. The state-of-the-art plant is an object of this present examination.

(b) Process shutdown to the safety state

Most abnormal events, which are caused by some process deviation or some control failure and which may escalated subsequently to accidents, can be guided to the safe state by manually or automatically stopping the process operation. Boiling accidents and radiolytically produced H_2 explosions are exceptions that require the continuous cooling and sweeping operations, respectively, because they are caused by the decay of radioactive material. These accidents resemble a LOCA at an NPP and are considered to cause greater consequences than many other accidents. However, reprocessing spent fuel are cooled for several years, for more than 4 years at the Rokkasho plant, for example, so the decay heat decrease to 2.7W per kg of spent fuel. Therefore, escalation of the initiating events in these accidents is slow and an operator can implement safety measures with a sufficient time margin, as is shown in chapter III.

(c) Slow escalation of an accident

Most initiating events escalate slowly to an accident. Even after an accident occurs, the consequence also escalates slowly in most cases. Example is shown in chapter III. Accordingly, the reliability of operator’s actions not only for preventing an accident, but also mitigating the consequences will be much higher than that in a NPP owing to this time margin.

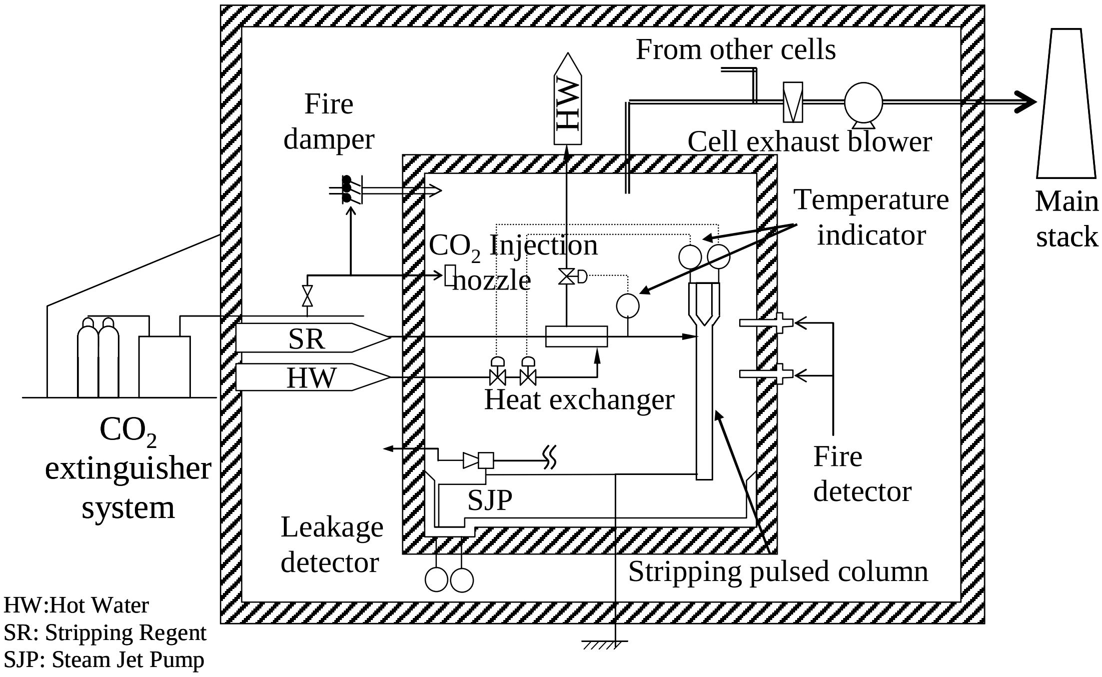
(d) Confinement for mitigating the consequence

The following confinement (see Fig.1), as opposed to the containment of an NPP, is the basis of the safety design in a reprocessing plant.

Reduced pressure control from the atmosphere, building, cell, and vessel in that order using the individual ventilation system

Reduction of radioactive aerosols in ventilation air through high efficiency particulate air (HEPA) filters installed in the individual ventilation systems and the subsequent release from a high stack

Figure 1. **Confinement structure of a reprocessing plant**



(2) Definition of the bounding consequence

The bounding consequence is different from the all uncontrolled and unmitigated consequence of an accident. The latter unmitigated consequence is determined without taken any safety measures into consideration, regardless of their reliability. On the other hand, the bounding consequence relies on highly reliable measures, which may include passive robust system, some operator actions and so on. As mentioned above, the reliability of operator actions will be quite high if there is a sufficient margin of time for an operator to correct his error.

The bounding consequence used here has a likelihood corresponding to that of the severe accident. Although it should be derived by the deterministic method and sound engineering judgment, the PRA result with respect to the frequency of occurrence is very helpful.

Although various types of accidents have to be assessed using PRA, the accident scenarios are very simple compared to those of the severe accident. For this reason, we had developed QSA and refer to the frequency of occurrence derived from QSA in this study.

III. EXAMPLES OF BOUNDING CONSEQUENCE

The below examples are provided to clarify the concept of the bounding consequence. The safety measures used here are nearly the same as those of the Rokkasho plant.

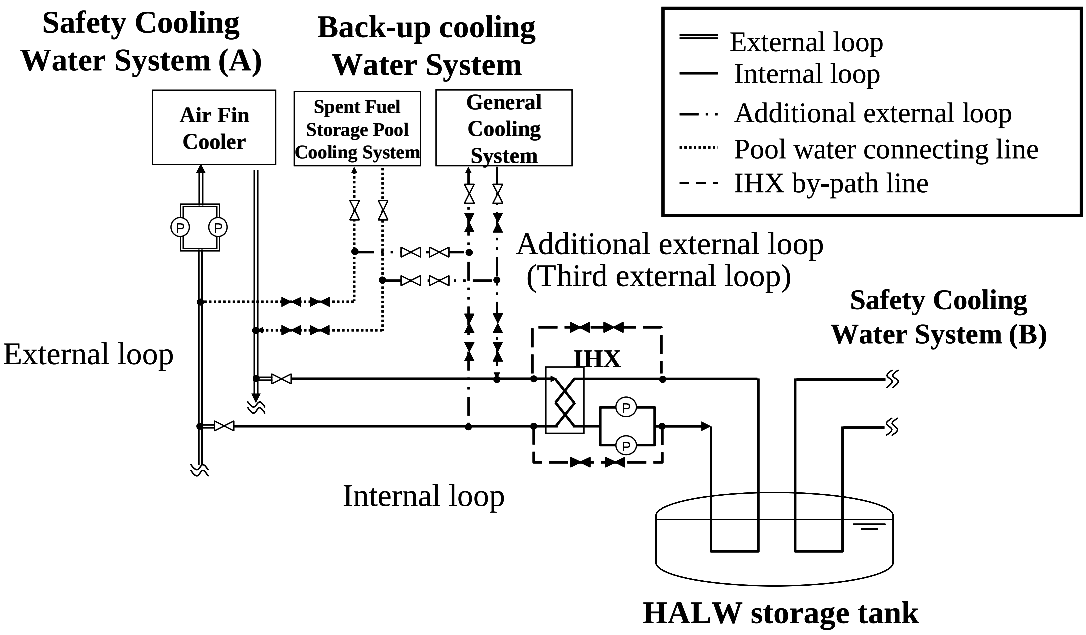
Safe storage of high-level liquid waste (HLLW) in tanks is one of the most important safety issues in a reprocessing plant, because most radioactive materials in solution are present in those tanks. Such tanks are used as buffers before HLLW is vitrified. If the cooling function of the tank is lost completely, the solution will begin to boil due to the decay heat of radioactive material, and radioactive aerosols in the steam will be discharged into the environment, since a large amount of steam will damage the HEPA filter. Due to the consequence importance of safety, PRAs were performed in the UK[1], France[2] and Japan[3] to confirm the adequacy of the design of the cooling system.

(1) Safety measures

It is assumed that two tanks and one spare tank (120m^3 capacity per tank) are installed. The decay heat of HLLW is 5kW/m^3 which was calculated assuming a 45,000MWd/t burn-up, 6-year cooling on average and 0.4m^3 HLLW/(t of spent fuel). In the event of a total cooling-loss, the HLLW would boil after 15 hours.

The cooling system is shown schematically in Fig.2, which is composed of two independent external cooling lines A and B, and two independent internal cooling lines a and b. Two pumps are installed in each cooling line, one running and the other on standby. Those lines A-a and B-b are connected to two independent electric power supply lines A and B, each with its respective backup diesel generator (DG). Each line has 100% cooling capacity.

Figure 2. **Model cooling system of HLLW tank**



(a) Cooling in normal operation

Although only one line A-A-a or B-B-b is sufficient to maintain safety, both lines are working.

(b) Cooling in an emergency

In the event that two internal pumps (one running and the other on stand-by) should fail, external cooling water can be introduced into a corresponding internal line bypassing the heat exchanger through manual valve operation.

In the event that an internal line (pump failure or leakage) should fail, the HLLW can be transferred by means of a steam-jet pump to a spare tank with independent internal lines.

If the function of two-lines is lost, and at least one internal line has no leakage, cooling water from the cooling system for the spent fuel storage pool can be introduced into the internal line. The pool cooling system also has two independent lines, each with 100% cooling capacity, and is independent from the HLLW cooling system.

(2) Accident scenario

The initiating event is a stoppage of normal operation. Since two lines are working in normal operation, two or more failures are necessary to cause the initiating event. In the event of the failure of all the effective emergency measures described above subsequent to the initiating event, a total cooling loss occurs.

(3) Calculation of the occurrence frequency

The failure rates relating to components such as the electric power supply, pump, instrumentation and control, and human reliability used in our PRAs are primarily taken from those prepared for American NPPs, where the types of components and working environments resemble one another. Some parameters were based on the operational experience of the Sellafield and Tokai reprocessing plant.

The frequency of the total cooling loss was calculated as 5×10^{-8}/y. Example is shown in chapter IV.

(4) Derivation of the bounding consequence

On the assumption of the continuation period of boiling of the two tanks (240m^3) for 24 hours, the effective dose for the nearest offsite public is obtained as 30mSv based on the five factor formula[4] and Gaussian plume model with the following parameters.

a) MAR x DR = 1m^3 HLLW per 1 hour boiling (600kW)

b) ARF = 10^{-4}

c) LPF = 1 (although LPF for the ventilation duct without HEPA filter is reported[2] as 4×10^{-2})

d) RF = 1 (conservative value)

e) Relative concentration X/Q (for 24 hours) = 5.8×10^{-7} (s/m^3) at 97%

meteorology at the Rokkasho plant with a 150 m stack

IV. QUANTITATIVE SAFETY ASSESSMENT (QSA)

Based on the FTA results of utility facilities, etc. obtained from detailed PRA, a QSA method, which can systematically evaluate the risks of various accidents that are postulated at the reprocessing plant (Hereinafter such a evaluated risk distribution is referred to as a “risk profile”.) and the importance of systems, components and human action, has been developed.

(1) Outline of QSA[5]

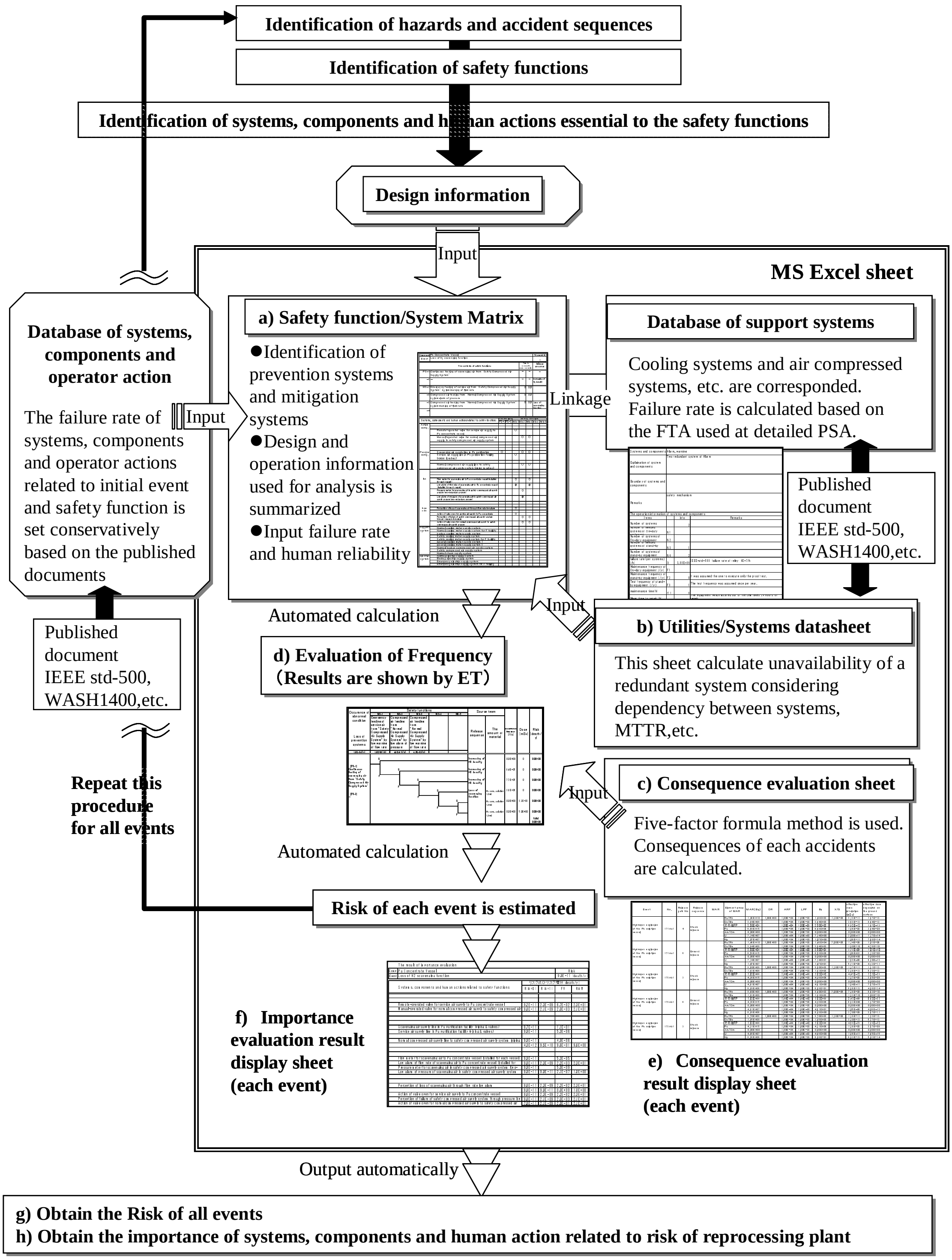
QSA is possible to use the MS Excel sheet for the evaluation of simplified PRA. The formulas corresponding FTA are inputted beforehand in the Excel sheet. Therefore, an analyst could evaluate only by adding design information related to initiating events and safety functions. Thus, this method provides high consistency throughout all assessment works of more than six hundreds events. It consist of several MS Excel sheets as working sheets, (a) Safety function/System Matrix sheet, (b) Utilities/Systems datasheet, (c) Consequence evaluation sheet, and as output sheets, (d) Event tree display sheet for each event, (e) Consequence evaluation result display sheet for each event, (f) Importance evaluation result display sheet for each event (g) Risk of all events display sheet, (h) Importance related to risk of reprocessing plant.

First of all, accident sequences and safety functions should be identified. Afterward, systems, components, and human action related to safety functions are identified from the design and operation information. As thus far explain, there is no difference from detailed PRA procedure. Design and operation information used for reliability analysis is summarized in (a) Safety function/System Matrix sheet. Analyst inputs prescribed failure rates for system, component and human action related to initiating events and safety function into the (a) Matrix selecting from a list “Database of system, component and human action” that have been set conservatively based on the published documents (IEEE-std 500, NUREG-1363, etc.). Human error rate was determined based on detailed PRA. In case of a redundant system such as blowers, analyst enters the operating information such as component type, redundancy, maintenance period and test interval, and so on into (b) Utilities/Systems datasheet in order to calculate the system’s unavailability. The calculated result is entered into (a) Safety function/System Matrix.

Unavailability of such support systems as utilities, which related to many events, has been set in (a) Safety function/System Matrix sheet beforehand, with simplified fault tree equation based on detailed PRA results. The other information such as time margin to accident occurrence from initiating event, relation of initiating events and detection should be entered in (a) Safety function/System Matrix sheet. Finally by running macro program of it, calculated result is automatically shown in form of event tree in (d) Event tree display sheet.

Occurrence frequency, consequence and risk of all accidents is summarized in (g) Risk of all events display sheet by accumulating QSA for all events. Total risk of reprocessing plant and risk profile will be obtained from this spreadsheet.

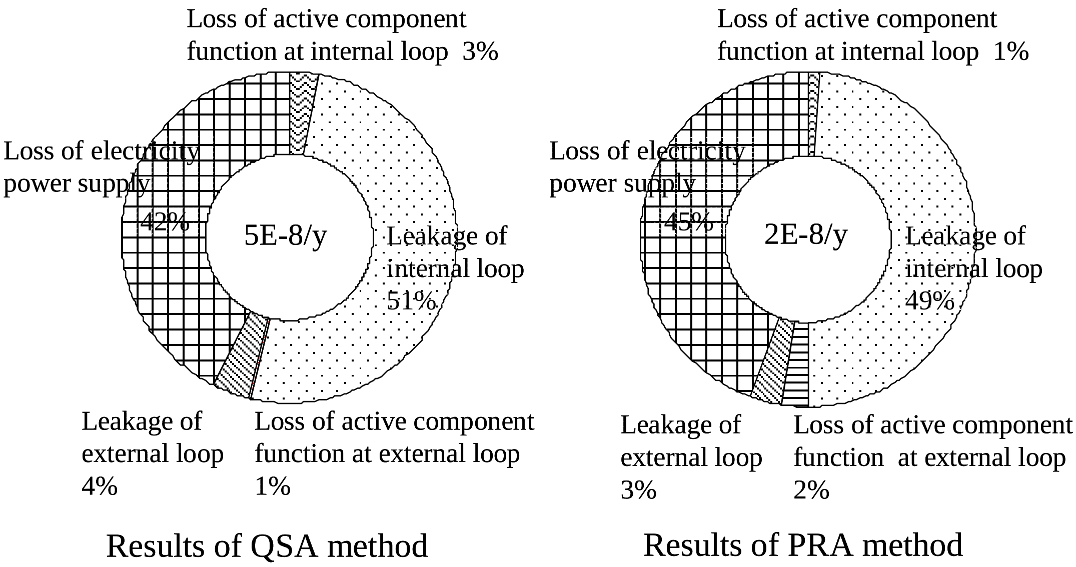
Figure 3. **Flow chart of QSA procedure**



(2) Comparison of the Results of QSA and Detailed PRA Method

As results of the QSA method and the detailed PRA, frequency and fraction of initiating events of "cooling loss of HLLW tank" are shown in Fig 4. The frequency calculated by the QSA method is 5E-8/year and well agrees with the result 2E-8/ year of detailed PRA method.

Figure 4. **Comparison of the result of cooling loss HLLW tank**



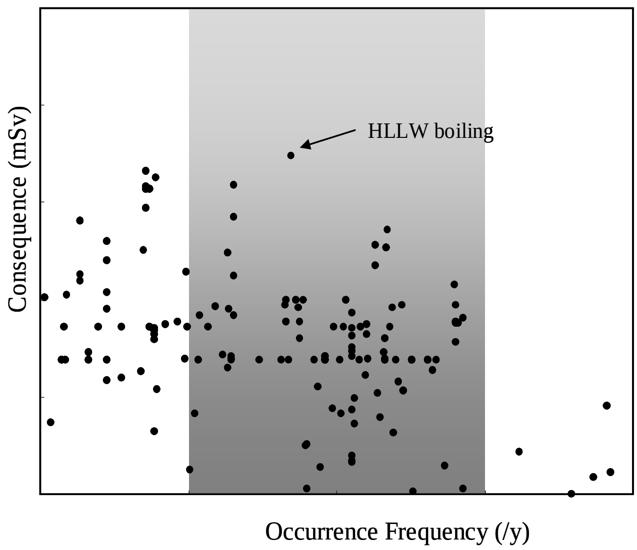
V. IDENTIFICATION OF THE BOUNDING ACCIDENT AND APPLICATION

A risk profile will be obtained from the QSA and shown on a figure of distribution with X-axis of occurrence frequency and Y-axis of consequence.

We consider it is a useful tool to identify the bounding consequences of the reprocessing plant.

An example of preliminary risk profile is shown as Fig. 5.

Figure 5. **Example of a Risk Profile**



(1) Identification of the bounding accident

The profile shows that bounding consequences of various accidents such as HLLW boiling, in-cell solvent fire, criticality accident, and the like in a range of occurrence frequency corresponding to a severe accident at a nuclear power plant. And we find that the bounding consequence of HLLW boiling is the largest among all in this range.

(2) Application of the bounding accident information

To build a common consensus about the bounding accident among concerned parties will encourage regulatory body to introduce such an idea for more effective regulation with scientific rationality.

We are going to study about follow issues.

a) Performance goal of the reprocessing plant will be discussed referring the bounding accident information.

b) Rational inspection activities will be performed by using the performance goal which endorsed by the regulatory body.

c) Additionally the study of bounding accidents can contribute to substantial development for accident management strategy as reprocessing operators.

References

[1] P. W. Ball, G. Hensley, “Reliability analysis and the use of probabilistic risk assessment in the storage of highly radioactive liquid wastes, “Paper presented at the Fourth EuReDatA Conf., Venice, Mar. 1983.

[2] J. P. Mercier, F. Bonneval, M. Weber, “Application of the Probabilistic Approach to the UP3A Reprocessing Plant,” IAEA-TECDOC-711, 95 (1993).

[3] T. Miyata, et al., “Application of probabilistic safety assessment to Rokkasho Reprocessing Plant (II), The occurrence frequency of boiling accident of highly active liquid waste,” Transactions of the Atomic Energy Society of Japan, 7, 85 (2008)

[4] “Nuclear Fuel Cycle Facility Accident Analysis Handbook,” NUREG/CR-6410, March 1998.

[5] K. Takebe, et al., “The Experience of Risk Assessment and its Future Utilization at Rokkasho Reprocessing Plant,” Workshop on Fuel Cycle Safety: Past, Present and Future, OECD NEA, Oct. 16-18, 2007

THE NEED TO STUDY OF BOUNDING ACCIDENT IN REPROCESSING PLANT

September 27-29, 2011

Mr. Satoshi Segawa, Mr. Kunio Fujita

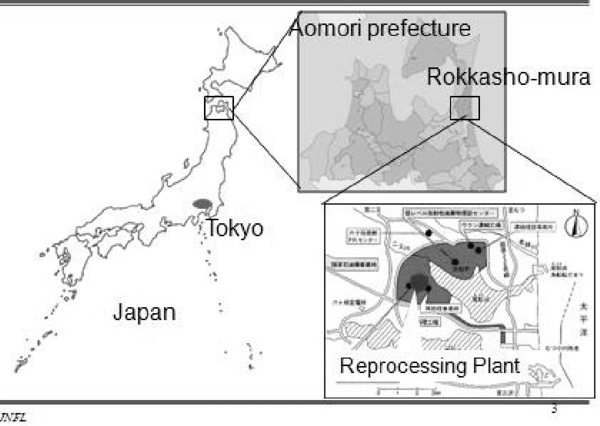
Safety Technology Office, Japan Nuclear Fuel Limited (JNFL).

JNFL 1

Contents

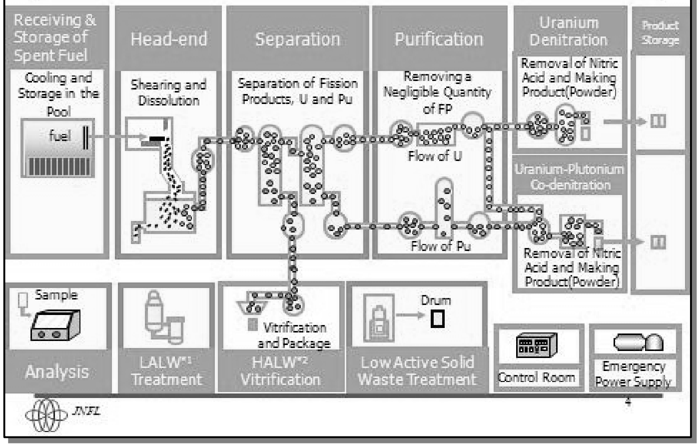
JNFL 2

Location of Rokkasho Reprocessing Plant



JNFL 3

The outline of Rokkasho Reprocessing Plant



Purpose of this study

- Back ground
  - In a NPP, the definition of the severe accident is clear.
  - In a reprocessing plant, various types of accident are postulated.
    (there is no consensus about like the severe accident)
  - The consequences of individual accidents* do not appear to have been sufficiently examined.

* These accident's occurrence frequency are extremely low depending on the robust design.
For this reason, we have assumed these accidents dose not occur in deterministic safety assessment.

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Purpose of this study

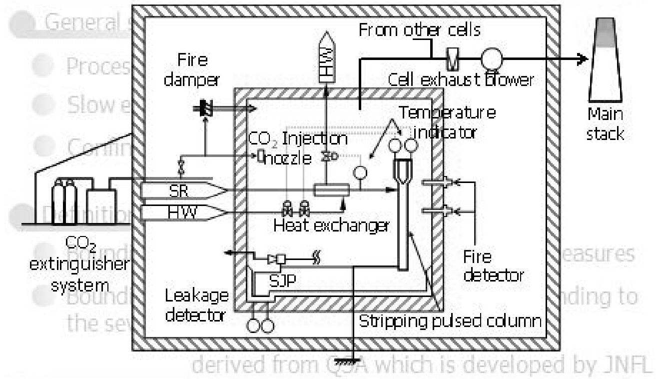
- purpose
  - To rationalize and reinforce the safety management and regulation
  - ⇨ An examination of bounding consequence and to identify bounding accidents should be helpful

⇩

Bounding consequence
Using the five factor formula in consideration of robustness of safety functions
Occurrence frequency of individual accident
Using the QSA (Quantitative Safety Assessment) which has been developed by JNFL to assess a occurrence frequency of many accidents systematically

JNFL 6

Explanation of the bounding consequence



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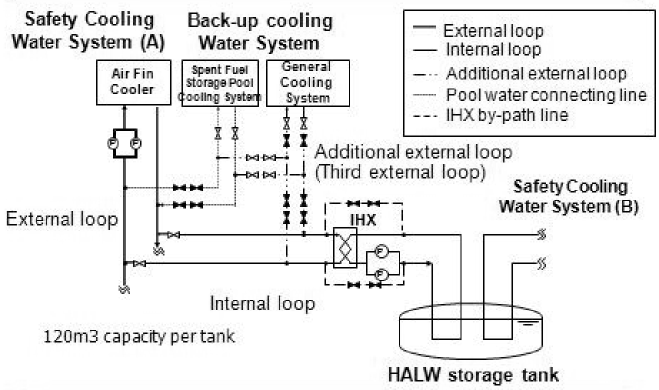
Example of bounding consequence

JNFL 8

Safety measures



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Source term evaluation method

- Source term evaluation for reprocessing plant has adopted the five factor formula* developed by USA which is used in QSA.
- It is also adopted in QSA

$$ST = MAR \times DR \times ARF \times RF \times LPF$$

Where,
MAR:Material At Risk
DR:Damage Ratio
ARF:Airborne Release Fraction
RF:Respirable Fraction
LPF: Leak path Factor

* "Nuclear Fuel Cycle Facility Accident Analysis Handbook," NUREG/CR-6410 (1998).

Dose evaluation for the general public

- The radiation dose for the general public is calculated based on conventional Gaussian Plume model.

$$Di = ST \times R \times X/Q \times H$$

Di:Effective Dose due to inhalation
ST:Source Term
R:Inhalation ratio
X/Q:Relative concentration
H:Coefficient for ingestion and inhalation

Development and application of the Quantitative Safety Assessment (QSA)

Development of QSA

Detailed PSA

- FTA results of utility facilities
- Human reliability Analysis, etc.

Comparison of results of Simplified PSA and Detailed PSA Method

Results Incorporated

QSA has been developed

- It can evaluate systematically evaluate the risk
- It can evaluate the importance of systems, components and Human action
- It is possible to use the MS Excel sheet for all evaluation
- The formula corresponding FTA are inputted in the sheet

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Outline of QSA procedure 1

QSA consist of several MS Excel sheet as follow,

Input sheets
(a) Safety function/System Matrix
(b) Utilities/Systems datasheet
(c) Consequence evaluation sheet

Output sheets
(d) Event tree display sheet for each event
(e) Consequence evaluation result display sheet for each event
(f) Importance evaluation result display sheet for each event
(g) Risk of all events display sheet
(h) Importance related to risk of reprocessing plant

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Outline of QSA procedure 2

Identification of hazards and accident sequences

Identification of safety functions

Identification of systems, components and human actions essential to the safety functions

Design information

Design and Operation Information input and summarized in (a) Safety function/System Matrix

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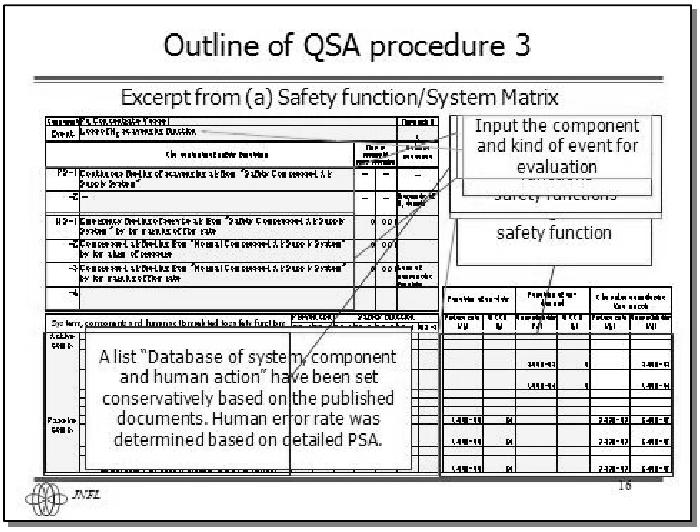
Outline of QSA procedure 3
Excerpt from (a) Safety function/System Matrix
Input the component and kind of event for evaluation
safety function
A list "Database of system, component and human action" have been set conservatively based on the published documents. Human error rate was determined based on detailed PSA.
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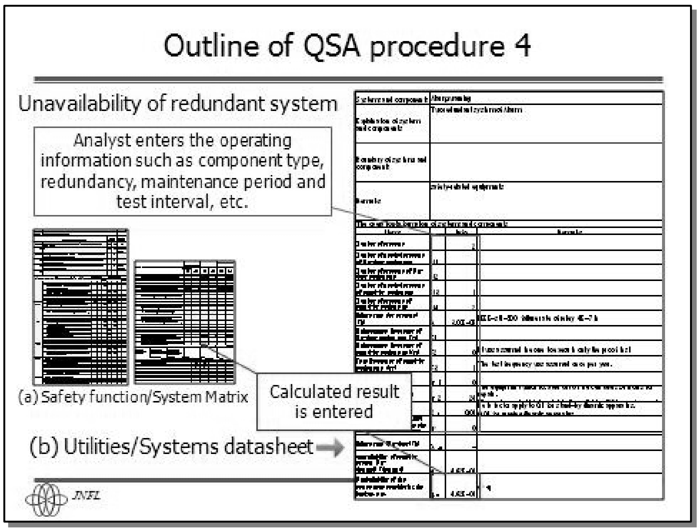
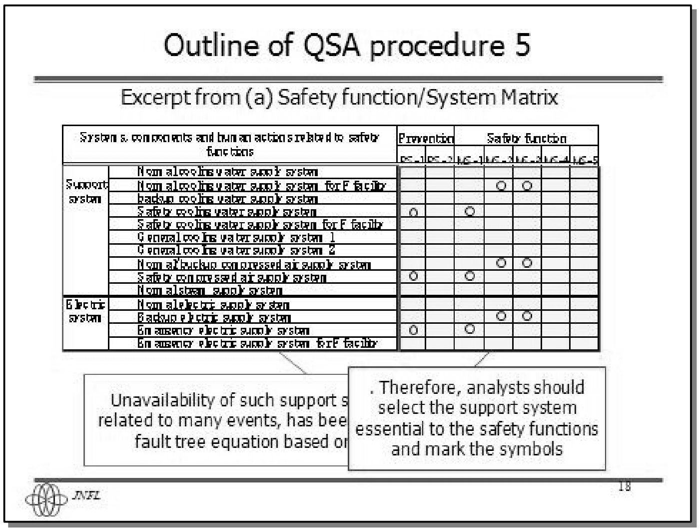

Outline of QSA procedure 4
Unavailability of redundant system
Analyst enters the operating information such as component type, redundancy, maintenance period and test interval, etc.
Calculated result is entered
(a) Safety function/System Matrix
(b) Utilities/Systems datasheet
JNFL


Outline of QSA procedure 5
Excerpt from (a) Safety function/System Matrix

Systems, components and human actions related to safety functions		Prevention		Safety function				
		PS-1	PS-2	MS-1	MS-2	MS-3	MS-4	MS-5
Support system	Normal cooling water supply system							
	Normal cooling water supply system for F facility				○	○		
	backup cooling water supply system							
	Safety cooling water supply system	○		○				
	Safety cooling water supply system for F facility							
	General cooling water supply system 1							
	General cooling water supply system 2							
	Normal/backup compressed air supply system				○	○		
	Safety compressed air supply system	○		○				
	Normal steam supply system							
Electric system	Normal electric supply system							
	Backup electric supply system				○	○		
	Emergency electric supply system	○		○				
	Emergency electric supply system for F facility							

Unavailability of such support s
related to many events, has bee
fault tree equation based or
. Therefore, analysts should select the support system essential to the safety functions and mark the symbols
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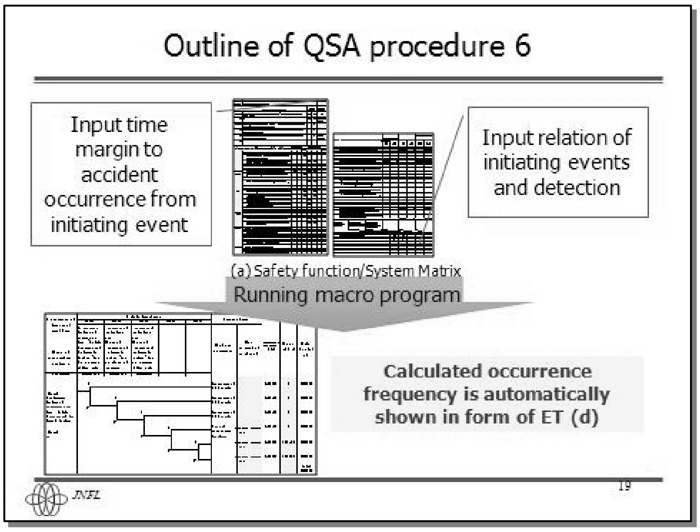
Outline of QSA procedure 6
Input time margin to accident occurrence from initiating event
Input relation of initiating events and detection
(a) Safety function/System Matrix
Running macro program
Calculated occurrence frequency is automatically shown in form of ET (d)
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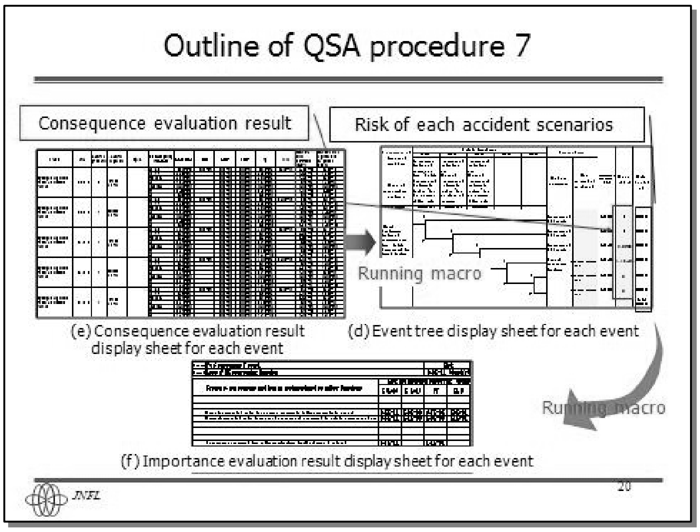
Outline of QSA procedure 7
Consequence evaluation result
Risk of each accident scenarios
Running macro
(e) Consequence evaluation result display sheet for each event
(d) Event tree display sheet for each event
Running macro
(f) Importance evaluation result display sheet for each event
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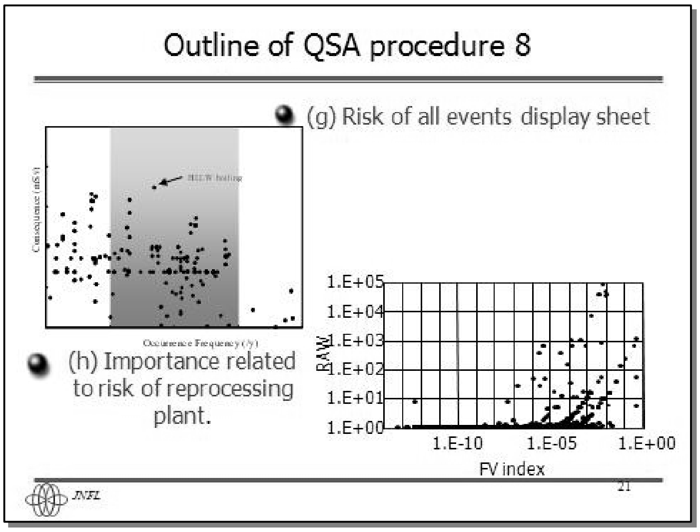
Outline of QSA procedure 8
(g) Risk of all events display sheet
HLLW boiling
Consequence (mSv)
Occurrence Frequency (/y)
(h) Importance related to risk of reprocessing plant.
1.E+05
1.E+04
1.E+03
1.E+02
1.E+01
1.E+00
RAW
1.E-10
1.E-05
1.E+00
FV index
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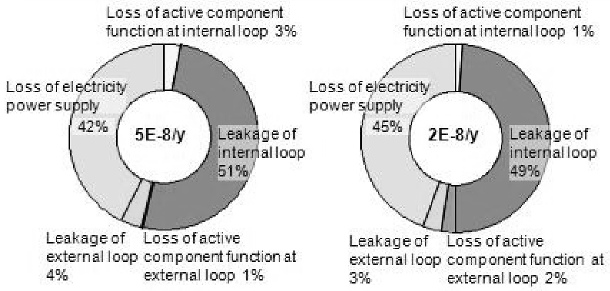
Comparison of the Results of QSA and Detailed PSA



Results of QSA method   Results of PRA method

Result of QSA almost agree with PRA

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conclusion

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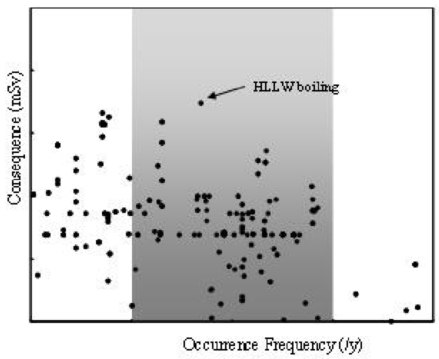
A risk profile



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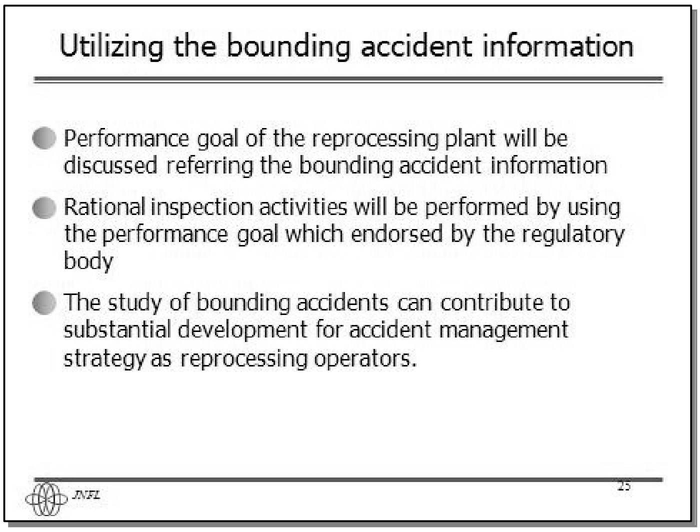
Utilizing the bounding accident information
Performance goal of the reprocessing plant will be discussed referring the bounding accident information
Rational inspection activities will be performed by using the performance goal which endorsed by the regulatory body
The study of bounding accidents can contribute to substantial development for accident management strategy as reprocessing operators.
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STUDY OF HYDROGEN CONSUMPTION REACTION CATALYZED BY PD IONS IN THE SIMULATED HIGH-LEVEL LIQUID WASTE.

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Abstract - To ensure the safety for storage of high-level liquid waste (HLLW) in tanks is one of the most important safety issues in a reprocessing plant since almost all radioactive materials under processing are collected in these tanks.

Accordingly the behavior of radiolytically formed hydrogen (H_2) in these tanks is one of key issues and has been studied by several researchers because it might cause an explosion. They reported that not all of H_2 formed in HLLW comes out in the gas phase because H_2 is consumed by some unclarified secondary reaction which may be caused by the irradiation and/or by the catalytic effect of certain fission product (FP) in HLLW.

In order to clarify such effect, we carried out the experiments using the simulated high level liquid waste (SHLLW) with and without palladium (Pd) group ions under irradiation and non-irradiation conditions. As a result, it was found that H_2 consumption reaction is not caused by radiation as was understood so far but is caused by a catalytic effect of Pd ion in SHLLW. That is, H_2 is reacting with HNO_3 and forming H_2O and NOx.

Using the catalytic reaction rate constant measured in the experiments, the analysis showed that the H_2 concentration in the gas phase of an HLLW tank does not reach its explosion limit of 4% even if the sweeping air stops for a long time.

I. INTRODUCTION

To ensure the safe storage of high-level liquid waste (HLLW) in tanks is one of the most important safety issues in a reprocessing plant since most radioactive materials under processing are present in these tanks.

The major causes jeopardizing the safe storage are cooling loss leading to boiling of HLLW.

In the present study the behavior of H_2 was examined aiming to increase the understanding of the safety of tank storage because explosion of hydrogen may be one of the cause of leading to loss of cooling system.

The behavior of radiolytically (by means of radiolysis) formed hydrogen (H_2) in these tanks was reported that not all of H_2 formed in HLLW comes out in the gas phase[1-6] because H_2 is consumed by some unclarified secondary reaction.

Now it has been considered that the H_2 consumption reaction in HLLW is mainly caused by the radiation effect. Though, the components of HLLW are included of α nuclides and β nuclides, the H_2 consumption reaction is not applied as α radiolytically formed hydrogen because Kuno et al. found that the remarkable consumption was not observed in the $Pu(NO_3)_4$ solution.[7]

In this report, we described cause of H_2 consumption reaction and their quantitative effect in HLLW, respectively (Described in Chapter. II) and analysis using the H_2 consumption reaction rate in HLLW (Described in Chapter. III).

II. STUDY OF H_2 CONSUMPTION REACTION

Two candidates of mechanism are envisaged as the H_2 consumption reaction. One is the radiation effect, nd the other is the catalytic effect of FP components in HLLW.

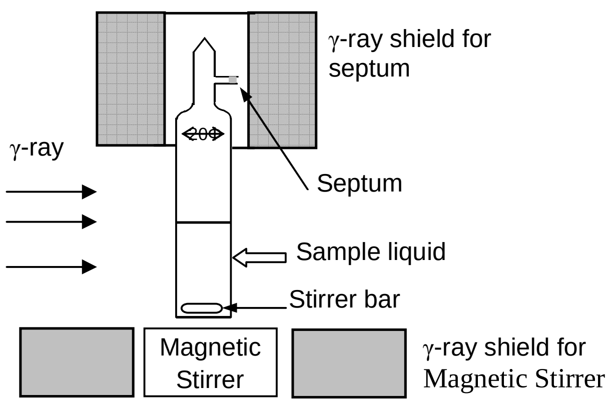
II.A. Radiation Effect

In order to confirm the radiation effect we carried out the following experiments.

Experimental apparatus is shown schematically in Fig. 1. Two conditions (still-standing and the agitation) of pure water and HNO_3 were introduced into glass ampoules. The ampoules were sealed and were irradiated with ^{60}Co γ-rays. Gas was sampled through the septum by a syringe at every one irradiation hour and H_2 concentration was measured by the gas chromatography.

After the end of irradiation, the gas and liquid phases were mixed by shaking the ampoules to obtain H_2 equilibrium between the both phases and the H_2 yield was calculated from the concentration measured by the gas chromatography and its gas volume.

Figure 1**. Vertical cross section of apparatus to measure G(H_2) of solution**



In case of the stir case, the radiation effect for H_2 consumption reaction, which has been thought that H2 is consumed by the reaction with OH radicals, will not appear because H_2 is quickly released in the gas phase of the ampoule from the solution. Contrarily, in case of the still-standing case, the radiation effect will appear if it works because H_2 gas retention time is longer than the stir case.

Hydrogen yields from the described samples are plotted in Fig. 2. The results for HNO_3 solution are quite different from those for pure water. As can be seen from Fig. 2b, the still-standing sample with high volume solution, a significant quantity of H_2 is accumulated in the liquid phase of HNO_3 solution during irradiation although it can be easily released into the gas phase by shaking. After all, the H_2 yield of the stir case is equal to the still-standing case after shaking of the solution. That is, H_2 consumption reaction does not work in the HNO_3 solution.

On the other hand, in case of the pure water, H_2 yields of the still case are smaller than the stir case. This means the radiation effect for H_2 consumption reaction is worked in the pure water. In pure water, the rapid consumption reaction proceeds through

$$H_2 + OH \rightarrow H + H_2O, \quad k = 4.2\times10^7\ (\mathrm{mol}/\ell)^{-1}\ s^{-1}$$

where OH radical is produced by radiation.[8] Therefore, H_2 produced in the liquid phase is consumed by the above mentioned reaction

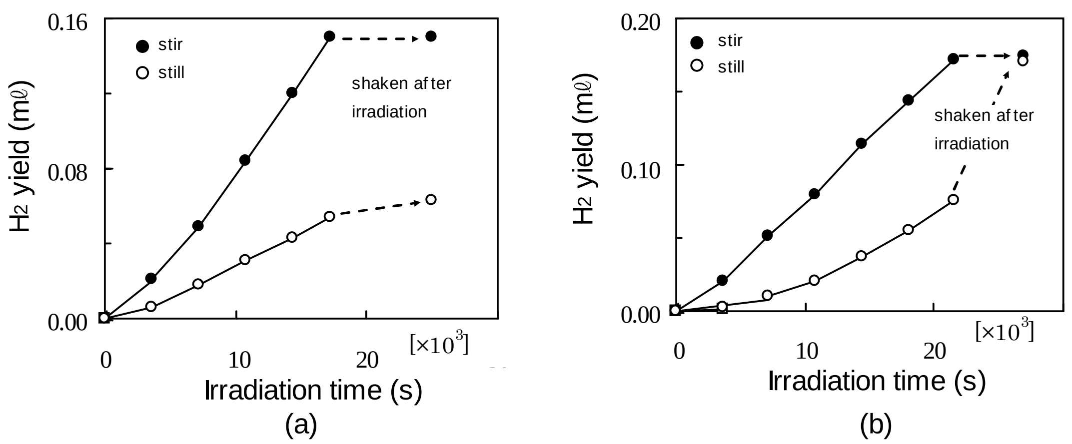
On the other hand, the reason why the radiation-induced consumption reaction does not seem to work in nitric acid solutions are explained by following reaction which has higher reaction rate constant than the above equation.

$$.\ OH + HNO_3 \rightarrow NO_3 + H_2O, \quad k = 1.4\times10^8\ (\mathrm{mol}/\ell)^{-1}\ s^{-1}$$

In nitric acid solutions, OH radical consumes efficiently with molecular HNO_3 forming NO_3 radical.[9]

Thus the radiation effect for H_2 consumption reaction does not work in HLLW which contains nitric acid solutions in case that there is no catalytic effect of FP (Pd) as examined in the following figure.

Figure 2. **H_2 yields by γ-irradiation of (a) pure water, (b) HNO_3(2 mol/ℓ) of different voluemes under still-standing and stirred conditions**



II. B. Catalytic Effect of FP (Pd)

In order to confirm the catalytic effect of FP components, we carried out the following experiments.

In this experiment SHLLW without palladium (Pd) group ions and SHLLW with Pd group ions were used. In addition, the solution containing Pd, rhodium, and ruthenium ions together was prepared.

Apparatus to measure the H_2 consumption rate is shown schematically in Fig. 3. Sample solution was taken in a glass ampoule, and 1.2 % H_2/air was bubbled in solution.

The ampoule was then sealed and was mounted horizontally on a shaker. The ampoule was shaken along the longitudinal. The shaking was stopped at a regular interval to sample gas for the measurement of H_2 concentration by the gas chromatography.

Figure 3**. Glass ampoule before (A) and after (B) the heat-seal used to measure H_2 consumption reaction rate**

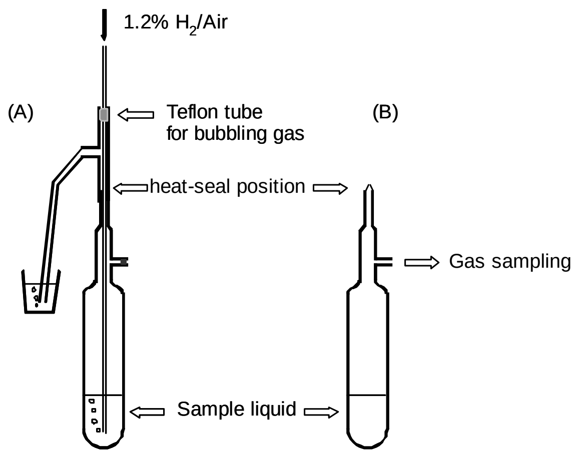
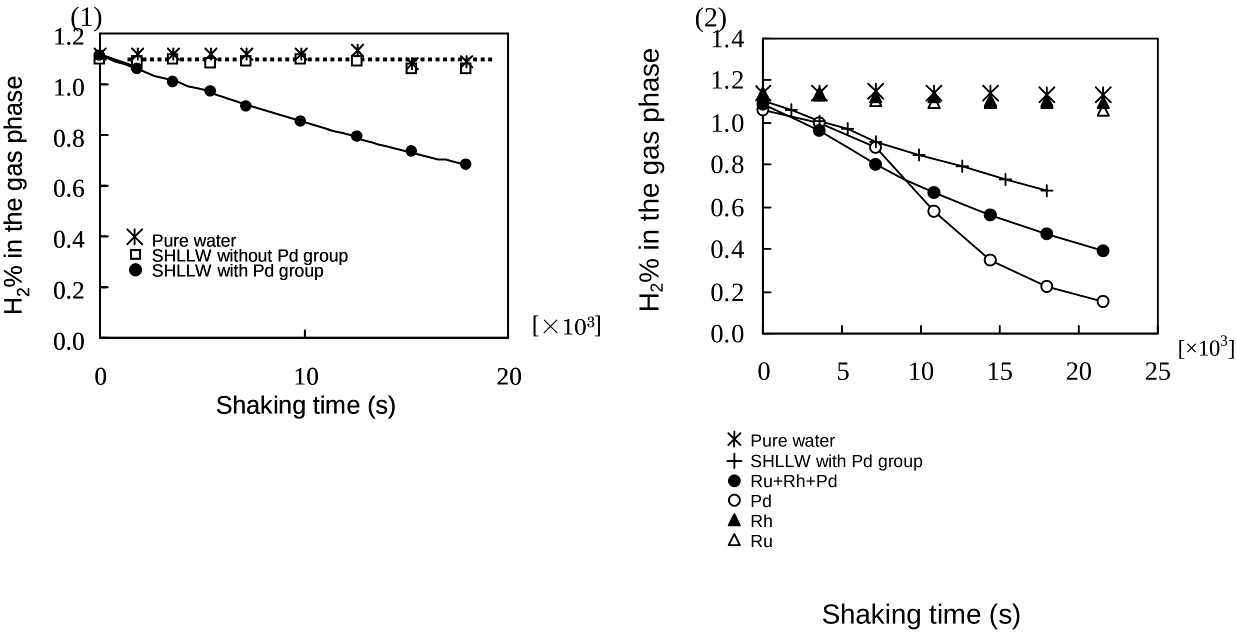


Figure 4(1) shows the results of H_2 consumption test for various solutions of SHLLW with Pd group ions. The results for pure water and SHLLW without Pd group ions are also given for reference. It can be seen that H_2 does not decrease with time in pure water or in SHLLW without Pd group ions, Contrarily, H2 decreases clearly in case of SHLLW with Pd group ions.

Figure 4**. Results of H_2 consumption**

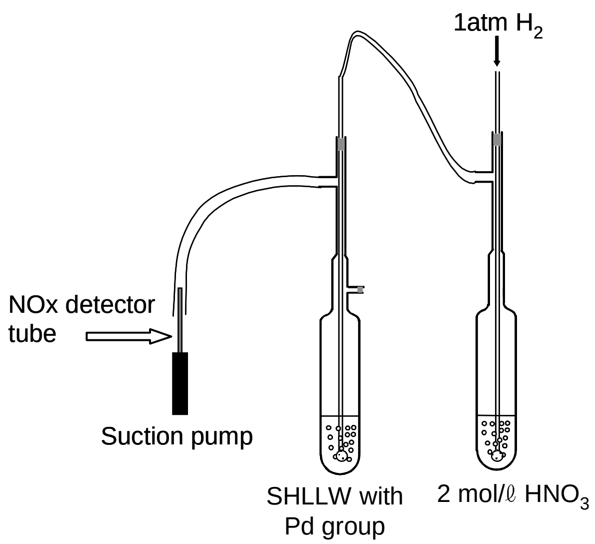


Since the difference of SHLLW with Pd group ions from SHLLW without Pd group ions is the presence of palladium group ions, these ions should be involved in the H_2 consumption reaction. Figure 4(2) shows the results of H_2 consumption for the solutions containing palladium group ions individually.As can be seen in the figure, all the solutions containing Pd ions show remarkable H_2 consumption reaction. On the other hand, the solutions containing Ru and Rh ions but not Pd ion have no or insignificant effect. Therefore, H_2 consumption reaction occurs in dissolver solution and extracted liquid waste because Pd ions are contained in these solution.

(1) Reactants and Products

In the H_2 consumption reaction, H_2 is thought to be oxidized in HNO_3 forming H_2O, NOx and possibly others. In order to confirm this assumption, experiments shown in Fig. 5 were performed.

Figure 5**. Apparatus to examine the reactants and products in the H_2 consumption reaction**



Hydrogen gas was bubbled in SHLLW with Pd group ions. In order to prevent the vaporization of the solution with the bubbling, which may lead to acidity change, H_2 gas was saturated beforehand with the vapor by passing through 2 mol/ℓ HNO_3.

Hydrogen bubbling was stopped intermittently, and part of the solution was sampled for pH measurement after dilution to determine the molar acidity change.

The NOx detector tube was put in the outlet stream, since NOx is expected to come out with H_2. After 100 mℓ of the outlet gas was inhaled, the length of the color change along the tube was observed to determine the concentration of NOx.

The pH of the solution increased with H_2 bubbling time which means that the content of HNO_3 decreased. On the other hand, the major part of the outlet gas is NO, and the remaining are composed of NO_2. From the results of the gas detector tube, the NOx yields in the outlet gas approximately correspond to the decrement of HNO_3 in the solution.

That is, we can infer that H_2 react with HNO_3, forming H_2O and NOx owing to the catalytic effect of Pd ions.

The overall reaction (stoichiometry) for the H_2 consumption is presumed as follows.

$$3\,H_2 + 2HNO_3 \rightarrow 4\,H_2O + 2NO$$

$$2\,H_2 + 2HNO_3 \rightarrow 3\,H_2O + NO + NO_2$$

(2) Effects of Nitrous Acid concentration

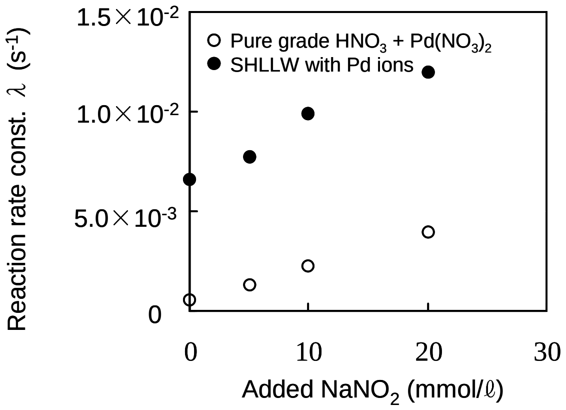
In a HLLW tank, the solution is irradiated by decay radiations with a dose rate of around 3.8 Gy/s producing nitrous acid constantly.

In order to confirm the effect of NO_2, SHLLW with Pd group ions and Pd ion solution in HNO_3 were taken in the glass ampoules, and then $NaNO_3$ was added. H_2/Air gas was bubbled in the solution, and immediately the H_2 consumption test was performed without bubbling. The same apparatus shown in Fig. 3 was used. . Hydrogen consumption rate constant λ was measured.

The effects of NO_2 on the H_2 consumption rate constant are shown in Fig. 6.

It can be seen that nitrous acid enhances the reaction. The difference of reaction rate constant between Pd ion solution in HNO_3 and SHLLW with Pd group ions arises from concentration of Pd ions.

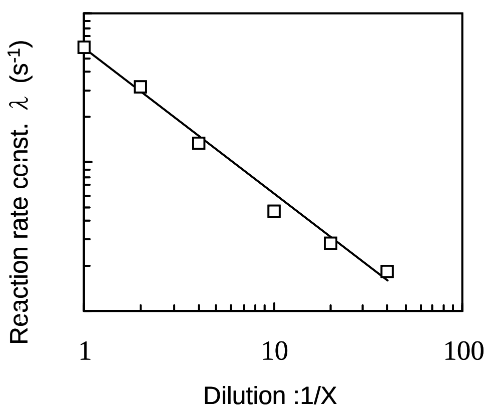
Figure 6**. Effect of $NaNO_2$ addition on the H_2 consumption rate constant**



(3) Effects of Pd Concentration

In order to examine the Pd concentration effects, the following solutions were prepared. SHLLW with Pd group ions was diluted with 2 mol/ℓ HNO_3 in various ratios up to 1/40. The same apparatus and procedure shown in Fig. 3 was used. The effects of Pd ion concentration are shown in Figs. 7. The rate constant decreases roughly linearly on log vs. log scale with decreasing Pd ion concentration.

Figure 7. **Effect of Pd ion concentration on the H_2 consumption reaction rate constant**



III. ANALYSIS OF H_2 ACCUMULATION IN CASE OF SWEEIPING-AIR LOSS IN A HLLW TANK

(1) Mathematical Model to Calculate H_2 Concentration

Equations describing the change of H2 concentration in the gas and the liquid phases are given as follows (see Fig. 8).

$$\frac{dC_G V_G}{dt} = \alpha H_2 \qquad (1)$$

$$\frac{dC_L V_L}{dt} = -\alpha H_2 - \lambda C_L V_L + \frac{GDV_L \rho}{A} \qquad (2)$$

Where,

C_G and C_L: H_2 concentrations in the gas phase and liquid phase, respectively (mol/ℓ)

α_{H2}: Net H_2 transfer rate from the liquid to the gas phase (mol/s)

V_G and V_L: Gas and liquid phase volumes, respectively (ℓ)

λ: H_2 consumption rate constant (s^{-1})

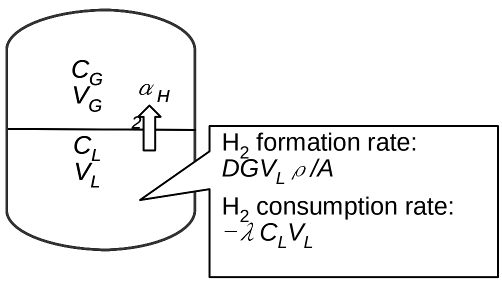
G: G-value of H_2 for HLLW (molecules/100 eV)

D: Dose rate in HLLW caused by decay of radioactive materials (Gy/s)

ρ: Density of HLLW (kg/ℓ)

A: Avogadro constant (=6.0×10^{23} molecules/mol)

Figure 8. **Parameters used to show the H_2 concentration change in an HLLW tank in case of sweeping-air loss**



Not only C_G but also C_L is assumed to be nearly uniform in a tank, which means vigorous agitation of the liquid phase. Such agitation would enhance the release of H_2 formed in the liquid phase to the gas phase. Owing to this assumption it is not necessary to consider the condition of convection of the solution due to heat unbalance. When the sweeping air stops, C_G increases with time and will approach the a steady-state value. On the assumption of that C_G and C_L approach the steady-state values. dC_GV_G/dt and dC_LV_L/dt are treated as 0. Accordingly, the following equation holds.

$$\lambda C_L V_L = \frac{GDV_L\rho}{A} \qquad (3)$$

$$\frac{C_L}{C_G} = h \qquad (4)$$

where, h is solubility of H2 in HLLW (-), that is, (mol/ℓ H_2 in liquid)/(mol/ℓ H_2 in gas) under equilibrium

From eqs. (3) and (4),

$$C_G = k\frac{GD\rho}{Ah\lambda} \qquad (5)$$

where, k is the unit conversion factor (100eV/J).

(2) Calculation of the Steady-State H2 Concentration

Using the values determined in the present experiments, the H_2 concentration in the gas phase of a HLLW tank in case of sweeping function loss can be calculated as follows.

Substituting the following values in eq. (5), the steady-state concentration C_G (the maximum possible value) is obtained as follows:

C_G = kGDρ/(Ahλ) = 1.1×10^{-4} to 1.3×10^{-5} mol/ℓ, that is, 0.3 to 0.04% (=$C_G\times22.7\times T/273\times100$; T=308 to 353K)

k = 6.25×10^{16} (100eV/J): the unit conversion factor. 1eV=1.602×10^{-19}J

D = 3.8 Gy/s, which is based on the following conditions. The HLLW of 0.4 m^3 is produced by processing of 1 t spent fuel (45,000 MWd/t, 6-year cooling). From the ORIGEN-2 results, the decay heat by α-radiation is 0.30 Gy/s and that by β·γ-radiation 3.5 Gy/s. As the sum, D is obtained as 3.8 Gy/s.

G = 0.033 H_2 molecules/100 eV, which is calculated as follows. G(H_2) by α-radiation is 0.17[13] and that by β·γ-radiation 0.021. Using these G(H_2) values and the above dose rate ratio, G is obtained as 0.033.

h = 0.011 (Measured value of SHLLW without Pd group)

λ = 0.014 to 0.12 (equation (A-1) of APPENDIX, T=308 to 353K)

ρ = 1.29 kg/ℓ (Measured value of SHLLW with Pd group)

As a result of this calculation, H_2 concentrations in the gas phase is 0.3 to 0.04% at T=308 to 353K.

IV. CONCLUSION

The present study using the simulated solution showed that H_2 reacts with HNO_3 forming H_2O and NOx owing to catalytic effect of Pd ions. The results seem to apply to actual HLLW. For instance, Pd-catalyzed H_2 consumption reaction is probably responsible for the low H_2 evolution from actual solution.

The analytical study using the present experimental results suggested that the H_2 concentration in the gas phase of HLLW tank does not reach its explosion limit of 4% even if the sweep air stops for a long time. Nevertheless, confirmation of our theory using actual solution and further actual HLLW tanks would be necessary to introduce these results for the safety assessment, the accident management, and the design of fuel cycle facilities.

The above mentioned confirmations will give credence to this theory, and facilitate reflecting results to the regulation, which provide rationality of the design of facilities, the enhancement of safety, of fuel cycle facilities.

APPENDIX

Effect of Temperature

1. Experimental

Temperature effect on the H_2 consumption rate constant was examined using SHLLW with Pd group ions. For this purpose a constant-temperature box was mounted on the shaker shown in Fig. 3, and the sample solution sealed in a glass ampoule was placed in it. The temperature in the box was controlled from room temperature to 353 K. The reaction rate constant was measured at each fixed temperature by the method given in II.B.

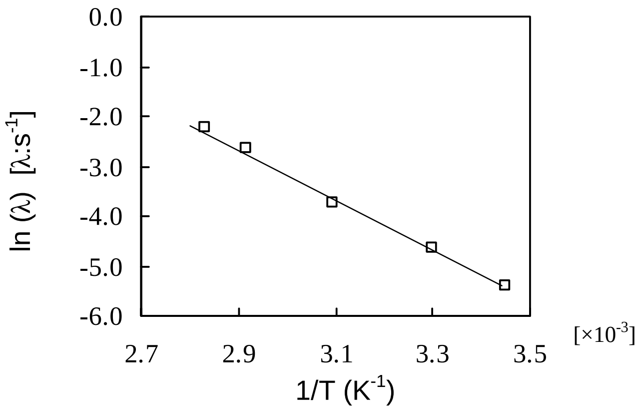
2. Results and discussion

The results are shown in Fig. 9 in the form of an Arrhenius plot. The reaction rate increases remarkably with increasing temperature; the rate at 353 K (λ=0.107 s^{-1}) is 24 times that at 290 K (λ=0.0045 s-1). The activation energy is calculated as Ea=4.2×104 J/mol (0.44 eV). These results give the following equation for the reaction rate constant, λ (s^{-1}), as a function of temperature, T (K).

$$\lambda = 2.24\times10^{5}\exp(-5{,}100/T) \qquad \text{(A-1)}$$

If the room temperature changes, for instance, by 2 K, λ changes 10 %. All λs described in this report contain this kind of fluctuation caused by room temperature change (283~300 K throughout a year).

Figure 9. **Temperature effect on the H_2 consumption rate constant**



References

[1] R.BECKER et al., "Radiolytically Generated Hydrogen from Purex Solutions," IAEA-SM-245/13 (1979).

[2] S.SPECHT, "Bemerkungen zur Wasserstoff-Freisetzung aus Salpetersauren Spaltproduktlösungen, " KfK-2940, p.196 (1980).

[3]. N.NAKAGIRI and T. MIYATA, "Evaluation of Value for Hydrogen Release from High-Level Liquid Waste," J. At. Energy Soc. Japan (in Japanese), 36, 744 (1994).

[4] N.NAKAGIRI and T. MIYATA, ibid. 37, 1119 (1995).

[5] N.NAKAGIRI and T. MIYATA, ibid. 38, 992 (1996).

[6] N.NAKAGIRI and T. MIYATA, ibid. 39, 1062 (1997).

[7] Y. KUNO et al., "Radiolytically Generated Hydrogen and Oxygen from Plutonium Nitrate Solution," J. Nucl. Sci. Technol., 30, 919 (1993).

[8] G. V. Buxton et al., "Critical Review of Rate Constants for Reactions of Hydrated Electrons, Hydrogen Atoms and Hydroxyl Radicals (OH/O^-) in Aqueous Solution," J. Phys. Chem. Ref. Data, 17, 513 (1988).

[9]. Y. Katsumura et al., "Pulse Radiolysis Study of Aqueous Nitric Acid Solutions. Formation Mechanism, Yield and Reactivity of NO_3 Radical," J. Phys. Chem., 95, 4435 (1991).

[10]. P. Neta, R. E. Huie and A. B. Ross, "Rate Constants for Reactions of Inorganic Radicals in Aqueous Solution," J. Phys. Chem. Ref. Data, 17, 1027 (1988).

[11]. H. Block, "Nitrous Acid, Hyponitrous Acid and Their Salts" in Mellor's Comprehensive Treatise on Inorganic and Theoretical Chemistry, Vol. 8, Suppl. II, Nitrogen, Part II, Longmans, London (1967), p. 353.

[12] P.-Y. Jiang et al., "γ- radiolysis Study of Concentrated Nitric Acid Solutions," J. Chem. Soc. Faraday Trans., 90, 93 (1994)

[13] N.E. Bibler, "Curium-244 α Radiolysis of Nitric Acid. Oxygen Production from Direct Radiolysis of Nitrate Ions," J. Phys. Chem., 78, 211 (1974).

[14] "Chronological Scientific Table" (Rika Nenpyo), National Inst. Natural Sciences, National Astrological Observatory of Japan (2009).

[15] W. F. Linke and A. Seidell, "Solubilities of Inorganic and Metal Organic Compounds," 4th Edition, Vol. １, Amer. Chem. Soc., p.1078 (1958).

Study of Hydrogen Consumption Reaction Catalyzed by Pd Ions in the Simulated High-Level Liquid Waste

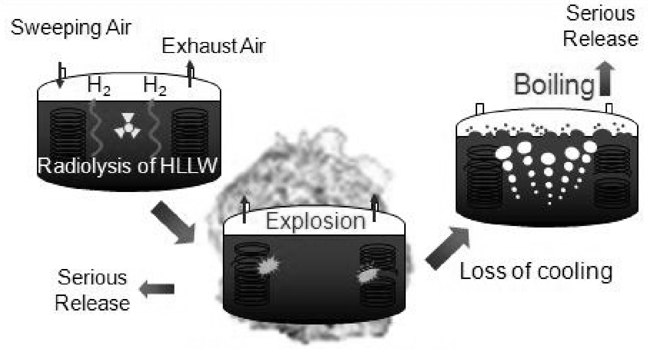
J N F L

Oct. 27 ~ 29, 2011
OECD/NEA WGFCS Workshop

Safety Technology Office, JNFL

Contents

Background

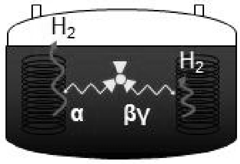


Safety jeopardized by H_2

HLLW: High-Level Liquid Waste

Background (cont.)

- Radiation effect for H_2 consumption
- α-ray has no significant effect*
- Mechanism of the radiation effect is not sufficiently studied.



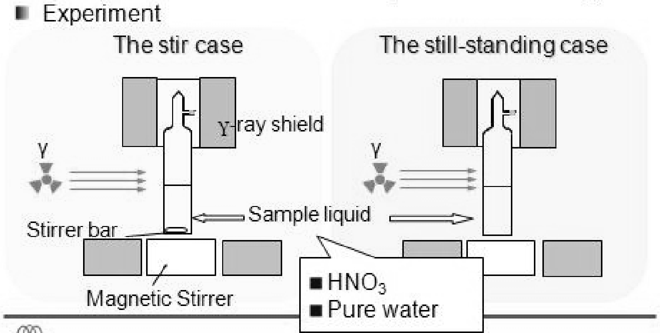
* Y. KUNO et al., "Radiolytically Generated Hydrogen and Oxygen from Plutonium Nitrate Solution," J. Nucl. Sci. Technol., 30, 919 (1993).

Purpose

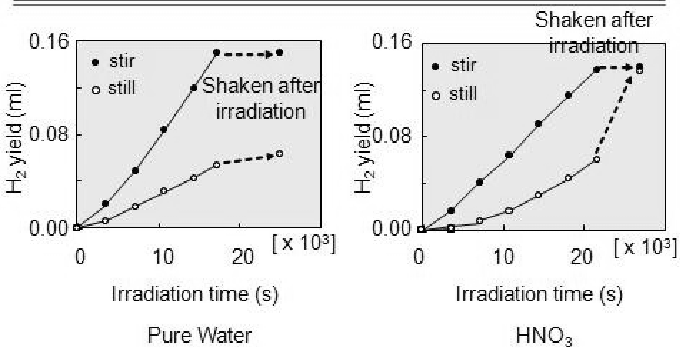
- Reveal the cause of H_2 consumption reaction
- Proposal of the mathematical model to calculate H_2 concentration
- Calculate steady-state H_2 concentration by proposed model in case of sweeping-air loss in a HLLW tank

Radiation Effect -Purpose and Experiment-

- Purpose
  Confirmation of the radiation effect in pure water and HNO_3
- Experiment



Radiation Effect -Result and Discussion-



Radiation Effect -Result and Discussion-

Pure Water (***H2 consumption reaction is worked***)

$H_2 + OH\cdot \rightarrow H\cdot + H_2O$

Consumption

$k = 4.2\times10^7 (mol/l)^{-1}s^{-1}$

Radiation effect has been thought that H_2 is consumed by the reaction with OH radicals.

HNO_3 (***H2 consumption reaction is not worked***)

$H_2 + OH\cdot \rightarrow H\cdot + H_2O$

$k = 4.2\times10^7 (mol/l)^{-1}s^{-1}$

Consumption

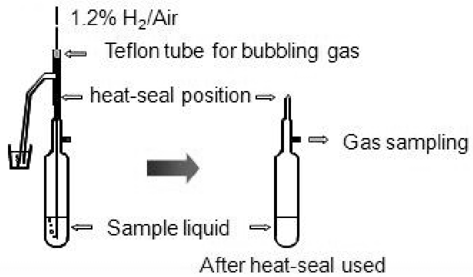
$OH\cdot + HNO_3 \rightarrow NO_3\cdot + H_2O$

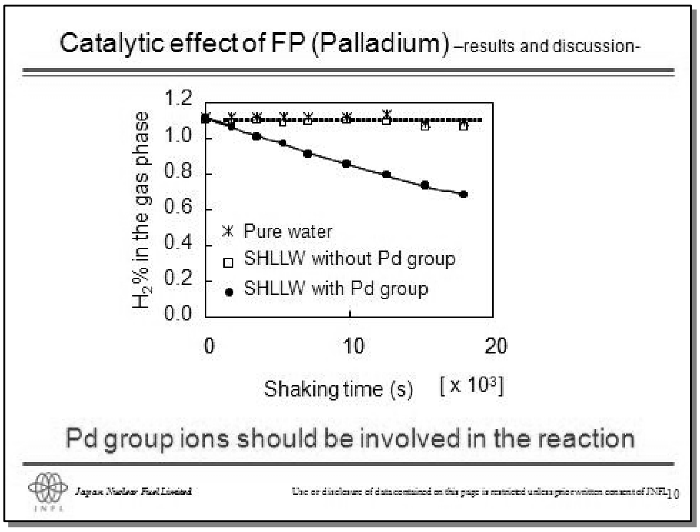
$k = 1.4\times10^8 (mol/l)^{-1}s^{-1}$

The radiation effect for H2 consumption reaction does not work in HLLW if there is no catalytic effect

Catalytic effect of FP (Palladium) -Purpose and Experiment-

- Purpose
  Confirmation of the catalytic effect of FP components
- Experiment




Catalytic effect of FP (Palladium) –results and discussion-
H2% in the gas phase
1.2
1.0
0.8
0.6
0.4
0.2
0.0
0
10
20
Shaking time (s) [x 10³]
Pure water
SHLLW without Pd group
SHLLW with Pd group
Pd group ions should be involved in the reaction
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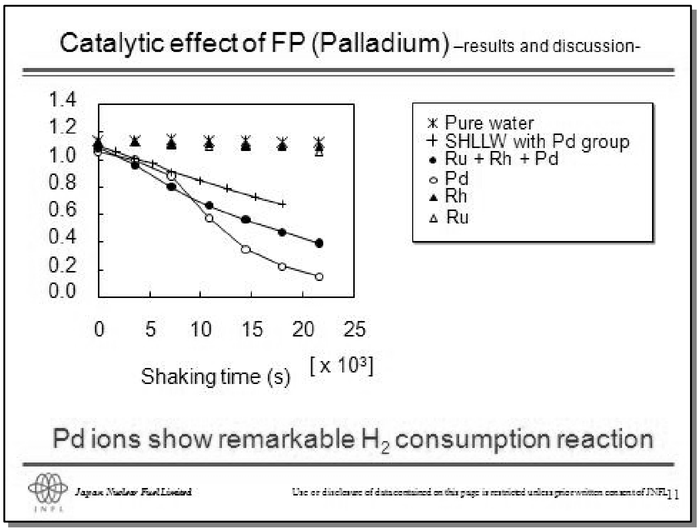
Catalytic effect of FP (Palladium) –results and discussion-
1.4
1.2
1.0
0.8
0.6
0.4
0.2
0.0
0
5
10
15
20
25
Shaking time (s) [x 10³]
Pure water
SHLLW with Pd group
Ru + Rh + Pd
Pd
Rh
Ru
Pd ions show remarkable H2 consumption reaction
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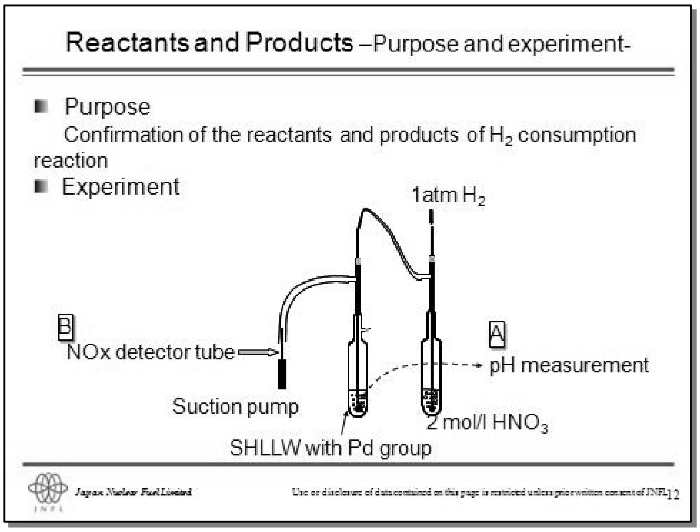
Reactants and Products –Purpose and experiment-
Purpose
Confirmation of the reactants and products of H2 consumption reaction
Experiment
1atm H2
B
NOx detector tube
Suction pump
A
pH measurement
2 mol/l HNO3
SHLLW with Pd group
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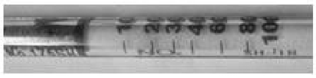
Reactants and Products –Results and discussion-

A
- pH of the solution increased with H_2 bubbling
- HNO_3 decreased

NO_x



B
- Major part of the outlet gas is NO, remaining is NO_2

Correlation confirmation
- NOx yields = HNO_3 decrement

B A

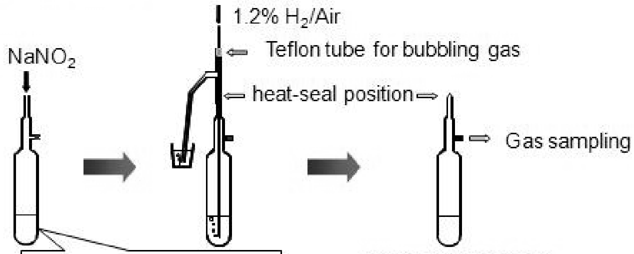
$$3H_2 + 2HNO_3 \rightarrow 4H_2O + 2NO$$

$$2H_2 + 2HNO_3 \rightarrow 3H_2O + NO + NO_2$$

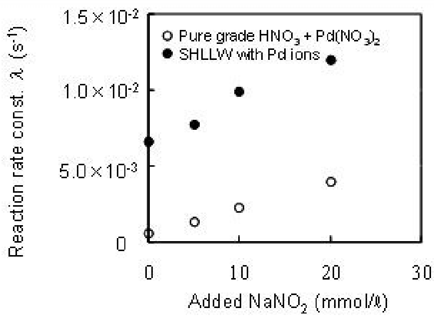
H_2 is consumed by reaction with HNO_3

Effect of Nitrous Acid concentration –Purpose and experiment-

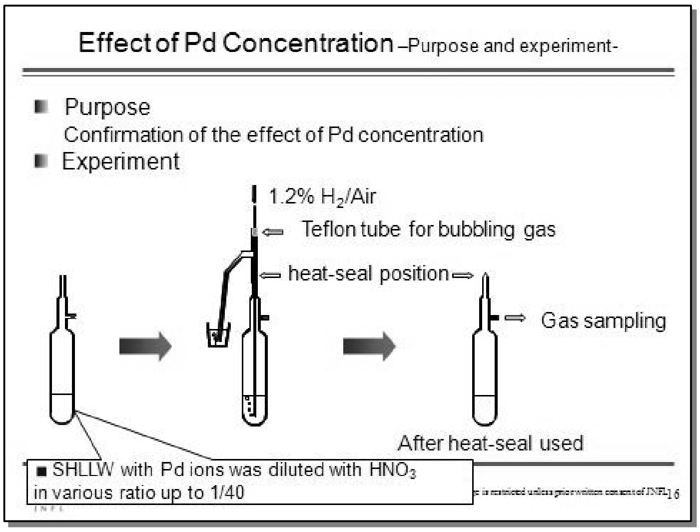
- Purpose
  Confirmation of the effect of NO_2 in H_2 consumption reaction
- Experiment

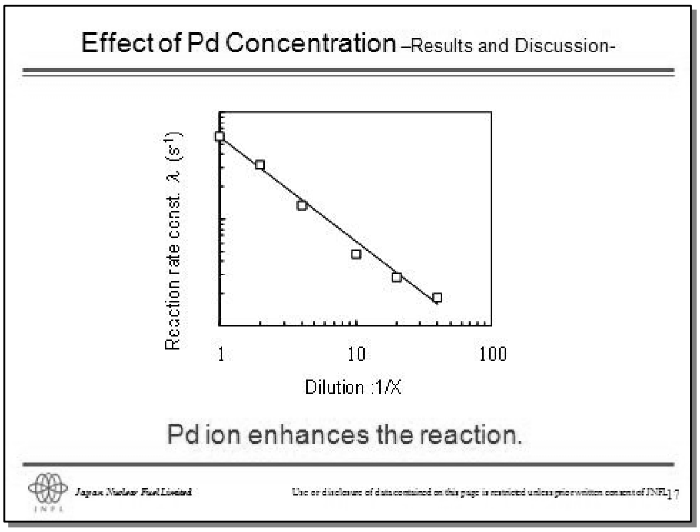


Effect of Nitrous Acid concentration –Results and discussion-



Nitrous acid enhances the reaction.


Effect of Pd Concentration –Purpose and experiment-
Purpose
Confirmation of the effect of Pd concentration
Experiment
1.2% H_2/Air
Teflon tube for bubbling gas
heat-seal position
Gas sampling
After heat-seal used
■ SHLLW with Pd ions was diluted with HNO_3 in various ratio up to 1/40
16


Effect of Pd Concentration –Results and Discussion-
Reaction rate const. λ (s^{-1})
1
10
100
Dilution :1/X
Pd ion enhances the reaction.
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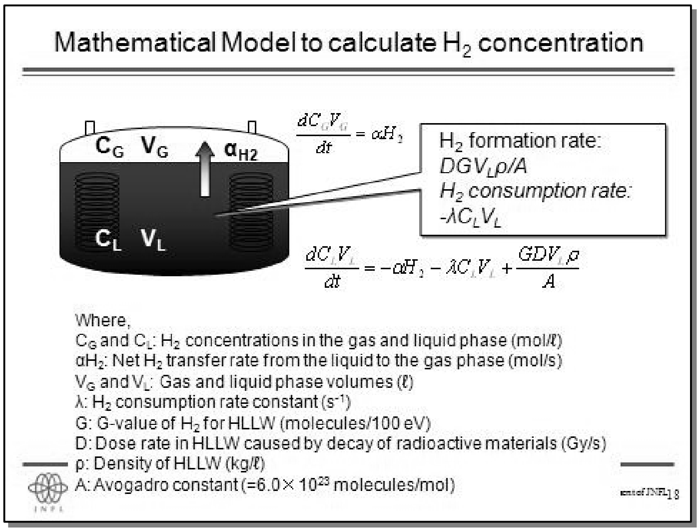

Mathematical Model to calculate H_2 concentration
C_G V_G α_{H2}
C_L V_L
$\frac{dC_G V_G}{dt} = \alpha H_2$
H_2 formation rate: $DGV_L\rho/A$
H_2 consumption rate: $-\lambda C_L V_L$
$\frac{dC_L V_L}{dt} = -\alpha H_2 - \lambda C_L V_L + \frac{GDV_L\rho}{A}$
Where,
C_G and C_L: H_2 concentrations in the gas and liquid phase (mol/ℓ)
αH_2: Net H_2 transfer rate from the liquid to the gas phase (mol/s)
V_G and V_L: Gas and liquid phase volumes (ℓ)
λ: H_2 consumption rate constant (s^{-1})
G: G-value of H_2 for HLLW (molecules/100 eV)
D: Dose rate in HLLW caused by decay of radioactive materials (Gy/s)
ρ: Density of HLLW (kg/ℓ)
A: Avogadro constant (=6.0×10^{23} molecules/mol)
18

Mathematical Model to calculate H_2 concentration (cont.)

Assuming that C_G and C_L approach the steady-state,

$$\frac{dC_G V_G}{dt} = \frac{dC_L V_L}{dt} \approx 0$$

Furthermore, solubility of H_2 in HLLW, $h=C_L/C_G$

Accordingly, the following equation holds.

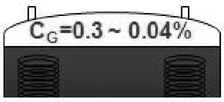
$$C_G = k\frac{DG\rho}{Ah\lambda}$$

Where, k is the unit conversion factor, 6.25×10^{16} (100eV/J)

Calculation of Steady State H_2 Concentration

$$C_G = k\frac{DG\rho}{Ah\lambda}$$

D: 3.8Gy/s
G: 0.033 H_2 molecules/100eV
h=0.011
(measured value of SHLLW without Pd group)
λ=0.014 to 0.12
($\lambda=2.24\times10^5\exp(-5,100/T)$, T=308 to 353K)
ρ=1.29kg/ℓ
(measured value of SHLLW with Pd group)
A=6.0× 10^{23} molecules/mol



H_2 concentration does not reach its explosion limit 4%.

Conclusion

- H_2 reacts with HNO_3 forming H_2O and NO_x owing to catalytic effect of Pd ions.
- Mathematical model to calculate H_2 concentration is proposed.
- Calculated H_2 concentration does not reach its explosion limit of 4%.
- Confirmation of our theory using actual solution and actual HLLW tanks would be necessary to utilize these result for the safety assessment, the accident management, and the design of fuel cycle facility.

RISK ASSESSMENT IN LONG-TERM STORAGE OF SPENT NUCLEAR FUEL

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Abstract - This paper presents probabilistic risk-informed approaches that the Nuclear Regulatory Commission (NRC) staff is planning to consider in preparing regulatory bases for long-term storage of spent nuclear fuel (SNF) for up to 300 years. Due to uncertainties associated with long-term SNF storage, the NRC is considering a probabilistic risk-informed approach as well as a deterministic design-based approach. The uncertainties considered here are primarily associated with materials aging of the canister and SNF in the cask system during long-term storage of SNF. This paper discusses some potential risk contributors involved in long-term SNF storage. Methods of performance evaluation are presented that assess the various types of risks involved. They include deterministic evaluation, probabilistic evaluation, and consequence assessment under normal conditions and the conditions of accidents and natural hazards. Some potentially important technical issues resulting from the consideration of a probabilistic risk-informed evaluation of the cask system performance are discussed for the canister and SNF integrity. These issues are also discussed in comparison with the deterministic approach for comparison purposes, as appropriate. Probabilistic risk-informed methods can provide insights that deterministic methods may not capture. Two specific examples include stress corrosion cracking of the canister and hydrogen-induced cladding failure. These examples are discussed in more detail, in terms of their effects on radionuclide release and nuclear subcriticality associated with the failure. The plan to consider the probabilistic risk-informed approaches is anticipated to provide helpful regulatory insights for long-term storage of SNF that provide reasonable assurance for public health and safety.

1. Introduction

Applying both a probabilistic risk-informed approach and deterministic approach helps the staff confirm the adequacy of existing regulatory bases or augment the regulatory bases to cover long-term storage of SNF. Such reviews are performed to provide reasonable assurance for public health and safety. This paper presents some probabilistic risk-informed and deterministic approaches that the NRC staff is planning to investigate in its review of long-term storage of SNF for up to 300 years. Due to uncertainties associated with long-term SNF storage, the NRC staff is considering both the probabilistic risk-informed and deterministic design-based approaches. The uncertainties considered here are primarily associated with materials aging. This paper discusses some potential technical issues associated with materials aging for the cask system. The cask system that stores SNF consists of components, such as the canister, SNF and overpacks.

Deterministic safety assessment methods that apply conservative assumptions may not provide performance insights that a probabilistic risk-informed performance based method may identify. A conservative deterministic method may not fully address insights that uncertainties may have during long-term materials aging for each failure mode. A probabilistic risk-informed approach allows the safety assessment of the cask system to be performed using realistic assumptions that incorporate uncertainty analyses and thereby assess the system's performance under a variety of conditions.

This paper considers some performance criteria in the probabilistic risk-informed approach under normal, accident, or natural hazard conditions. Methods of performance evaluation are also considered. And, some potentially important technical issues associated with long-term materials aging in the canister and SNF are discussed for the canister and SNF integrity. These issues are also discussed when comparing the deterministic approach, as appropriate. Two example issues, marine stress corrosion cracking (SCC) of the stainless steel canister and hydrogen embrittlement of Zircaloy cladding are discussed in more detail, in terms of their potential effects on radionuclide release and nuclear subcriticality, as risk-informed performance measures.

2. Potentially important technical issues associated with long-term materials aging

In the probabilistic risk-informed approach, three questions on risk are asked: (i) what can go wrong? (ii) How likely is it? and (iii) What are the consequences? Accounting for uncertainties associated with long-term materials aging to the canister and SNF, the NRC staff's technical approach will, for example, consider the following:

- Preventing nuclear criticality, due to degradation of SNF, the canister , or neutron absorbers
- Preventing unacceptable release of radioactive material (i.e., confinement), due to degradation of the seal, the canister, and SNF
- Avoiding excessive radiation dose rates and doses (i.e., radiation shielding), due to degradation of shielding material
- Maintaining the retrievability of the contents under SNF degradation.
- Physical protection and security
- Maintenance programs

During its evaluation, the staff will assess the following:

- SNF specifications – not limited to type of SNF, uranium mass loading, cladding material, specification of damaged SNF, enrichment, heat load as a function of cooling time and heat dissipation, source term, dimensions, burnups and loading curves, inerting atmosphere, maximum storage time, and operating history.
- Structural capability of the cask system to withstand loads under accident conditions and natural phenomena events
- Heat removal under normal, loading, off-normal, and accident conditions

- Maintaining confinement
- Compliance with regulatory dose limits in air and direct radiation from the cask system
- Maintaining nuclear subcritical condition

3. Methods of performance evaluation of cask system

The staff is planning to consider the following methods of performance evaluation of the cask system.

Deterministic modeling techniques which have been widely used to support design-based regulatory requirements. Examples include laboratory test results and rigorous numerical modeling in the structural, thermal and criticality assessment. A deterministic modeling technique typically consists of bounding analyses that obtain their basis from engineering and scientific methods and experimental test results that are expected to bound the potential accident conditions. Consensus standards or regulatory guides provide examples of acceptable approaches to meeting deterministic regulatory requirements.

The probabilistic modeling technique is an extension of deterministic methods. It is based on extended event identification and the associated failure mechanisms, the probability and probability cut-off of the failure mechanisms, the uncertainties and variability, and incorporation of the system consequence analysis. The probabilistic modeling of the overall cask system is consistent with the risk-informed approach. This enables optimizing inspection programs and identifying mitigation (or inspection and remediation) techniques for the design and operation of the cask system. The probabilistic modeling of the overall cask system will also help in early identification of potentially risk-significant issues. Example computer codes that may be applicable to the risk assessment of the cask system include SAPHIRE [1], MELCOR [2], MACCS2 [3], RSAC [4], and PCSA [5]. Applications of these codes to cask systems are available in NRC [6] and EPRI [7] reports. The probabilistic modeling of the overall cask system can be fixed in time or modeled as time-dependent.

The probability of external initiating events or event sequences affecting the cask performance after long-term storage are assessed under normal, various accident and natural hazard conditions. Examples of the external initiating events include aircraft hazards and seismic events. Examples of event sequences include cask tip-over, weld failure, heatup, or SNF failure. Maintaining subcritical SNF conditions is assessed. The assessment of nuclear subcriticality may include (i) moderator exclusion, (ii) configuration stability of the SNF assemblies and internal structures, (iii) effectiveness of neutron absorber, and (iv) burnup credit. The consequence considered is dose (or cancer fatality) to the worker or to the public from radionuclide release.

4. Consideration of potential technical issues in canister and SNF

Failure mode and effect analyses are integral to safety assessments. For the list below, some potentially important factors in the probabilistic risk-informed approach are emphasized for each failure mode.

- Mechanical failure of the stainless steel canister – Canister breach from mechanical puncture resulting from excessive impact stresses. A cut-off probability is used to exclude breach at welds as a function of impact stress [6]. The breached area will affect the magnitude of the radionuclide release fraction. Depending on the magnitude, the impact failure may lead to a configurational change in internal structure that affects the nuclear subcriticality assessment, and the retrievability of the SNF materials.
- SCC of stainless steel canister – In a marine (coastal) environment with salt deposits on the canister from salt water droplets in the air, the weld area may develop SCC [8-14]. Conservatively assuming no inspection and remediation, the crack opening area will affect the

magnitude of the radionuclide release fraction. Inspection/remediation and design mitigation processes (e.g., stress relief by heat treatment) may exclude the potential for SCC.

- Cladding failure – Failure could occur by impact stress, creep, or hydrogen embrittlement (e.g., [6] and [15]). Cladding failure may affect the magnitude of the radionuclide release fraction, and may lead to internal structure configuration change.
- Degradation of SNF matrix – Volume expansion associated with the oxidation/hydration of the UO_2 matrix may lead to crack/unzip of defective cladding [16]. The oxidation/hydration may occur with either residual moisture inside the intact canister or from intruded moisture inside the failed canister. This cladding failure may affect the magnitude of the radionuclide release fraction and challenge the retrievability of the SNF materials, and lead to the configurational change in internal structure. The fine-grained and porous rim structure near the cladding of the high burnup UO_2 pellet (above about 60 MWd/MTU) may impact the magnitude of the radionuclide release fraction [6], [17].
- Degradation of neutron absorber – Corrosion of neutron absorbers (e.g., aluminum alloys or borated stainless steel) will affect the effectiveness of nuclear subcriticality control. However, for a criticality event to occur some degree of water moderation would be needed.

5. Two example technical issues

Below describes two specific example issues, which the NRC staff is planning to consider in preparing the risk-informed regulatory bases.

(1) SCC of the stainless steel canister in coastal environments

SCC at of non-stress-relieved welds in a stainless steel canister located in coastal environments may be initiated if the relative humidity (RH) in the air is sufficiently high and the amount of salt deposits is of a sufficient amount to form aggressive and sufficient aqueous conditions. The weld area may have residual tensile stress resulting from the closure welding process. These two conditions of RH and salt deposits are functions of the canister surface temperature. After long time periods, the canister temperature will decrease as the radioactivity inside the canister gradually decays, increasing RH. The temperature, RH and the amount of salt deposits will not homogeneously distribute on the canister surface because the SNF configuration and air flows between the canister and the concrete overpack are not uniform. Considering all these factors, finite probabilities of SCC exists at various times, locations and type of cask designs. The probability could be low enough for SCC to be screened out of a probabilistic evaluation, especially with appropriate inspection and remediation.

When through-the-weld SCC occurs, radionuclide releases may escape the confinement barrier as aerosol driven by the pressure of inert fill gas and fission gas inside the canister. The release rates are affected by the opening area of the canister surface. The SCC area density per weld area of the canister can be estimated conservatively by [18]:

$$\delta = C\,\sigma/E \qquad (1)$$

δ: crack areal density (m^2/m^2)

σ: applied stress (MPa)

E: Young’s modulus (MPa)

C: geometric constant

This formula could be applied to the cladding degradation discussed below in the second example issue. An example calculation for stainless steel using equation (1) is the crack mean areal density per unit weld area is approximately 1.2×10^{-3} for 170-310 MPa of applied stress, $(193\text{-}207) \times 10^{3}$ MPa of Young's modulus [18]. The weld area fraction is about $10^{-2} - 10^{-1}$ [19]. In an example canister surface area of about 30 m^2, the surface opening area will become $3.6 \times (10^2 - 10^3)$ mm^2. The model in equation (1) is very conservative, assuming a distribution of uniform crack size. In reality, the number and size of cracks are likely to be smaller. On the other hand, larger cracks may form, depending on the incipient crack size. In any case, this calculated area is obviously larger than that allowed for leak tight [20]. The rate of SCC has not been established.

Through tight cracks, the radionuclide release could be slowed [7, 21]. In the aforementioned many computer models for performance evaluation, the cracks formed will lower the magnitude of the radionuclide release fraction, compared with that from bare SNF in the source-term assessment of radionuclide release [22].

In addition to radionuclide release due to SCC, there are other technical issues as aforementioned. The moisture and oxygen will intrude through the cracks into the canister. The UO_2 matrix will be oxidized or hydrolyzed [23-26]. With the volume expansion associated with the oxidation and hydration, initially defective cladding may be cracked or unzipped. This may challenge retrievability of the SNF materials, and may lead to a configurational change in the internal structure. Also, Zircalloy may continue to corrode forming oxide which will also impose stress on cladding. Other internal components may corrode continuously, including neutron absorber, basket materials, and the canister internals. The likelihood of this occurring is very low, because corrective actions should have occurred prior to this stage.

(2) Hydrogen effects on cladding integrity

During reactor operation, the cladding metals, mainly Zircaloy, corrode in the reactor coolant. The corrosion process introduces hydrogen into the Zircaloy. Hydrogen can degrade the strength of Zircaloy by overall embrittlement caused by a dispersion of radially-oriented hydrides (perpendicular to hoop stress) [27]. The hydrides formed during reactor operation are mostly circumferential hydrides (parallel to hoop stress). Circumferential hydrides may not affect the strength significantly, depending on the magnitude of severity. However, circumferential hydrides are known to be radially-reoriented in the presence of appropriate applied stress and temperature [27]. The other hydrogen effects are potential delayed-hydride cracking (DHC). The small cracks developed on the inner or outer surfaces of the cladding may lead to crack propagation assisted by hydrogen diffusion to the crack tip forming radially-oriented hydrides at the crack tip. The mechanism has not been proven to exist under the dry storage conditions, although data are available under reactor operational conditions. If it happens under dry storage conditions, it will likely be limited to higher burnup SNF (e.g., above 60 GWd/MTU). The crack density and size from hydrogen embrttlement of hydride reorientation and DHC can be similarly assessed on a conservative basis as the case of SCC of stainless steel described above in equation (1). The values of crack opening from the model need to be compared with those used in determining the release fraction from experimental work [28]. Lorentz, et al. [28] conducted burst tests by heating a cladded SNF rod, allowing an opening area of about 1.6 cm^2 ($\sim 10^{-1}$ fraction of the total cladding surface). If the calculated value of the opening area by cladding cracking induced by hydrogen embrittlement is smaller than the area used in the independent experiments by Lorenz, et al. [28], the release fraction will not be increased with further cladding cracking by embrittlement. Otherwise, the release fraction from the UO_2 matrix to the canister inside will be affected by the embrittlement. The current regulation for SNF requires that the cladding must be protected during storage against degradation that leads to gross rupture or the SNF needs to be otherwise confined.

Elam, et al. [29] assessed the effects on nuclear subcriticality caused by the configuration changes due to cladding failure. The main assumption in this study is full water flooding in the cask. The report presented the reactivity for various SNF rod conditions. For uniform burnup of 45 and 75 Gwd/MTU collapsed SNF rods, the variations of the neutron multiplication factor, Δk_{eff}, were not significant for this changed cladding configuration.

6. Summary

Some potential technical and environmental issues are discussed, which are associated with uncertainties during long-term SNF storage for up to 300 years. The uncertainties considered are primarily from materials aging. The NRC staff's plan is to incorporate probabilistic risk-informed approaches to confirm the adequacy of the existing regulatory framework or revise the regulatory framework, as appropriate, for long-term SNF storage. This paper addressed:

- Examples of potential technical issues under consideration, including nuclear subcriticality, radionuclide release, radiation shielding, and SNF materials retrievability.
- Methods for performance evaluation to assess the types of risk, including deterministic, and probabilistic risk-informed evaluation under normal accident and natural hazard conditions.
- Consideration of performance evaluation, some potentially important technical issues in the probabilistic risk-informed approach for the canister and cladding integrity. They include mechanical failure of the canister, stress corrosion cracking of the canister, cladding failure, degradation of the SNF matrix, and degradation of neutron absorber. For comparison, the consequence of deterministic approach for these issues are also addressed, as appropriately.
- Two specific example cases, stress corrosion cracking of the canister and cladding failure by hydrogen embrittlement , with respect to radionuclide release and nuclear subcriticality.

The plan to consider the probabilistic risk-informed approach helps ensure a sound regulatory framework remains in place to ensure public health, safety and protection of the environment.

Disclaimer

The NRC staff views expressed herein are preliminary and do not constitute a final judgment or determination of the matters addressed or of the acceptability of any licensing action that may be under consideration at the NRC.

References

[1] Idaho National Engineering Laboratory, "System Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) Version 6.0, Saphire Reference Manual." Idaho Falls, Idaho: Idaho National Engineering Laboratory, 1998.

[2] Gauntt, R. O., et al., "MELCOR Computer Code Manuals," NUREG/CR-6119, Rev. 2, Washington, DC: NRC, 2009.

[3] NRC, "Code Manual for MACCS2," NUREG/CR-6613, Vol. 1, Washington, DC: NRC. 1998.

[4] Wenzel, D. R., "The Radiological Safety Analysis Computer Program (RSAC-5) User's Manual, WINCO-1123."Rev. 1. Idaho Falls, Idaho: Idaho National Engineering Laboratory. 1994.

[5] Dasgupta, B., R. Benke, B. Sagar, R. Janetzke, and A. Chowdhury, "PCSA Tool Development, Progress Report II," Version 2, San Antonio, TX: Center for Nuclear Waste Regulatory Analyses, 2002.

[6] U.S. Nuclear Regulatory Commission (NRC), "A Pilot Probabilistic Risk Assessment Of a Dry Cask Storage System At a Nuclear Power Plant." NUREG-1864, Washington, DC: NRC. 2007.

[7] EPRI, "Probabilistic Risk Assessment (PRA) of Bolted Storage Casks, Updated Quantifications and Analysis Report, EPRI 1009691, Palo Alto, California: EPRI, 2004.

[8] Albores-Silva, O. E., E. A. Charles, and C. Padovani, "Effect of Chloride Deposition on Stress Corrosion Cracking of 316L Stainless Steel Used for Intermediate Level Radioactive Waste Containers," The Fourth International Workshop on Long Term Prediction of Corrosion Damage in Nuclear Waste Systems, the Belgian Nuclear Research Centre (SCKNCEN) and the Belgian Agency for Radioactive Waste and Enriched Fissile Materials (ONDRAF/NIRAS), 2010, special issue of *Corrosion Engineering, Science and Technology*, 2011.

[9] Shirai, K., M. Wataru and T. Saegusa, "Long-Term Containment Performance test of Metal Cask," Proceedings of 2011 International Radioactive Waste Management Conference (IHLRWMC), Albuquerque, New Mexico, April 10-14, Paper No. 3332, LaGrange Park: Illinois, American Nuclear Society, 2011a.

[10] Shirai, K., J. Tani, T. Arai, M. Wataru, H. Takeda, and T. Saegusa, "SCC Evaluation of Multi-Purpose Canister," Proceedings of 2011 International Radioactive Waste Management Conference (IHLRWMC), Albuquerque, New Mexico, April 10-14, Paper No. 3333, LaGrange Park: Illinois, American Nuclear Society, 2011b.

[11] Tani, J., K. Shirai, M. Wataru and T. Saegusa, "Stress Corrosion Cracking of Stainless Steel Canister of Concrete Cask," International Seminar on Interim Storage of Spent Fuel (ISSF) 2010, Central Research Institute for Electric Power Industry (CRIEPI), Tokyo, Japan: CRIEPI, 2010.

[12] Caseres, L and T. M. Mintz, “Atmospheric Stress Corrosion Cracking Susceptibility of Welded and Unwelded 304, 304L, and 316L Austenitic Stainless Steels Commonly Used for Dry Cask Storage Containers Exposed to Marine Environments,” NUREG/CR-7030, Washington, DC: NRC, 2010.

[13] EPRI, “Climate Corrosion Considerations for Independent Spent Fuel Storage Installations in Marine Environments,” EPRI 1013524, Palo Alto, California: EPRI, 2006.

[14] EPRI, “Effects of Marine Environments on Stress Corrosion Cracking of Austenitic Stainless Steels,” EPRI 1011820, Palo Alto, California: EPRI, 2005.

[15] EPRI, ”Spent Fuel Transportation Applications – Assessment of Cladding Performance,” EPRI 1015048, Palo Alto, California: EPRI, 2007.

[16] Cunnane, J., W. Ebert, M. Goldberg, R. Finch, and C. Mertz, “Yucca Mountain Project Report, Waste Form Testing Work,” Argonne National Laboratory (ANL), Chemical Technology Division, Argonne, Illinois: ANL, 2003.

[17] Spino, J., M. Coquerelle, and D. Baron, “Microstructure and Fracture Toughness Characterization of Irradiated PWR Fuels in the Burnup Range of 40-67 GWd/t,” Proceedings of the Technical Committee Meeting IAEA on Advances in Fuel Technology. 1996.

[18] Gwo, J., T. Ahn, and X. He, “Modeling Disruptive Events Using the β-SOAR Model: Levels of β-SOAR Model Flexibility in Applications and Initial Insights,” Proceedings of 2011 International Radioactive Waste Management Conference (IHLRWMC), Albuquerque, New Mexico, April 10-14, pp. 867-874, LaGrange Park: Illinois, American Nuclear Society, 2011.

[19] ASM International (The Materials Information Society), *ASM Handbook*, Volume 2. Properties and Selection: Nonferrous Alloys and Special-Purpose Materials, Materials Park, Ohio: ASM International, 1990.

[20] Institute for Nuclear Materials Management, “American National Standard for Leakage 12782 Tests on Packages for Shipment of Radioactive Materials,” ANSI N14.5, 1997.

[21] Sprung, J. L., et al., “Reexamination of Spent Fuel Shipment Estimates,” NUREG/CR-6672, Vol. 1, SAND2000-0234, Albuquerque, NM: Sandia National Laboratory; Washington DC: NRC. 2000.

[22] Kamas, S., T. Ahn and M. Bailey, “Pre-closure Safety Analysis: Sensitivity Studies and Preliminary Dose Results,” Proceedings of the 11th International High-Level Radioactive Waste Management Conference (IHLRWM), April 30 - May 4, Las Vegas, NV, pp.1096-1100, American Nuclear Society, LaGrange Park, Illinois: American Nuclear Society, 2006.

[23] Ahn, T. and S. Mohanty, “Dissolution Kinetics of Commercial Spent Nuclear Fuels in the Potential Yucca Mountain Repository Environment,” NUREG-1924, NRC ADAMS ML 083120074, Washington, DC: NRC, 2008.

[24] Ahn, T., “Dry Oxidation and Fracture of LWR Spent-fuels,” NUREG-1565, NRC ADAMS ML040150720, Washington, DC: NRC, 1996.

[25] Wasywich, K. M. and C. R. Frost, “Behavior of Used CANDU Fuel Stored in 150 °C Moisture-Saturated Air,” 3rd International High level Radioactive Waste Management (IHLRWM) Conference, Las Vegas, Nevada, April 12 – 16, Proceedings of High Level Radioactive Waste Management, Vol. 1, pp. 1166–1173, La Grange Park, IL, American Nuclear Society, 1992.

[26] Taylor, P., R. J. Lemire, and D. D. Wood, “The Influence of Moisture on Air Oxidation of UO_2 : Calculations and Observations, Proceedings of the Third International Conference, “High Level Radioactive Waste Management,” Las Vegas, Nevada, April 12-16, 1993, La Grange Park, Illinois: American Nuclear Society, 1992.

[27] Chung, H. “Understanding Hydride- and Hydrogen-Related Processes in High-Burnup Cladding in Spent-Fuel-Storage and Accident Situations, Proceedings of 2004 International Meeting on LWR Fuel Performance, Orlando, Florida, September 19-22, 2004, Paper No. 1064, LaGrange Park: Illinois, American Nuclear Society.

[28] Lorenz, R. A., et al., “Fission product Release from Highly Irradiated LWR Fuel,” NUREG/CR-0722, Oak Ridge National Laboratory, pp. 48-80, Washington, DC: NRC, 1980.

[29] Elam. K. R., J. C. Wagner and C. V. Parks, “Effects of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks, NUREG/CR-6835, Washington, DC: NRC, 2003.

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Why Risk Assessment in Long-Term Storage of Spent Nuclear Fuel?

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Office of Nuclear Material Safety and Safeguards, U.S.A.

For
OECD/NEA Workshop
Safety Assessment of Fuel Cycle Facilities –
Regulatory Approaches and Industry Perspectives
September 27 - 29, 2011, Toronto, Canada

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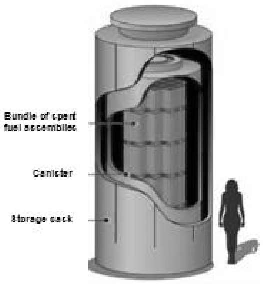
Cask System for Spent Nuclear Fuel (SNF) Storage



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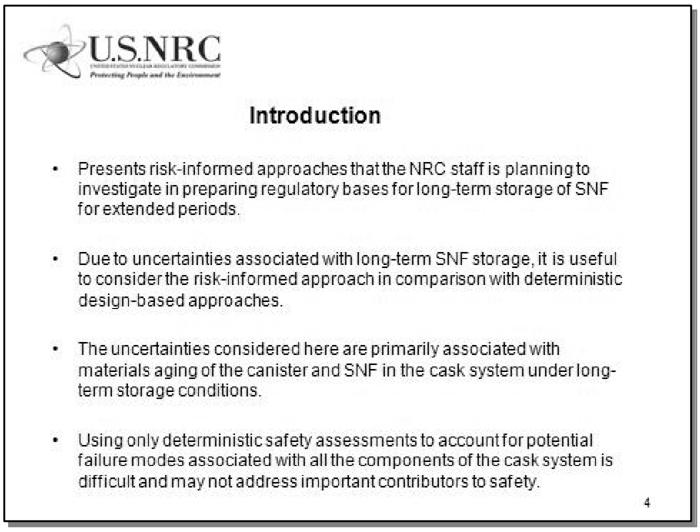
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Introduction
• Presents risk-informed approaches that the NRC staff is planning to investigate in preparing regulatory bases for long-term storage of SNF for extended periods.
• Due to uncertainties associated with long-term SNF storage, it is useful to consider the risk-informed approach in comparison with deterministic design-based approaches.
• The uncertainties considered here are primarily associated with materials aging of the canister and SNF in the cask system under long-term storage conditions.
• Using only deterministic safety assessments to account for potential failure modes associated with all the components of the cask system is difficult and may not address important contributors to safety.
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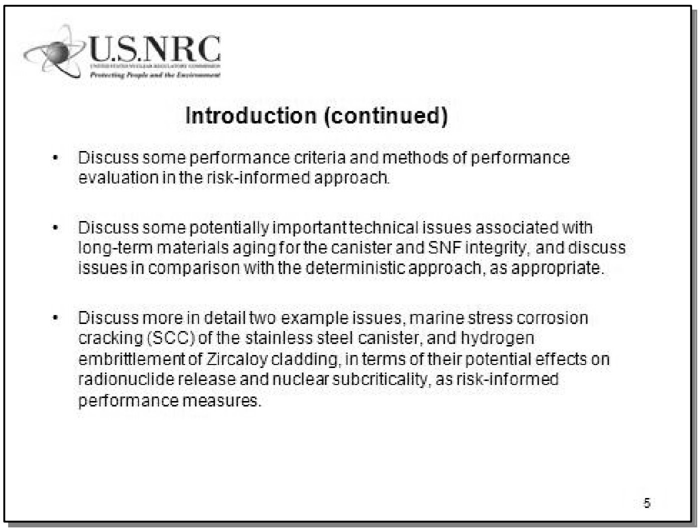
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Introduction (continued)
• Discuss some performance criteria and methods of performance evaluation in the risk-informed approach.
• Discuss some potentially important technical issues associated with long-term materials aging for the canister and SNF integrity, and discuss issues in comparison with the deterministic approach, as appropriate.
• Discuss more in detail two example issues, marine stress corrosion cracking (SCC) of the stainless steel canister, and hydrogen embrittlement of Zircaloy cladding, in terms of their potential effects on radionuclide release and nuclear subcriticality, as risk-informed performance measures.
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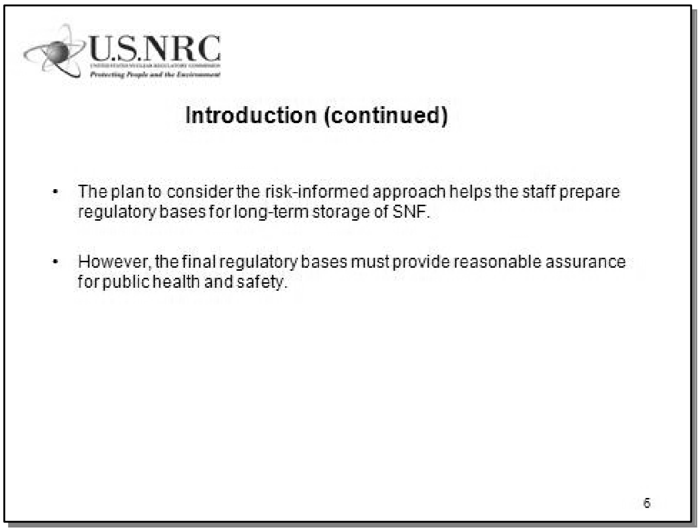
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Introduction (continued)
• The plan to consider the risk-informed approach helps the staff prepare regulatory bases for long-term storage of SNF.
• However, the final regulatory bases must provide reasonable assurance for public health and safety.
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Some Potentially Important Technical Issues Associated with Long-Term Materials Aging

- Three questions on risk: (i) What can go wrong? (ii) How likely is it? and (iii) What are the consequences?

- Preventing nuclear criticality, due to degradation of SNF, the canister, or neutron absorbers
- Preventing unacceptable release of radioactive material (i.e., confinement), due to degradation of the seal, the canister, and SNF (e.g., canister monitoring, and site boundary dose assessment)
- Avoiding excessive radiation dose rates and doses (i.e., radiation shielding), due to degradation of shielding material
- Maintaining the retrievability of the contents under SNF degradation

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Some Methods of Performance Evaluation of Cask System

- Deterministic modeling techniques, widely used to support design-based regulatory requirements:
- laboratory test results and rigorous numerical modeling in the structural, thermal and criticality assessment
- typically bounding analyses
- consensus standards or regulatory guides

- Probabilistic modeling technique, an extension of deterministic methods:
- event identification and the associated failure mechanisms,
  the probability and probability cut-off of the failure mechanisms,
  the uncertainties and variability, and incorporation of the system consequence analysis
- consistent with the risk-informed approach, and enables optimizing and identifying mitigation techniques for the design and operation of the cask system that are a function of the associated risk to the public
- help with the early identification of potentially risk-significant issues

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Some Potentially Important Issues in Canister and SNF

- Mechanical failure of the stainless steel canister:
- mechanical puncture from impact stress will cause canister breach
- cut-off probability is for exclusion
- the breached area affecting the magnitude of the radionuclide release fraction
- a configurational change in internal structure that affects the nuclear subcriticality assessment, and the retrievability of the SNF materials
- the breached canister may or may not be acceptable in the deterministic approach

- SCC of stainless steel canister:
- in a marine (coastal) environment with salt deposits on the canister welds due to salt water droplets in the air
- the crack opening area affecting the magnitude of the radionuclide release fraction
- the design mitigation process for SCC exclusion (e.g., by applying compressive stress), which may be also accepted in the deterministic approach

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Some Potentially Important Issues in Canister and SNF (continued)

- Cladding failure:
- failure by impact stress, creep, or hydrogen embrittlement
- the crack opening area affecting the magnitude of the radionuclide release fraction and configurational change in internal structure
- In the deterministic approach, this failure may or may not be acceptable (e,g., separate confinement requirement)

- Degradation of SNF matrix:
- volume expansion associated with the oxidation/hydration of the UO_2 matrix, cracking/unzipping defective cladding. The oxidation/hydration may occur with either residual moisture inside the intact canister or from intruded moisture inside the failed canister
- affecting the magnitude of the radionuclide release fraction, challenging the retrievability of the SNF materials, and configurational change in internal structure

- Degradation of neutron absorber – corrosion of neutron absorbers (e.g., aluminum alloys or borated stainless steel)

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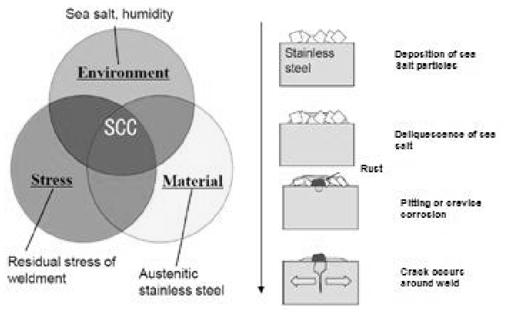
Specific Example Case 1: Canister SCC



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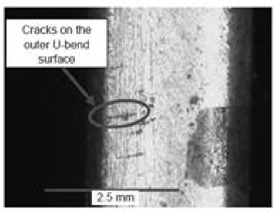
Specific Example Case 1: Canister SCC (continued)



Typical SCC Behavior of a 304 Stainless Steel Single U-Bend Specimen Exposed for 1 Month at a Temperature of 176 °C [350 °F]

(Caseres and Mintz, NUREG/CR-7030, 2010)

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Specific Example Case 1: Canister SCC (continued)

- The SCC of the stainless steel canister is considered when the relative humidity (RH) in air is sufficiently high, the amount of salt deposits is of a sufficient amount, and the surface temperature is low enough to allow deliquescence. This can form aggressive and aqueous conditions at welds to initiate SCC. In longer time periods, the temperature will decrease as the radioactivity inside the canister gradually decays, increasing RH on the outside canister surface.
- The weld area has residual tensile stress remained from the closure welding or fabrication process.
- The SCC area density per weld area:
  $\delta = C\,\sigma/E$

  δ: crack areal density (m^2/m^2); σ: applied stress (Mpa);
  E: Young's modulus (MPa); C: geometric constant

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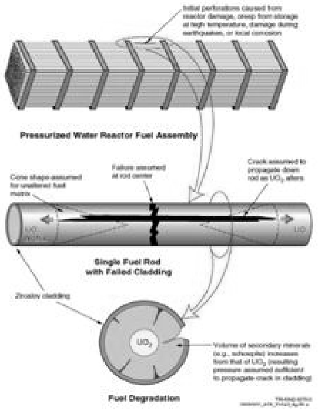
Specific Example Case 2: Hydrogen effects on cladding integrity



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Specific Example Case 2: Hydrogen effects on cladding integrity (continued)

- Hydride reorientaion: circumferential hydrides (parallel to hoop stress of cladding) may be radially reoriented in the presence of appropriate temperature and stress, causing embrittlement.
- Delayed-hydride cracking: the small cracks developed on the inner or outer surfaces of cladding, leading to crack propagation assisted by hydrogen diffusion to the crack tip forming radially-oriented hydrides at the crack tip.
- Nuclear criticality caused by the configuration changes due to cladding failure seems to be controlled from model studies.

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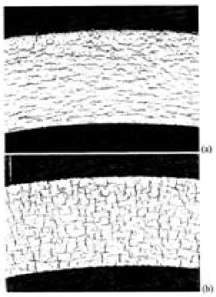
Specific Example Case 2: Hydrogen effects on cladding integrity(continued)

Hydride reorientation from circumferential (a) to radial (b) direction to hoop stress (Yagnik, et al., 2004); cladding thickness of ~0.6 mm



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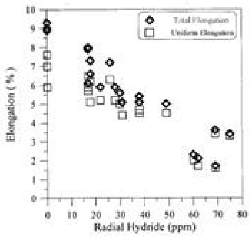
Specific Example Case 2: Hydrogen effects on cladding integrity (continued)

Effects on mechanical property, ductility loss



(Yagnik, et al., 2004)

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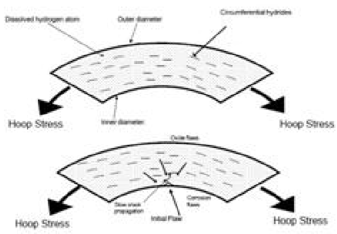
Specific Example Case 2: Hydrogen effects on cladding integrity (continued)

Delayed-hydride cracking



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UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Summary

- Some potential technical and environmental issues are discussed, which are associated with uncertainties during long-term SNF storage. The uncertainties considered are primarily from materials aging, and the risk-informed approaches are emphasized in identifying potential issues.
- The paper addressed: types of potential risk, methods for performance evaluation, and some potentially important technical issues in the risk-informed approach for the canister and cladding integrity. For comparison, the deterministic approach for these issues are also addressed, as appropriate. Two specific example cases were illustrated, for canister SCC and cladding failure by hydrogen embrittlement, with respect to radionuclide release and nuclear subcriticality.
- The plan to consider the risk-informed approach helps the staff prepare regulatory bases for long-term storage of SNF. However, the final regulatory bases must provide reasonable assurance for public health and safety.

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SPECIAL SESSION

PRESENTATION AND DISCUSSION ABOUT LESSONS LEARNT FOR FCFS FROM THE FUKUSHIMA ACCIDENT

Overview of Fukushima Accident and Regulatory Issues for FCFs after the Accident
Y. UEDA *(JNES, Japan)*

Following the Fukushima Accident: Actions in Progress in France
D. Conte, L. Tabard *(ASN, France)*

UK Response to Fukushima: Specific Reference to Fuel Cycle Facilities
N. Blundell *(ONR, UK)*

U.S. Response to Fukushima
J. Kinneman *(USNRC, USA)*

Complimentary Safety Assessment of the Fuel Cycle Facilities in France
P. Nocture *(AREVA, France)*

OVERVIEW OF FUKUSHIMA ACCIDENT AND REGULATORY ISSUES FOR FCFS AFTER THE ACCIDENT

Y. UEDA ***(JNES, Japan)***

NO PAPER AVAILABLE

JNES

Overview of Fukushima accident and regulatory issues for FCFs after the accident

OECD/NEA Workshop
Safety Assessment of Fuel Cycle Facilities – Regulatory Approaches and Industry Perspectives
Toronto, Canada, 27 – 29 September 2011

Yoshinori UEDA
Japan Nuclear Energy Safety Organization (JNES)

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Contents

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1. Overview of Fukushima accident

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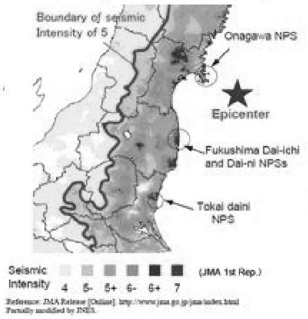
2011 off the Pacific coast of Tohoku Earthquake

- Occurred 14:46 March 11, 2011
- Magnitude:9.0 Mw
- Epicenter location: 38° 10''N and 142° 86''E, and 23.7km in depth



Reference: JMA Release [Online], http://www.jma.go.jp/jma/index.html
Partially modified by JNES.

Source: Fire and Disaster Management Agency

- East coast of northern area in the main island of Japan is seriously damaged
- As of September 7, 15,774 people are dead and 4,227 people are missing according to the Fire and Disaster Management Agency

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Summary of Fukushima Dai-ichi NPS

	Unit 1	Unit 2	Unit 3	Unit 4	Unit 5	Unit 6
	BWR-3	BWR-4	BWR-4	BWR-4	BWR-4	BWR-5
PCV Model	Mark-1	Mark-1	Mark-1	Mark-1	Mark-1	Mark-2
Electric Output (MWe)	460	784	784	784	784	1100
Max. pressure of RPV	8.24MPa	8.24MPa	8.24MPa	8.24MPa	8.62MPa	8.62MPa
Max. Temp of the RPV	300°C	300°C	300°C	300°C	302°C	302°C
Max. Pressure of the CV	0.43MPa	0.38MPa	0.38MPa	0.38MPa	0.38MPa	0.28MPa
Max. Temp of the CV	140°C	140°C	140°C	140°C	138°C	171°C(D/W) 105°C(S/C)
Commercial Operation	1971,3	1974,7	1976,3	1978,10	1978,4	1979,10
Number of D/G	2	2*	2	2*	2	3*
Electric Grid	275kV x 4				500kV x 2	
Plant Status on Mar. 11	In Operation	In Operation	In Operation	Refueling Outage	Refueling Outage	Refueling Outage

* One Emergency DG is Air-Cooled

Source: Application document of license for establishment of NPP

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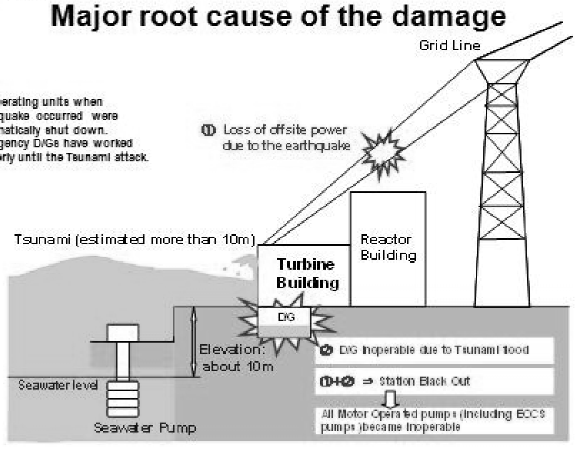
Major root cause of the damage

Note:
- All operating units when earthquake occurred were automatically shut down.
- Emergency D/Gs have worked properly until the Tsunami attack.



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Collapsed Tower

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Tsunami getting over seawall

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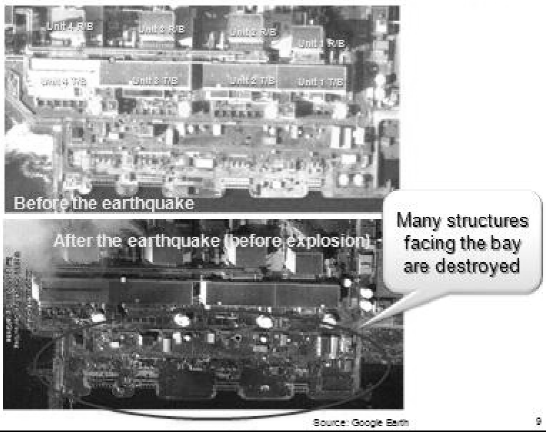
Satellite view of Fukushima Dai-ichi NPS



Source: Google Earth

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Current status estimation of the NPS

	Unit 1	Unit 2	Unit 3	Unit 4
Fuel in RPV	Damaged Fuel was melted	Damaged Fuel was melted	Damaged Fuel was melted	-
RPV	There might be small leak path at the bottom head	There may be rupture at the bottom head	There may be rupture at the bottom head	Not damaged
PCV	Leak tightness is kept to some extend, although small leak path may exist	There may be rapture at the S/P	Leak path may exist in D/W	Not damaged
Reactor Bldg.	Hydrogen explosion occurred on March 12	The blow-out panel was opened due to the explosion in unit 3	Hydrogen explosion occurred on March 14	Building damage was identified on March 15

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INES rating

- NISA issued provisional INES ratings, based on "What is known" at the time.
- At first, following units were rated as Level 3 based on "Defense in Depth" criteria about 10 hours later from the earthquake.
  Fukushima Dai-ichi unit 1, 2 and 3, Fukushima Dai-ni Unit 1, 2 and 4
- In the evening on March 12, the rating of Fukushima Dai-ichi Unit 1 was re-evaluated to Level 4 base on the "Radiological Barriers and Control" criteria.
- On March 18, Fukushima Dai-ichi Unit 1, 2 and 3 were re-rated to Level 5 based on "Radiological Barriers and Control" criteria because the fuel damage was highly possible. Fukushima Dai-ichi Unit 4 was evaluated to Level 3 based on the "Defense in Depth" criteria.
- On April 12, Fukushima Dai-ichi NPS was revised Level 7 based on the "People and Environment" criteria, as a result of discharged estimation.
- Official rating will be done after cause and countermeasures are identified.

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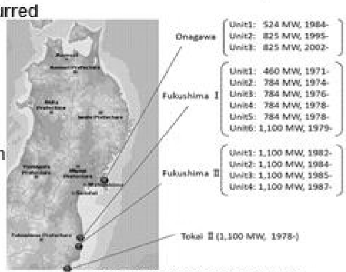
Nuclear reactors near epicenter of the earthquake

March 11, 14:46, The earthquake occurred

- 11 reactors under operation were automatically shut down
  - Onagawa 1,2,3
  - Fukushima Dai-ichi 1,2,3
  - Fukushima Dai-ni 1,2,3,4
  - Tokai Dai-ni
- 3 reactors under periodic inspection
  - Fukushima Dai-ichi 4,5,6



Location of the Nuclear Installations

Around 1 hour later, after tsunami hit the NPSs above

- Following reactors went to cold shut down
  - Onagawa 1,2,3 : External power and sea water pumps were alive
  - Fukushima Dai-ichi 5,6: Emergency DG was alive
  - Fukushima Dai-ni 1,2,3,4: External power was alive
  - Tokai Daini: Emergency DG was alive

- **The problems came with Fukushima Dai-ichi 1,2,3 and 4.**

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Responses at other NPSs (1/2)

1. Emergency Safety Measures

- NISA directed all electric power companies and JAEA to implement emergency safety measures. (30 March)
- Based on the report from each electric utilities and JAEA, NISA has confirmed that emergency safety measures had been appropriately implemented.(6 May)

2. Hamaoka NPS shutdown

- The government requested Chubu Electric Power Company to halt the operation of all units of Hamaoka NPS due to high possibility of large-scale tsunami resulting from the envisioned earthquake within mid to long term countermeasures. (6 May)

3. Preparatory Measures for Response to Severe Accidents

- NISA directed all electric power companies and JAEA to implement preparatory measures for response to severe accidents taking into account the accident at Fukushima Dai-ichi NPS . (7 June)
- Based on the report from each electric utilities and JAEA, NISA has confirmed that the preparatory measures had been appropriately implemented.(18 June)

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Major responses at other NPSs (2/2)

4. Comprehensive evaluation of safety of existing NPPs based on the Fukushima Dai-ichi NPP incident

- NSC directed NISA to report on comprehensive evaluation of safety on robustness against external events, which are beyond design basis, for all existing NPPs. (6 July, at the 50th extraordinary meeting of NSC)
- Government announced to introduce the safety evaluation above with a new procedure and rule referring to the stress test introduced at European countries. (11 July)
- NISA proposed an evaluation method and implementation plan about the comprehensive evaluation. (21 July, at the 55th extraordinary meeting of NSC)

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Outline of Emergency Safety Measures

Phase	Emergency Safety Measures	
	Short Term	Mid Term
Expected Time to Completion	Done	One to three years
Goals (Desired Level / Extent)	Preventing fuel damage and spent fuel damage even if (1)AC power supplies, (2)seawater cooling functions and (3)spent-fuel storage pool cooling functions are all lost.	Enhancing reliability of emergency safety measures (short term) (Securing/speeding up achievement of cold shutdown; measures against tsunami)
Examples of Specific Measures	【Securing Equipment】 ● Deploying power generator vehicles (to support cooling reactors and spent fuel pools) ● Deploying fire engines (to supply cooling water) ● Deploying fire hoses (to secure water supply routes from freshwater tanks, seawater pits, etc.) 【Preparing Procedural Manuals, Etc.】 ● Preparing procedural manuals for emergency responses utilizing the above-mentioned equipment 【Training to Respond】 ● Implementing training for emergency responses based on the procedural manuals 【Measures Against Flooding】 ● Measures to prevent flooding at reactor buildings assuming approx. 15-meter-high tsunami	【Measures Against Assumed approx.15-Meter Tsunami】 ●Building seawalls ●Installing water-tight doors 【Measures to Secure/Speed Up Achievement of Cold Shutdown】 ●Installation of air-cooled diesel power generators ●Securing back-up electric motors for seawater pumps ●Actions needed for other necessary equipment

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Outline of Preparatory Measures for Response to Severe Accidents

Followings are measures to be implemented in the direction.

1) Secure working environment in the main control room

(Measures should be established so that the Main Control Room has a secure working environment with radiation protection, etc. by enabling the emergency ventilation and air conditioning facilities (recirculation system) to run on power supplied by a power supply vehicle in case all AC power supply is lost during an emergency .)

2) Secure means of communication inside the NPS in case of emergency

3) Secure supplies and equipment such as high-level radiation protective gear, and develop a system for radiation dose management

4) Establishing measures to prevent hydrogen explosion

5) Deploy heavy machinery for removing rubble

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2. Regulatory issues for FCFs after the accident

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2.1 Major responses at FCFs

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Major responses at FCFs[1] (1/2)

1. Emergency Safety Measures[2]

- NISA directed licensees of a reprocessing facility (JAEA and JNFL) to implement emergency safety measures. (1 May)
- Based on the reports from licensees and on-site inspections, NISA has confirmed that emergency safety measures had been appropriately implemented.(15 June)

2. Measures at Reprocessing Plants that Take into Account Severe Accident Response in NPSs[2]

- NISA instructed licensees of a reprocessing facility (JAEA and JNFL) to implement measures that take into account severe accident response in NPSs. (15 June)
- Based on the reports from licensees and on-site inspections, NISA has confirmed that the measures had been appropriately implemented. (8 July)

[1] No significant event, e.g. those which are subject to the reporting criteria, fire, injured person, has occurred at any FCFs due to the earthquake on March 11.
[2] Both measures 1 and 2 above were applied only to a reprocessing facility.

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Major responses at FCFs (2/2)

3. Comprehensive evaluation of safety of existing NPPs based on the Fukushima Dai-ichi NPP incident

- NSC directed NISA to report on comprehensive evaluation of safety on robustness against external events, which are beyond design basis, for all existing NPPs. (6 July, at the 50th extraordinary meeting of NSC)
- NISA proposed an evaluation method and implementation plan about the comprehensive evaluation. In this proposal NISA gave a following statement for FCFs. (21 July, at the 55th extraordinary meeting of NSC)

NISA examines to perform such kind of evaluation for FCFs and other related facilities separately.

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2.2 Emergency Safety Measures

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Outline

◆NISA conducted an exploration of the necessity and content of emergency response measures for various FCFs such as fuel processing and interim storage of spent nuclear fuel.

◆As a result, it was determined that the need for emergency safety measures in the event that tsunami and other phenomena induced loss of AC power, decay heat removal capability, as well as hydrogen accumulation prevention capability (hereinafter referred to as "total AC power loss"), due to the presence of highly radioactive solutions dissolved spent fuel, and the generation of hydrogen through a reaction between radioactivity and solutions such as highly-radioactive liquid waste.

◆Accordingly, NISA issues instructions to, and requires a response from, reprocessing plant licensees (JAEA and JNFL), while will be also revising the Rules for Reprocessing of Spent Fuel [1] (hereinafter referred to as "Rules for Reprocessing"), in order to ensure the effectiveness of such a process.

◆Furthermore, NISA issues instructions on further multiplexing emergency power generations at reprocessing facilities, in order to enhance the reliability of AC power sources.

◆With regard to FCFs other than reprocessing facilities, loss of power does not pose a safety concern.

[1] Ministerial ordinances related to the Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors

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Reason to pose no safety concern for other FCFs

Type of FCF	Reason
Fuel processing fabrication (U fuel fabrication, MOX fuel fabrication, U-enrichment)	The facilities have no cooling function by electric power and no necessity to consider H_2 explosions. For MOX fuel fabrication facilities it was confirmed that in the loss of power supply Pu decay heat can be removed with natural condition.
Interim storage of spent nuclear fuel	The facility under construction adopted metal casks which are cooled by natural air convection. Therefore, the facility has no cooling function by electric power and no necessity to consider H_2 explosions.
Radioactive waste disposal	The facilities has no cooling function by electric power and no necessity to consider H_2 explosions.
Radioactive waste management	

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Required items (1/3)

1. Emergency Safety Measures Taking into Account Tsunami and Other Phenomena

Measures listed below shall be taken, as emergency safety measures, in the event that a facility loses total AC power etc. due to tsunami and other phenomena, in order to prevent damage to spent fuel and control the discharge of radioactive materials while at the same time seeking to recover total AC power for the reprocessing facilities.

Additionally, an operational safety program shall be reviewed in accordance with future amendment(s) of the Rule for Reprocessing and an approval for the amendment of the operational safety program shall be applied for.

1) Implementation of Emergency Inspections

Implement emergency inspection of equipment and facilities in response to emergency situations caused by tsunami and other phenomena

2) Inspection and drilling of the Emergency Preparedness Plan

Inspect the emergency preparedness plan and conduct a drill, assuming total AC power loss

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Required items (2/3)

3) Securing of Power Sources in Emergencies
Secure an alternative power source to rapidly supply the necessary electric power in case of unavailability of emergency power due to power loss to the reprocessing facilities

4) Long-Range Measures Against Total AC Power Loss in Emergencies
In the event that all conventional AC power sources are lost, implement long-range measures for the total recovery of AC power sources

5) Implementation of Necessary Measures for the Time Being Taking into Account the Structure and other Aspects of Each Reprocessing Facility

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Required items (3/3)

2. Securing Multiple Stand-by Emergency Power Systems

In order for two emergency power systems to be operational at all times (includes the number of emergency power systems in the event that emergency power generation equipment in another building can supply power to the required emergency main lines, if there are multiple buildings within the reprocessing facility), review an operational safety program and develop plans to deploy the systems.
If a certain amount of time is required to bring the additional emergency power systems into operational state, the plan should also provide for responses during this time.

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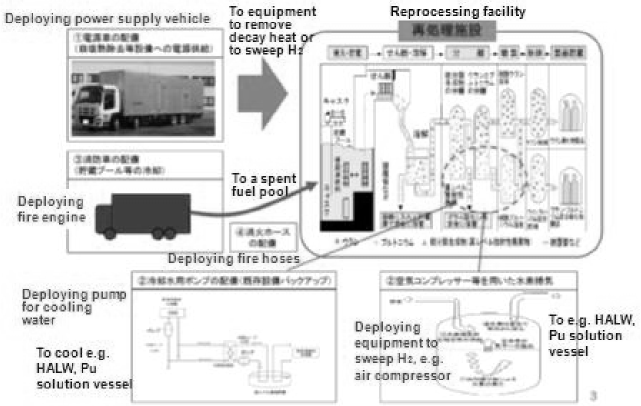
Example of safety measures for total AC power loss



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2.3 Major items implemented in emergency safety measures

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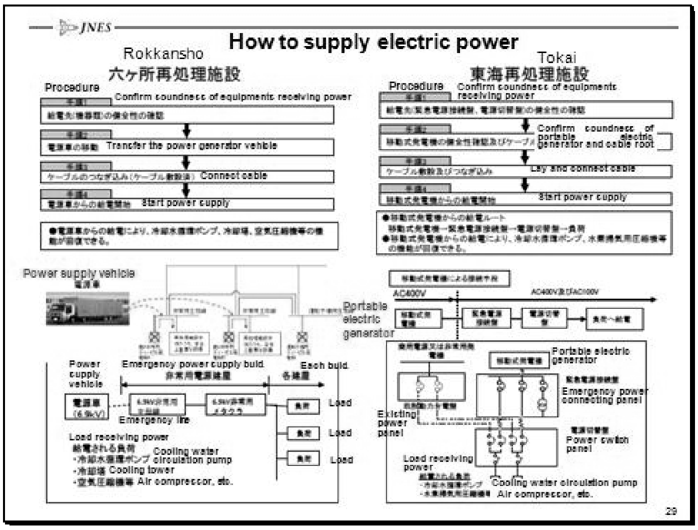

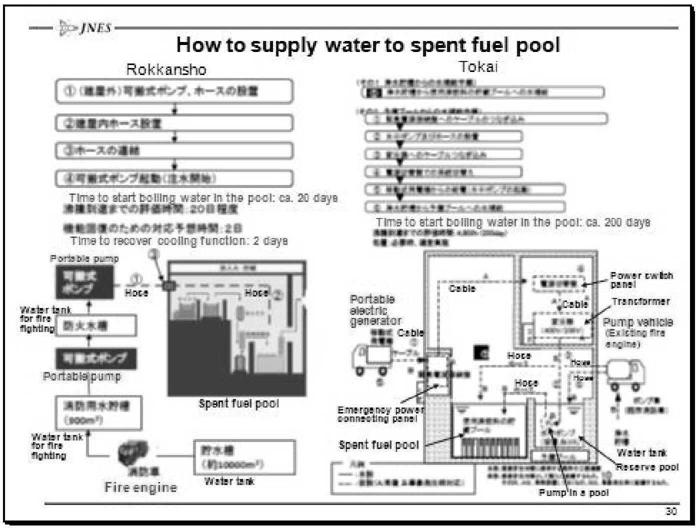

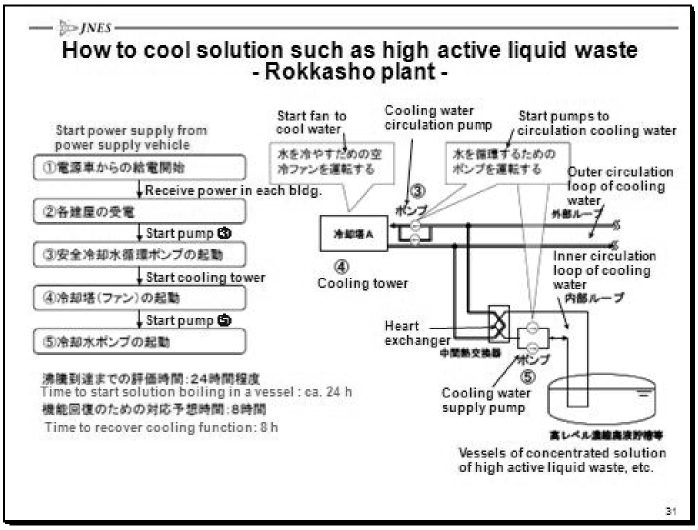
JNES
How to cool solution such as high active liquid waste
- Rokkasho plant -
Start power supply from power supply vehicle
①電源車からの給電開始
Receive power in each bldg.
②各建屋の受電
Start pump ③
③安全冷却水循環ポンプの起動
Start cooling tower
④冷却塔(ファン)の起動
Start pump ⑤
⑤冷却水ポンプの起動
沸騰到達までの評価時間:24時間程度
Time to start solution boiling in a vessel : ca. 24 h
機能回復のための対応予想時間:8時間
Time to recover cooling function: 8 h
Start fan to cool water
水を冷やすための空冷ファンを運転する
Cooling water circulation pump
Start pumps to circulation cooling water
水を循環するためのポンプを運転する
Outer circulation loop of cooling water
外部ループ
③ ポンプ
冷却塔A
④ Cooling tower
Inner circulation loop of cooling water
内部ループ
Heart exchanger
中間熱交換器
ポンプ
⑤ Cooling water supply pump
高レベル濃縮廃液貯槽等
Vessels of concentrated solution of high active liquid waste, etc.
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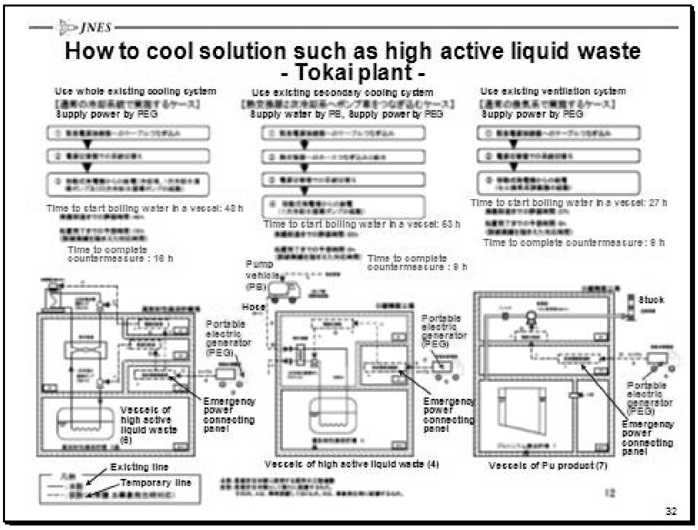
JNES
How to cool solution such as high active liquid waste
- Tokai plant -
Use whole existing cooling system
Supply power by PEG
Use existing secondary cooling system
Supply water by PB, Supply power by PEG
Use existing ventilation system
Supply power by PEG
Time to start boiling water in a vessel: 48 h
Time to complete countermeasure : 16 h
Time to start boiling water in a vessel: 63 h
Time to complete countermeasure : 9 h
Time to start boiling water in a vessel: 27 h
Time to complete countermeasure : 8 h
Pump vehicle (PB)
Hose
Portable electric generator (PEG)
Emergency power connecting panel
Vessels of high active liquid waste (8)
Vessels of high active liquid waste (4)
Vessels of Pu product (7)
Stack
Existing line
Temporary line
12
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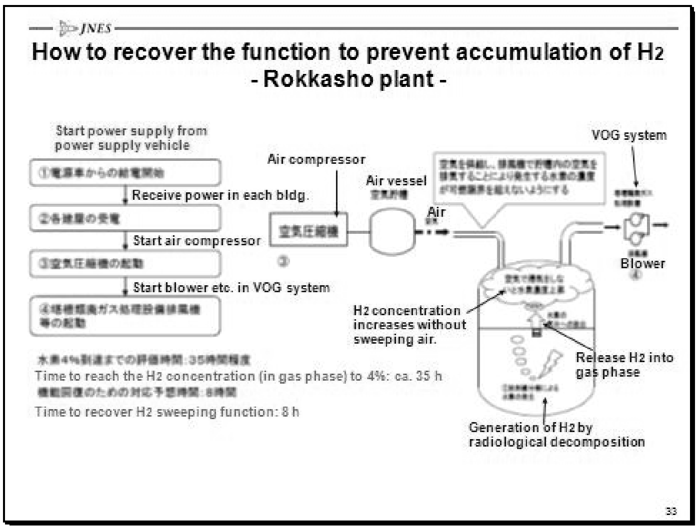
JNES
How to recover the function to prevent accumulation of H2
- Rokkasho plant -
Start power supply from power supply vehicle
①電源車からの給電開始
Receive power in each bldg.
②各建屋の受電
Start air compressor
③空気圧縮機の起動
Start blower etc. in VOG system
④塔槽類廃ガス処理設備排風機等の起動
水素4%到達までの評価時間:35時間程度
Time to reach the H2 concentration (in gas phase) to 4%: ca. 35 h
機能回復のための対応予想時間:8時間
Time to recover H2 sweeping function: 8 h
Air compressor
空気圧縮機
③
Air vessel
空気貯槽
Air
VOG system
Blower
④
H2 concentration increases without sweeping air.
Release H2 into gas phase
Generation of H2 by radiological decomposition
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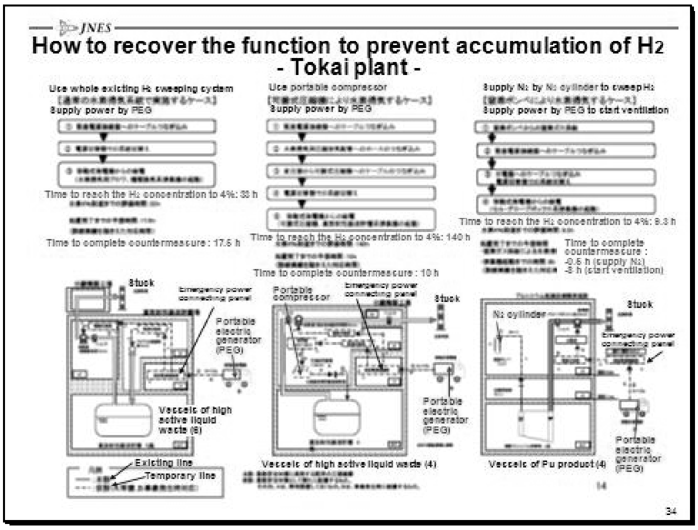

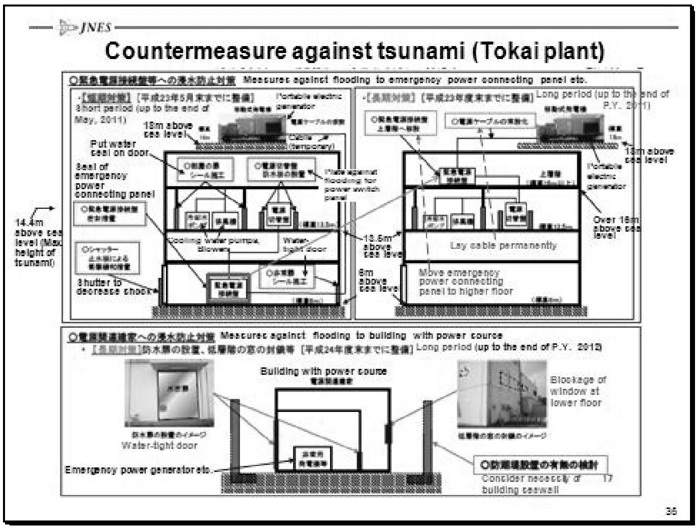

Outline of the mid- and long-term plan for Rokkasho plant

Mid- and long-term measures enhancing reliability (measures which start immediately but take a long time to accomplish them)				
Securing reserve equipments for quick removing decay heart and preventing H2 accumulation, Installing emergency power supply with large capacity to operate safety functions of a facility		Protective measures for tsunami		
Securing reserve equipments, e.g. pump	Installing emergency power supply with large capacity	Water-tight buildings and other facilities	Flood barrier	Building seawall
- Securing water source outside the site and related equipments etc. to inject water etc. (within ca. 3 months) - Equipments etc. to inject water to cooling coil by fire engine etc. (ca. within 1 year) - Deploying an air compressor with engine (within ca. 1 month)	- Deploying additional two power generator vehicles with 2000kVA (within F.Y. 2011) - Increase emergency power generator (3 to 4 years after accomplishment of detailed design)	- (no effect due to a tsunami [1])	- (no effect due to a tsunami [1])	- (no effect due to a tsunami [1])

[1] The site is located at 55 m above sea level and 5 km far from the sea coast.

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Outline of the mid- and long-term plan for Tokai plant

Mid- and long-term measures enhancing reliability (measures which start immediately but take a long time to accomplish them)				
Securing reserve equipments for quick removing decay heart and preventing H_2 accumulation, Installing emergency power supply with large capacity to operate safety functions of a facility		Protective measures for tsunami		
Securing reserve equipments, e.g. pump	Installing emergency power supply with large capacity	Water-tight buildings and other facilities	Flood barrier	Building seawall
Additional two pump vehicles for a cooling function (within ca. 6 months)	- Increase emergency power generator (within ca. 3 years)	- Moving emergency power connecting panel to higher floor (within 1 year) - Installing a water-tight door to a building with power source, blockage of a window at lower floor (within ca. 2 years)	-	Considering necessity of building seawall (within F.Y. 2011)

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Summary of the emergency safety measures (1/3)

◆The Tokai plant has been under PSR (Periodic safety review). (The PSR is planed to be continued up to March 2012.) The Rokkasho plant has been under pre-service tests (commissioning stage). Its main processes, e.g. shearing, resolution has not been operating and there is no plan to operate such main processes for the time being. (The commissioning stage is scheduled to completed in October 2012.)

◆This time the emergency measures were implemented for parts of the facilities which are operating and need safety measures, e.g. vessels with HALW or Pu solution etc., spent fuel pool. For remained parts of both facilities the measures from the same viewpoints are supposed to be implemented before their full operation. An appropriateness of measures will be also confirmed by NISA before their operation.

◆NISA has confirmed that emergency safety measures had been appropriately implemented based on the reports from licensees and on-site inspections. The on-site inspection for Rokkasho plant was performed on 1 and 2 June and that for Tokai on 6 and 7 June.

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JNES

Summary of the emergency safety measures (2/3)

◆Followings are typical items confirmed by NISA and JNES

- Appropriateness of equipments for safety measures

(No missing of equipments which need safety measures)

-Time to start solution boiling in a vessel and that to reach the H_2 concentration in gas phase to 4% (including appropriateness of equations and parameters used in the evaluations)

-Time to accomplish safety measures (Safety measures should be accomplished before starting solution boiling and reach the H_2 concentration in gas phase to 4%.)

-Capacity of a power generator vehicle or a portable electric generator

-Specification of cables (length etc.) and other relating equipments

-Operational procedure to supply power or water, manual and training, etc.

-Appropriateness of preparation and storage of equipments for safety measures

-Maintenance plan for equipments, etc.

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JNES

Summary of the emergency safety measures (3/3)

◆JNFL and JAEA revised their operational safety program and submitted an application to NISA for approval on 30 May and 3 June respectively.

◆The following were added to operational safety programs as items related to the organization of a system to ensure the safety of the reprocessing facility in the event of loss of all AC power and other functions:

i. Establish the necessary plan for carrying out actions to ensure the safety of the reprocessing facility in the event of loss of all AC power and other functions

ii. Deploy personnel needed to carry out actions to ensure the safety of the reprocessing facility in the event of loss of all AC power and other functions

iii. Train personnel needed to carry out actions to ensure the safety of the reprocessing facility in the event of loss of all AC power and other functions

iv. Deploy power supply vehicles and other equipment and supplies needed to carry out actions to ensure the safety of the reprocessing facility in the event of loss of all AC power and other functions

v. Assess these measures regularly, as well as implement the necessary measures based on these assessment results

In addition, operational safety programs were amended to ensure that 2 emergency generators were operable at all times.

◆NISA approved the revised operational safety programs on 15 June.

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JNES

2.4 Measures at Reprocessing Plants that Take into Account Severe Accident Response in NPSs

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JNES

Outline

◆NISA directed all electric power companies and JAEA to implement preparatory measures for response to severe accidents taking into account the accident at Fukushima Dai-ichi NPS . (7 June)

◆NISA instructed licensees of a reprocessing facility (JAEA and JNFL) to implement measures that take into account severe accident response in NPSs. (15 June)

◆Followings are measures to be implemented in the instruction.

1) Securing a work environment in a control room

(Verify that there are measures to ensure radiological and other protection for, and the preservation of, the work environment in the control room, in order to more smoothly carry out operations in the control room in an emergency, even in the event of loss of all AC power.)

2) Securing communication within a reprocessing facility in an emergency

3) Securing materials and equipment such as high-level radiation protective suits, and organizing a radiation control system

4) Deployment of heavy machinery for rubble removal

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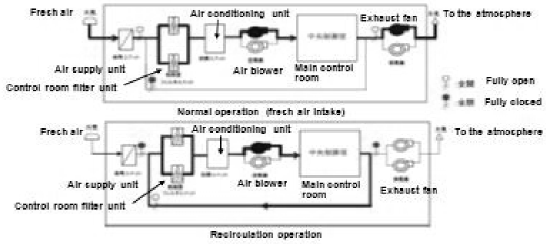
JNES

Major items implemented (1/7)

Securing a working environment in a control room - In the case of Rokkasho -

- Even in the event of a station blackout, the supply capacity of the power source vehicles that have been deployed in accordance with the emergency safety measures is sufficient to power the main control room's emergency ventilation and air conditioning system equipment (the recirculation system).
- The procedure manual required for operating the main control room's emergency ventilation and air conditioning system equipment during emergencies was prepared.
- The necessary materials and equipment, including replacement parts for the filters (high-performance particulate filters) of the main control room's emergency ventilation and air conditioning system equipment, are prepared.

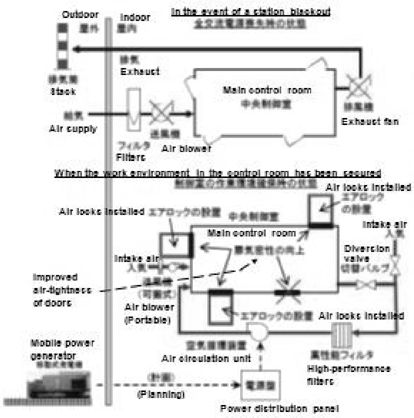


43

JNES

Major items implemented (2/7)

Securing a working environment in a control room - In the case of Tokai -



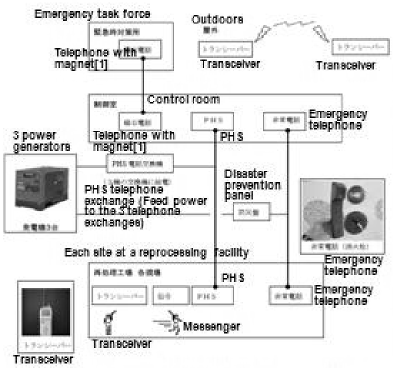
- As no emergency ventilation and air conditioning system equipment is installed in the recirculation system, ventilation and air conditioning equipment featuring iodine removal filters will be installed by the end of this fiscal year.
- In the event of a station blackout before such equipment has been installed, the valves in the air supply and exhaust lines will be closed manually in order to eliminate the air stream lines to and from the control room. In addition, workers in the control room will wear half-face masks. To prevent contamination, air locks will be provided for the control room doors. Procedure manuals were prepared for these measures.

44

JNES

Major items implemented (3/7)

Securing communication within a reprocessing facility in an emergency - In the case of Rokkasho -



[1] Telephone working with small battery, i.e. without power supply from outside

- The communications equipment that is used normally, such as on-site PHS and paging devices, will function on batteries, etc. for approx. one to three hours after a station blackout. In preparation for long-hour blackout, three power generators will be deployed by the end of July. Before the power generators become available, a means of communication will be secured by using transceivers (dry cell battery-operated) and messengers.

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JNES

Major items implemented (4/7)

Securing communication within a reprocessing facility in an emergency
- In the case of Tokai -

- On-site PHS and paging devices, etc. will become unusable in the event of a station blackout and tsunami. As a measure against such a situation, transceivers (operated by rechargeable or dry cell batteries) and portable wireless devices (rechargeable) will be secured as an alternative means of communication. In addition, relocation of on-site PHS equipment, etc. to higher place (the disaster prevention and management building scheduled to be built) is being planned.

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JNES

Major items implemented (5/7)

Securing materials and equipment such as high-level radiation protective suits, and organizing a radiation control system

- Lead-laced high-level radiation protective suits, full-face masks, Tyveks, personal dosimeters, contamination survey meters, ionization chamber survey meters, etc. are prepared in quantities sufficient to respond to accidents.
- In addition, based on the lessons learned from this accident, tungsten-laced high-level radiation protective suits are prepared. (In Rokkasho reprocessing plant)

[Example of a tungsten-laced high-level radiation protective suit]

Weight: approx. 18 kg
Shielding ability: equivalent to a reduction of about 20% of the dose of exposure

- Conventionally, Rokkasho reprocessing plant have had an agreement that allows licensees of nuclear energy related activity to cooperate with one another in lending materials and equipment as well as dispatching personnel during emergencies.
- In Tokai reprocessing plant, when there are shortages in materials, equipment, or radiation control personnel necessary for radiation control, supplementary materials and equipment and additional personnel assistance are available from the other sections of the laboratory as well as other laboratories belonging to organizations in the Ibaraki area. In addition, a system in which 18 nuclear sites in the Tokai area will cooperate during emergencies has been established in accordance with an agreement concluded between the nuclear sites.

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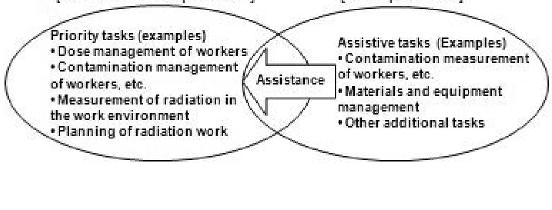
JNES

Major items implemented (6/7)

Securing materials and equipment such as high-level radiation protective suits, and organizing a radiation control system (continued)

- A system has been established to enable radiation control personnel to focus exclusively on priority tasks during emergencies by having other personnel perform the supportive tasks, such as dose management of workers, etc. and materials and equipment management. In addition, the monitoring and operating personnel for each process will be educated about radiation work. By introducing such systems and education, a system has been established to assist radiation control personnel during emergencies.



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Major items implemented (7/7)

Deployment of heavy machinery for rubble removal

- Heavy machinery will be deployed to remove rubble scattered by the earthquakes or that expected to be brought about by tsunamis.

[An example at Rokkasho reprocessing plant]

Wheel loader

Specifications

Overall length: approx. 6.1 m
Overall width: approx. 2.2 m
Height: approx. 3.1 m
Weight: approx. 6.7 t
Maximum breakout force: approx. 6,800 kgf
Bucket capacity: 1.3 m^3

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Summary of the measures at reprocessing plants that take into account severe accident response in NPSs (1/2)

◆Based on results of verification conducted so far, such as on site inspection of the implementation status of specific measures, NISA finds that measures reported by reprocessing operators are being carried out appropriately.

◆With regard to JNFL's planned installation, such as that of iodine removal filters in the emergency ventilation and air conditioning system for the control room, and measures planned by JAEA prior to the facility going online, such as the installation of a recirculation-type emergency ventilation and air conditioning system for the control room, NISA will stringently verify the implementation status of these actions through on-site inspections and other means prior to the facilities going online.

◆NISA will also subject to close scrutiny the implementation status of measures to be completed in the future such as relocation of facilities, deployment of materials and equipment, the organization of operational procedure that takes the previous 2 activities into account and the fact of ongoing improvement through training and other means, using safety inspections and other methods.

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Summary of the measures at reprocessing plants that take into account severe accident response in NPSs (2/2)

◆In the coming days, NISA will continue to work on even greater accident management readiness for beyond design basis events[1], from the standpoint of further improving the safety of reprocessing facilities.

◆Furthermore, NISA will encourage reprocessing operators to continue working on necessary improvements in the future, and pursue further enhancement of these measures.

[1] Design basis events refer to events that have been selected to consider in designing, and in assessing the design of a reprocessing facility. These events were selected through an identification and analysis of many events with the potential to place a reprocessing facility in an abnormal state.

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3. Further activities for FCFs

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Further activities for FCFs (1/2)

1. Emergency safety measures for the remained parts of a facility

◆For remained parts of the two facilities[1], measures from the same viewpoints are supposed to be implemented prior to the facilities going on full-operation. NISA will confirm an appropriateness of measures prior to the full-operations.

◆The Rokkasho plant is scheduled to complete the active test (commissioning stage) by October 2012 before its commercial operation. Therefore, emergency safety measures for its remained parts, which are main processes, e.g. shearing, dissolution , solvent extraction, denitration, and the confirmation of them by NISA will be conducted in the period.

2. Improvement of accident managements[2]

◆An improvement of accident management on the occasion of occurrences of beyond design basis events is planed to be conducted from the viewpoint of further enhancing safety of a reprocessing facility.

◆A preparation for this activity, e.g. discussions on how to select events and conditions in evaluations, is now underway.

[1] Refer the second paragraph in page 38.
[2] In the evaluation report of "measures at reprocessing plants that take into account severe accident response in NPSs " an intention to improve accident managements was announced. (18 June) (refer page 51)

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Further activities for FCFs (2/2)

3. Comprehensive evaluation of safety of existing NPPs based on the Fukushima Dai-ichi NPP incident

◆NSC directed NISA to report on comprehensive evaluation of safety on robustness against external events, which are beyond design basis, for all existing NPPs. (6 July, at the 50th extraordinary meeting of NSC)

◆NISA proposed an evaluation method and implementation plan about the comprehensive evaluation. In this proposal NISA gave a following statement for FCFs. (21 July, at the 55th extraordinary meeting of NSC)

NISA examines to perform such kind of assessment for FCFs and other related facilities separately.

◆Discussions on the assessment are underway.

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We would like to express our most sincere appreciation for your assistance.

ARIGATO

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Supplemental information

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Major information in English

Title (Date)	URL
Regarding the Implementation of Emergency safety measures for the Other Nuclear Power Stations considering the Accident of Fukushima Dai-ichi and Dai-ni Nuclear Power Stations (March 30)	http://www.nisa.meti.go.jp/english/files/en20110427-5.pdf
The Implementation of Emergency Safety Measures at Reprocessing Facilities Taking into Account the Accident at Fukushima Dai-ichi Nuclear Power Station, Tokyo Electric Power Co. Inc. (May 1)	http://www.nisa.meti.go.jp/english/files/en20110505-1.pdf
Regarding the Confirmed Results for the Implementation of the emergency safety measures for other Nuclear Power Stations Based on the Accident in Fukushima Dai-ichi Nuclear Power (May 1)	http://www.nisa.meti.go.jp/english/files/en20110512-1.pdf
Regarding Implementation of Preparatory Measures for Severe Accidents in Other NPSs Taking into Account the 2011 Accident at Fukushima Dai-ichi NPS of Tokyo Electric Power Co. Inc. (June 7)	http://www.nisa.meti.go.jp/english/press/2011/06/en20110615-1.pdf
Regarding the Approval of the Amendment to Operational Safety Programs Related to the Reorganization of a System to Implement Activities for Protecting Reprocessing Facilities (June 15)	http://www.nisa.meti.go.jp/english/press/2011/07/en20110722-8.pdf
Regarding Verification Results of the State of Implementation of Preparatory Measures for Response to Severe Accidents in Other NPSs Taking into Account the Accident at Fukushima Dai-ichi NPS (June 18)	http://www.nisa.meti.go.jp/english/press/2011/07/en20110729-2.html
Regarding the Results of Verification into the Implementation Status of Measures at Reprocessing Plants that Take into Account Severe Accident Response in Nuclear Power Stations (July 8)	http://www.nisa.meti.go.jp/english/press/2011/08/en20110822-2.html

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Abbreviation

FCF	Fuel Cycle Facility
HALW	High Active Liquid Waste
JAEA	Japan Atomic Energy Agency
JNES	Japan Nuclear Energy Safety Organization
JNFL	JAPAN NUCLEAR FUEL LIMITED
NISA	Nuclear and Industrial Safety Agency
NPP	Nuclear Power Plant
NPS	Nuclear Power Station
NSC	Nuclear Safety Commission of Japan

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Lessons learned from the accident thus far (1/3)

Japanese government report to the IAEA Ministerial Conference on Nuclear Safety identified 28 lessons in 5 categories.

Lessons in category 1

Strengthen preventive measures against a severe accident

(1) Strengthen measures against earthquakes and tsunamis
(2) Ensure power supplies
(3) Ensure robust cooling functions of reactors and PCVs
(4) Ensure robust cooling functions of spent fuel pools
(5) Thorough accident management (AM) measures
(6) Response to issues concerning the siting with more than one reactor
(7) Consideration of NPS arrangement in basic designs
(8) Ensuring the water tightness of essential equipment facilities

Lessons in Category 2

Enhancement of response measures against severe accidents

(9) Enhancement of measures to prevent hydrogen explosions
(10) Enhancement of containment venting system
(11) Improvements to the accident response environment
(12) Enhancement of the radiation exposure management system at the time of the accident
(13) Enhancement of training responding to severe accidents

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Lessons learned from the accident thus far (2/3)

Japanese government report to the IAEA Ministerial Conference on Nuclear Safety identified 28 lessons in 5 categories.

Lessons in Category 2 (continued)

Enhancement of response measures against severe accidents

(14) Enhancement of instrumentation to identify the status of the reactors and PCVs
(15) Central control of emergency supplies and equipment and setting up rescue team

Lessons in Category 3

Enhancement of nuclear emergency responses

(16) Responses to combined emergencies of both large-scale natural disasters and prolonged nuclear accident
(17) Reinforcement of environmental monitoring
(18) Establishment of a clear division of labor between relevant central and local organizations
(19) Enhancement of communication relevant to the accident
(20) Enhancement of responses to assistance from other countries and communication to the international community
(21) Adequate identification and forecasting of the effect of released radioactive materials
(22) Clear definition of widespread evacuation areas and radiological protection guidelines in nuclear emergency

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Lessons learned from the accident thus far (3/3)

Japanese government report to the IAEA Ministerial Conference on Nuclear Safety identified 28 lessons in 5 categories.

Lessons in Category 4
Reinforcement of safety infrastructure

(23) Reinforcement of safety regulatory bodies
(24) Establishment and reinforcement of legal structures, criteria and guidelines
(25) Human resources for nuclear safety and nuclear emergency preparedness and responses
(26) Ensuring the independence and diversity of safety systems
(27) Effective use of probabilistic safety assessment (PSA) in risk management

Lessons in Category 5
Thoroughly instill a safety culture

(28) Thoroughly instill a safety culture

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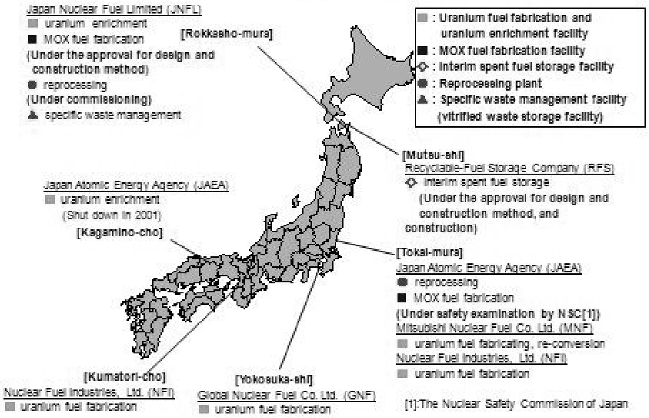
Fuel Cycle Facilities in Japan



62

FOLLOWING THE FUKUSHIMA ACCIDENT: ACTIONS IN PROGRESS IN FRANCE

D. Conte, L. Tabard ***(ASN, France)***

NO PAPER AVAILABLE

asn

Following the Fukushima accident :

Actions in progress in France

ASN/DRC
D. Conte, L. Tabard

asn

First actions started in France

- **The ASN asked French licensees to undertake stress tests that are called "Complementary safety assessments" (CSA)**
  - Request from the Prime minister to the ASN
  - Request from the European Commission to organize « stress tests »
  - Specifications of stress tests written based on the WENRA specifications
  - Specifications imposed by ASN regulatory decisions of the 5th May 2011 (one decision for each concerned licensee)
- **The ASN has undertaken specific inspections linked to stress tests to control the conformity of nuclear installations**
  - All NPP + non-NPP that were determined as priorities for 2011
  - Themes of the inspections : stress tests subjects
  - about 2 to 4 days for each nuclear site - a total of 110 inspection-days to be performed before the end of 2011
- **The Ministry of the Interior has evaluated the crisis management in France**

asn

Stress tests

- **All French nuclear installations are submitted to stress tests, according to the following prioritization :**
  - In 2011 : all NPP (58 reactors)
  - Non-NPP : from 2011 to 2012
- **Specifications of stress tests :**
  - Common provisions
  - Specific provisions for NPP and for non-NPP
- **Six issues :**
  - Flooding risks
  - Earthquakes
  - Loss of electrical power
  - Loss of cooling systems
  - Accidental situations management
  - Management of subcontractors
- **Stress tests are complementary to safety approaches implemented by licensees under the ASN control**
- **Goal : determine whether improvements are necessary for each nuclear installation**

French non-NPP facilities

- **90 facilities**
- **21 licensees**
- **Various types of installations**
  - Research reactors
  - Medical radionuclides manufacturer
  - Research laboratories
  - Uranium enrichment plants
  - Fuel manufacturers
  - Fuel reprocessing plants
  - Nuclear waste storages
  - Plants under decommissioning (all kinds of facilities : graphite-gas reactors, laboratories, fuel cycle plants…)
  - Irradiators

French non-NPP facilities

❑ **Approach to prioritize the non-NPP facilities based on the following criteria :**

- The degree up to which the plant could be damaged or disabled by each of the 5 stress-tests issues, and its sensitivity to cliff effect
- The maximum quantities of radioactive materials that could be involved in an accident

⇒ Quote of the facilities

❑ **The quote resulted in a 3-level prioritization :**

- Stress tests to achieve in 2011
- Stress tests to achieve in 2012
- Feedback to take into account while assessing the next periodic safety review (PSR)

French non-NPP facilities

Licensees	Stress tests in 2011	Stress tests in 2012	Feedback in next PSR
CEA	5	9	22
AREVA group	14	1	1
EDF (except NPP)		10	6
ILL (High Flux Reactor)	1		
CIS BIO		1	
Iter Organization		1	
Others			10
TOTAL	**20**	**22**	**39**

Types of facilities	Stress tests in 2011	Stress tests in 2012
Research reactors	5	4
Fuel cycle facilities	14	1
Plants under decommissioning	1	10
Waste treatment		3
Research plants		3
Nuclear storages		1
TOTAL	**20**	**22**

French non-NPP facilities

Stress tests for the French fuel cycle facilities :

2011 :

- **The La Hague site (UP2-400, HAO, ELAN IIB, STE2, UP3-A, UP2-800 and STE3 plants)**
- **The Tricastin site (Comurhex, TU5/W, EURODIF, GB II and SOCATRI plants)**
- **The MELOX facility**
- **The FBFC plant (Romans site)**

2012 :

- **The CERCA plant (Romans site)**

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Provisional calendar for 2011

- 5 May 2011 : ASN regulatory decisions imposing stress tests specifications
- 1st June 2011 : deadline for sending of licensees methodologies to achieve stress tests
- 6 July 2011: review of the licensees methodologies by Standing Groups of experts (reactors, laboratories, fuel cycle facilities)
- 19 July 2011 : notification of the ASN opinion concerning the licensees methodologies
- 15 September 2011 : deadline for licensees to send stress tests for facilities determined as priorities for 2011
- 15 October 2011 : deadline for performing the specific inspections concerning facilities determined as priorities for 2011
- 8, 9 et 10 November 2011 : review of the licensees stress tests by groups of experts
- End of November 2011 : ASN conclusions concerning stress tests
- December 2011: ASN regulatory decisions concerning facilities determined as priorities for 2011

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Results of the review of licensees methodologies

The ASN estimated that the licensees methodologies were globally satisfying, but asked that the licensees take into account some very specific issues.

For example, the ASN asked the AREVA group to take into account, on the Tricastin site, the hypothetical accidental situation of the loss of the protections against flooding because of the break of the dams of the Donzère canal.

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WGFCS – 26 September 2011 – actions in progress in France following the Fukushima accident 10

Thank you for your attention

WGFCS – 26 September 2011 – actions in progress in France following the Fukushima accident 11

UK RESPONSE TO FUKUSHIMA: SPECIFIC REFERENCE TO FUEL CYCLE FACILITIES

N. Blundell ***(ONR, UK)***

NO PAPER AVAILABLE

UK Office for Nuclear Regulation (ONR)

UK Response to Fukushima

Specific Reference to Fuel Cycle Facilities

September 2011

Office for Nuclear Regulation
An agency of HSE

UK Response to Fukushima

Specific Reference to Fuel Cycle Facilities

Neil Blundell
HM Principal Inspector (Nuclear)
OECD NEA WGFCS Workshop on Safety Assessment of Fuel Cycle Facilities
Toronto Canada September 2011

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Presentation overview

- Fukushima Event
- UK Response (ONR and Government)
- Interim Report (Conclusions & Recommendations)
- Relevance to Workshop
- Post Interim Report Events and Link to Stress Tests
- Extents of Stress Test
- UK Severe Weather examples

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11 March 2011

- Magnitude 9 earthquake
- Subsequent tsunami
- 14-15m Fukushima 1
- Loss of cooling – loss of containment
- Flooding

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UK Response

- Set up Redgrave Court Incident Suite (RCIS)
- Advice to SAGE and COBR
- SAGE - Scientific Advisory Group for Emergencies
- COBR - Cabinet Office Briefing Room
- Links with International Stakeholders
- Prompt assurance of UK fleet

  'Protecting People and Society'

  17000 UK Nationals in Japan

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Secretary of State Request

- Identify any lesson to be learnt by the UK nuclear industry
- Co-operate and co-ordinate with international colleagues, to include 'stress test' requirements
- Interim report by the middle of May 2011
- Final report within 6 months

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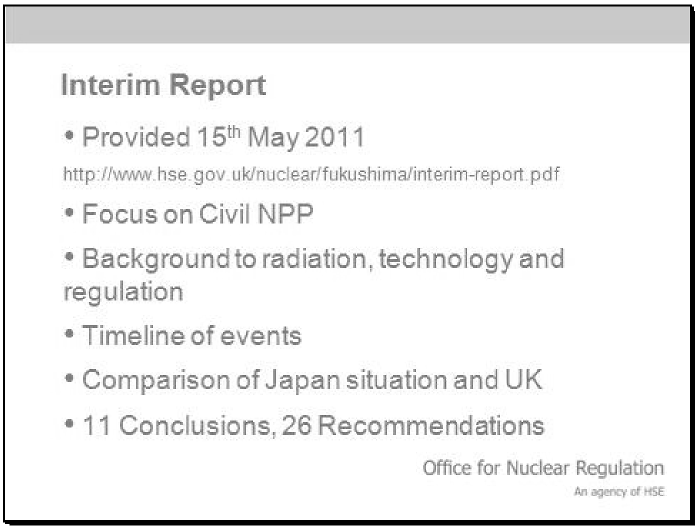
Interim Report
• Provided 15th May 2011
http://www.hse.gov.uk/nuclear/fukushima/interim-report.pdf
• Focus on Civil NPP
• Background to radiation, technology and regulation
• Timeline of events
• Comparison of Japan situation and UK
• 11 Conclusions, 26 Recommendations
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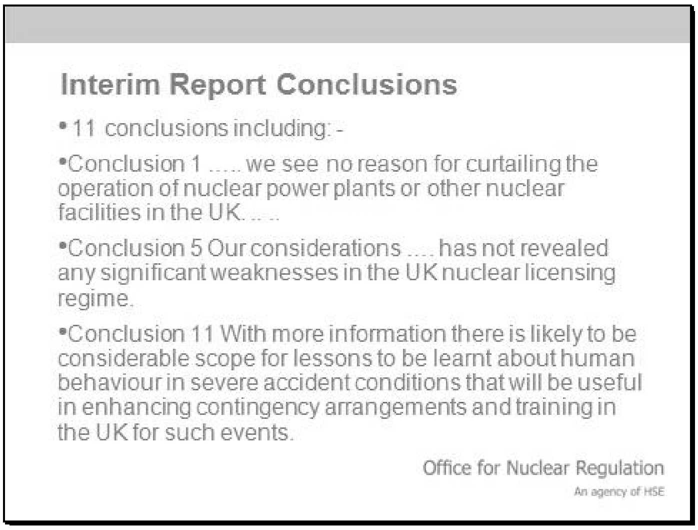
Interim Report Conclusions
• 11 conclusions including: -
•Conclusion 1 ..... we see no reason for curtailing the operation of nuclear power plants or other nuclear facilities in the UK. .. ..
•Conclusion 5 Our considerations .... has not revealed any significant weaknesses in the UK nuclear licensing regime.
•Conclusion 11 With more information there is likely to be considerable scope for lessons to be learnt about human behaviour in severe accident conditions that will be useful in enhancing contingency arrangements and training in the UK for such events.
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26 Interim Report Final Recommendations
• Although there were 26 recommendations the UK regulatory regime is robust
•The recommendations derive from the UK's philosophy of continuous improvement
•It would be complacent to suggest that we had nothing to learn from the Fukushima accident or that safety in any walk of life could not be improved.
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Interim Report Recommendations

- General
- Relevant to the Regulator
- Relevant to the Nuclear Industry
- Way Forward

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Relevance to this Workshop

Safety Assessment Approach

Recommendation 5 …ONR should undertake a formal review of the Safety Assessment Principles … particularly for "cliff-edge" effects.

Off-site Infrastructure Resilience

Recommendation 9 …the UK nuclear industry should review what lessons can be learnt from ….Fukushima-1 and Fukushima-2 sites.

Impact of Natural Hazards

Recommendation 10 …………… initiate a review of flooding studies…. to confirm the design basis and margins for flooding at UK nuclear sites, and whether there is a need to improve ….

Spent Fuel Strategies

Recommendation 12 ……ensure the adequacy of any new spent fuel strategies compared with the expectations in the Safety Assessment Principles ……..

Seismic Resilience

Recommendation 15 …….. the UK nuclear industry should consider any implications for improved understanding of the relevant design and analyses.

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Relevance to this Workshop (2)

Safety Case

Recommendation 25 The UK nuclear industry should review, and if necessary extend, analysis of accident sequences for long-term severe accidents. This should identify appropriate repair and recovery strategies to the point at which a stable state is achieved, identifying any enhanced requirements for central stocks of equipment and logistical support.

Way forward

Recommendation 26 A response to the various recommendations in the interim report should be made available within one month of it being published. These should include appropriate plans for addressing the recommendations. Any responses provided will be compiled on the ONR website.

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Post Interim Report Events

- Responses received in line with Rec. 26
- IAEA Fact Finding mission led by UK Chief Inspector
- Engagement to define Stress Test
- Ongoing receipt of submissions
- Production of the Final Report

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Final Report Contents

- Update to Interim Report
- Incorporation of information from IAEA and other sources
- Latest position regarding stress test
- Inclusion of all UK Nuclear Facilities

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On-going ONR activities

- Provision of stress test National Report
- Participation in European peer review
- Subsequent report for emerging information
- Additional missions to Japan
- IAEA action plan from Ministerial Conference

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EC Stress Test

- European Council 24/25 March – safety of all EU nuclear plants should be reviewed
- WENRA developed specification
- ENSREG agreed
- Review design
- Check consequences LOOP/cooling
- Consider severe accident management issues

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UK Extent of EC Stress Test

- Duty holders to perform by Oct 11
- UK also undertaking non-NPP same criteria
- National regulators produce national report by Dec 11
- WENRA task force peer reviews national reports by Apr 12

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UK Extreme Weather Examples – Flooding (1)

Event

- June 2007 following a period of prolonged heavy rainfall a number of buildings on an inland site, involved in the processing of enriched uranium, encountered 'significant' water ingress, in some areas with minor water ingress in others.
- Local controls to deal with the water ingress were put in place in accordance with Site Contingency Plan for Severe Weather Conditions.
- Under normal rainfall conditions the rainwater flows into the 1800mm dia. drains across the site and 'backs up' until the level causes 'flow' down the interconnections into 900mm dia. drains and away to the 'outfall'.

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UK Extreme Weather Examples – Flooding (2)

Analysis

•Under 'torrential conditions' the 1800mm dia drains provides a holding volume, which will give time for the rain to 'ease off' and also for the water to drain down the 900mm dia. drain towards the outfall.

•In exceptional circumstances the 1800mm drains will back up completely, the 900mm drains will not be able to cope with the volume of water and the 'lower end' to the east of the site may flood. This may result in drains in buildings backing up and in some cases manhole/inspection covers being lifted

•*Development of the site in 'recent' years, including several large buildings and storage rafts may have contributed to the additional flow through site surface water drains, where previously the rainfall would percolate through soil, sub-soil etc. and not enter the site surface water system.*

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UK Extreme Weather Examples - High Winds

Taken directly from Construction News 17March 1994 :

FALLING SCAFFOLDING KILLS BUILDER AT SELLAFIELD SITE.

•A construction worker was trapped and crushed to death by collapsing scaffolding at the Sellafield nuclear power site on Sunday. Safety inspectors believed high winds caused the fall which killed John Graham from Workington.

•Five other workers were injured during the collapse and Health and Safety Executive investigators returned to the Amec run site yesterday (Wednesday) to take witness statements.

•Mr Graham was helping build a store room on the EPS (2) encapsulation plant phase of the nuclear complex in Cumbria.

•A British Nuclear Fuels spokeswoman said: 'It appears that a combination of high winds and the heavy weight of the scaffolding led to the tragedy.'

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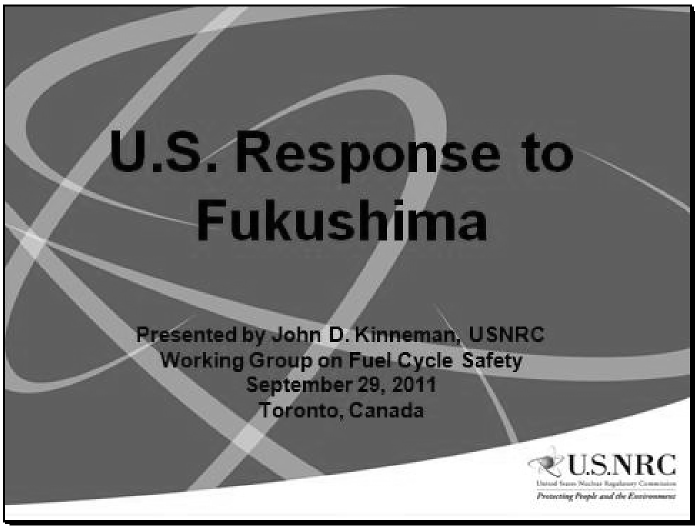
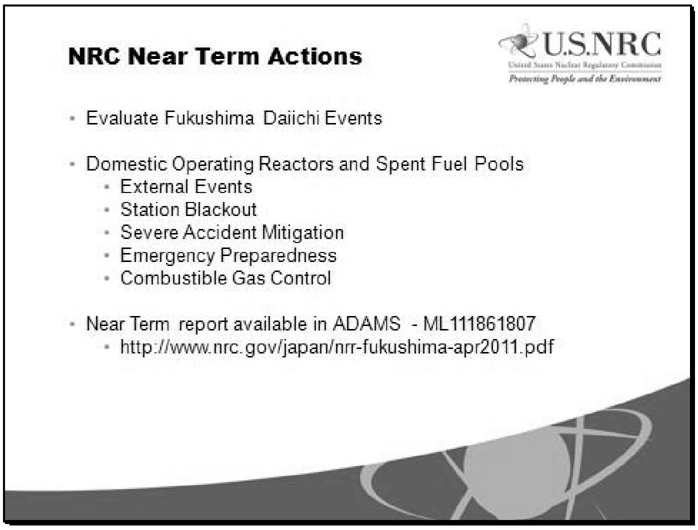
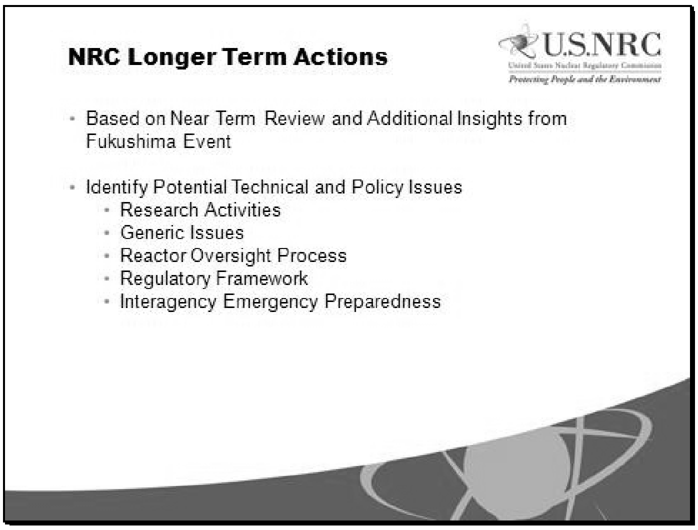
U.S. RESPONSE TO FUKUSHIMA

J. Kinneman ***(USNRC, USA)***

NO PAPER AVAILABLE


U.S. Response to Fukushima
Presented by John D. Kinneman, USNRC
Working Group on Fuel Cycle Safety
September 29, 2011
Toronto, Canada
U.S.NRC
Protecting People and the Environment


NRC Near Term Actions
U.S.NRC
Protecting People and the Environment
• Evaluate Fukushima Daiichi Events
• Domestic Operating Reactors and Spent Fuel Pools
• External Events
• Station Blackout
• Severe Accident Mitigation
• Emergency Preparedness
• Combustible Gas Control
• Near Term report available in ADAMS - ML111861807
• http://www.nrc.gov/japan/nrr-fukushima-apr2011.pdf


NRC Longer Term Actions
U.S.NRC
Protecting People and the Environment
• Based on Near Term Review and Additional Insights from Fukushima Event
• Identify Potential Technical and Policy Issues
• Research Activities
• Generic Issues
• Reactor Oversight Process
• Regulatory Framework
• Interagency Emergency Preparedness


U.S. Actions Related to Fuel Facilities
U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment
Currently no safety concerns with the fuel cycle facilities
Verify that the licensees' mitigation strategies for each of the licensing basis events are:
Properly implemented
Prevention and/or mitigation strategies appropriate for the consequence
Licensing basis events:
Seismic hazards
Flooding hazards
Wind and tornado loading
Extended loss of AC power and emergency power
Fire impacts
Evaluate the adequacy of emergency prevention and/or mitigation strategies for consequences of selected beyond licensing basis events

COMPLIMENTARY SAFETY ASSESSMENT OF THE FUEL CYCLE FACILITIES IN FRANCE

P. Nocture ***(AREVA, France)***

NO PAPER AVAILABLE

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Complementary Safety Assessments (CSA) of the Fuel Cycle Facilities in France

AREVA Methodology

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Sites submitted to CSA

- La Hague: **Reprocessing**
- Melox: **MOX Fuel Fabrication**
- Tricastin: **Conversion and Enrichment**
  - EURODIF, SET: Enrichment
  - AREVA NC: De-conversion, storage
  - SOCATRI
  - COMURHEX Conversion
- Romans/Isère: **Fuel fabrication**
  - FBFC/ CERCA

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AREVA Methodology

- **Objectives :**
  - Determine the extent of facility resistance to external natural phenomena postulated beyond the current requirements of design basis, and to the postulated loss of some safety functions, with special care on potential cliff-edge effects
  - Verify that resources that can be called-up are sufficient to limit the consequences of severe accidents that may happen simultaneously on several facilities and to limit the impacts of releases into the environment
  - If needed, identify the arrangements to reinforce the organization, or the equipment

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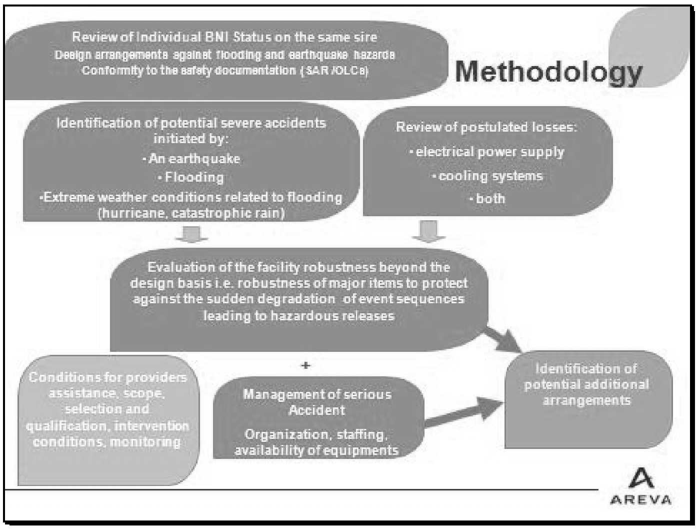

Installation Robustness 1/2

- Goal: the qualification of the barrier robustness by focusing on the deciding factors
- **The robustness assessment is based on:**
  - Available design margins,
  - The diversity of existing independent barriers
  - The multiple redundancies of protection systems (and associated required utilities)),
  - The various technical and organizational means if the intervention delays are compatible with the kinetics of the scenario which is looked at
- The quantification of the threshold above which the key barriers and components cannot fulfill the behavior requirements (and leads to a severe accident)

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Installation Robustness 1/2

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- The quantification of the threshold above which the key barriers and components cannot fulfill the behavior requirements (and leads to a severe accident)

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Installation Robustness 2/2

- The appreciation of margins available for those essential structures and equipments will rest on the assessment of a group of experts, ouside AREVA and recognized by their peers:
  - Experienced in the civil works, and the structure and equipment resistance
  - Members of the AFPS (French Association of Earthquake Engineering)
  - Assessment based of the technical analysis prepared for the studies of serious event scenarios, including design basis documents and the ones use for the periodic safety reassessment of the facilities
  - Input from experts' visits inside the facilities

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Accident Management

- Goal: the analysis of the technical and organizational means that need to be implemented during a beyond design accident that impacts one or several facilities of the same site:
  - To reach a minimum safety level
  - To limit the accident consequences on the site environment
- Identification:
  - The technical and human means require at the site levels
  - The common support means proposed by the Group (+ management of supply)
  - The adequacy of means to the needs
  - The delays for implementing the means with account of
    - the degradation by the initiating event of the infrastructures and equipments (e.g. instrumentation, lack of electrical power) and
    - the constrains to the intervention by consequential events (fire, explosion, criticality event...)

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AREVA main conclusions

- For AREVA front-end facilities :
  - The facility designs provide for a quick return to the safe state.
  - Major replacement facilities in construction (GBII, COMURHEX II) with safety enhancement already implemented
  - No need for active safety functions to maintain the safe state: no strong need for technical means to provide electrical power and cooling in a short period of time (but loss of surveillance instrumentation)
- For back-end facilities :
  - Strong robustness to postulated natural aggressions, in particular in case of an earthquake
  - All the hazards resulting from the CSA scenarios are those used for the facility design
  - For those facilities, keeping the safe state requires to recover some active safety functions in a limited time period but CSA shows that adequate arrangements can face the worse common mode situations
- Launch of additional studies on the crisis management:
  - Need to assess the mitigation means in case of major events affecting the whole site at the same moment
  - Communication and information
  - Better identification of human and technical AREVA resources that can be mobilized
  - Arrangements to enhance the training and readiness for a lasting situation.

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A National Request and Schedule…

- 23 mars 2011: Prime Minister's request for safety audits and improvements of all nuclear facilities at the light of Fukushima event
- 5 May 2011: Detailed ASN "Decisions"
- 1 June 2011: AREVA detailed methodology sent to ASN
- 15 September 2011: AREVA CSA sent to ASN

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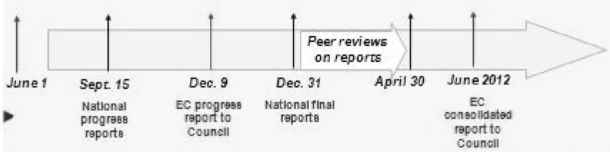
… within an EU process

- Timing: assessments undertaken by operators before June 1, 2011



- Scope: extreme natural hazards (earthquakes, flooding…) and their consequences + National specific demands, i.e. in France, the level of sub-contracting

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CSA Scope

a) Initiating events

- Earthquake
- Flooding

b) Consequences of the loss of safety functions following any conceivable event on the facility site

- Loss of electrical power,
- Loss of cooling
- Combination of both

c) Operational management of severe accidents

For b) and c) earthquake, tsunami , flooding (whatever origin), extreme weather conditions

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Review of the BNIs Status

- **Based on existing information in available reference documentation:**
- **Description of the site and its environment**
  - Geographical location
  - Environment
  - Access
  - Seismic and flooding hazards
  - Description of the site activities
  - History
- **Description of the facilities**
  - Site or facility organization (in operational states and accident conditions)
  - Inventory of hazardous (radioactive and chemical) materials
  - Inventory of hazards and conditions to keep the facilities safe
  - Presentation of the accidents defined in the emergency plan
  - Conformity of the facility to its reference basis

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Adaptation of « Severe Accident » to FCFs 2/3

- **1/ Hazard quantification :**

  Quantities of hazardous material (radioactive or chemical) which could be released during the accident :
  - Authorized quantities for facilities that are not planned for shut-down or are planned for commissioning (e.g.: La Hague, MELOX, GBII, ...)
  - Quantities which are really in the facilities that are planned for a closed shut-down (e.g.: GBI, COMURHEX Pierrelatte, ...),
  - Residual quantities for shut-down facilities (e.g.: UP2-400).
- **2/ Active and passive safety functions**
- **3/ Identification of the gravity of consequences:**
  - For events affecting the environment, the gravity is estimated with account of the capabilities and delays of remediation of the induced negative effects (pollutions, contamination,...)
  - Effect on people (radiological and toxicological effects)

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Main Steps

For all the BNIs of the same site:

- Review of the status including
  - The provisions already taken in the design basis
  - The verification of the conformity to the current design requirements
- Adaptation of the concept of « severe accidents »
- Assessment of the robustness of key structure/equipment during postulated beyond current design scenarios
- Assessment of means that can be raised in case of severe accident occurring on several installations for limiting the consequences

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Identification of potential complementary arrangements

- To prevent serious accident
- To manage the accident crisis:
  - Adequacy of the intervention promptness with the kinetics of the situation degradation
  - Duration of the mobilization of special intervention means

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