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Operational and Regulatory Aspects of Criticality Safety

Workshop Proceedings
19-21 May 2015
Albuquerque, New Mexico
United States

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Organisation de Coopération et de Développement Économiques
Organisation for Economic Co-operation and Development

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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

Operational and Regulatory Aspects of Criticality Safety

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ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 35 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.

The OECD member countries are: Australia, Austria, Belgium, Canada, Chile, the Czech Republic, Denmark, Estonia, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Israel, Italy, Japan, Korea, Latvia, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Slovak Republic, Slovenia, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The European Commission takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1 February 1958. Current NEA membership consists of 31 countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, Norway, Poland, Portugal, Russia, 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;
- 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

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of senior scientists and engineers with broad responsibilities for safety technology and research programmes, as well as representatives from regulatory authorities. It was created 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 among NEA member countries. The main tasks of the CSNI 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 reach 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 priority of the CSNI 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-operative mechanisms with the NEA Committee on Nuclear Regulatory Activities (CNRA), which is responsible for issues concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with other NEA Standing Technical Committees, as well as with key international organisations such as the International Atomic Energy Agency (IAEA), on matters of common interest.

ACKNOWLEDGEMENTS

Gratitude is expressed to the United States Nuclear Regulatory Commission (US NRC) for hosting the workshop, and to the workshop technical committee, the session chairpersons and the workshop participants for their effort and co-operation.

Organising committee

The workshop organising committee under Robert Johnson's lead was as follows:

Name	Country	Organisation
Robert Johnson	United States	NRC
Consuelo Alejano	Spain	CSN
Gregory Chapman	United States	NRC
José M. Conde	Spain	ENUSA
Stéphane Evo	France	IRSN
Clive Ingram	United Kingdom	ONR
Eric Létang	France	IRSN
Véronique Lhomme	France	IRSN
Tetsuo Nakata	Japan	S/NRA
Kotaro Tonoike	Japan	JAEA
Tom Hiltz	United States	DOE
Eckhard Westermeier	Germany	BfS
Hatsumi Yoshida	Japan	S/NRA
Olli Nevander	International organisation	NEA

EXECUTIVE SUMMARY

During its 2013 annual meeting, the Working Group on Fuel Cycle Safety (WGFCS) identified the need to organise information exchange on regulatory and operational criticality safety practices used in OECD countries. This resulted in the organisation, by the Committee on the Safety of Nuclear Installations (CSNI), of a WGFCS workshop on operational and regulatory aspects of criticality safety (ORACS). The workshop was held on 19-21 May 2015 in Albuquerque, New Mexico in the United States.

It is important for all nuclear programmes involving significant amounts of fissile nuclear material (e.g. a large enough quantity of material such that a criticality accident could credibly occur) to maintain a complete and thorough nuclear criticality safety (NCS) programme. The goal for this workshop was to organise an information exchange on this topic to include both regulatory and industry perspectives from both OECD and non-OECD countries.

During two and half days, discussions were organised around presentations of 24 papers. On the third day, the workshop attendees toured the Sandia National Laboratories Facility. The workshop was attended by about 55 participants from 8 countries.

The following discussion summarises and documents the proceedings of the ORACS workshop.

Summary and conclusions

During the workshop, presenters and participants discussed various issues related to criticality safety at fuel cycle facilities (FCFs). Topics like national safety requirements and regulatory perspectives, identification of the main operational practices/controls and challenges for preventing inadvertent criticality events, lessons learnt from operating experience and potential regulatory gaps and identification of needs for R&D (in terms of codes, experimental data, uncertainty/bias assessment methods) were discussed during the workshop.

In the workshop, the following topics were identified as themes for future consideration:

- Facilitating the sharing and using of operating experiences and existing criticality event information in national databases at the international level, such as integrating national criticality event information into the international Fuel Incident Notification and Analysis System (FINAS) database operated by International Atomic Energy Agency (IAEA). FINAS is managed by IAEA and NEA according to the wishes of the database Steering Committee.
- Identifying the advantages and disadvantages of deterministic and risk-informed approaches to NCS assessments and determining the appropriate balance between the two approaches as well as examining how the various countries apply defence-in-depth (DiD) and double-contingency principles.
- Addressing technical competency and complacency in the field of nuclear criticality safety at FCFs and developing guidance on the consideration of human and organisational factors in nuclear criticality safety programmes.

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LIST OF ABBREVIATIONS AND ACRONYMS

CEA	Commissariat à l'Énergie Atomique et aux Énergies AlternativesCSNI Committee on the Safety of Nuclear Installations
DCP	Double contingency principle
DiD	Defence in depth
DOE	Department of Energy (United States)
FCF	Fuel cycle facilities
FINAS	IAEA/NEA Fuel incident and notification and analyses system
GRS	Gesellschaft für Anlagen-und Reaktorsicherheit
IAEA	International Atomic Energy Agency
IRSN	Institut de Radioprotection et de Sûreté Nucléaire (France)
ISA	Integrated safety analysis
NCERC	National Criticality Experiments Research Center
NCS	Nuclear criticality safety
NEA	Nuclear Energy Agency
NRC	Nuclear Regulatory Commission (United States)
OECD	Organisation for Economic Co-operation and Development
ORACS	Operational and regulatory aspects of criticality safety
TSO	Technical support organisation
WGFCS	Working Group on Fuel Cycle Safety

1. Introduction

Criticality safety is an integral component of ensuring the continued safe operation of current reactor and fuel cycle facilities, as well as addressing new and interesting challenges that are expected in the design of anticipated future nuclear facilities. Fuel cycle facilities (FCFs) represent a broad range of operations including uranium enrichment, fuel fabrication, radioisotope production, reprocessing, decommissioning, waste management (storage, handling, maintenance, and disposal of fissile materials including spent and damaged fuel) and transportation. All these operations exhibit the need to maintain a complete and thorough nuclear criticality safety (NCS) programme. The workshop discussions are intended to bring significant regulatory and operational aspects of these programmes into focus.

2. Objective

Working under the mandate of the CSNI, the objective of the WGFCS is to advance the understanding for regulators, technical support organisations (TSO) and operators of relevant aspects of nuclear fuel cycle safety in NEA member countries.

The objective of this workshop was to discuss and review current national activities, plans and strategies for maintaining and improving operational and regulatory approaches to criticality safety. Safety authorities and their TSO, fuel cycle facilities (FCF) operating organisations and international organisations were invited to share information on their approaches, practices and current developments.

3. Background

Traditionally, NCS Programmes have focused heavily on the preparation of deterministic criticality safety assessments based on computational modelling of normal and anticipated abnormal conditions to demonstrate sub criticality. These calculations form the basis for safety limits and controls embodying a DiD approach to safety known as the double contingency principle. In this deterministic approach, an operator assumes the failure of each control barrier occurs, without regard for the likelihood of occurrence, and demonstrates the system remains subcritical under the worst-credible condition that results. This is a very conservative approach that has resulted in a long record of safety but with the imposition of very large safety margins.

In some countries, a risk-informed and performance-based approach has been recently introduced in operations and regulations to ensure safety of FCFs. The integration of probabilistic risk analysis methods with the traditional deterministic approaches employed in criticality safety practice has resulted in numerous complexities and challenges, especially in facilities handling and processing fissile materials due to the complexity of the physical and chemical forms and the diversity of their processes and hazards. For the probabilistic approach this added complexity makes it even more important to focus on the regulatory aspects of criticality safety. In addition, the integration of risk assessment requires operational aspects of criticality safety to be considered in addition to the analytical elements.

4. Workshop structure and contents

This workshop presented a unique opportunity to discuss criticality safety from an operational and regulatory perspective, with the objective to review current national activities, plans and strategies for maintaining and improving operational and regulatory approaches to criticality safety. The workshop centred on the following focus areas: national regulatory approaches, operational NCS analysis and operational NCS implementation.

The workshop was divided into five sessions: an opening session; three technical sessions; and a final summary session. Each session resulted in a panel session where information exchange and further discussion were encouraged.

4.1 Session 1: National regulatory approaches

This session was chaired by Stéphane Evo (IRSN, France).

Seven papers were presented during Session 1, on the national regulatory approaches to criticality safety. American, Spanish, French and German approaches were presented, as well as the IAEA Criticality Safety Guide SSG-27. Some application cases were also presented (Spanish approach and its implementation in Spanish facilities, Implementation of geometry control in France…). The key points raised during the session 1 are summarised hereafter:

- Since the criticality risk may lead to high consequences to workers, main efforts focus on prevention of the criticality risk rather than mitigation of the criticality accident. Nevertheless, participants agree there is a need to be prepared for any mitigation action.
- The double contingency principle (DCP) is widely used for criticality safety assessment, but its application or interpretation may differ from one country to another. In particular, some discussions were raised about knowing whether a single parameter with two or more controls may comply with the DCP.
- The general methodology of the defence-in-depth (DiD) principle is well understood and shared by participants. Nevertheless, its application to criticality safety of fuel cycle facilities is somewhat challenging. For instance, some countries underlined some differences in the interpretation of the third level of the DiD principle. Discussions are still ongoing at the IAEA on the adaptation of this principle to nuclear criticality safety.
- All national criticality safety approaches have at least a deterministic component. But, for some countries, this approach is supplemented by a risk-informed or a probabilistic approach.
- There is a generalised concern about maintaining competences of criticality staff. To achieve this objective, the existence of training programmes in criticality safety and of criticality experimental facilities is a key issue and should be maintained over time.
- No criticality accident has been reported since the Tokaimura accident in 1999. Thus, the danger to slip slowly in a form of complacency increases. Some participants have drawn attention on the need for everyone not to fall in a kind of routine and to stay “uneasy” with the criticality risk.
- The different presentations show that there is generally no prescriptive k_{eff} limit (acceptability criterion) in regulations and there can be differences with k_{eff} value in different countries. The k_{eff} limit varies in particular according to the studied situations, the controlled parameters, the safety margins, the sensitivity of the calculated k_{eff} to the different parameters of the study and the validation bias.
- The importance of the interfaces with other technical areas (such as fire hazard, seismic hazard, chemical process engineering…) has been underlined.
- The involvement of criticality specialists during the lifetime of facilities handling fissile material has been emphasised.
- Sharing the feedback of operating experience between licensees, regulators and TSO contributes to maintaining the knowledge and the competencies in criticality safety.
- The criticality safety approach for final disposal has been questioned. Indeed, the safety margins or requirements could be commensurate to the consequences of a critical excursion during the post-closure phase.

4.2 Session 2: Operational nuclear criticality safety (NCS) analysis

The session was chaired by José M. Conde (ENUSA Industrias Avanzadas, Spain). Nine papers were presented during Session 2, which was devoted to operational considerations in criticality safety analysis. A wide range of fuel cycle facilities and issues were covered in the session, including the fuel cycle front-end: operational safety and experience in fuel fabrication facilities; the back-end: spent fuel storage and transportation issues; spent fuel disposal; and criticality control of degraded cores. The key points raised are summarised hereafter:

- The Integrated Safety Analysis (ISA) performed by fuel cycle facilities in some countries has demonstrated itself to be a powerful tool useful to identify the steps in the facility processes where safety enhancements are needed.
- However, there are concerns on the regulatory side because experience shows that the ISA assessment is incomplete sometimes. Examples were presented of event sequences at some facilities that were not identified, and hence not addressed in the ISA process.
- Although the ISA assessment process is currently required in several countries, with similar aims and objectives, some differences in the implementation processes were identified, which may lead to differences in the ISA scope and results.
- A common criticality safety analysis weakness identified in several instances is the lack of participation of criticality specialists in the facilities' design changes evaluation procedure, resulting in deficient criticality safety assessments. The need for embedding the criticality safety experts in the modifications process was stressed.
- The relevance of lessons learnt from the internal and external operating experience was emphasised. The need to share operating experience by means of specific databases (such as FINAS) and expert fora was highlighted.
- In the same line of thought, a discussion was held about the need to better analyse and understand criticality-related events and the adequate ways to extract and learn the applicable lessons. The discussion included considerations on management of data in existing event databases to increase the value added, and on the improvement and promotion of data sharing among countries.
- The oversight practices were also analysed, and the role of inspectors in ensuring criticality safety was discussed. The need for a proper and thorough training of inspectors was enhanced.
- There is a need to improve the coherence between the contents of spent fuel cask storage and transport certificates. This issue was identified already some time ago, and work is ongoing to address it both at the national and international level.
- While the use of a risk-informed safety approach with a probabilistic component has become a common practice in fuel cycle facilities during the last years (at least those regulated under US NRC 10CFR70), a similar evolution has not been implemented for spent fuel storage and transportation so far. Efforts are ongoing in several countries towards this goal. However, a limited advance in this matter has been achieved since the WGFCS reviewed this issue in the Workshop on Safety Assessment of Fuel Cycle Facilities (Toronto, 2011).
- Methodological approaches to perform as-loaded criticality safety analysis of spent fuel casks are being developed, which are able to demonstrate additional reactivity margins existing in casks loaded in the past when compared with the analysis based on the design basis fuel.
- The additional margins may be used to address margin needs for transportation of old casks, as well as to support direct disposal of casks. Credit to the neutron absorption in ambient isotopes may be needed to justify the latter, but a lack of nuclear data in this field is identified.

4.3 Session 3: Operational nuclear criticality safety (NCS) implementation

This session was chaired by Véronique Lhomme (IRSN, France) and co-chaired by Eckhard Westermeier (BfS, Germany). Six papers were presented during this session by a representative of the Nuclear Regulatory Commission (NRC) from the United States, operators from Sweden (Swedish Nuclear Fuel and Waste Company - SKB), United States (Los Alamos National Laboratories) and France (French Alternative Energies and Atomic Energy Commission – CEA) and representatives from the French technical support organisation (IRSN). Key points from these presentations are summarised here after:

- When operating conditions in a nuclear facility change, there is a need for new safety analyses. This was demonstrated in the presentation from Fredrik Johansson which illustrated the changes in the operating conditions during the last 30 years in the Swedish intermediate Storage of Spent Nuclear Fuel (Clab). Particularly when the changes have been introduced gradually, it is important to periodically re-do original analyses completely and make a new overall assessment. Therefore, it is also important to use methodologies and codes which are state of the art. Moreover, gradual changes do not make the management of an organisation aware of safety gaps. We should be aware that a lot of accidents happen when people are working according to old routines and unaware that the conditions have changed.
- Timothy Sippel highlighted that it is important to learn from the past operating and events experience in order to prevent future accidents from occurring. However, it is the experience of the author that events with similar root causes and characteristics continue to occur. His findings and the lessons learnt from the events are that in some cases an operation or credible upset condition was not considered by the criticality safety analyst or credible upsets were considered, but were incorrectly considered incredible. Before a new operation with fissile material or a change of an existing operation, it shall be determined, that the entire process is subcritical under both normal and credible abnormal conditions. To do this it is obviously necessary to understand what the "credible abnormal conditions" are. Only then can the analyst determine what controls are needed to ensure that the entire process is subcritical. Another key cause of accidents and events is the failure to maintain existing safety controls and enforce procedures. The best analysis possible and best possible controls and procedures will not prevent accidents if these controls are not maintained and procedures are not followed.
- These lessons learnt from past events were completed with those presented by Jean-Paul Daubard resulting from the global analysis, conducted by the IRSN, of the 135 significant events related to criticality occurred in the French civil FCFs and reported to the nuclear safety authority from 2005 to 2014. One major issue which was underlined was the importance of human and organisational failures (inadequate or insufficient operating documents, non-compliance with procedures or instructions…) as the root cause of the majority of the events. Actually, a significant proportion of these failures are due to an underestimation or to a lack of knowledge regarding the difficulties for the operators in performing their tasks. This can result in an inappropriate technical and organisational criticality risk management which does not allow dealing with situations other than those concerning normal plant's operation.
- Besides technical matters, taking care of human and operational factors in the framework of nuclear criticality safety often requires a lot of efforts from operators to put in place, operate and, maintain a strong and robust criticality safety organisation in their facilities.
- A good example of what can be implemented by operators in terms of criticality safety management was provided by Georges Kyriazidis (CEA, France) who described the organisation in place regarding the ten French research centres operated by the CEA. This organisation is based on a "three levels" operational line (CEA general directorate level, Research Centres directorate level and facilities level) working in parallel with a control line, totally independent from the operational line, established at each Center directorate level. In this framework, senior

criticality experts and qualified engineers in terms of criticality are responsible of criticality calculations and criticality safety management. In addition, the necessary documentary organisations as well as the training practices, including criticality exercises, were detailed.

- Another important point is about maintaining skills and proficiency of workers during all the operation time of a facility, whatever the changes or transitions that occur. The case presented by Steven D. Clement (Los Alamos National Laboratories), about the operating experience of the relocation and reconstitution of the Los Alamos Critical Experiment Facility (TA-18), illustrated that challenge. The approach was in particular to utilise, during two years, other similar operational nuclear facilities (e.g. the French CEA Research Center of Valduc) to train the U.S. scientists pending the construction of the new facility (the National Criticality Experiments Research Center – NCERC). Besides, the criticality improvement measures (seismic protection, pre-action fire suppression, vault design…) associated to the TA-18 mission relocation and the construction of the NCERC was presented.

- Finally, it also appears crucial that operational criticality safety implementation should take into account the diversity of working situations and complexity of their organisational interfaces. This was demonstrated with the presentation of Carine Hebraud and Lise Menuet (IRSN). This presentation established that only a socio-technical approach can allow defining a relevant set of criticality safety rules favouring efficient and safe criticality activities. This approach should be based on a comprehensive risk analysis, combining technical aspects as well as in-depth knowledge of staff operating practices according to the specific context in which operations are performed. According to this, based on the operating experience feedback of an event occurred, in 2012, in a French FCF producing fuel for pressurised water reactors, the speakers emphasised how important it is to have criticality experts and local management “in the field”, to firstly get a better understanding of the complexity of work situations that may be encountered and also to improve understanding of the criticality risk to workers and make them willing to comply with the provisions of prevention.

5. Conclusions and recommendations

5.1 Session 1: National regulatory approaches

Regarding the discussions and the papers presented during the session 1, it could be recommended to develop further actions, such as working groups, on the following topics.

1. To define a “benchmark criticality safety case” and to compare national approaches on this benchmark.
2. To report on the different applications/interpretations of the DCP.
3. To deal with the application of the DiD in criticality safety.
4. To exchange on risk-informed or probabilistic approaches.
5. To define a minimum training programme to maintain the vigilance and the competences of people involved in criticality safety.

5.2 Session 2: Operational nuclear criticality safety (NCS) analysis

Based on the discussions and the papers presented during session 2, the following recommendations can be forwarded:

1. Although ISA has consolidated as a powerful tool for oversight of the facility operation, and to identify facility weaknesses, attention must be paid to its completeness regarding the specific event sequences assessed.

2. The thorough and systematic review of internal and external operational experience is a key factor for criticality safety improvement.
3. Regarding this criticality-related operational experience, NEA and IAEA might want to analyse if the VIBS database (Germany) can improve the usefulness of the FINAS database.
4. Criticality experts are instrumental in the fuel cycle facilities safety management; they should be consistently involved in the facility modifications processes.
5. The importance of the work towards implementation of risk-informed safety approaches for the back-end facilities should be emphasised.
6. While not new, the need to harmonise the requirements of the different back-end facilities and conditions is emphasised again, especially regarding storage and transportation.

5.3 Session 3: Operational nuclear criticality safety (NCS) implementation

The conclusions and outputs from Session 3 can be summarised as follows.

- When having to re-do criticality analyses, it is not enough to put additional analyses on top of the existing ones. It is always necessary to conduct a thorough revision of the NCS analysis, meeting the latest international standards and guidelines and taking into account state of the art of science and technology.
- It is much more likely that a criticality incident will occur due to some unforeseen event or lack/violation of safety procedures than to errors in the criticality calculations. Actually, human and organisational failures (inadequate or insufficient operating documents, non-compliance with procedures or instructions, lake of communication…) are the most important root causes of the majority of the criticality events occurred in FCFs.
- Maintaining skills and proficiency of workers during all the operation time of a facility, e.g. by the way of training, good communication and sharing of operating experience, is crucial to fight complacency as well as lack of awareness when changes happen gradually in processes and operations.
- Nothing can replace implementing in-field competences in criticality safety management. A good practice could be to put in place both an operational management line and NCS control management line independent of one another.
- An efficient criticality risk analysis should combine technical, organisational and human aspects (socio-technical approach), in order to define appropriate measures for controlling the risks encountered. This analysis should be based on an in-depth knowledge of staff operating practices required for operation and for criticality risk control, as well as a detailed knowledge of the specific context in which operations are performed.
- Learning from past experience is essential in order to improve operational nuclear criticality safety implementation and prevent future criticality accidents from occurring. This emphasises the importance of developing databases (e.g. the FINAS database) reporting on criticality events and of sharing lessons learnt from operating experience based on a relevant analysis of these events.

Appendix 1.
Programme of the conference

Tuesday 19 May 2015

Opening of sessions
(R. Johnson)

Keynote address
(M. Moury)

Goals and targets of OECD/NEA workshop
(O. Nevander)

Session 1: National regulatory approaches
Session Chair: Stéphane Evo, Institut de Radioprotection et de Sûreté Nucléaire (IRSN)

1) Christopher Tripp, US NRC
A Comparison of Criticality Safety Regulatory Practices in the United States and IAEA Safety Series Guide 27

2) Consuelo Alejano, Consejo de Seguridad Nuclear (CSN)
Regulatory Aspects of Criticality Safety in Fuel Cycle Facilities in Spain

3) Stéphane Evo, IRSN
Status of French Regulation concerning Nuclear Criticality Safety

4) Robert Kilger, Gesellschaft für Anlagen-und Reaktorsicherheit (GRS)
Criticality Safety Related German Industrial Standards from the DIN Series

5) Aurélie Bardelay, IRSN
Issues about Implementation of Geometry Control to Maintain Nuclear Criticality Safety

6) Fitz Trumble, US DOE
The DOE Criticality Safety Support

7) Ramon Gater, IAEA
IAEA Safety Standards and Criticality Control in Nuclear Fuel Cycle Facilities

Session 2: Operational Nuclear Criticality Safety Analysis
Session Chair: Jose M. Conde, Enusa Industrias Avanzadas (ENUSA)

1) Julio Lopez-Marquez, ENUSA
Nuclear Safety Management at the Juzbado Fuel Fabrication Facility

2) Florian Rowold, GRS
Statistical Evaluation of Criticality-Related Events in the Fuel Cycle Facilities Included in the GRS and BfS VIBS Database

3) Julio Lopez-Marquez, ENUSA
Fractionated Dosage of Hydrogenated additives in the ceramic Pellets Fabrication Process

Wednesday 20 May 2015

Session 2: Operational Nuclear Criticality Safety Analysis
Session Chair: Jose M. Conde, Enusa Industrias Avanzadas (ENUSA) (Cont'd)

Keynote address
by M. Bailey

4) Julio Lopez-Marquez, ENUSA
Use of MAVRIC-SCALE Sequence to compute Transmission Factors for Criticality Alarm System Applications

5) Kaushik Banerjee, ORNL
Criticality Safety Assessment for As-loaded Spent Fuel Storage and Transportation Casks

6) John M. Scaglione, ORNL
A Potential New Approach to Demonstrating Criticality Safety of Spent Fuel Storage and Transportation Casks

7) Kotaro Tonoike, Japan Atomic Energy Agency
Criticality Control of Fuel Debris – TMI-2 Review and Fukushima

8) Vladimir Sobes, ORNL
Validation Study for Crediting Chlorine in Criticality Analyses for US Spent Nuclear Fuel Disposition

9) Kaushik Banerjee, ORNL
Subcriticality Demonstration Options for Direct Disposal of Dual-purpose Canisters

Session 3: Operational Nuclear Criticality Safety Implementation
Session Chairs: Véronique Lhomme, IRSN and Co-Chair Eckhard Westermeier, BfS

1) Fredrik Johansson, Sweden K B
From Geometrical Safe Configuration to the Burn-up Credit

2) Timothy Sippel, US NRC
A Comparison of Recent Events in the USA with Historical Criticality Accidents

3) Jean Paul Daubard, IRSN
Major Lessons Learnt by the Global Analysis of Significant Events Related to Criticality Declared in France between 2005-2014

Thursday 21 May 2015

Session 3: Operational Nuclear Criticality Safety Implementation
Session Chairs: Véronique Lhomme, IRSN and Co-Chair Eckhard Westermeier, BfS (Cont’d)

Keynote address
by S. Pickering

4) Georges Kyriazidis, CEA
Criticality Safety and Organisational Principles at the CEA

5) Steven D. Clement, Los Alamos National Laboratories
Lessons Learnt and Management Perspective for the United States’ Only General-Purpose Critical Experiments Facility

6) Lise Menuet, IRSN and Carine Hébraud, IRSN
Criticality Risk Management: Why Analysis of Operating Practices Maters?

Closing remarks and workshop summary
Robert Johnson, US NRC and Olli Nevander, NEA

Appendix 2.
List of registered participants

France

Aurélie	BARDELAY
Jean-Paul	DAUBARD
Stéphane	EVO
Emmanuel	GAGNIER
Carine	HEBRAUD
Georgios	KYRIAZIDIS
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Véronique	LHOMME
Lise	MENUET
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Marissa	BAILEY
Lawrence	BERG
Alan	BLAMEY
Cornelia	BLANTON
Gregory	CHAPMAN
A. Eagon	CHOLAKIAN
Glenn	CHRISTENBURY
Steven	CLEMENT
Marilyn	DIAZ
William	DOANE
Stanley	FITCH
Dale	GOVAN
Margie	KOTZALAS
Kevin	HAHN
Tom	HILTZ
Robert	JOHNSON
Mark	LEE
Matthew	MOURY
Jeremy	MUNSON
Nicholas	PETERKA
Steven	PETRAS
Kenneth	REIL
JM	SCAGLIONE
Tim	SIPPEL
Garrett	SMITH
Vladimir	SOBES
Chris	TRIPP
Fitz	TRUMBLE
John	WAGNER
Kent	WOOD

Jordan

Mohammad Ara’of Aghani	ALRWASHDEH	

Nigeria

Frederick Efe	OMOMA	
Chizoba	ORJI	
Olabode	RASHEED	

International organisations

Ramon	GATER	IAEA
Olli	NEVANDER	NEA

Appendix 3

Conference papers

IAEA Safety Standards and Criticality Control in Nuclear Fuel Cycle Facilities

Presented at ORACS 2015 organised by OECD-NEA and hosted by US-NRC

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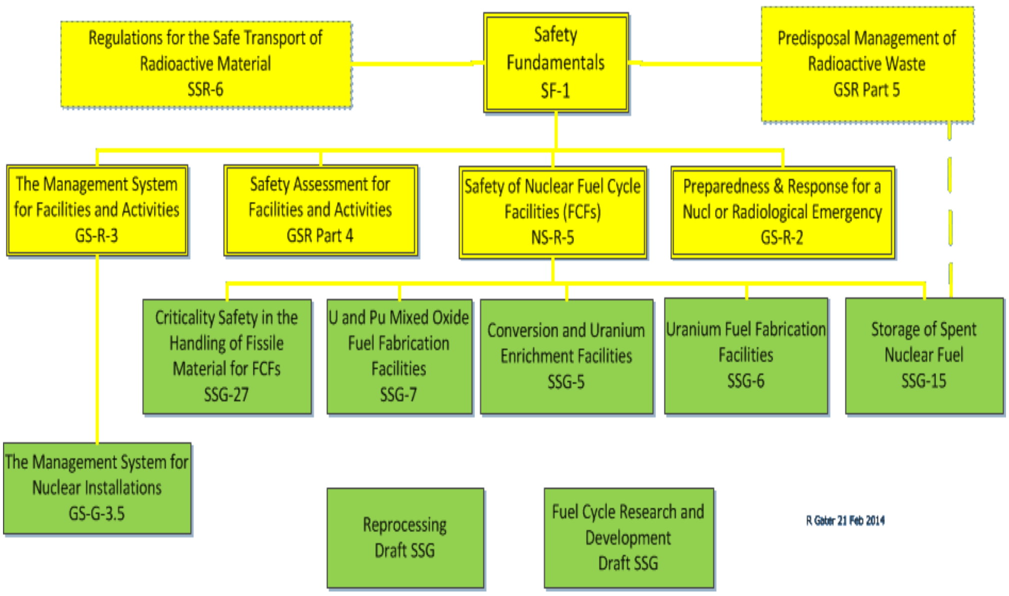
Until 2005, the IAEA published little specific guidance on maintaining criticality safety in Nuclear Fuel Cycle Facilities (NFCFs), apart from a brief mention of the double contingency principle in safety reports. Recognising developments in safety analysis and demands for consistency with reactors, the IAEA started to promote the application of Design Basis Analysis to NFCFs alongside the double contingency principle, in 2005. The IAEA published its first safety requirements document covering NFCFs (known as NS-R-5) in 2008. Such was the importance of filling the gap in IAEA standards that new documents, containing some criticality safety guidance, have been published every two years since then. With more safety guides and a revised safety requirements document in preparation, this paper summarises the direction being taken by the IAEA safety standards regarding the maintenance of criticality safety in NFCFs, to review what has been achieved and permit discussion on the areas which need improvement and the areas (hopefully) to leave alone.

The increase in the number of safety standards covering NFCFs has happened at a time when the IAEA has been restructuring most of its other safety standards, so it is opportune to begin by outlining the new framework in which the new NFCF standards reside. Figure 1 shows the structure of the standards currently issued or planned, with the safety requirements documents and the safety guides shown in different colours.

Figure 1 – The Structure of IAEA Safety Standards Covering NFCFs



IAEA Fundamentals

The top level document is SF-1, which states the fundamental safety objective and establishes ten fundamental safety principles that govern the requirements developed in the IAEA standards and guides. The publication of SF-1 was endorsed by a large number of international bodies, including the World and Pan-American Health Organisations (WHO and PAHO), EURATOM and the Food and Agriculture Organisation (FAO) and thus reflects a broad international consensus on nuclear safety principles.

> **Fundamental Safety Objective**
>
> To protect people and the environment from harmful effects of ionizing radiation.
>
> **Ten Fundamental Safety Principles in SF-1**
>
> 1. Responsibility for safety
> 2. Role of government
> 3. Leadership and management for safety
> 4. Justification of facilities and activities
> 5. Optimization of protection
> 6. Limitation of risks to individuals
> 7. Protection of present and future generations
> 8. Prevention of accidents
> 9. Emergency preparedness and response
>
> Protective actions to reduce existing or unregulated radiation risks

Below SF-1 sits an integrated and consistent set of safety requirements documents, which establish the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements are governed by the objective and principles of the Safety Fundamentals. If the requirements are not met, measures must be taken to reach or restore the required level of safety. The format and style of the requirements facilitate their use for the establishment, in a harmonized manner, of a national regulatory framework.

Most of the safety requirements documents have a number of safety guides associated with them and the Specific Safety Guides (SSGs) associated with NS-R-5 are described later in this paper.

IAEA Safety Requirements

There are two types of IAEA requirements document; Generic Safety Requirements (GSRs) that apply to all types of facility handling atomic radiation and Specific Safety Requirements (SSRs) that used to be called Nuclear Safety Requirements. The safety requirements for NFCFs are contained in NS-R-5. These are expressed as ‘shall’ statements and the main body of NS-R-5 contains nearly twenty “shall” statements relating directly to the control of criticality. Appendices in NS-R-5 cover various specific facility types. Appendix I concerns Uranium Fuel Fabrication and establishes ten additional requirements relating directly to criticality safety. The other appendices in NS-R-5 cover Mixed-Oxide Fuel Fabrication, Conversion and Enrichment and in 2014 appendices covering Reprocessing and R&D Facilities were added.

One example of the criticality requirements is; *for the prevention of criticality by means of design, the double contingency principle shall be the preferred approach.* This means that new designs need to include sufficient safety factors to require at least two unlikely, independent & concurrent changes in process conditions before a criticality accident is possible. NS-R-5 treats the double-contingency and design basis approaches as equivalent, stating that; *Design criteria for all relevant parameters shall be specified for each operational state of the facility and for each <u>design basis accident or equivalent</u>.*

It is important to note that the principle of defence in depth is also a requirement that applies to NFCFs; *the concept of defence in depth shall be applied at the facility for the prevention and mitigation of accidents.* For instance, the first level of defence in depth requires *prevention of abnormal operation and failures* and the fourth level includes *prevention of accident progression*. Defence in depth must be addressed in a manner that is proportionate to the hazard presented by a particular NFCF.

Another example, where the double-contingency principle is quoted in IAEA requirements, concerns the bounding fissile composition to be used when assessing the safety of an enrichment facility. Normally it is the maximum uranium enrichment[1] authorized in any part of the facility which shall be used *unless the impossibility of reaching these compositions is demonstrated in accordance with the double contingency principle*.

One feature of the "shall" statements is that they must be assessed and the assessment documented in the safety case.

IAEA Safety Guides

There are a number of specific safety guides under NS-R-5, all written to help operators, regulators and support organisations to apply the requirements in specific situations. All of the specific safety guides provide guidance on the control of criticality in NFCFs. Two new guides, covering reprocessing and R&D facilities, have been drafted for publication in several months' time.

The safety document that covers criticality as its main topic is SSG-27, published in 2014. It applies to all facilities and activities except reactors and critical assemblies. It summarises the general principles of criticality safety and draws attention to many anomalies that have been recorded in criticality phenomena. It states that a criticality safety assessment is needed and recommends that safety limits should be derived either;

– From value of K_{eff} in which case uncertainties, errors and potentially significant non-linearity in variation of K_{eff} with other parameters should be considered.

– From a critical value[2] of one or more control parameters, applying safety margins to derive the safety limits.

SSG-27 does not recommend a specific figure for the safety margin, simply stating $K_{eff} < 1$, but consideration should be given to uncertainties and significant non-linearity[3]. After describing these concepts, later chapters in SSG-27 provide guidance on measures for ensuring criticality safety (such as the setting of safety limits) and guidance on the process of criticality safety assessment.

The IAEA recommends a hierarchy of safety measures to ensure criticality safety, with preference given to safety measures closer to the top the following list;

(1) Passive engineered safety measures using passive components to ensure sub-criticality. Such measures are highly preferred because they provide high reliability, cover a broad range of criticality accident scenarios, and require little operational support to maintain their effectiveness as long as ageing aspects are adequately managed. Human intervention is not necessary. Advantage may be taken of natural forces, such as gravity, rather than relying on electrical, mechanical or hydraulic action. Like active components, passive components are subject to degradation[4] and to human error during installation and maintenance activities. They require surveillance and, as necessary, maintenance.

(2) Active engineered safety measures using active components such as electrical, mechanical or hydraulic hardware to ensure sub-criticality. Active components act by "sensing" a process variable

1. For MOx facilities it is the maximum plutonium isotopic composition, plutonium content and uranium enrichment (if 235U > 1%).

2. The critical value is the value of a control parameter that would result in the system no longer being reliably known to be subcritical.

3. Some believe that the IAEA should specify a limiting value of Keff in standards for NFCFs (for example, Keff << 0.98), although this might tempt some designers to approach the limiting value in new facilities.

4. Such degradation may be due to seemingly random effects such as chemical phenomena.

important to criticality safety (or by being actuated through a control system) and providing automatic action to place the system in a safe condition, without the need for human intervention. Active engineered safety measures should be used when passive engineered safety measures are not feasible. However, active components are subject to random failure and degradation and to human error during operation and maintenance activities. Therefore, components of high quality and with low failure rates should be selected in all cases. Fail safe designs should be employed, if possible, and failures should be easily and quickly detectable. The use of redundant systems and components should be considered, although these do not prevent common cause failure. Active engineered components require surveillance, periodic testing for functionality and preventive and corrective maintenance to maintain their effectiveness.

(3) Administrative safety measures, SSG-27 lists several types of administrative controls that could be used if passive and active means are not feasible, many of which are procedurally-based. The criticality assessment should provide a demonstration that credible deviations have been exhaustively studied and combinations understood. Specialists in human performance and human factors should be consulted and the derived controls documented and independently reviewed by a knowledgeable person.

Use of this list should not preclude use of manual interventions for defence-in-depth, as additional measures may be provided for mitigation.

Reassuringly, the safety guide states that personnel knowledgeable in criticality safety should perform the safety analyses and conduct independent reviews.

Future Developments

The IAEA is developing a number of generic services and workshops to support NFCF safety activities, and a review of the existing safety requirements in NS-R-5 is underway.

(1) For reactors, the IAEA supports a consistent approach to safety through an active programme of peer review missions. In these missions[5], the IAEA and international experts conduct thorough reviews of the implementation of safety standards. The equivalent service for NFCFs is called Safety Evaluation During Operation (SEDO). To date, there have been only a small number of SEDO missions but it is expected that the number of such missions will increase in future.

(2) The IAEA will hold a Technical Meeting on Safety Analysis and Safety Documents for Fuel Cycle Facilities in Vienna, from 4 to 8 May 2015 at which topics, such as defence in depth and criticality accident detection can be discussed. The IAEA also intends to develop materials that could be used to support training in criticality safety for NFCFs.

Example of Quantified Grading

An exempted quantity of fissionable materials in the licensed site is defined as an inventory of fissionable materials, as follows:

less than 100 g of ^{233}U, or ^{235}U, or ^{239}Pu, or of any combination of these three isotopes in fissionable material combined in any proportion, OR

an unlimited quantity of natural or depleted uranium or natural thorium, if no other fissionable materials nor significant quantities of graphite, heavy water, beryllium, or other moderators more effective than light water are allowed in the facility, OR

less than 200 kg in total of natural or depleted uranium or natural thorium if some other fissionable materials are present in the facility, but the total amount of fissile nuclides in those fissionable materials is less than 100 g.

Facilities operating with exempted quantities of fissionable materials are exempt from the requirements of this document.

5. IAEA peer reviews are called “Operational Safety Review Teams (OSART)” for power reactors and “Integrated Safety Assessment for Research Reactors (INSARR)”.

(3) In terms of new safety documents for NFCFs, work has started on a complete review of NS-R-5. Some rationalisation of requirements relating to criticality, currently replicated in different appendices for specific facility types, has already been achieved. Areas in which more work is needed include;

(4) Definition of Design Extension Conditions (DEC) for NFCFs; following the accident at Fukushima Dai-ichi, IAEA Member States have agreed requirements for nuclear facility designs to maintain their safety beyond the traditional design basis. Reactor conditions involving significant core damage, station-black-out, and anticipated transients without SCRAM are now included in the list of DECs which new power reactors will need designing against, but there is no such equivalent list for NFCFs. Three alternative possible definitions of a DEC for NFCFs follow, all describing events of low-frequency having high-consequences;

- Definition 1, in terms of an internal system failure: At least two failures are required to reach DEC. In principle, an accidental criticality could be defined as a DEC for NFCF.
- Definition 2, hybrid of system failure and consequence: Conditions leading to consequences exceeding either the highest off-site reference levels for protection of the public in emergency exposure situations (i.e. 100 mSv integrated over 7 days or in a year, conservatively assessed) or to an unintended relocation of a substantial quantity of radioactive material within the facility which places a demand on the integrity of the remaining physical barriers.
- Definition 3, purely in terms of consequence: Conditions leading to a requirement for long-term and large-scale public protection actions, like evacuation of large populations or large areas needing food production and consumption restrictions.

These are areas where IAEA standards previously left more discretion in the implementation of defence in depth. Would any of these definitions lead to requirements for heavy shielding of facilities that previously were unshielded, to protect the public?

(5) The attention given to safety assessments and resources provided for designs and modifications should always be proportionate to the hazards. The application of the graded approach is especially important for NFCFs which can have hazards that vary in magnitude by several orders of magnitude. However, the concept of grading is very difficult to describe using requirements in the safety standards. Instead, examples and quantities are needed to explain the graded approach. If this cannot be done within NS-R-5, a separate safety report on grading may be needed. An example of a quantitative approach to grading is given in the box.

(6) Most IAEA documents covering NFCFs provide large numbers of requirements and much advice on management, which focus on control by procedures. However, supervision has been a contributory factor in many accidents and supervision relates more to behaviour than to procedures. Supervision does not receive the same attention as management in the IAEA safety documents covering NFCFs. Management is frequently inspected and reviewed, whereas supervision is not. Supervision is important to the implementation of many administrative control procedures and it would be useful to identify examples where supervision has a role in criticality safety.

(7) Requirements concerning the production of safety analysis reports and performing safety reviews for existing NFCFs could be stronger. It is clear that a safety analysis report should exist for all facilities where there is a risk of inadvertent criticality. Analyses for several facilities may be grouped together, but the safety analyses should identify all the systems important to safety and clearly state any required operating limits or conditions. The role of acceptance criteria in relation to criticality safety analyses may need more definition. One expected output from the IAEA meeting in May 2015 will be improved guidance on the contents and format safety analyses, to include criticality safety analysis. The IAEA may arrange a consultancy meeting later in 2015 or in 2016 to draft a report following this meeting.

(8) There may be a need for more guidance in relation to criticality accident management, detection, monitoring systems and post-accident response.

Opinions expressed in this paper are those of the author. These aspects of criticality safety and assessment should be discussed and feedback from expert groups such as ORACS is welcomed.

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A COMPARISON OF CRITICALITY SAFETY REGULATORY PRACTICES IN THE UNITED STATES AND IAEA SAFETY SERIES GUIDE 27

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ABSTRACT

Criticality safety regulatory requirements and practices vary somewhat between countries because of their specific histories and regulatory philosophies. This paper presents the regulatory requirements and practices for commercial fuel facilities in the United States (US) and compares these to the International Atomic Energy Agency (IAEA) Safety Series Guide that relates to criticality safety (SSG-27, "Criticality Safety in the Handling of Fissile Material"). Regulations regarding criticality safety for commercial US fuel cycle facilities are found in Title 10 of the Code of Federal Regulations (CFR) Part 70. The few criticality safety requirements in Part 70 are high-level, and therefore guidance has been established to provide acceptable methods of demonstrating compliance with those requirements. This guidance incorporates national standards to the extent practical. This paper compares regulatory requirements and guidance for criticality safety for US fuel facilities to those in SSG-27. Specifically, this paper will address deterministic and probabilistic approaches to criticality safety, exemptions from criticality analysis and monitoring, subcritical margin and validation, mitigation of consequences, configuration management and change control, and the level of prescriptiveness. In addition, the role of operating experience in regulatory practices and exemptions from regulatory requirements will be discussed. This will demonstrate how the overall US regulatory practices compare to international guidance and how the US allows exemptions when justified from selected regulations.

I. OVERVIEW OF NRC REGULATIONS AND REGULATORY GUIDANCE

Part 70 of Title 10 of the Code of Federal Regulations (denoted 10 CFR Part 70) deals with the licensing of special nuclear material (SNM). SNM as defined in NRC regulations includes plutonium, ^{233}U, and uranium enriched in ^{233}U or ^{235}U. This paper will be confined to those facilities possessing greater than a critical mass of enriched uranium or plutonium—defined under Part 70 as a mass of 700 grams ^{235}U at any enrichment or 1500 grams if enriched to no more than 4wt% ^{235}U, 450 grams of plutonium or any combination of the above, or half such quantities if moderators or reflectors composed of graphite, heavy water, or beryllium are present—and engaged in uranium enrichment or the processing or fabrication of enriched uranium, prior to such material being irradiated in a reactor.

Such facilities are required to meet "Subpart H" of Part 70, which requires, among other things, that they perform an integrated safety analysis (ISA) and demonstrate compliance with certain "performance requirements" specified in 10 CFR 70.61. These requirements establish certain levels of acceptable risk for radiological and chemical exposure events. In this framework, risk is defined as likelihood times consequence, so that "high-consequence" events must be rendered "highly unlikely" and "intermediate-consequence" events "unlikely" consistent with the "risk matrix" reproduced below. Engineered and administrative controls (known as *items relied on for safety*, or IROFS) must be designated to the extent necessary to achieve an appropriate combination of likelihood and consequence, either by limiting likelihood (prevention) or limiting consequence (mitigation) or by some combination thereof. Management measures must then be established to ensure that IROFS will be available and reliable to perform their intended safety functions.

Consequence	**Likelihood**		
	Highly Unlikely (1)	**Unlikely (2)**	**Not Unlikely (3)**
High (3) TEDE ≥ 1 Sv worker TEDE ≥ 0.25 Sv public	Acceptable 3	Unacceptable 6	Unacceptable 9
Intermediate (2) TEDE ≥ 0.25 Sv worker TEDE ≥ 0.05 Sv public	Acceptable 2	Acceptable 4	Unacceptable 6
Low (1) TEDE < 0.25 Sv worker TEDE < 0.05 Sv public	Acceptable 1	Acceptable 2	Acceptable 3

Figure 1

Risk matrix from NUREG-1520 (based on 10 CFR Part 70); numbers in parentheses refer to likelihood and consequence scores. Their product is the risk score, indicated in the nine central boxes; a risk score ≤ 4 is needed to meet the performance requirements of Part 70.

This requirement to demonstrate acceptable risk defined by the performance requirements constitutes what the NRC means by "risk-informed" regulation. Because licensees have the ability to choose whatever controls they deem advisable as long as they achieve this goal, this also embodies "performance-based" regulation. Criticality has the potential to exceed the 1 Sv (100 rem) threshold for being a "high-consequence" event and therefore must be rendered "highly unlikely." The likelihood terms of unlikely, highly unlikely, and credible, are not defined in the regulation, but each licensee must propose an ISA method—subject to regulatory approval—that includes definitions of these terms and means of demonstrating that it meets them. The ISA may be based on quantitative risk assessment methods, but it may also evaluate risk qualitatively.

For criticality, there is the additional performance requirement that it must demonstrate subcriticality under normal and credible abnormal conditions. This must be met independent from the requirement to demonstrate acceptable risk because, as stated in the Statements of Consideration accompanying issuance of the rule [1], criticality is deemed unacceptable even if the consequences to workers can be shown to be negligible (as when substantial shielding may be present). New facilities and new processes at existing facilities (relative to the date of the revised rule) must also meet the double contingency principle (DCP). A last key provision dealing with criticality safety requires the licensee to establish and maintain a criticality monitoring and alarm system and associated emergency response activities whenever a critical mass of SNM as defined above is present. These requirements to demonstrate subcriticality, meet double contingency, and provide for a criticality alarm system, are deterministic, in that they are imposed regardless of the risk as determined as part of the ISA.

Because of the relatively non-prescriptive nature of such risk-informed and performance-based regulation, additional guidance in the form of NUREG-1520, "Standard Review Plan for License Applications for Fuel Cycle Facilities" [2] has been developed. The Standard Review Plan is guidance on one acceptable way that licensees may meet the regulations, but is not mandatory. Topics include performance and documentation of an ISA, implementation of an NCS Program, technical practices for criticality analyses and calculations, expectations for control of various parameters, and responding to emergencies. This was first issued in 2002 to accompany the new regulation, but as experience was gained, several additional topics needing clarification were identified, resulting in the issuance of several Interim Staff Guidance (ISG) documents. The ISGs of greatest interest to NCS are listed as references, and consist of ISG-01 on qualitative determination of likelihood (in the ISA), ISG-03 on the relationship of the risk-informed ISA requirements to the deterministic NCS requirements, and ISG-10 on justifying the minimum margin of subcriticality. ISG-01 and -03 were incorporated into a 2010 revision of NUREG-1520, and ISG-10 is being incorporated into the current draft that will be Rev. 2.

II. COMPARISON BETWEEN NRC REGULATORY POSITIONS AND IAEA SSG-27

Regulation is a progressive endeavor, and hence regulatory positions will be refined over time. The comparison that follows is based on Chapter 5 of NUREG-1520, Rev. 1 (published in May 2010) and the ISGs mentioned above. While there are some incremental changes in Rev. 2, which was published in draft form in May 2014, the discussion is mainly based on the current official version, Rev. 1. Similarly, SSG-27 was published in final form in May 2014, and the discussion that follows is based on that version, though many of the current observations were first made during an IAEA workshop in February 2014, "Workshop on Criticality Safety in the Handling of Fissile Material for Fuel Cycle Facilities," which involved an earlier draft of SSG-27.

As NUREG-1520 implements 10 CFR Part 70, SSG-27 implements NS-R-5, "Safety of Nuclear Fuel Cycle Facilities," Rev. 1 (published May 2014). The current paper is not concerned with NS-R-5.

II.A Deterministic and Probabilistic Approaches to Criticality Safety

The traditional methods of ensuring criticality safety in fuel facility operations have by any objective standard been highly successful. There have been 22 known process criticality accidents [3] between 1953 and 1999. Most of the 22 accidents worldwide have involved solutions of plutonium or high-enriched uranium, whereas only two of them involved low-enriched uranium (defined under NRC regulation as < 20 wt% ^{235}U). Only 7 of these 22 accidents occurred in the US, and only one of those occurred in a commercial fuel facility, at Wood River Junction in 1964. The most recent accident in the US was at the Idaho Chemical Processing Plant in 1978. Thus, the most recent accident in the US occurred more than 35 years ago, and the most recent accident at a commercial US plant 50 years ago. None of the US accidents involved low-enriched uranium, and the enrichment of the two that occurred elsewhere exceeds that used in the current US commercial nuclear industry.

The reasons for this track record are varied, but the history of criticality safety and reforms put in place following the accidents of the first few years suggest that the defense-in-depth provided by adherence to the double contingency principle, and the large safety margins resulting from the conservative nature of criticality analysis methods, deserve much of the credit. (While factors such as the increased reliance on passive geometry are often credited, we note that many processes do not rely on geometry control and many still rely extensively on administrative controls.) These large safety margins are ensured by an analysis methodology that can be characterized as *deterministic*—by which we mean that all parameters that are not specifically controlled, and all parameters upon the loss of their controls, are assumed to be at their most reactive credible conditions, regardless of the likelihood of actually attaining such conditions. This can lead to very unrealistic modeling assumptions, such as assuming full flooding upon the loss of moderation control, or filling a storage array with the maximum allowable quantity of material at the maximum enrichment, or assuming that a spill takes on spherical geometry, optimum moderation, and full water reflection. These very conservative assumptions can be extremely limiting on plant operations, so in recent years more effort has been expended to refine what constitutes the "worst credible" condition. Examples have been efforts to characterize the maximum moisture content of air following fire sprinkler activation, or the maximum density of oxide powder.

Under 10 CFR Part 70, and consistent with the American national standard ANSI/ANS-8.1, risk must be limited "*by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical*". This fundamental requirement is normally implemented by application of the double contingency principle, which states that "*process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible*." As stated in NRC guidance [4], the spectrum of "credible abnormal conditions" that must be shown to be subcritical is interpreted in the context of double contingency, and so consists of all credible single events or chains of related events. While the consideration of upsets may therefore be truncated once one has shown compliance with double contingency, it is also seen from the two requirements that a nuclear system must be shown to be subcritical upon occurrence of any single credible event, regardless of its likelihood. The only way it can be dismissed from being shown to be subcritical is to show that it is not credible.

A *probabilistic* methodology, conversely, is one in which sequences of events, or combinations of conditions, are evaluated if they exceed a certain minimum likelihood. Conditions may also be evaluated conservatively, but generally not as conservatively as in a fully deterministic approach. There will be some consideration given to the likelihood of attaining a given condition and it will be more likely to model realistic, rather than worst credible, conditions. The focus of a probabilistic analysis is generally to show that the likelihood of criticality is acceptable, so it will seek to determine that sequences of events (terms *accident sequences*) are highly unlikely, rather than show that combinations of conditions are subcritical.

The attempt to satisfy both risk-informed and deterministic requirements is discussed in NRC guidance document ISG-03. Some of the associated challenges were discussed in [5] and [6]. Reference [5] in particular identified several difficulties experienced in trying to apply probabilistic methods to criticality hazards. Those difficulties included verifying completeness of hazard and IROFS identification, determining likelihood qualitatively, demonstrating independence, accounting for initiating and enabling events, and factoring in safety margin. More recently identified challenges have involved the basis for screening out events that are considered "not credible" and identifying which design characteristics are appropriate to designate as IROFS.

NRC requirements, as seen above, include elements of both a probabilistic and deterministic nature; both are addressed in NUREG-1520 (mainly probabilistic in Chapter 3 and deterministic in Chapter 5). SSG-27 states that "criticality safety assessments have generally been based on a deterministic approach," which it defines as being based on "conservative rules and requirements." If those rules and requirements are satisfied, the risk may be "inferred" to be acceptably low. However, it allows that it is "common to complement the deterministic approach to criticality safety assessment with a probabilistic approach" (Sections 4.1 and 4.2 of SSG-27). Thus, a probabilistic approach is seen as merely complementary or supplementary, and the main focus is on the deterministic. Accordingly, there is considerable guidance in SSG-27 concerning the traditional deterministic approach to criticality, and very little if any concerning the probabilistic approach. (Some care must be taken with regard to terminology. SSG-27 includes within the scope of a deterministic approach "a qualitative judgement of the likelihood of failure," which NRC guidance would consider part of the probabilistic approach, because it is a qualitative description of probability. NUREG-1520 refers to the "index method," in which likelihood scores are added together to obtain an overall likelihood index for a particular accident sequence, as "semi-quantitative." Regardless of the terminology used, SSG-27 says almost nothing with regard to either qualitative or quantitative risk assessment.)

Another important aspect of both deterministic and probabilistic methods is defense-in-depth. In a purely probabilistic approach, the choice of controls does not matter, as long as they are of sufficient depth of meet whatever likelihood goal has been established. However, in 10 CFR Part 70, there is a statement that the facility design must incorporate a preference for engineered over administrative control; NRC guidance further defines a preferred control hierarchy, preferring: (1) engineered over administrative control; (2) passive over active engineered; (3) enhanced over simple administrative; (4) diversity over redundancy; and (5) fixed geometry over other controlled parameters. SSG-27 does not mention this, except to say that defense-in-depth should include passive safety features and double contingency (which it classifies as “fault tolerance”). While NRC guidance agrees that passive safety control is preferable, including this in the definition of defense-in-depth is problematic, as it is not always possible to engineer passive safety into a facility. Both NRC guidance and SSG-27 agree that criticality control may be based on one or more parameters, but that diversity in parameters is preferable. In the US, there has been a robust debate concerning whether such single-parameter control can be said to constitute double contingency protection [7]. The NRC has historically considered double contingency to be able to be based on single-parameter control [8], though as stated above, this is not preferable.

A final distinction between NRC guidance and SSG-27 concerns what NUREG-1520 refers to as the “reliability and availability qualities” (R&AQs). The lists provided in Chapter 3 of NUREG-1520 and Section 3.51 of SSG-27 are very similar. The R&AQs include such factors as safety margin, control type, complexity, independence, reliability, diversity, redundancy, demand rate, fail-safe or self-revealing status, management measures, and feedback from operating experience. These technically apply to qualitative assessments of likelihood, which constitute a qualitative approach to probabilistic analysis. However, they could similarly be treated as deterministic criteria, the satisfaction of which may be “inferred” to result in acceptably low risk. Thus we see the dividing wall between a qualitative risk-informed approach and a deterministic approach can be very thin.

II.B Exemptions from Analysis and Monitoring Requirements

The requirement for having criticality detectors—and not necessarily the alarm horns (whose presence may depend on the potential dose from neighboring areas)—depends on the quantity of material present or available. The rule limits the required area of coverage to those areas having greater than a critical mass of uranium or plutonium (same definition as in Section I). Besides the mass limits, there may be other considerations that justify not requiring a criticality alarm system. While 10 CFR 70.24 requires a criticality alarm system whenever the above mass limits are exceeded, 10 CFR Part 70 makes allowance for exemption from any part of the regulations at the NRC’s discretion. An exemption may be granted if authorized by law, if the NRC determines it will not endanger life, property, or the common defense and security, and if it is otherwise in the public interest. Although no specific guidance has been developed, licensees have often been granted exemptions based on risk arguments showing that the risk of criticality is negligible or ‘not credible.’ A common example is in outdoor solid UF_6 cylinder storage yards, where uranium concentration is very diffuse (such as low-level waste or contaminated soil), or where finished fuel is stored in its approved shipping configuration (due to the very conservative nature of transportation requirements). ANSI/ANS-8.3-1997, which is widely used in Department of Energy (DOE) facilities, allows a determination to be made on the need for criticality alarm coverage based on evaluation of the overall risk. This is not consistent with the NRC’s regulations, which require the NRC’s prior approval for any areas for which an exemption is sought.

SSG-27 takes the approach of defining specific criteria for excluding criticality alarms from areas containing greater than a “safe mass” (which is not defined, but considered half a minimum critical mass). These criteria include when criticality is not credible, when there is no appreciable risk benefit or when the overall risk to workers may be increased, when shielding reduces dose to acceptable levels, or when fuel is stored in shipping configuration. In the case of shielding, it states that even without an evacuation alarm, a means of detecting the criticality should still be provided. This approach is similar to that taken in ANSI/ANS-8.3, except it provides greater definition of the licensee’s evaluation of the need for coverage.

A related concept is the threshold for requiring criticality analysis. There is no minimum threshold stated in the rule or guidance, but the NRC has allowed licensees to exclude those areas having less than a certain mass (ranging from 15g to 350g ^{235}U). SSG-27 allows exemption for areas involving low fissile mass or low isotopic concentration—without defining those quantities—or those that are “fissile exempt” under transportation requirements. In both cases, exemption from requiring a criticality alarm system and exemption from criticality analysis, consideration should be given not just to quantities present under normal conditions, but also to those that might be present under credible abnormal conditions.

II.C Subcritical Margin and Validation

The topic of margin is complicated by a lack of consistency in terminology regarding the various types of margin. In NRC guidance (e.g., Ref [2] and [9]), the term *safety margin* has been taken to mean margin in macroscopic system parameters, and *subcritical margin* to mean margin in neutron multiplication, k_{eff}. For example, safety margin may be the margin between the amount of mass in a system and the amount of mass needed to attain criticality. Even with this distinction, is this the mass under normal conditions or abnormal conditions? Is it the actual mass present or the mass assumed in calculations? Under what conditions is the mass needed for criticality determined? For subcritical margin, there are several contributors that may go into the margin, including a bias and bias uncertainty calculated from benchmark experiments, an administrative margin for unknown uncertainties (in NRC guidance referred to as the *minimum margin of subcriticality*), a margin for extrapolating outside the validated area of applicability, etc. So it is very important to clearly define what margin is being discussed.

Both NRC guidance and SSG-27 recognize that the margins relied on to provide assurance of subcriticality may be expressed in terms of system parameters, k_{eff}, or both. 10 CFR 70.61(d) states that assurance of subcriticality includes ‘use of an approved margin of subcriticality for safety,’ without defining what this *margin of subcriticality for safety* consists of. For other regulated areas, such as transportation and spent fuel pools, NRC rules and guidance specify a maximum allowed k_{eff} (such as 0.95 used in analyzing transportation packages, 0.95 or 0.98 used for spent fuel pools as in 10 CFR 50.68). For fuel facilities, the margin may vary between facilities, or from process to process within a facility, and is approved on a case-by-case basis. FCSS-ISG-10 [10] describes several criteria that NRC staff should consider in deciding on the acceptability of the proposed margin—conservatism in calculational models, validation methodology and results, system sensitivity and uncertainty, knowledge of neutron physics, and the likelihood of abnormal conditions. In discussing conservatism in calculational models, the ISG recognizes that margin may be expressed as a combination of margin in k_{eff} and margin resulting from the conservative assumptions and technical practices used in setting up those models. Those assumptions and practices should be described in the license application and may form part of the approved margin of subcriticality for safety.

SSG-27 also recognizes that margin may be expressed compositely, and includes a similar set of factors to be considered in setting safety margins. Section 3.18 of SSG-27 enumerates these as uncertainty in the system’s conditions, the probability or rate of change of system conditions, and consequence of a criticality accident. Use of the term *safety margin* here is indicative of the appearance that this pertains more to setting safety limits in the facility than to defining limits for performing criticality analyses. Uncertainty in the modeled conditions is normally handled in criticality analyses by adding sufficient conservatism to ensure (in the words of NUREG-1520) that the “uncertainty and variability in operating parameters” is suitably bounded. This concerns uncertainty in the actual conditions, rather than in the calculational method, and is therefore a different kind of margin than the subcritical margin discussed herein. Similarly, the probability or rate of change of system conditions is an important consideration, but is more applicable to operating or safety margin than to subcritical margin. Consequence had not been recognized as one of the factors to be considered in FCSS-ISG-10, but was added as part of the Yucca Mountain review [11]. There, a reduced subcritical margin was justified based on the lack of any significant dose from a post-closure criticality accident. In similar fashion, transportation casks have historically been certified based on a subcritical margin of 0.05, which exceeds the margin at many fuel facilities, because of increased potential dose to the public due to the closer proximity. This idea is further reflected in the NRC's endorsement of the U.S. standard ANSI/ANS-8.10, which allows for a relaxation in margins and layers of protection when the potential dose is reduced due to the presence of thick shielding.

Validation is an important part of ensuring adequate subcritical margin, because it has a direct bearing on our confidence in the calculational method. SSG-27 and NRC guidance are largely consistent on the expectations for criticality code validation. Both allow for the extrapolation of the validated area of applicability where benchmark data is lacking, for example, although the NRC has traditionally placed limits on that extrapolation (e.g., Ref. [9] limits extrapolation to ±10% of the range covered by the benchmarks). Practices related to validation vary widely in the US, as reflected by the NRC's two exceptions to ANS-8.24-2007 in Reg. Guide 3.71 [12]: (1) the NRC has historically had a prohibition on the use of so-called "positive bias" (defined herein as meaning that the calculated k_{eff} exceeds the experimental k_{eff}), and (2) the NRC does not consider it appropriate to reject benchmark "outliers" based solely on statistical considerations. These exceptions do not so much reflect a true disagreement with SSG-27 as the fact that NRC guidance is considerably more detailed than SSG-27 in regard to validation.

II.D Mitigation of Consequences

As stated above in regard to the subcriticality requirement of 10 CFR 70.61(d), criticality must foremost be prevented. Consequence mitigation is an important part of the protection strategy for many hazards found at fuel facilities, but criticality is not one of them. This does not mean that *no* provision is made for mitigating the dose from an inadvertent criticality, but merely indicates that almost all of the emphasis is on prevention. On this point, 10 CFR 70.61(d) is worded as follows:

…the risk of nuclear criticality accidents must be limited by assuring that under normal and credible abnormal conditions, all nuclear processes are subcritical…Preventive controls and measures must be the <u>primary means of protection</u> against nuclear criticality accidents.

This was put in place to ensure that, even when dose consequences to workers and the public are minimized, such as where shielding or great distance is present, criticality is still unacceptable. The other performance requirements are only concerned with the product of likelihood and consequence, but 70.61(d) establishes that criticality must be prevented regardless of the consequence. The occurrence of a criticality accident represents a gross loss of control, as well as an environmental problem. Thus, 70.61(d) may be considered a *deterministic* performance requirement that is in addition to the *probabilistic* or *risk-informed* performance requirements preceding it. This is entirely consistent with the process analysis requirement in ANS-8.1, although there has been substantial debate in the industry lately as to how this requirement is to be interpreted in light of the text of ANS-8.10, which allows for relaxation of certain requirements when heavy shielding is present. Consistent with this is the language that follows the subcriticality requirement that 'preventive controls and measures' are the 'primary means of protection.' Note that this does not preclude the use of mitigation, but merely requires safety to be based primarily on prevention. In fact, mitigation is normally expected, in the form of having a criticality accident alarm system (CAAS) and associated emergency response procedures, which are primarily associated with ensuring timely evacuation and medical intervention. These measures are required under 10 CFR 70.24, and expanded upon in NRC-endorsed standards, most notably ANSI/ANS-8.3 and -8.23.

Only shielding and distance prior to an accident (and not following evacuation) are considered mitigation from the standpoint of the performance requirements of 10 CFR 70.61. This is because, unlike in a slowly evolving chemical release or reactor accident, the release of radiation in excess of the thresholds of 10 CFR 70.61 can occur almost instantaneously. The mitigative measures required deterministically, such as a CAAS and emergency response, may protect additional individuals, but will do nothing to protect individuals present at the scene of the excursion, and will therefore not prevent the high-consequence threshold from being exceeded. For this reason, such measures are not classified as items relied on for safety, though they are very important to protect health and safety.

This issue of mitigating the dose from a criticality accident forms the largest area of divergence between NRC regulations and guidance and SSG-27. The safety philosophy of SSG-27 is based on five levels of defense-in-depth that include first prevention and then mitigation. These five layers of protection (from Table 1 of SSG-27) are summarized below:

I. Prevention of deviations from normal operations

II. Detection and interception of deviations to prevent anticipated abnormal occurrences from escalating to accident conditions

III. Control of events within the design basis (or equivalent) to prevent a criticality accident

IV. Mitigation of accident consequences beyond the design basis (or equivalent), and keeping the consequences of a criticality accident as low as practicable

V. Mitigation of consequences due to a release of radioactive material

The application of these five levels to criticality safety was the subject of considerable discussion at the IAEA workshop in February 2014. The origin of these five levels appears to be based on a reactor transient model, in which a process whose reactivity is controlled and closely monitored is subject to a slowly-evolving transient. Such a transient may permit several opportunities to interrupt the transient or take mitigative measures in response to a transient. This is not characteristic of a typical criticality accident (though it may occur in steady-state criticality), which makes fitting criticality safety into such a framework challenging. For example, under the NRC's Part 70 regulatory framework, plant conditions are classified as normal conditions, credible abnormal conditions, and accidents. Normal conditions are those in which all safety controls function as designed. Credible abnormal conditions arise as the result of a contingency, and are required to be subcritical. Thus, in accordance with the double contingency principle, there are in general two preventive layers of protection. However, the requirement as stated in the double contingency principle is that there be "*at least two…changes in process conditions*" before a criticality accident is possible. As stated in NUREG-1520 and FCSS-ISG-01, it is not always necessary or sufficient to have two layers of protection (controls) needed to achieve this. Double contingency "*may necessitate one, two, or more than two controls depending on the possible conditions that can lead to criticality*." An example of where a single control may be sufficient could be a very robust favorable geometry column, or a siphon break to prevent backflow, provided there is no identifiable credible means of failure. An example where more than two controls may be necessary could be reliance on dual independent sampling for transfers from favorable to unfavorable geometry. Because of the complex administrative nature of sampling and the strong potential for common mode failure, normally dual sampling would only be credited as one leg of double contingency (the other typically being some kind of engineered device, such as an inline radiation monitor). As the above examples show, these controls may be passive or active engineered, or may be administrative. Active monitoring of the system to interrupt an undesirable transient condition is much less prevalent in a fuel facility than in a reactor.

Following two independent and unlikely contingencies, criticality may occur. Following an accident, criticality alarms (unless they have been determined to be unnecessary) would trigger evacuation and other emergency response activities to minimize further doses to personnel. Thus, there are only three required layers of protection against criticality in the NRC's framework—(1) a preventive layer (explicit control, natural and credible course of events, etc.); (2) a second preventive layer; and (3) a mitigative layer consisting of criticality monitoring and associated emergency response activities.

The concept of a 'design basis' doesn't truly apply to a criticality accident, as this concept is normally associated with hazards resulting in offsite consequences. (For this reason, the term 'or equivalent' was added after 'design basis.')

One other significant difference between SSG-27 and the NRC's framework is that SSG-27 has expectations to include the following in the design of nuclear processes: (1) designing for the "slow progression" of an accident, to enable intervention prior to reaching criticality (Section 3.10 of the SSG);

(2) designing to allow automatic or human termination of critical excursions (Sections 3.4, 6.22, 6.27, 6.28, and 6.58), and (3) designing shielding for consequence mitigation (Section 3.13). While these are technically sound practices and good design goals, none of them are required by NRC's rules or recommended by its guidance. The focus in SSG-27 on intervention (both prior to and after an accident) and mitigation exceed what is required for adequate protection in the NRC's regulations and guidance, which, as discussed above, focus mainly on prevention.

II.E Configuration Management and Change Control

Partly in response to significant fuel cycle events (most notably, the Sequoyah Fuels accident in 1986 and GE-Wilmington near-criticality in 1991), 10 CFR Part 70 was revised to require a more formal configuration management program, facilitated by listing all items relied on for safety (IROFS).[6] Loss of configuration control has been prominent in more recent fuel facility events, as well as several of the fuel facility criticality accidents, notably Tokai-mura. SSG-27 mentions the importance of configuration management and change control in several places, including the need for periodic inspection of plant operations and thorough review of modifications (Sections 2.12, 2.14, and 3.37). SSG-27 does not contain detailed guidance for how these goals should be accomplished. NRC regulations and guidance have been far more prescriptive in specifying a structured framework for configuration management and change control, mainly due to the aforementioned events. 10 CFR 70.72(a) states that licensees must have a configuration management system that applies to the entire facility (specifically the site, structures, processes, systems, equipment, components, computer programs, and activities of personnel), and is not just confined to safety controls. 10 CFR 70.72(b) and (c) require that changes must be evaluated prior to being implemented, and specify what changes may be made without prior NRC approval (in the form of a license amendment).

While the configuration management program applies to the entire facility, not just safety controls, the designation of controls as IROFS is central to having a controlled '70.72 change process.' This is because the criteria listed in 10 CFR 70.72(c) are mainly expressed in terms of accident sequences and IROFS, which are listed in a licensee's ISA Summary that is submitted to the NRC for approval upon applying for a license (or, for pre-existing facilities, was required to be submitted four years after issuance of the rule). It is instructive to paraphrase those criteria below. Licensees may make changes to their facilities without prior NRC approval if the change does not:

- Create new types of accident sequences exceeding the performance requirements of §70.61 and which have not previously been described in the ISA Summary
- Use new processes, technologies, or control systems for which the licensee has no prior experience
- Remove without at least an equivalent replacement of the safety function, an item relied on for safety necessary for compliance with the performance requirements of §70.61
- Alter any sole item preventing or mitigating an accident sequence
- Is not otherwise prohibited by regulation, license condition, or order

Even though licensees may make changes that meet all of these criteria without obtaining a license amendment from the NRC, they are required to submit a list of facility changes and any corresponding changes to the ISA Summary annually.

6. It must be noted that not all controls relied on for criticality safety need be identified as IROFS. Under Part 70, those controls needed to demonstrate that a licensee has met the performance requirements must be identified as IROFS. However, controls relied on to meet the double contingency principle do not have to be IROFS; there is no requirement that both sets of controls be identical. The difficulties this has led to have been acknowledged in [5].

Despite this structured framework, the management of facility changes in a way that ensures adequate regulatory oversight while allowing reasonable operational flexibility has proven to be complex and challenging. This has given rise to issuance of Regulatory Guide 3.74, “Guidance for Fuel Cycle Facility Change Processes” [13] to clarify the 70.72 criteria. Several issues that have proven problematic concern the definitions of “new types of accident sequences,” “new processes, technologies, or control systems,” “prior experience,” “equivalent replacement of the safety function,” and “sole item preventing or mitigating an accident sequence.” The question of which features of a facility or process included in a criticality safety analysis must be considered controls, and which controls need to be designated as IROFS, has not been as straightforward as may at first be thought. Criticality safety analyses often contain very detailed system models, giving rise to the criticism that a strict interpretation of 10 CFR 70.61(e), which specifies that controls needed to meet the performance requirements be designated as IROFS, would require everything in the facility to be an IROFS. Clearly, such an outcome would defeat the intent of having a distinctive class of items warranting special regulatory and licensee attention. Some licensees have taken the approach of minimizing the number of controls designated as IROFS to increase their flexibility to make changes. Others have taken the approach of making all, or nearly all, controls IROFS to reduce the number of event reports and apparent significance of control failures. This issue arises in distilling a large collection of criticality controls down to a reasonable set of IROFS for credible events, but also affects the determination of which events are to be considered credible. NUREG-1520 contains the guidance with regard to credibility that “*the ‘not credible’ nature of an event must not depend on any facility feature that could credibly fail to function or be rendered ineffective as a result of a change to the system*” and “*such a demonstration of ‘not credible’ must be convincing despite the absence of designated IROFS*.” The conundrum this leads to is the possibility of deciding that an event is incredible based on certain controls which are assumed to be present, and then using this as justification for not designating those controls as IROFS. Historically the basis for incredibility has not been well-documented, as the ISA Summary tends to focus only on the remaining credible accident sequences, and so not recognizing those controls as IROFS could make it more likely they would be degraded or removed in the future, which could make what was formerly supposedly incredible become credible. Finally, the more stringent requirements concerning so-called “sole IROFS” has led some licensees to avoid their use, and instead rely on multiple controls that may result in reduced safety. Typically, “sole IROFS” tend to be among the most robust engineered controls, such as favorable geometry. Whereas the NRC’s guidance states that reliance should be based on favorable geometry wherever practical, in cases where there’s only one such barrier available, licensees may choose to establish control on a less preferable parameter (such as concentration or reflection) to avoid using geometry control as a sole IROFS. This is an example where well-intentioned regulation—seeking to encourage greater defense-in-depth—can have the unintended consequence of leading licensees to conditions of reduced conservatism. The various issues associated with configuration and change control have been recognized and led to efforts to develop national and international standards (draft ANSI/ANS-57.11, “Integrated Safety Assessments for Fuel Cycle Facilities,” proposed ISO standard “Nuclear Criticality Safety Dimensions”).

II.F Level of Prescriptiveness

When 10 CFR Part 70 was revised in 2000, a primary motivation was to increase confidence in the margin of safety by making fuel facility regulation more risk-informed and performance-based [14]. The first goal was discussed at length in Section II.A. The second goal, that of being performance-based, was to be achieved by identifying performance requirements for preventing and mitigating accidents, requiring performance of an ISA (including the identification of accidents, IROFS, and management measures), and requiring reporting of program and facility changes to the NRC annually, while allowing licensees to make certain changes without prior NRC approval [1]. The first few of these strategies are in line with the philosophy of performance-based regulation, namely establishing certain performance criteria to ensure acceptable risk, but leaving it up to licensees to determine the means of meeting them. In short, the regulations specify *what* overarching safety goals must be met, but do not tell licensees *how* to meet them.

The latter strategies are in recognition that there is a twofold approach to ensuring safety: a *programmatic* approach and a *technical* approach.

The evolution of fuel facility regulation has been one of gradual reductions in prescriptiveness. The first major step was the issuance of the current revision to Part 70 in 2000. The second was the development of guidance to enshrine this risk-informed and performance-based regulation. The next was review of the ISA Summaries and on-site ISA documentation, and this has been followed by gradual refinement in the guidance to address areas where guidance was found to be insufficient, as well as removing prescriptive acceptance criteria later found to be unnecessary.

The performance based, non-prescriptive regulatory framework contained in Part 70 applies to all fuel cycle facilities, whether enrichment or fuel fabrication facilities. The ISA requirements in particular, in Subpart H of Part 70, are generic in that they can be applied to a wide variety of facilities across the nuclear industry (and are being extended to conversion facilities regulated under Part 40 as well). Currently this approach is limited to facilities at the front end of the fuel cycle (starting with uranium conversion and ending prior to irradiation). Transportation of fresh fuel and transportation, processing, and disposal of spent fuel are not included. In particular, transportation requirements tend to prescribe more conservatism because of the higher potential criticality dose consequences to members of the public. Despite this, within the fuel cycle arena, the NRC's goal has been to develop guidance that is as generic as practical. Licensees have the latitude to design their programs, analysis methods, control systems, and management measures appropriate to their facilities. Licensees are permitted to develop their own methods for performing ISAs, including individual criteria for likelihood and consequence determination, for example, provided they meet the performance criteria of 10 CFR 70.61. Another example is the tradeoff between operational flexibility and complexity. Performing very detailed criticality analyses and calculations may allow reductions in conservatism, resulting in economic gains, while performing more conservative bounding analyses may reduce demands on staff resources and the number of controls that need to be established. Conversely, SSG-27 contains much more prescriptive guidance for different types of facilities, including conversion and enrichment facilities, fuel fabrication, spent fuel operations, reprocessing, waste management and decommissioning, transportation, and research and development. The NRC has considered the essential elements of an NCS Program to be the same regardless of the type of facility; since its guidance mainly programmatic, it is feasible to make this guidance rather generic.

III. OPERATING EXPERIENCE IN FUEL CYCLE FACILITY CRITICALITY SAFETY

In the non-prescriptive environment of performance-based regulation, consideration of operating experience is crucial as an objective gauge against which compliance with the performance criteria can be verified. For fuel cycle facilities in this incipient stage of performance-based regulation, the difficulties are twofold: (1) fuel facilities are very diverse, as are the methods that have been developed for analyzing them, and (2) the risk criteria involve such low levels of likelihood that there will be very few occurrences of significance. The amount of relevant operating data from a particular fuel facility will necessarily be very scant, and data such as it exists is likely to be anecdotal and not easily generalized. Operating data from other facilities may be of dubious applicability. A major difference between fuel facilities and power reactors is that in the case of fuel facilities there is no large ensemble of nearly identical facilities from which to draw general conclusions. Thus, the only means of determining if such low likelihoods are being met is to wait a prohibitively long time and count the number of similar events that actually occur (and of course, when events do occur it usually leads to design or operational changes, skewing any such count).

This feedback, where process designs are subject to gradual improvement as control failures and events reveal safety vulnerabilities, is crucial to ensuring that facilities meet the performance requirements, given that accurate prior knowledge of specific failure frequencies and probabilities is not generally available. The safety program required by 10 CFR 70.62 includes maintaining records of IROFS or management measures failures, a main purpose of which should be to assess the adequacy of control measures and provide feedback to the risk assessment process on a continuing basis.

Nevertheless, the NRC is currently formulating an Operating Experience program applicable to fuel cycle facilities, which is envisioned to be structured similar to the NRC's Operating Experience program for reactors. While this program is still under development, it is envisioned to encompass both safety and security and have as a main goal the collection of industry experience to inform regulatory decision-making as well as communicating generic concerns to the industry in a timely manner. One example is a generic letter on the treatment of natural phenomena at fuel cycle facilities (in response to the events at Fukushima), which is currently under development.

IV. EXEMPTIONS FROM NRC REGULATORY REQUIREMENTS

Regulations are intended to provide an adequate envelope of safety, but may not necessarily cover all possible circumstances. Room therefore must be provided to allow for exceptions. Staff is permitted under 10 CFR 70.17 to grant an exemption from the regulations of Part 70, either at the request of an applicant or its own initiative, provided the change is authorized by law, does not endanger life or property or the common defense and security, and is otherwise in the public interest. In practice, most of the exemptions that have been granted with regard to NCS involve exemptions to the criticality monitoring and alarm requirement in 10 CFR 70.24(a). This regulation requires an alarm system in all areas exceeding a critical mass as defined in the regulations (and as discussed in Section I of this paper). In practice, exemptions have often been granted where it can be shown that there is no credible pathway to criticality, despite the presence of a critical mass, or if criticality is credible, that the risk is negligibly low, such that there is no tangible benefit, or the risk due to evacuation and disruption of operations would exceed any slight benefit that may be realized.

V. CONCLUSION

Understanding approaches to criticality safety in different countries and different regulatory environments can provide useful insights for all practitioners, and takes on increasing importance as international cooperation increases. While the underlying best practices are very similar, salient differences are seen with regard to methods for risk assessment, safety margin, and emphasis on mitigation rather than prevention. When considering the approaches discussed in this paper, it is necessary to remember that the different approaches have all contributed to the exemplary worldwide track record of safety, in that there has not been a criticality accident since 1999.

References

1. "Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material." 65 FR 56211 (September 18, 2000).
2. NUREG-1520, Standard Review Plan for License Applications for Fuel Cycle Facilities," Rev. 1, U.S. Nuclear Regulatory Commission, 2010.
3. LA-13638, "A Review of Criticality Accidents," Los Alamos National Laboratory, 2000.
4. FCSS-ISG-03, "Nuclear Criticality Safety Performance Requirements and Double Contingency Principle," U.S. Nuclear Regulatory Commission, 2005.
5. C. Tripp, *et al*., "Transition from a Traditional Deterministic Nuclear Criticality Safety Framework to a Risk-Informed Integrated Safety Analysis Approach: Methodology," ICNC 2007, St. Petersburg, Russia, 2007

6. K. Morrissey, *et al*., “Transition from a Traditional Deterministic Nuclear Criticality Safety Framework to a Risk-Informed Integrated Safety Analysis Approach: Integrated Safety Analysis Summary,” ICNC 2007, St. Petersburg, Russia, 2007
7. S. Monaghan, *et. al*., “The Language of the Process Analysis Requirement and the Double Contingency Principle,” Trans. Am. Nuc. Soc., **105**, 551 (2011).
8. C. Tripp and D. Morey, “The Double Contingency Principle and Subcriticality Revisited,” Trans. Am. Nuc. Soc., **106**, 505 (2012).
9. NUREG/CR-6698, “Guide for Validation of Nuclear Criticality Safety Calculational Methodology,” U.S. Nuclear Regulatory Commission, 2001.
10. FCSS-ISG-10, “Justification for Minimum Margin of Subcriticality for Safety,” U.S. Nuclear Regulatory Commission,” 2006.
11. C. Tripp, “A Risk-Informed Approach to Subcritical Margin for Applications Within and Beyond Fuel Cycle Manufacturing,” Nuclear Criticality Safety Division Topical Meeting, Richland, WA, 2009.
12. RG.371, “Nuclear Criticality Safety Standards for Fuels and Material Facilities,” Rev. 2, US Nuclear Regulatory Commission, 2010.
13. RG-3.74, “Guidance for Fuel Cycle Facility Change Processes,” US Nuclear Regulatory Commission, 2012.
14. “Domestic Licensing of Special Nuclear Material; Possession of a Critical Mass of Special Nuclear Material.” 64 FR 41338 (July 30, 1999).

International Workshop on Operational and Regulatory Aspects of Criticality Safety

REGULATORY ASPECTS OF CRITICALITY SAFETY IN FUEL CYCLE FACILITIES IN SPAIN

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ABSTRACT

The main Fuel Cycle Facilities (FCFs) in Spain are a Fabrication Facility and three Interim Spent Fuel Storage Installations (ISFSIs) in operation, and one more in the licensing process. Additionally a Centralized Temporary Storage (CTS) is under construction, scheduled to be in operation in 2017. This paper addresses the criticality safety regulatory aspects for each of them:

I. Juzbado Fuel Fabrication Facility: Nuclear Criticality Safety (NCS) approach based in the conservative and specific modelling of the fissile material configurations in the facility. The Integrated Safety Analysis (ISA) of the facility has been required and recently finalized, with results in terms of criticality severity margins, and Items Relied On For Safety (IROFS) being identified and implemented.

II. Interim Spent Fuel Storage Installations, with two coexisting spent fuel cask concepts: dual purpose metallic casks and multipurpose canisters, operated with different overpacks for storage or transportation conditions. Specific NCS analysis for each storage and transportation safety case, with burnup credit in transportation cases.

III. CTS Facility where the fuel will be unloaded from the transportation casks and stored in the facility specific canisters, which design is being finalized.

• INTRODUCTION

The Regulation on nuclear and radioactive facilities, "Reglamento sobre instalaciones nucleares y radiactivas", approved in 1999 and modified in 2008, is the national regulatory framework for FCFs. CSN has developed specific standards consistent with US regulations providing guidance to comply with safety requirements to prevent nuclear criticality events.

As a general principle, the Criticality Safety approach in Spanish FCFs is based in deterministic analysis of the specific nuclear material configurations to ensure that they remain subcritical under normal, off-normal and accident conditions, unless at least two unlikely independent events occur. Risk informed elements are being incorporated into the methodology to identify weaknesses and areas for improvement within the previous deterministic frame. The following sections describe the specific situation in each FCF.

Figure 1. Fuel Cycle Facilities in operation in Spain

- **JUZBADO FUEL FABRICATION FACILITY**

This Fuel Fabrication Facility in operation since 1985, manufactures LWR fuel assemblies (PWR 17x17, 16x16, 15x15, 14x14, BWR GE11, GE12/14,GNF2, VVER-440) starting the fabrication process with the reception of enriched uranium powder being brought from UK or USA. There is no conversion process performed in the facility. Table 1 shows main fabrication steps:

Table 1

Main fabrication steps in the Juzbado fuel fabrication facility

Area	Main Steps
Ceramic Area: powder>pellet>rod 	UO2 powder reception, transfer to drums Powder mix with additives (mixers) Powder compressing to obtain “green pellets” Pellet sintering to achieve nominal density Geometry rectifying
Mechanical Area: rod>fuel assembly 	FA assembly FA inspection and washing FA storage and transportation cask loading

The facility was licensed in 1985, and a comprehensive set of regulations, inspections and Periodic Safety Reviews (PSRs) ensure the objective of maintaining safety throughout the operating life of the facility:

- PSRs, mandatory every ten years, systematically assess the cumulative effects of facility ageing, design modifications, operating experience, new standards, technical development and siting aspects.
- A Basic Inspection Program is established by CSN identifying the minimum required periodic inspections to determine whether the facility is operating safely and securely in accordance with regulatory requirements, and to identify indications of declining safety or safeguards performance. Specific inspection procedures are issued by CSN for every safety topic, addressing objectives and required inspection frequency. A Criticality Safety Inspection is required once a year, additional to any other performed to follow/review criticality relevant design modifications or events.

Criticality Safety requirements are contained in the Facility licensing documentation:

- Technical Specifications
  - Criticality Alarm System
- Safety Analysis Report (Spanish Regulation on Nuclear and Radioactive Facilities consistent with NUREG-1520: Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility - Final Report)
  - Chapter 7 “Nuclear Safety (Criticality Control)” supported by the “Nuclear Criticality Safety Report”.

7.1 Basic principles.

7.2 Technical Requirements

7.3 Administrative Requirements

7.4 Variation range of FA design parameters (density, enrichment, geometry).

7.5 Criticality Safety Requirements applicable to the Juzbado facility processes.

7.6 Nuclear Criticality Safety Report chapters.

7.7 References

The regulatory Criticality Safety approach in the Juzbado Fuel Fabrication Facility is based in the NCS analysis and evaluation through conservative and specific modelling (composition, mass, geometry, moderation) of the fissile material configurations in all the manufacturing processes involving nuclear material, assuming any expected normal and accident operating conditions. NCS analysis includes the interfaces with the transportation NCS requirements for incoming UO_2 powder casks and outgoing manufactured fuel assembly’s transportation casks.

➢ Nuclear Criticality Safety Report (NCSR):

NCSR is not an official licensing document but the input for the SAR (chapter 7). The results of NCS calculations for every step of the manufacturing process are directly extracted to SAR in terms of *Controlled Parameters* (Mass, Geometry and/or Moderation depending on the process), *Safety Limits* and *Control Type* (administrative or passive/active engineering controls) required to ensure the established Safety Limits. Normal and accident conditions are analysed, covering any expected mass, geometry and moderation condition in each process.

The report consists of 18 chapters where detailed reactivity calculations are performed for every fabrication phase involving fissile material:

- Specific code system sequences SCALE/ CSAS25/26/2X validation.
- Generic assessment for powder, pellets, rods…basic configurations. Maximum allowed values for key safety parameters.
- Pellet and cladding diameter uncertainties effect on fuel assembly reactivity
- Specific assessment of each step process:
  - ✓ Powder storage area
  - ✓ Powder preparation and green pellets fabrication process
  - ✓ Green and sintered pellets storage. Sintering process.
  - ✓ Oxidation process
  - ✓ Rectification process and rod loading.
  - ✓ Solid waste treatment
  - ✓ Liquid waste treatment
  - ✓ Rod storage
  - ✓ Rod inspection operations
  - ✓ Final assembly of PWR and BWR fuel assemblies.
  - ✓ Final inspection of PWR and BWR fuel assemblies
  - ✓ Washing and drying process of PWR fuel assemblies
  - ✓ Storage of PWR and BWR fuel assemblies
  - ✓ Transfer carts for nuclear material
  - ✓ Chemical Laboratory

The report has been thoroughly reviewed and evaluated by CSN to verify that:

- The Calculations are performed:
  - ✓ with conveniently validated analytical methods
  - ✓ accounting for uncertainties derived from validation process, as well as those from calculation
  - ✓ at a 95 percent probability, 95 percent confidence level.
- The Assumptions cover any anticipated sequence, including in the report each parameter necessary to reproduce any calculation. Assumptions and simplification should be enough conservatives.
- The Results in terms of estimated ratio of neutron production to neutron absorption and leakage (k-effective) do not exceed the values stated according to operating condition and system to guarantee subcriticality:
  - ✓ $K_{eff} < 0.90$ for normal operating conditions and criticality control not geometrical.
  - ✓ $Ke_{ff} < 0.95$ for normal operating conditions and geometrical configuration maintained by means of structures.

✓ $K_{eff} < 0.95$ for accident conditions

✓ $K_{eff} < 0.98$ for anticipated conditions of very low probability

Figure 2 includes an example of NCSR results for a specific configuration in the UO_2 Powder Storage Area: six UO2 powder drums diameters analysed (from 15 to 35 cm) in the storage area in four different possible configurations (geometry parameter) on the storage roller rows (CND1, 2, 3 and 4) to comply with NCSR assumptions for the worst expected conditions.

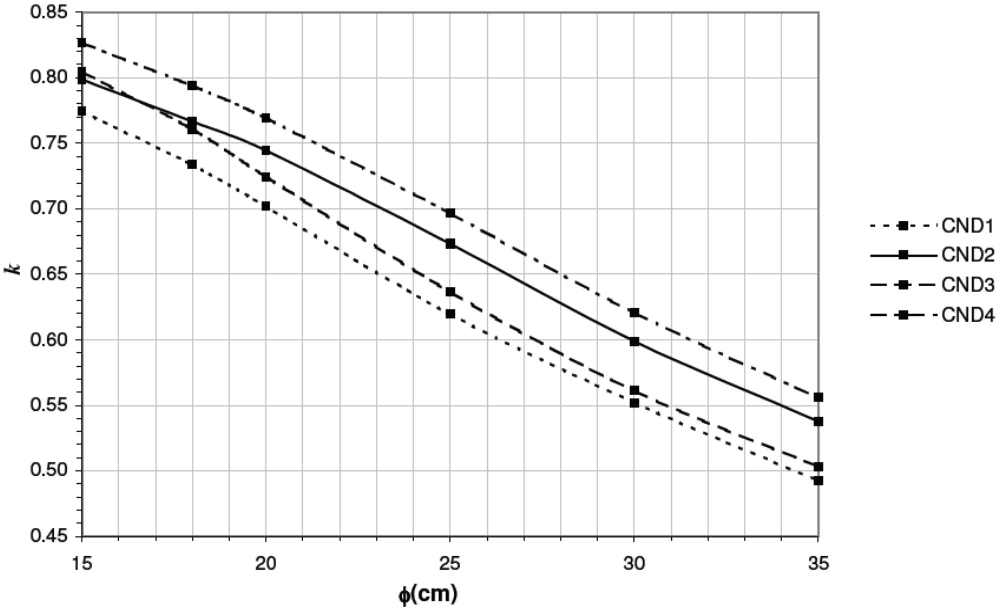

Figure 2. Results of the NCS Analysis for different diameters and drums configuration in the Powder Storage Area. *Criticality Safety Report. Juzbado Fuel Fabrication Facility.*

CSN accepted this Criticality Safety Analysis methodology and NCSR is the main basis for the evaluation of any design modification involving processes where fissile material is handled.

Some recent relevant design modifications evaluated on the light of the NCSR and requiring its modification:

- Evaluation of the NCS analysis for a new BWR fuel bundle design fabrication in the Juzbado facility (GNF2): impact in NCSR (new calculations for the new bundle specific geometry) and derived modifications in SAR Chapter 7.
- Evaluation of the NCS analysis of the modification for the Powder Storage extension: impact in Powder Storage NCSR chapter and derived modifications to SAR Chapter 7.
- Evaluation of the NCS analysis for the new equipment lay-out in the mechanical area: impact in NCSR and derived modifications to SAR Chapter 7.
- Evaluation of the NCS for a new Oxidization Furnace in Juzbado facility: impact in Oxidation Process NCSR chapter and corresponding modifications to SAR Chapter 7.
- Evaluation of the NCS analysis for a new PWR Rods Storage Lay-out: impact in Rod Storage NCSR chapter and corresponding modifications to SAR Chapter 7.
- Evaluation of the NCS analysis for the use of a new UO2 powder drum design: impact in Storage Area NCSR chapter and corresponding modification to SAR Chapter 7

➢ Safety Guide for Design Modifications:

Specific not compulsory guidelines for the regulation of the design modifications process were recently issued by CSN in the Safety Guide 3.1*"Modifications in Nuclear Fuel Fabrication Installations"*. The implementation in the facility procedures is on-going.

➢ Integrated Safety Assessment (ISA) :

The Integrated Safety Analysis (ISA) required to the facility has been recently completed. For each analysed sequence where fissile material is involved, risk in terms of Criticality Severity Margin has been calculated as a ratio between the Subcriticality Margin (0.98 – k-eff) associated to the nominal conditions and the Subcriticality Margin associated to the analysed sequence.

As a result of this assessment, Items Relied On For Safety (IROFS) have been identified and design modifications and improvements in the process implemented to decrease the potential risk and/or the probability of the sequences analysed, and to increase the safety margins in the facility operation.

IROFS particularly relevant for NCS have been identified in the manufacturing process of green pellets, and specifically in the blending and homogenization nodes where hydrogenated additives are involved:

- ✓ A fractioned dosage of hydrogenated additive has been implemented in blenders and homogenizers, acting as an active engineering control to prevent the occurrence of uniform and non-uniform over-moderation sequences. This topic will be developed in a specific paper in this workshop.

- **INTERIM SPENT FUEL STORAGE INSTALLATIONS (ISFSIS)**

The *Nuclear Safety Council's Safety Safety Instruction IS-29, of 13th October 2010, on safety criteria at spent fuel and high-level radioactive waste storage facilities* is the regulatory national framework for these facilities, with requirements in accordance with *NUREG-1567 "Standard Review Plan for Spent Fuel Dry Storage Facilities"* developing *10CFR.72.124"Criteria for Nuclear Criticality Safety".*

All Spanish ISFSIs are licensed as a design modification of the corresponding NNP: the licensee is authorized to dry store the irradiated fuel in previously licensed casks, within the specific nuclear plant site. The *Nuclear Safety Council Safety Instruction IS-20, of January 28th 2009, establishing safety requirements relating to spent fuel storage casks* regulates the storage casks licensing process in accordance with *NUREG-1536 "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility"*

Table 2
Interim Spent Fuel Storage Installations in Spain

NPP	Fuel type	Cask model	Licensed capacity: Casks/fuel assemblies	Cask type	1st loading
Trillo	PWR	DPT (ENSA)	80/1680	Dual purpose metal cask	2002
José Cabrera	PWR	HI-STORM 100Z (Holtec)	12/384	Metal canister in concrete overpack	2009
Ascó	PWR	HI-STORM 100 (Holtec)	32/1024	Metal canister in concrete overpack	2013
Garoña	BWR	ENUN52B (ENSA)	32/1664	Dual purpose metal cask	Under licensing

There are three ISFSIs in operation and one more, the first one for BWR fuel, in the licensing process and two coexisting cask concepts:

- dual purpose metallic casks
- multipurpose metallic canisters, operated with different overpacks for storage (concrete) or transportation (metal) operations

In all cases, storage and transportation authorizations are mandatory before the first fuel loading in the canister is performed.

Figure 3. ISFSI at Trillo NPP site

A comprehensive and conservative NCS evaluation of the cask loaded with the Design Basis Fuel is required, with conveniently validated analytical methods to support the licensing process by demonstrating that the k-effective including all biases and uncertainties at a 95 percent probability, 95 percent confidence level, should not exceed 0,95 under all normal, off-normal and accident conditions. No specific criticality safety condition is derived from the cask NCS evaluation for the cask storage in the ISFSI, neither for the subsequent cask transportation, but those related to the allowed Design Basis Fuel in the cask. Fuel requirements for the Loading Map of the cask are contained in the NPP Technical Specifications.

3.1 INTERIM SPENT FUEL STORAGE INSTALLATION AT TRILLO NPP SITE

Trillo NPP (1987, 1066 Mwe) is a PWR German design plant with the spent fuel pool located inside the containment building, what leads to a reduced spent fuel storage capacity. The on-site ISFSI was licensed in 2002 to allow the storage of up to 80 dual purpose (storage and transportation) casks located in a dedicated building. Cask design (DPT) is similar to that of the STC cask designed by NAC (USA).

Figure 4. ISFSI at José Cabrera NPP site

NCS cask safety cases, both storage and transportation, assume fresh fuel and no damaged fuel is allowed in any of the 21 available positions in the DPT cask.

Neutron absorber credit is taken in NCS cask models. The most reactive safety case is the transportation accident condition, where total flooding is assumed within the cask and within the fuel rod (flooded gap).

3.2 INTERIM SPENT FUEL STORAGE INSTALLATION AT JOSÉ CABRERA NPP SITE

The ISFSI at José Cabrera NPP site (150 Mwe, 1968-2006), was licensed in 2009 for the storage of the full inventory of spent fuel pool after permanent shut down in April 2006.

Canister-type multipurpose casks (MPCs) were selected, located in an open-air storage area within the plant site. The spent fuel is loaded in 12 MPCs in its concrete storage casks, with a capacity of 32 irradiated fuel assemblies each. The system designed by Holtec (USA) use different overpacks to store (HI-STORM 100Z) or transport (HI-STAR 100) the MPC.

Cask Safety Analysis Reports for Storage and Transportation include specific modelling and criticality calculations for the cask loaded with the Design Basis 14x14 PWR Fuel. Damaged fuel is allowed in eight peripheral positions of the cask, and specific conservative models (bare rods, loss of geometry) are analyzed with reactivity penalizing results for k-effective. Credit to neutron absorbers is taken up to 90% of its B-10 concentration, what requires comprehensive qualification tests to verify the presence and uniformity of the absorber.

Figure 5. ISFSI at Ascó NPP site

Fresh fuel assumption is taken for the storage case, with credit to the spent fuel pool boron concentration during the fuel loading, and burnup credit (BUC) is taken to comply with the NCS acceptance criteria for the transportation safety case. Loading curves (enrichment/ minimum burnup) for damaged and undamaged fuel are calculated applying a BUC methodology which assumes bounding uniform burnup axial profiles for the 14x14 PWR fuel, fuel assemblies exposed to control rod insertion during full power operation and credit only to major actinides (9 isotopes). A conservative misload analysis was required, what exempts from the requirement to perform out-of-core burnup measurements, in accordance with ISG-8 "*Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks"* revision 3.

3.3 INTERIM SPENT FUEL STORAGE INSTALLATION AT ASCÓ NPP SITE

Ascó NPP (Unit-I: 1032 Mwe, 1982; Unit-II: 1027 Mwe, 1985) is a two units Westinghouse-PWR 17x17 fuel design plant whose spent fuel pool capacity was recently reached. An on-site Interim Spent Fuel Storage Installation was licensed in 2013 to allow the storage of up to 32 casks (32 fuel assemblies capacity each) in an open-air storage grounded on two concrete slabs, one for each NPP unit.

The same MPC system from Holtec-USA as for José Cabrera was selected for the 17x17 PWR Ascó fuel, with HI-STORM 100 for dry storage and HI-STAR 100 for transportation overpacks. The cask design allows the load of up to eight damaged fuel assemblies, what has been evaluated through specific conservative models, resulting in a k-effective penalty.

Neutron absorbers credit is taken up to 90% of its B-10 concentration, what requires specific qualification tests of the absorber material.

Figure 6. Cask transfer at Ascó NPP site

The NCS evaluation contained in the cask SAR, assumes fresh fuel for the storage safety case, taking credit from the spent fuel pool boron concentration during the cask fuel loading operations, required in the NPP Technical Specifications.

BUC is taken to comply with the NCS acceptance criteria for the transportation safety case. The BUC applied methodology assumes specific bounding axial burnup distributions conservatively determined for the 17x17 PWR fuel irradiated in Ascó, four different configurations in respect to presence of inserts (such as burnable poison or control rods) during depletion, credit to major and minor actinides and to a short number of fission products (up to 25 isotopes), supported by the corresponding validation process. A conservative misload analysis was required, what allows the exemption from the requirement to perform out-of-coreburnup measurements, in accordance with ISG-8 revision 3.

Loading curves (enrichment/burnup) for damaged and undamaged fuel in each of the four in-reactor insert configurations are calculated. NCS evaluation ensures that fuel bounded by these curves (minimum burnup for a given enrichment) do not exceed 0.95 k-eff under all normal, off-normal and accident conditions.

3.4 INTERIM SPENT FUEL STORAGE INSTALLATION AT GAROÑA NPP SITE

A new application for a Spanish design dual purpose cask for 8x8 BWR fuel, and for the corresponding ISFSI of Garoña NPP (BWR-3 reactor, 466 Mwe, 1970), is being licensed. It will be the first interim spent fuel storage facility licensing process for BWR fuel in Spain. In the first stage of this application, which scope no credit is taken from burnup neither from Gadolinium.

The ENUN52B metallic dual purpose cask system from ENSA (Spain) was selected for Garoña ISFSI, with 52 fuel bundles capacity and no option for damage fuel loading. The licensing process of the cask design has recently finished, with specific authorization for storage and certificate of compliance for transportation.

- **CENTRALIZED TEMPORARY STORAGE**

The Spanish General Radioactive Waste Plan establishes the Government Policy about Radioactive Waste Management and Nuclear Installations Decommissioning. High Level Waste and Nuclear Spent Fuel management priority is the interim storage at a centralized storage facility. Generic design was approved by CSN and Villar de Cañas site was approved by the Government. The facility is under construction and it is expected to be in operation in 2017.

At the CTS Facility the fuel will be unloaded from the transportation casks coming from ISFSIs in the country, and loaded in a different canister specific to the facility, canister design is being finalized. A bounding NCS analysis for the different fuel designs stored (PWR 17x17, 16x16, 14x14, BWR 8x8, 9x9, 10x10) will be required as well as a bounding NCS analysis for the different loaded casks (DPT, HI-STAR 100, ENUN52B…) arriving to the CTS Interim Loaded Cask Storage.

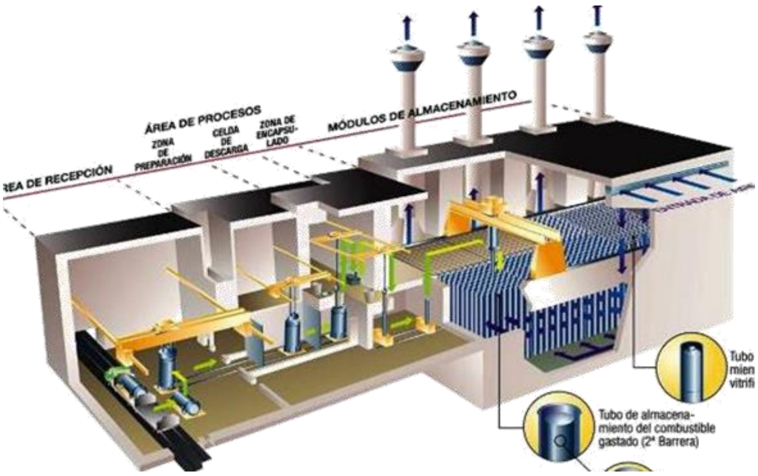

Figure 7. Conceptual design of the CTS main process

A Spent Fuel and Radioactive Waste Laboratory is also projected in the CTS facility, provided with several concrete and metallic cells, as well as glove boxes to perform studies on spent fuel and other wastes in support of R&D objectives for long term storage and disposal. An NCS analysis of this laboratory will be required.

STATUS OF FRENCH REGULATIONS CONCERNING NUCLEAR CRITICALITY SAFETY

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ABSTRACT

Hitherto, in France, licensees used to demonstrate compliance with Fundamental Safety Rule I.3.c. This Rule sets out the principles for demonstrating Nuclear Criticality Safety in all nuclear facilities (excluding reactors). Although there is a good level of consensus between the licensees and the Regulator, this Rule is now somewhat dated (it came into force in 1984) and does not cover all aspects. The French Nuclear Safety Authority (ASN) used the 2006 Law on Transparency and Nuclear Safety as the catalyst for a thorough revision and updating of its regulatory documentation. Thus, in 2010, ASN decided to set up a “criticality working group” involving ASN, the licensees (AREVA, CEA, EDF…) and the Technical Safety Organization (IRSN). This group was tasked with drafting a regulatory text (a “resolution”) and a criticality guide. The objectives were to update Rule I.3.c, clearly defining the scope for reactors and transport packages of fissile material, to further develop areas not covered in the existing rule and to take account of the feedback and lessons learned from past events.

Thus, this paper presents the status of French regulations concerning Nuclear Criticality Safety (NCS). Firstly, it focuses on the existing Rule I.3.c and then explains how this will be modified by the forthcoming improvements to the resolution and the criticality guide. It in particular addresses the scope of the new regulations (concerning reactors and the transport of fissile material), how the principle of defence in depth is to be applied to criticality safety, the double contingency principle, control modes, the reference fissile medium, the organization, the role of the Criticality Engineer, admissibility criteria, validation, etc.

1. INTRODUCTION

Fundamental Safety Rule I.3.c **[1]**, dealing with Criticality Safety in nuclear facilities, was issued in October 1984. This very synthetic text reflected the know-how of the time and, since then, has been the starting point for every criticality safety demonstration in France. Although this Rule has rarely been called into question, it became clear that it needed to be revised if it was to conform to the new nuclear safety regulatory context.

Thus, the French Nuclear Safety Authority for “civil” facilities (ASN) decided in 2010 to create a working group with two major objectives:

- To draft a regulatory text, that is a “resolution” **[2]**, for Nuclear Criticality Safety;
- To draft a nuclear criticality safety guide.

This paper presents the status of Nuclear Criticality Safety in France, from the beginning of the nuclear activities to the present day, the organization of the Criticality working group, the main issues of the Criticality Safety resolution and some of the issues addressed by the Criticality Safety guide.

2. THE REGULATORY SITUATION PRIOR TO RECENT IMPROVEMENTS

Until recently, regulatory documentation in France related to criticality safety in nuclear facilities has been relatively succinct.

On one hand, there was Fundamental Safety Rule I.3.c, which is a guide of just 4 pages, written by Criticality experts and the French Authority in October 1984. Although the Rule was not a regulatory text, it was widely followed by French licensees. Among other principles, the Rule stipulates the double contingency principle, defines the main requirements associated with criticality control modes (mass, geometry, concentration, moderation and/or neutron absorber control) and the reference fissile medium. These requirements are included in the new criticality resolution and are described later on.

On the other hand, from a regulatory point of view, there was only one article on criticality safety. Article 45 of the "31st December 1999" Act, dedicated to the nuclear safety of facilities, was an exact copy of the double contingency principle as written in Fundamental Safety Rule I.3.c. Otherwise, the other regulatory texts concerning facilities contained no further provisions concerning criticality safety.

With regard to nuclear facilities for defence activities, the regulations differ from those applicable to civil facilities. Today, there is an Act dating from 2007 which is the counterpart of the "31st December 1999" Act for civil facilities. According to the 2007 Act, the licensees have to apply the double contingency principle, as for the "31st December 1999" Act. In practice, the licensees of "defence" facilities had been applying Fundamental Safety Rule I.3.c since long before. For the time being, there are no plans to improve criticality safety in the regulations for "defence" facilities.

3. REVISION OF FRENCH NUCLEAR SAFETY REGULATIONS

3.1 National texts

The legal system applicable to nuclear facilities was revised in depth by Act 2006-686 of 13th June 2006 on transparency and security in the nuclear field, called the "TSN" Act and its application decrees, in particular Decree 2007-1557 of 2nd November 2007 concerning nuclear facilities and the regulation of nuclear safety in the transport of radioactive substances, called the "Nuclear facilities Procedures" Decree.

The provisions of the TSN Act have been codified in the Environment Code and underpin the Nuclear facilities licensing and regulation system.

The Nuclear facilities Procedures Decree defines the framework in which Nuclear facilities procedures are carried out and covers the entire lifecycle of a Nuclear facility, from its authorization decree to commissioning, final shutdown and decommissioning. Finally, it explains the relations between the Minister responsible for nuclear safety and ASN in the field of Nuclear facilities safety.

The decree clarifies the applicable procedures for adoption of the general regulations and for issuing individual resolutions concerning Nuclear facilities. It defines how the Act is implemented with regard to inspections and administrative or criminal sanctions. Finally, it defines the particular conditions for application of certain regimes within the perimeters of the Nuclear facilities.

3.2 General technical regulations

The general technical regulations provided for by the Environment Code comprise all the general texts laying down the technical rules concerning nuclear safety, whether binding (ministerial orders and ASN statutory resolutions) or non-binding (circulars, basic safety rules, ASN guides).

Following publication of the TSN Act of 13th June 2006, ASN initiated an overhaul of the general regulations concerning Nuclear facilities.

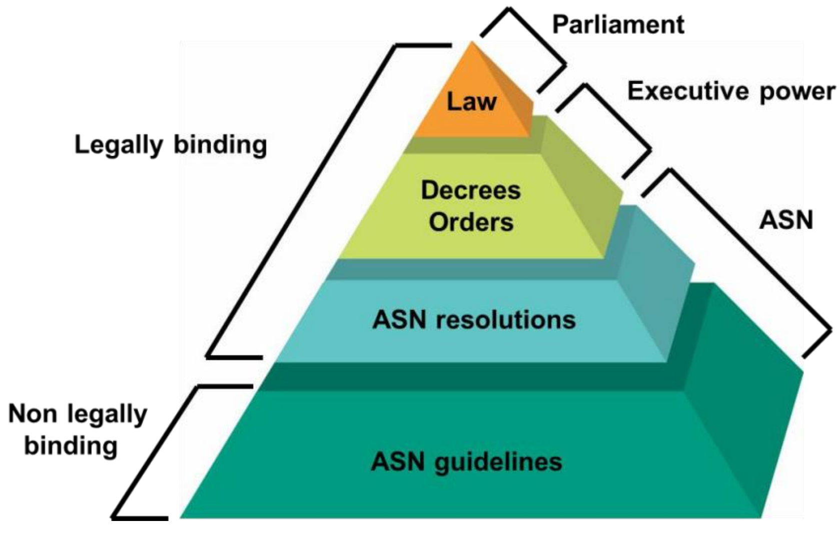
The entry into force of the Order of 7th February 2012 setting the general rules concerning Nuclear facilities, known as the "Nuclear facilities" Order, is a major step forward in the overhaul of Nuclear facilities regulations with regard to the majority of its provisions. It defines the essential requirements applicable to Nuclear facilities for protection of the interests listed in the Act: public safety, health and sanitary conditions, protection of nature and the environment.

It significantly reinforces the regulatory framework applicable to Nuclear facilities because, in the light of operating experience feedback, it clarifies numerous requirements from previous orders and provides a legal foundation for several ASN requirements, for example those formulated further to the analysis of the stress tests and imposed on the licensees following the Fukushima accident.

In particular, it addresses the following subjects: organization and responsibility, demonstration of nuclear safety and emergency situation preparedness and management.

Concerning the criticality risk, it specifies that "to control nuclear chain reactions, the licensee demonstrates that the provisions made are able to prevent the untimely occurrence of criticality".

In order to clarify the Order of 7th February 2012, ASN defined a program of fifteen statutory resolutions which are the subject of public consultation and require approval by the Minister in charge of nuclear safety. Among these resolutions, one was related to the control of the criticality risk within Nuclear facilities. The following figure shows the regulatory "pyramid".



3.3 Criticality safety resolution and guide

In 2010, the French Nuclear Safety Authority (ASN) decided to create a working group with two major objectives:

- To draft a regulatory text, that is a "resolution", for nuclear criticality safety;
- To draft a nuclear criticality safety guide.

This “criticality working group” acted under the authority of ASN, but technical leadership was entrusted to the Technical Safety Organization (IRSN). Besides ASN and IRSN, this group comprised a number of representatives from the licensees (AREVA, CEA, EDF…), primarily criticality experts. The objectives were to update Rule I.3.c, clearly defining the scope for reactors and fissile material transport packages, to further develop areas not covered in the existing rule (criticality accidents management and mitigation, organization of licensees for control of the criticality risk, etc.) and to take account of the feedback and lessons learned from past events (notably the Tokai Mura accident, but also various incidents which had occurred in France in the past).

The criticality resolution was drawn up by a sub-group consisting of representatives from ASN and IRSN. The process to develop the regulatory text involved several ASN in-house reviews as well as consultation of the public and the licensees and therefore lasted about 3 years.

The criticality guide drafting process is still ongoing. The objective of this guide is to clarify, specify and detail the legally binding criticality provisions and requirements which appear in the resolution. It involves several sub-groups, comprising criticality experts from ASN, IRSN and the various licensees, who were assigned to the following technical fields:

- general provisions to prevent criticality
- mass control mode
- design criteria and codes qualification
- licensee organization, operator training
- decommissioning operations
- criticality accidents
- instrumentation
- transportation casks scope
- reactors scope.

4. MAIN ISSUES OF THE CRITICALITY SAFETY RESOLUTION

The Criticality Safety resolution (2014-DC-0462) was signed on 7th October 2014 and will come into force on July 1st, 2015.

4.1 Scope of the resolution

In the past, the scope of RFS I.3.c had not been clearly defined with regard to reactors and fissile material packages within nuclear facilities.

Although reactors were excluded from the scope of RFS I.3.c, it was commonly admitted that this exclusion concerned the reactor cores, rather than the operations outside the cores. In any case, the double contingency principle applied to the reactors, since it was required by the “31st December 1999” Act. One of the many challenges faced by the “Criticality Working Group” was also to clearly define to what extent the criticality decision should cover the reactors, in particularly the various core states. Thus, throughout the four years of work on the guide and the resolution, extensive discussions took place regarding the possibility of including core states within the scope, until the fission chain reactions monitoring system and reactivity control resources could be deployed. Indeed, before these systems are deployed, the core must not become inadvertently upper-critical. Thus, the debate focused on whether a core in a shutdown state and without any reactivity control means could be demonstrated as being safe by applying the Criticality resolution. The discussions brought to light no major technical obstacle to implementing the Criticality resolution for these shutdown states. Nevertheless, the change in the approach to these core states was considered to be too important and needed further discussion. Thus, ASN decided to include core loading in the scope of the resolution, but not the shutdown states, in particular the states for which the core is completed even if no reactivity control means are available. The possibility of extending this scope could be reviewed later.

For the fissile material packages, some licensees were in the past over-reliant on a certificate of approval to justify the subcriticality of all the operations involving these packages. It should be recalled that the criticality safety demonstration for transport is based on compliance with the IAEA regulations for the safe transport of radioactive material. For transportation, the “normal” and “accident conditions of transport”, in isolation and in an array of packages must be examined, as defined by the regulations. Although these transport conditions are very conservative, the conditions encountered in the facilities might be not covered by the transport demonstration. For instance, a package may be handled without its shock absorbers, at a height greater than 9 m, whereas the IAEA regulations require a 9 m drop test in transport conditions, which may include the shock absorbers. In addition, the neutron interactions with other fissile units may be not covered by the transport demonstration.… Finally and probably most importantly, the loading or unloading of the package and the associated failures (misloading, etc.), are specific to the facility, and cannot be covered by the transport regulations, from a criticality point of view. For all these reasons, the operations involving packages of fissile material in plants or storage facilities are within the scope of the Criticality resolution. Nevertheless, the resolution enables the criticality demonstrations of the transport case to be used, provided that the licensee can prove that they are applicable to its facility.

Deep geological disposal (after closure) is also excluded from the scope since one cannot expect to have a criticality safety demonstration in the same way as for other facilities, and since the consequences are not the same after closure.

Finally, facilities for which criticality is physically impossible (for instance, owing to the composition and the chemical form of the fissile material) are excluded.

4.2 Defence in depth

For many years, the two objectives of the defence in depth principle, that is to prevent accidents and, if prevention fails, to limit their consequences, have been clearly identified in the Criticality Safety approach.

However, the definition of different levels regarding the Criticality risk is a far more delicate matter. The reasons for this include the following:

- The 5-level concept of defence in depth was to a large extent conceived for nuclear facilities with high potential energy and large quantities of radioactive materials, in order to protect the public, the workers and the environment against severe accidents with major off-site radiological consequences. When one reviews past criticality accidents, no significant release of radiological material is likely to occur in the event of a criticality accident (provided no other failure occurs). The main (lethal) danger is the neutrons and □ rays that could be received by the workers inside the facility, in the vicinity of the accident;
- For most facilities, a critical excursion would be a sudden phenomenon without precursors (the conditions leading to a critical situation may be more or less progressive, and may have some precursors, but not the chain reaction itself), and there is no means of “controlling” the first moments of a criticality accident (the consequences of which are already lethal in the immediate vicinity).

In the French Nuclear Facilities Act of 2012, the regulator defines the first four levels of defence in depth for Nuclear safety (the fifth level, which implies an off-site emergency response, is not in the Act since it is not the responsibility of the licensee, but of the public authorities). These four levels are:

- To prevent incidents, in particular by taking safety margins and using conservative assumptions,

- To detect any incident and to take measures to avoid any accident, and to restore the facility to a normal condition, or at least to a safe state,
- To control accidents, in order to restore the facility to a safe state,
- To manage accidents in order to limit the consequences.

The application of the first, second and fourth bullets entailed no particular difficulty in terms of Criticality Safety. However, it became apparent that the third bullet could not be applied directly to a criticality accident since a critical excursion is not a progressive phenomenon as underlined above. Thus, the initiation of a criticality accident cannot be controlled, for example in the same way as an outbreak of fire or a Loss Of Coolant Accident (LOCA) could be. Nevertheless, although there is strictly speaking no corresponding level for the criticality risk, the resolution makes provision for the prevention of a criticality risk in the event of any (non-criticality) accident taken into account in the nuclear safety demonstration. The idea behind this provision is that if a criticality accident were to occur during an accident (for instance, a fire), it could make it very difficult to restore the facility to a safe state (for instance, the fire could develop and lead to significant releases).

In short, the defence in depth principle has been applied to criticality safety, as follows:

- To prevent the criticality risk,
- To prevent any anomaly likely to jeopardize the control of criticality safety, and to restore the facility to a normal condition, or at least to a safe state,
- To limit the consequences of a criticality accident, when a conceivable accumulation of anomalies can lead to a criticality accident, and if it can bring significant benefits in terms of protection of people or the environment. The various means of limiting these consequences are:
  - Detection and alarm systems,
  - Mitigation means and procedures,
  - Appropriate radiological protection systems, allowing possible off-site intervention,
  - Procedures and means of evacuation and sheltering of the people present on the site.

Finally, there is provision for the prevention of a criticality risk in the event of any (non-criticality) accident taken into account in the nuclear safety demonstration.

4.3 Double contingency principle

To demonstrate compliance with the preventive aspect of defence in depth, the double contingency principle remains the required approach.

According to the resolution, a licensee shall demonstrate compliance with the double contingency principle, that is:

- A criticality accident shall in no case result from one single anomaly;
- If a criticality accident can result from the simultaneous appearance of two anomalies, it is then demonstrated that:
  - The two anomalies are independent;
  - The probability of occurrence of each of the two anomalies is sufficiently low;
  - Each anomaly is highlighted using appropriate and reliable means, allowing repair or implementation of compensatory measures in good time.

An anomaly is defined as an initiating event (i.e., internal failure, internal or external hazard such as earthquake, flooding, etc.) or a deviation from procedures.

In some cases, strict compliance with the double contingency principle is impossible. If this is so demonstrated, then it shall be proven that the technical and organizational provisions are sufficient to make accident scenarios extremely improbable. For instance, bridge crane may have unique components, which failure could result in a load drop. If criticality cannot be excluded in case of load drop, this situation does not comply with the double contingency principle, and the reliability of the bridge crane should be as high as possible to prevent the load drop risk.

To some extent, the implementation of the double contingency approach is both deterministic and "risk- informed" in its application in France.

The first step of the approach is in fact based on a choice of scenarios, each one corresponding to an anomaly as defined previously. Not all anomalies or scenarios are considered and "incredible" scenarios or accident situations not considered in the safety demonstration are, if relevant, rejected. In this context, "incredible" does not mean zero probability, although the experts agreed that the probability of the rejected scenario is low enough for the anomaly to be disregarded. In the case of accident situations beyond design basis, the cost of preventing the criticality risk would be very high or in any case too high for the limited benefit, given the probability of the situation. In addition, for the credible scenarios considered in the demonstration, the safety margin may be adjusted according to the degree of credibility. This is why this first step can be referred to as a "risk-informed" approach.

The second step of the approach is purely deterministic. The subcriticality of any system for all credible scenarios considered must be demonstrated. If this cannot be done, additional provisions shall be considered, in order to comply with the double contingency principle and rule out unacceptable scenarios. A third step shall consider whether there is any possibility of combining two anomalies leading to a criticality accident. If so, according to the double contingency principle, each anomaly has to be independent, have a sufficiently low probability of occurrence and be rapidly detectable. The second criterion (low probability of occurrence) is by far the most difficult and is based to a large extent on an expert assessment.

4.4 Criticality control modes and Reference Fissile Medium

The control of subcriticality is based on the limitation of physical and/or chemical parameters. The resolution thus defines five Criticality control modes that correspond to the parameters to which a limit applies:

- mass of fissile material,
- geometry,
- fissile concentration or content in given media considered to be homogeneous,
- moderator quantity or content,
- neutron poisoning.

These parameters are obviously not the only ones that take part in controlling sub-criticality. Some parameters are therefore related to the characteristics of the fissile material, such as the isotopic composition or enrichment, the specific gravity, the nature of the moderator and so on. Since these parameters are related to the fissile material, the resolution defines a Reference Fissile Medium that takes account of these parameters, rather than other control modes. A Reference Fissile Medium is a bounding medium for the fissile material handled in a unit and leads to the most restrictive limits for the chosen control mode(s). It is also important to underline the important connection between the choice of the control mode and the choice of the reference fissile medium.

In this process, account must also be taken of the environment of the fissile unit, that is the reflection conditions (concrete walls, etc.) and the distance from other fissile units (neutron interactions).

4.5 Acceptability criterion

There is no numerical value for the acceptability criterion. The resolution simply requires the effective multiplication factor (Keff), including all calculation uncertainties, to be lower than 1 with a sufficient margin. This margin has to be justified on the basis of the Keff sensitivity to studied parameters, of the studied configuration (normal or abnormal) and of the calculation assumptions.

4.6 Criticality engineers

The licensee shall set up a Criticality safety organization that provides “criticality engineers” with specific skills. The “criticality engineers” shall be expected to be able:

- To give technical advice before any modification or intervention which may have a criticality impact;
- To provide the management team with recommendations, even in the event of an emergency response;
- To participate in training of the staff;
- To participate in integrating feedback.

The Criticality engineers do not report directly to the management in charge of operating the facility.

They can rely on members of the operating team with Criticality skills. These skills are not of the same level as those of the Criticality engineers, who may provide these persons with technical support.

4.7 Training

The resolution requires periodic training of any person (operator, manager, criticality staff) involved in operations with fissile material. The training shall be appropriate to the facility and to the operations.

Training shall be renewed if there is a significant modification, from a criticality point of view, or in the event of a new assignment.

5. ISSUES OF THE CRITICALITY SAFETY GUIDE

The guide will not be legally binding. Its function is to explain and to develop the issues raised by the resolution and by other regulatory texts, as well as to validate the calculation tools.

The guide is still at the draft stage and requires validation. It is nonetheless the result of the discussions of the “criticality working group”.

5.1. Acceptability criteria and validation

In the same way as the resolution, the guide does not give any numerical value for the acceptability criterion.

The criterion is however expressed by means of these two formulas.

$$\mathbf{Keff + n\sigma \leq 1 - \Delta Kbias - \Delta Kms}$$

or

$$\mathbf{Keff + n\sigma + \Delta Kbias \leq 1 - \Delta Kms}$$

ΔKms, positive value corresponding to the safety margin
ΔKbias, positive value (possibly 0) corresponding to the validation bias
σ corresponding to the standard deviation for Mont Carlo codes
n defining the confidence interval for the calculated Keff.

ΔKms may be modulated depending on the studied scenario, given the operational margin assumptions considered in the calculation model. In practice, ΔKms is generally equal to 0.05 for normal configurations, and varies between 0.03 and 0.02 for abnormal configuration.

ΔKbias results from the validation analysis. The standard approach is to select representative benchmarks and to derive ΔKbias from the C-E discrepancies (calculated Keff – experimental Keff). If there is no representative benchmark, a standard bias must be proposed.

In France, n is generally equal to 3, which corresponds to a confidence interval of 99.7 %.

5.3 Decommissioning

Decommissioning operations often lead to new assumptions being formulated and new demonstrations, which are different from those of the operating period. Criticality control modes may be also different and may evolve during the various stages of decommissioning.

Feedback from decommissioning operations shows how important it is to be fully familiar with what happened during the operating period (type and quantity of fissile materials, operating events and so on). Moreover, feedback also shows that the residual mass of fissile material can be greater than expected. A prudent and graduated approach is thus highly recommended.

5.4 Instrumentation

All kinds of instrumentation (simple or sophisticated) are used for criticality safety purpose: scales, nuclear measurement systems, chemical analyses, etc. Depending on the importance of the instrumentation for safety, it must be validated for its field of application. Thus, the guide in particular recommends documenting:

- The description of the instrumentation and, as necessary, its surroundings (important for nuclear measurements),
- The field of application and the validation for this field,
- The measurement uncertainties,
- The calibration procedure,
- The periodic tests and controls.

This documentation is commensurate with the “sophistication” of the instrumentation.

5.5 Criticality accident

The last level of defence in depth entails limiting the consequences of a criticality accident should prevention fail. The guide will focus more specifically on the following three major issues in the event of a criticality accident:

- detection of the accident,
- evacuation and sheltering of people on the site,
- mitigation of the accident.

Identifying the type of management needed is a very difficult question, as there are so many types of criticality accident, in so many different configurations. Several questions in fact need to be addressed when defining a strategy for limiting the consequences of a criticality accident. These questions are:

- Is a criticality accident with multiple excursions credible? For a single excursion, except for a low transient, there is in fact no benefit to be gained from rapid detection and evacuation of people.

- How "credible" is a criticality accident?
- Where are the most credible locations of a criticality accident? Are the detection systems well located? Even for a "minimum" criticality accident?
- What would be the radiological consequences of an accident? The evacuation and sheltering procedures will depend on the answer. No radiological consequences could mean no evacuation and no sheltering procedure.
- How could the accident be mitigated? Are neutron absorbers available on the site?

The guide will not itself answer all these questions, but will encourage the licensees to ask themselves the right questions and to help them to answer them.

CONCLUSIONS

Recent developments in the French regulations have the particular merit of enshrining in law what was previously widespread practice, while at the same time accepting a degree of flexibility in application of the rules.

The new resolution and the forthcoming guide sought to clarify the scope and reposition the principles of criticality safety within the overall nuclear safety demonstration. These two documents should help achieve a clearer understanding of criticality safety in France.

Finally, the possibility to extend the scope of the resolution to more shutdown states of the nuclear cores has been written in the resolution and could be reconsidered in a near future.

ACKNOWLEDGMENTS

The authors would like to thank all the experts who took part in the French Criticality Working Group.

REFERENCES

1. Fundamental Safety Rule I.3.c, October 1984
2. ASN resolution 2014-DC-0462 of 7th October 2014 concerning the management of the criticality risk in nuclear facilities

The DOE Criticality Safety Support Group – A Retrospective Perspective

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ABSTRACT

The Department of Energy (DOE) Nuclear Criticality Safety Support Group (CSSG) came into being as a response to the DNFSB recommendation 1997-2 "*Criticality Safety*" which dealt with the continuation of criticality safety at defense nuclear facilities in the Department of Energy Enterprise. The DNFSB was concerned over the lack of capability management of practical experience pertinent to avoiding a criticality accident in non-reactor environments. One of the specific recommendations of 1997-2 was to "*Identify a core group of criticality experts experienced in the theoretical and experimental aspects of neutron chain reactions to advise on the above steps and assist in resolving future technical issues*". The CSSG, a group of 10 recognized experts in criticality safety, was chartered in late 1997to address the recommendation. Members of the CSSG are drawn from DOE employees and contractor staff to provide advice and technical support to help meet the criticality safety needs of DOE missions, including stockpile stewardship, materials stabilization, transportation, storage, facilities lifecycle (design through decommissioning), and waste disposal.

The CSSG is an integral part of the DOE Nuclear Criticality Safety Program (NCSP) developed to maintain and enhance the operational and technical criticality safety expertise and capability within the Department of Energy Enterprise. This paper outlines the history, purpose and continuing contribution of the CSSG as well as providing an understanding of the interfaces between the DOE CSSG, the DOE Criticality Safety Coordinating Team (CSCT), the ANS Nuclear Criticality Safety Division and the EFCOG Criticality Safety Subgroup.

Key Words: Criticality Safety, Criticality Safety Support Group, Department of Energy, Nuclear Criticality Safety Program

1. INTRODUCTION

More than fifteen years ago, in response to concerns raised by the Defense Nuclear Facilities Safety Board (DNFSB), the Department of Energy (DOE) established the Criticality Safety Support Group to serve as a technical advisory group. The group, whose purpose is to provide technical guidance to the DOE in the development and maintenance of competency in the field of nuclear criticality safety, has remained active and engaged throughout this 15+ year period. The CSSG has provided significant service in four distinct areas: development of competency within the criticality safety discipline, driving consistency in regulations and the practice of criticality safety, technical support to criticality safety programs, and programmatic support to the DOE's Nuclear Criticality Safety Program (NCSP). Over the years the CSSG has tackled some of the most pressing criticality safety issues, always with a mind toward ensuring DOE resources are used wisely and are providing real improvements in criticality safety “on-the–floor” operations.

2. DISCUSSION

A brief review of the history of the CSSG formation, accomplishments in each of the four service areas, interfaces between the CSSG and other groups, impacts attributable to the CSSG and the future of the CSSG will all be discussed.

2.1 CSSG History

To understand the state of criticality safety that prompted the formation of the DOE CSSG, one needs to go back to the early 1990s after the Cold War ended. DOE was reducing funding and beginning to close facilities related to the weapons production efforts in order to capitalize on the “peace dividend”. By the early 1990s all of the general purpose criticality experiment facilities had been closed with the exception of TA-18 at Los Alamos National Laboratory (LANL). There were serious considerations underway to close TA-18 as well. It was in this atmosphere that the Defense Nuclear Facilities Safety Board (with Dr. Herb Kouts as the primary driver) wrote DNFSB Recommendation 1993-2 (93-2). In this recommendation the DNSFB recognized both the likely detriment from closure of the experiments facility and raised the concern that due to these closures many of the criticality safety engineers would no longer have “hands on” experience with systems at or near the critical state. The DNFSB recommended to DOE that they keep the experimental capability active at LANL to ensure that criticality safety engineers received both the theoretical and experimental experience necessary to be effective in preventing a criticality safety accident. They also reminded DOE that there were still some discrepancies between the existing calculational models and the experimental results that could only be resolved by the capability to perform additional experiments. Based on this recommendation DOE kept the TA-18 facility funded and experiments continued.

By 1997 the DNFSB (again championed by Dr. Kouts) wrote a recommendation on criticality safety competence. In DNFSB recommendation 1997-2 (97-2)the DNFSB again noted the decline in personnel with first-hand experience with systems at or near the critical state, and noted that the large increase of criticality safety engineers were being trained on the job without practical experience and with an over-reliance on criticality computational techniques which led to overly complex analytical models being used. This was, in the DNFSBs perspective, causing reductions in the productivity of several DOE facilities. The DNFSB also expressed a concern that the decades long period without a criticality accident in the United States (see Figure 1) may be leading to a sense of complacency within DOE. In 1997-2 the DNFSB made nine sub-recommendations associated with: coordinating experimental activities, organizing calculations and experiments in criticality safety, developing a way to interpolate and extrapolate between these data, using this information to create guidance and bounding curves, developing a course of instruction in criticality safety which includes “hands on” experiments to serve as a foundation for criticality qualification, and establishing a group of technical experts to advise DOE on the accomplishment of these sub-recommendations and to help resolve future technical issues. This group of technical experts was formed and named the CSSG.

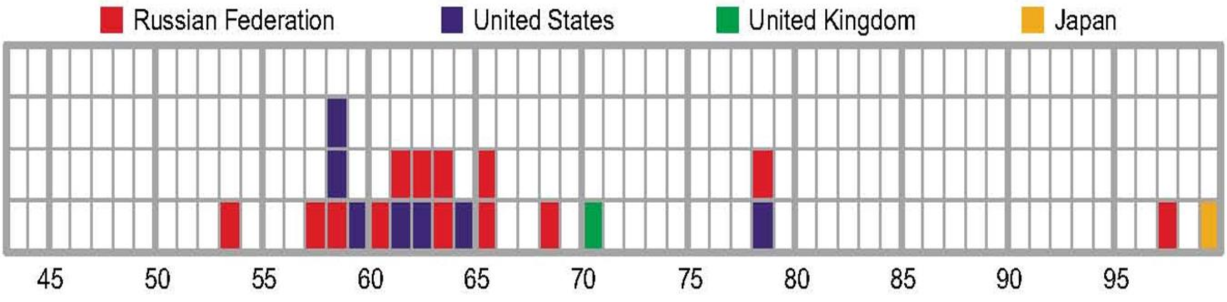

Figure 1. Chronology of process criticality accidents.

The CSSG was chartered in 1997 in direct response to 97-2 sub-recommendation 8 which stated DOE should "*Identify a core group of criticality experts experienced in the theoretical and experimental aspects of neutron chain reactions to advise on the above steps and assist in resolving future technical issues*". Several charter members of the CSSG participated in writing the DOE Implementation Plan in response to 97-2. The current charter of the CSSG can be found on the NCSP website at http://ncsp.llnl.gov/cssg/Revised-CSSG-Charter-August-2008.pdf. Table I shows the10 initial charter members of the DOE CSSG. The members were selected by DOE senior leadership to ensure that the group comprised a strong mix of those with theoretical, experimental and practical experience in the field of criticality safety. The CSSG membership policies and criteria, which are available at http://ncsp.llnl.gov/cssg/CSSG_Membership_Policy_Changes-06.pdf, requires at least15 years of experience in the field of criticality safety, demonstrated leadership and expertise in nuclear criticality safety including in the ANS Nuclear Criticality Safety Division (NCSD) and participation in ANSI/ANS-8 standards development.

Table I. Charter Members of the CSSG in 1997

Name	**Organization**	**Name**	**Organization**
Adolf Garcia, Chair	DOE-NE	Jim Morman, Dep Chair	ANL
Mike Westfall*	ORNL	Robert Wilson	DOE-EM
Tom McLaughlin	LANL	Tom Reilly*	SRS
Calvin Hopper*	ORNL	Rick Anderson*	LANL
Jerry McKamy*^	DOE HQ	Hans Toffer*	Hanford

*currently Emeritus

^currently DOE NCSP Manager

2.2 CSSG Accomplishments

Over the last 15 years the CSSG has fulfilled its charter responsibilities in four specific areas: development of competency within the criticality safety discipline, driving consistency in regulation and the practice of criticality safety, technical support to criticality safety programs, and programmatic support. In addition the CSSG has been instrumental in guiding the advancement of analytical methods for criticality safety, preserving historical criticality safety data and documents, and in developing training materials for criticality safety practitioners.

The CSSG charter explains how the CSSG is tasked with specific activities by the NCSP Manager and how the CSSG responds to those Taskings with Responses. These Responses, starting in 2006, are available on the NCSP website at http://ncsp.llnl.gov/cssgMain.html and are noted in brackets in the following sections. Accomplishments in each of the specific areas are described below.

2.2.1 Increasing Competency in Criticality Safety

As was noted in the DNFSB Recommendation 93-2, one of the most important competencies that the NCSP can provide to the criticality safety engineer is experience with systems at or near the critical state. The CSSG was a strong advocate for the continuation of critical experiments activities at LANL and the re-establishment of critical experiments capabilities at the Device Assembly Facility (DAF) once DOE decided to close TA-18 at LANL. During the planning for the move of the experimental capability the CSSG reviewed the machine capabilities that would be established at the DAF [2005, Tasking not available]. The CSSG also reviewed the criticality safety evaluations for the critical experiments and reviewed the determination of the need for criticality safety accident alarms at the facility [2005-04].

Realizing that this experimental capability, while important for furthering criticality safety research, was also an important learning tool for the criticality safety engineer, the CSSG was also involved in the development of a training course that incorporated this national facility -now called the National Criticality Experiments Research Center (NCERC). The CSSG was integral in assessing the needs, developing the outline, reviewing the content and periodically auditing the NCSP two week hands on critical experiments class [2006-03, 2009-03].

The CSSG has also reviewed the DOE Standards related to DOE Federal Criticality Safety Qualifications (DOE-STD-1173) providing input on the necessary competencies and documentation to ensure federal personnel remain competent in providing oversight of the contractor programs [2009-05].

The CSSG also had a major hand in developing the first ever training and qualification standard for criticality safety engineers DOE-STD-1135-99 which was later superseded by ANSI/ANS-8.26.

2.2.2 Increasing Consistency in the Regulation and Practice of Criticality Safety

Many DOE Orders and Standards as well as consensus standards (e.g., ANSI/ANS-8 series) provide significant latitude in the way that they can be implemented in the field. While this often provides needed flexibility to match implementation to risk, it can also lead to significantly different interpretations of these Orders and Standards at the Site level. The CSSG has provided guidance that can be used to help create consistency in the implementation of DOE Orders and Standards (which in the past were often developed without significant input by the criticality safety community). Several examples of topical areas where the CSSG, often in collaboration with subject matter experts from other disciplines, has provided guidance are discussed below.

- Development of the NCSET Modules available on the NCSP website (1999-current) at http://ncsp.llnl.gov/trainingMain.html
- Guidance and content for the development of Nuclear Criticality Safety Evaluations via DOE-STD-3007 which was authored by the CSSG [2004-2005], and for the upcoming 2015 revision of DOE-STD-3007;
- The proper role of criticality safety in Facility Categorization and recommendations for changes to the DOE-STD-1020 [2010-02];
- The proper balance of risk between seismic design guidance and criticality safety and recommendations for changes to the DOE-STD-1027 [2010-01];
- The proper balance of risk between fire protection and criticality safety [2013-01];
- A process for uniform criticality incident categorization [2009-02]; and
- Guidance for uniform roles and responsibilities for Criticality Safety Committees [2009-01].

In addition the CSSG reviews all DOE Orders and standards involving or tangentially involving criticality safety. These have included:

- CSSG review and comment on DOE Order 420.1B/C [2004, 2011-01];
- CSSG review and comment on DOE-STD-1189 [2007-05];
- CSSG review and comment on DOE-STD-3009 [2011-02, 2013-03-01];
- CSSG review and comment on the NCS Good Practices Guide

2.2.3. Providing Technical Support to Criticality Safety Programs

There are sometimes specific topical areas that manifest themselves at a particular site which either have wide applicability to the rest of the DOE Enterprise, or whose failure could impact the mission accomplishment of DOE. In these cases the CSSG can be brought in to provide guidance and technical assistance. Access to CSSG support is available to any part of DOE/NNSA via request to, and approval from the NCSP Manager. Examples of this technical support are provided below.

- CSSG review of the criticality safety approach used for pre-closure of the Yucca Mountain Site as part of their license application [2006-07];
- CSSG review of WTP and Hanford Tank Farms in regard to plutonium solids issues [2009-06];
- CSSG assessment of the preliminary criticality safety approach for the UPF facility, including reviews of the interaction of criticality and seismic [2011-04];
- CSSG assessments and direct technical support for the LANL criticality safety program [2005-tasking not available, 2011-06, 2013-02, 2014-01]; and
- CSSG review of the approaches used by Y-12 to define the Immediate Evacuation Zone (IEZ) [2007-07].

2.2.4 Providing Programmatic Support to the NCSP

Since the inception of the NCSP and the CSSG, the CSSG has provided guidance to the NCSP in terms of the overall DOE approach to criticality safety. This is reflected by the CSSG review of the NCSP Mission and Vision as well as the NCSP 5 and 10-year plans available from http://ncsp.llnl.gov/ncspMain.html and http://ncsp.llnl.gov/planMain.html, respectively.

In addition the CSSG provides a yearly prioritization of tasks proposed to be performed under the NCSP budget. Each year the CSSG reviews all the current and proposed tasks and provides the NCSP Manager a prioritized list of activities for each of the elements within the NCSP. This prioritization is based on the collective CSSG perception of the best use of the limited funds available to furthering the competencies of the criticality safety discipline within DOE. The CSSG is charged with providing this perception while keeping a balance between experiments, data (new and historical), tools, and training such that the criticality safety professional is best prepared to perform their function.

2.3 CSSG Interfaces With Other Organizations

The CSSG, by design, is a transparent organization. As was noted in section 2.0 the available CSSG Taskings and responses (excluding those identified as OUO / Internal Use) as well as CSSG minutes from meetings (more to be added) are available on the NCSP website. However the CSSG does not just passively post information, it actively engages with other organizations in an effort to provide updates on activities as well as learning of new issues or areas of concern within the discipline.

As an integral part of the NCSP the CSSG has interfaces with all the elements of the NCSP program (nuclear data thru the Nuclear Data Advisory Group, analytical methods, bounding sensitivity and uncertainty, integral experiments, information preservation and training). Interfaces with the other elements of the NCSP occur during yearly meetings of the NCSP to plan upcoming work and report accomplishments (usually held in the Spring) and the yearly meeting to discuss execution of the projects (usually held in the Fall). Some CSSG members are also engaged with, and in some cases are task managers for, activities in these other elements of the NCSP.

In addition the CSSG retains a close coordination with the DOE Criticality Safety Coordinating Team (CSCT) which is comprised of the Federal (Headquarters and Field) responsible entities at each of the DOE sites. This coordination is maintained via a cross pollination of the CSSG with several CSCT (DOE) members (see Table II) and an occasional joint meeting of the two groups. Typically the CSSG Chair or Deputy Chair attends the monthly CSCT teleconferences. This allows the CSSG to remain aware of issues that the individual Site federal oversight engineers may be facing.

Table II. Current Members of the CSSG

Name	Organization	Name	Organization
Fitz Trumble, Chair	URS	David Erickson, Dep Chair	SRNS
Adolf Garcia	DOE-ID	Robert Wilson	DOE-EM
Tom McLaughlin	LANL	Jim Morman	ANL
David Hayes	LANL	Kevin Kimball	Y-12
Dave Heinrichs	LLNL	Mikey Brady-Rapp	PNL

The CSSG also maintains close coordination with the Energy Facility Contractors Owner Group (EFCOG) criticality safety sub-group. This group is comprised of “end-users” of the criticality safety data, tools and training prepared by the NCSP and is made up of NCS managers and engineers from the various DOE Sites. Coordination with this group is via attendance at the EFCOG subgroup teleconferences (monthly) by the CSSG Chair or Deputy Chair as well as attendance at selected EFCOG technical meetings.

As was noted in section 2.1 the CSSG members are also closely involved with the ANS NCSD and are or have been active in the program, executive, education committees as well as serving on a number of the ANS-8 standards writing groups. This engagement with NCSD helps ensure that the CSSG members are aware of the perspectives and approaches used outside of the DOE Enterprise. It also facilitates sharing of information between the CSSG and the non-DOE criticality safety community.

2.4 Impact Attributable to the CSSG

Over the past 15 years, the CSSG has worked closely with the NCSP to develop, maintain, and enhance the practice of criticality safety within DOE by providing guidance on the data, tools, and training used by the criticality safety engineer. The CSSG strives to present information and guidance related to the prevention of criticality accidents in a balanced risk perspective, ensuring that regulations promulgated by the DOE are respectful of the limited resources available to the accomplishment of mission and are providing real improvements to safety. During this first 15 year period of the CSSG, the group has been a strong advocate for “doing the right thing” and not allowing political considerations to outweigh technical considerations. This has resulted in a much stronger application of the graded approach to criticality safety which is one of the fundamental underpinnings of the ANSI/ANS series 8 standards.

The CSSG has also championed the approach that criticality safety differs in no intrinsic way from other safety disciplines. While a criticality safety accident clearly can cause a fatality in a nearby worker, there are very few instances where a criticality accident would cause serious damage to a facility, or would impact co-located workers or the public. By putting criticality accident risk into perspective the CSSG continues to advocate for the regulation of criticality safety to follow rules based primarily on consequence and not political perception.

The CSSG also ensures that the current almost 40 year period (see Figure 1) without a process criticality accident in the United States does not develop into a sense of complacency within the DOE community.

2.5 Future of the CSSG

The CSSG has proven itself an important and integral part of the NCSP over the last 15 years. As the CSSG looks forward it has the staffing, expertise and mission to continue in that role. The CSSG membership criteria help ensure that the CSSG remains the most authoritative body on criticality safety within the DOE Enterprise. The CSSG has multiple Taskings underway in 2015 and these Taskings impact all four of the CSSG service areas. Table II provides the current membership of the CSSG supplemented by the Emeritus members noted in Table I. The Emeritus members continue to provide invaluable advice to the current CSSG members although they are no longer directly funded by the NCSP.

In 2014 the CSSG developed a strategic plan for the organization that is designed to ensure that the group’s capabilities and impacts are well known in the DOE leadership, technical programs, and regulatory writing bodies. The strategic plan is also intended to ensure that the work the CSSG performs is timely and focused on the most pressing needs of the DOE Enterprise.

As new and revised DOE policy and regulation relating to criticality safety or criticality safety’s interaction with other safety and operational disciplines is developed, the CSSG stands by ready to provide guidance and direction. The CSSG is also ready to address new technical issues that may arise in the DOE Enterprise as the future unfolds. A request for access to the CSSG capabilities is available thru contact with the NCSP Manager (Dr. Jerry McKamy) or any member of the CSSG.

3. CONCLUSIONS

Born during a time of concern over the ability of the Department of Energy to maintain capability in the field of criticality safety, the CSSG has provided crucial information, guidance and direction to the NCSP and the DOE Enterprise. The NCSP, with CSSG guidance, has been successful in stabilizing the loss of, and reinvigorating, criticality expertise and capability within the United States and has provided new data, tools, and training that directly support and enhance the practice of criticality accident prevention. The CSSG has helped ensure that the operations within the DOE Enterprise have been conducted safely and has pointed out where the potential for over-regulation could cause resources to be wasted without a commensurate reduction in risk. As the single most authoritative body on criticality safety within the DOE Enterprise, the CSSG is well positioned to continue that support role into the future and plans to continue its interactions with other criticality safety practitioners to ensure two way information flow. For over 15 years, the CSSG has delivered on the expectations of the group set by the DNFSB Recommendations 93-2 and 97-2 and plans to continue to do so into the future as long as the need remains.

ACKNOWLEDGMENTS

The CSSG gratefully acknowledges the funding provided by the Department of Energy which has allowed the group to make a meaningful difference in the conduct of criticality safety within the DOE Enterprise for the last 15 years.

REFERENCES

1. DNFSB Recommendation 93-2
2. DNFSB Recommendation 97-2
3. Nuclear Criticality Safety Program 2014 Mission and Vision

Criticality safety related German industrial standards from the DIN series

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ABSTRACT

In Germany, a number of standards of the DIN series are available which provide guidance on criticality safety on operations with nuclear fuel outside power reactors. These include standards covering general issues like double contingency principle and defence-in-depth, guidance on administrative measures to ensure criticality safety, and the use of burnup credit in wet spent fuel pools and dry transport and storage cask of spent fuel. Another standard provides guidance on criticality safety during the operational phase and the post-closure phase of a final repository for used nuclear fuels, including probabilistic analyses. Additional guidance is available for the validation of numerical tools being applied in criticality safety evaluations. This set of standards was recently reviewed, revised where necessary, and harmonized. A currently relevant topic is the treatment of fuels with very low burnup due to the immediate shutdown of eight German NPPs as a consequence of the Fukushima event.

Keywords: Criticality safety standard, regulatory framework, German industrial standard DIN

Introduction

In Germany, the primary regulation concerning the use of nuclear energy and nuclear materials is defined by the Federal Atomic Energy Act [1]. It is supported by subordinate regulatory frameworks, rules, standards and guidelines of various levels of legally binding character as well as technical detail. The most important of those is the Radiation Protection Ordinance [2], followed by the guidelines of the Reactor Safety Commission (RSK) [3] and the Nuclear Waste Management Commission (ESK) [4]. Also effective are the rules of the Nuclear Safety Standards Commission (Kerntechnischer Ausschuss – KTA) [5] for NPP and research reactors. In terms of transport of radioactive materials, the Act on the Transportation of Dangerous Goods [6], which regulations are based on IAEA TS-R-1/SSR-6 [7], has to be observed. Finally, there are the standards of the German Institute for Standardization (Deutsches Institut für Normung – DIN) [8], often providing the most benefit in practice due to their high level of technical detail.

The set of criticality safety related DIN standards is maintained by the DIN Standards Technical Committee "Criticality Safety" which is a panel of criticality safety experts. They are delegates from the Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety; from the Federal Office for Radiation Protection; from various Federal State Authorities; from technical support organisations and technical control boards; and from industry. Each single standard is being reviewed in a five years cycle, and is being confirmed, revised or withdrawn as considered appropriate by the panel. Most recently, all criticality safety related DIN standards have been reviewed, mainly revised, and are now available as topical final revision.

1. Corresponding author, on behalf of the DIN Standards Technical Committee NA062-07-45AA "Criticality Safety".

Figure 1 shows the network of corresponding criticality safety related DIN standards, embedded within other nuclear safety related DIN standards and further guidelines.

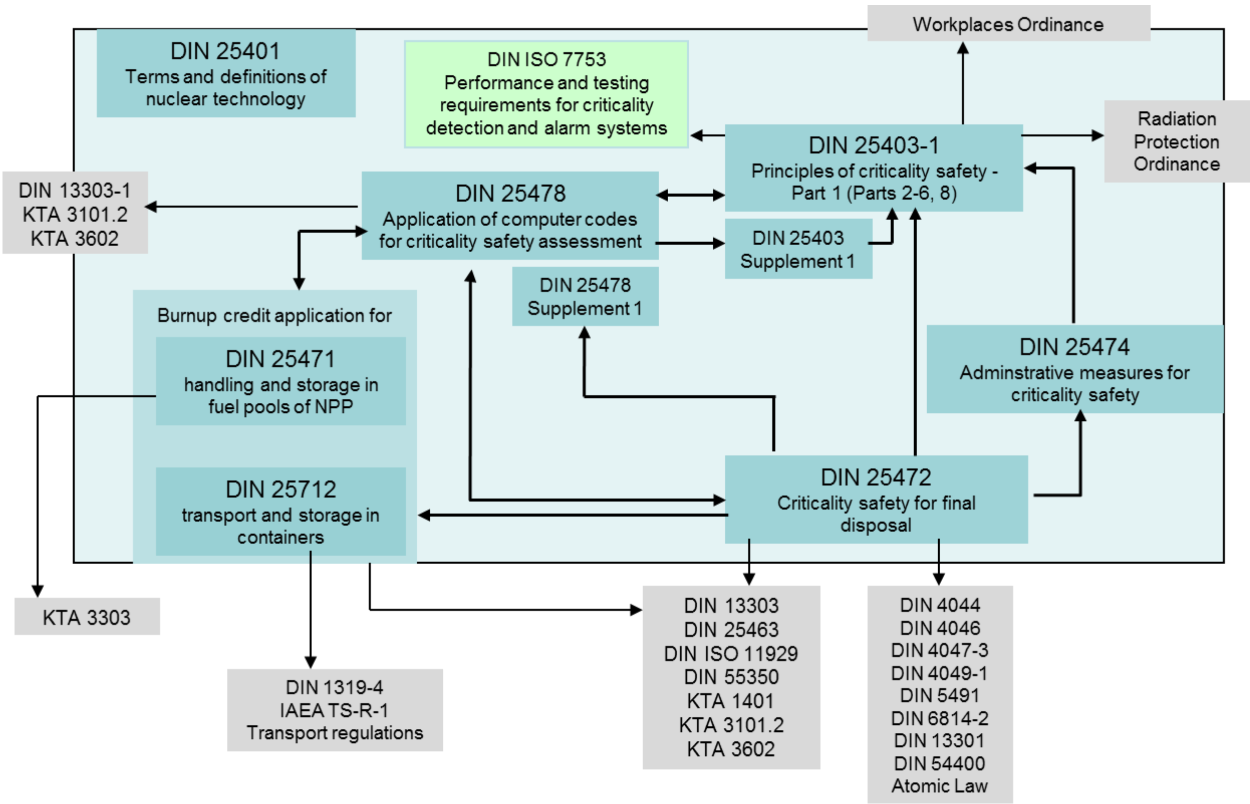

Figure 1: Network of DIN standards on criticality safety

The arrows in the figure depict cross references between the different standards, and other guidelines, like rule KTA 3602 which stipulates regulations for handling and storage safety of spent fuel at NPP fuel pools. Another example is the standard DIN 25463 on decay heat, or the Workplaces Ordinance which is a more general requirement not only assigned to nuclear but also conventional facilities. On the topic of criticality detection and alarm systems, the well-known international ISO 7753 standard was fully adopted and translated to DIN ISO 7753.

The criticality safety related DIN standards, depicted in Figure 1 within blue boxes, are discussed in the following. They apply to the handling of nuclear fuel in nuclear installations except facilities with nuclear reactors. (Note that in this context the cooling pond of a NPP is not considered as an immediate part of the nuclear reactor). With the exception of final disposal, for nuclear facilities in Germany the exclusion of criticality generally should be achieved by deterministic approaches.

Note that for the security of storage and transport of nuclear fuel outside nuclear installations, and for protection against actions by third parties, special further needs arise which are not covered by DIN standards.

DIN 25401 – Terms and definitions of nuclear technology

This standard is available as a final revision as of March 2015. It includes general basic terminology on physical and chemical properties of materials occurring in the framework of nuclear technology, especially concerning the nuclear chain reaction. Furthermore, terminology in the field of reactor design is specified there. Therefore this standard is not only related to criticality safety but related to nuclear safety in general.

DIN 25403 – Criticality safety in processing and handling of fissile materials - Part 1: Principles, and supplement 1

This standard is available as a final revision as of December 2013. It includes technical criteria and recommendations to ensure subcriticality in processing and handling of nuclear fuel. Part 1 of the standard includes general considerations, safety principles and criteria of criticality as its objective. It formulates the single failure criterion, also referred to as double contingency principle, and specifies special criteria and parameter as e.g. enrichment or moderation to be observed, in order to maintain criticality safety under normal and abnormal operation conditions. It also addresses the relevance of safety factors as well as safety measures to be taken. The standard does not address the effect of radiation on human beings or on matter, and the case of a criticality accident. The supplement 1 to this standard contains explanations of the physical properties of criticality of systems containing fissile materials.

DIN 25403 – Criticality safety in processing and handling of fissile materials - Parts 2 to 6, and 8: Criticality Data

These parts of the standard DIN 25403 have originally been issued between December 1991 and September 2000, and summarize criticality data of various fissile systems. They include tables and figures e.g. with values for minimum critical sphere mass, cylinder diameter or slab thickness of such systems. The systems are mainly chosen as conservatively bounding for realistic arrangements, e.g. pure ^{239}Pu for Plutonium systems.

The parts are concerned with homogeneous, reflected systems and are divided into 2) $^{235}U_{metal}$-light water systems; 3) $^{239}Pu_{metal}$-light water systems; 4) uranium dioxide-light water systems; 5) ^{239}Pu dioxide-light water systems; 6) ^{239}Pu nitrate-light water systems; 8) ^{235}U nitrate systems.

The former part 7), covering low enriched uranium dioxide rod lattices in light water, has been withdrawn in June 2014, as the determination of a critical sphere mass of a heterogeneous rod lattice by use of homogenized nuclear cross sections does no more represent the state of the art, and is in practice replaced by explicit three-dimensional Monte Carlo calculations. However, the criticality data for homogeneous systems as described in the other parts still remain of relevance.

DIN 25474 – Measures of administrative character for conservation of criticality safety in nuclear facilities excluding reactors

This standard is available as final revision as of June 2014. It addresses the technical design and defines a framework for administrative measures to comply with, to ensure criticality safety. It describes the hierarchy of measures and controls:

- passive measures
- active technical measures becoming automatically effective
- active technical measures becoming manually effective
- administrative measures and controls.

Possible administrative measures and controls are summarized, e.g. control of the enrichment of the fissile material composition, its mass, its density or its concentration, the degree of moderation, the chemical composition, control of permitted moderators and reflectors and many more. The designation of areas of certain control measures (e.g. moderation control), and transitions between different areas of control, are addressed.

In accordance with the Radiation Protection Ordinance, accountability and responsibility are assigned to the radiation protection officer of the facility. All persons who handle nuclear fuels or have the authority to issue instructions in handling nuclear fuels are to be instructed on the measures necessary to comply with criticality safety before starting work. All persons, including foreign personnel who are at risk from the consequences of a criticality incident, are to be instructed on the appropriate reaction in case of a criticality incident.

Furthermore, special address is also given to the designation and labelling of fissile material positions; to commissioning and periodic inspections of a facility; to fire fighting measures; to maintenance and rebuilding operations; to deviations from normal operation; and to measures for occurrence of criticality like evacuation and rescue.

DIN 25471 – Criticality safety taking into account the burnup of fuel elements when handling and storing nuclear fuel elements in fuel pools of nuclear power plants with light water reactors

This standard is available as final revision as of May 2009. It applies to the assessment of criticality safety, especially taking into account the fuel burnup in the handling and storage of fuel in the fuel pool of NPPs with light water reactors. It is in compliance with the requirements of the Nuclear Safety Standards Commission [5] rule KTA 3602 "Storage and Handling of Fuel Assemblies and Associated Items in Nuclear Power Plants with Light Water Reactors" already mentioned above.

Beside general requirements, this standard gives detailed guidance on the design of a fuel storage pool and its rack for the storage of irradiated and fresh fuel assemblies. The numerical determination of conservative nuclide inventories of irradiated fuel and the design-dependent neutron multiplication factor are described in-depth, yielding the deduction of a loading curve for a given storage system. Special focus is put to all tolerances and uncertainties to be taken into account in the safety analysis. Typically, the calculated neutron multiplication factor must not exceed 0.95 under normal and abnormal operation conditions, with conservative provision for all uncertainties and tolerances. If a parameter is a random variable, both the choice of a bounding approach or the definition of a one-sided 95%/95% tolerance limit for it are permissible.

The standard also summarizes and explains the parameters that influence criticality safety and addresses the incidents and accidents that are relevant for criticality safety. It also pays attention to the identification and verification of each unique fuel assembly and its burnup before being loaded into a storage position that requires a minimum burnup.

Three extensive technical annexes describe the practical procedure to apply numerical tools in accordance to the standard to a given storage system for irradiated light water reactor fuels. Examples are given for the potential choice of nuclides to be considered in the analysis, treatment of heterogeneous axial burnup profiles, consideration of uncertainties, requirement for code validation, and many more.

DIN 25712 – Criticality safety taking into account the burnup of fuel for transport and storage of irradiated light water reactor fuel assemblies in casks

This standard is available as a final revision as of April 2015. It applies to the assessment of criticality safety during transport and storage of irradiated fuel assemblies or other fuel arrangements (e.g. quivers for damaged fuel rods) from NPPs with light water reactors in dry casks as well as for the related handling operations. The standard includes requirements to ensure criticality safety during transport and storage of spent fuel and fuel rods from nuclear power plants with light water reactors in containers and in the associated loading and unloading operations taking into account the fuel burnup. This standard also contains provisions for and gives guidance on determining the burnup and to carry out checks on compliance with the required burnup for loading the fuel into a cask.

The overall structure and content of this standard is generally similar to DIN 25471. However DIN 25712 addresses dry cask storage rather than fuel pool storage, but since cask loading as well as accident conditions have to be considered, the cask is usually considered to be fully flooded with unpoisoned light water. In terms of axial burnup profiles and definition of loading curves, the annexes included go even beyond the level of detail within DIN 25471.

The standard allows for a general cask loading in terms of a loading curve, but also for individual loading of a unique cask loading pattern in a case by case study, if necessary. The latter option warranted by this standard might become relevant for some fuel assemblies from NPP units which have experienced the unplanned final shutdown decision in Germany after the Fukushima accident. As some units had commenced power operation after regular downtime briefly before, some fuel assemblies with very short irradiation time exist now, which currently cannot be loaded into recently approved transport and storage casks due to a high initial enrichment.

DIN 25472 – Criticality safety for final disposal of nuclear fuels to be discarded

This standard is available as final revision as of August 2012. It applies to the area of disposal of used nuclear fuel for the operational and post-closure phase of a repository and is applicable to all disused nuclear fuel of any composition, packaging and configuration being stored in the repository. This standard is also applicable to the activities required in the operational phase and procedures for handling or control of the casks with nuclear fuel to be discarded, irrespective of the type or burnup state of the fuel. In terms of account for fuel burnup, the criteria are similar to DIN 25712 in structure and content.

This standard is independent of site or host rock of the final repository. For the operational phase of a repository, compliance with its requirements ensure criticality safety when handling the disused nuclear fuel and ensure safe storage of these materials. During the post-closure phase of a repository, these requirements ensure that the probability of criticality and, if necessary, the consequences of a postulated critical excursion, are being sufficiently limited. Here, a site-dependent set of evolution scenarios, i.a. including loss of integrity and geometrical configuration, have to be identified and analyzed, to judge the exclusion, or estimate the probability of occurrence of a critical excursion–which is however still an undesirable event. Based on the assumed triggering event, also potential consequences of a critical excursion are to be analyzed.

A transition from the operational phase under full control to the post-closure phase where the repository is out of active human control and its evolution left to its own devices is defined. Application of probabilistic means is stipulated due to the long period of one million years to the future, during which sub-criticality has to be ensured [9]. A continuous increase in the allowable probability of occurrence of criticality of 10^{-6} at the time of final closure of the repository up to 10^{-4} at the (generally unknown) point in future time on which criticality cannot be excluded any more by deterministic means and thus is being postulated, is foreseen. A technical description of the approach of DIN 25472 is given e.g. in [10], and a closer justification of the relation mentioned before is given in Annex B of the standard.

Hence, based on the disposed nuclear fuel and the scenarios to be investigated in the analysis of the long term post-closure phase, it might become inevitable to postulate a critical excursion. Then according to the requirements listed in this standard, a proof is requisite that the consequences of the postulated excursion are sufficiently limited not to affect the barrier effectiveness and tightness. This represents a significant contribution to demonstrate that the overall risks are balanced during the commissioning, operational and post-closure phase, i.e. the risk minimization principle is satisfied. Thus, the safety criteria to be met in the post-closure phase of a repository are formulated such as to prevent an undue increase of the actual risks during the operational phase.

DIN 25478 – Application of computer codes for the assessment of criticality safety, and supplement 1

This standard is available as final revision as of June 2014. It applies to the use of computing systems in the assessment of criticality safety of fissile materials. It includes requirements for the choice of a computing system; the verification and validation status of a selected calculation system; the documentation of the verification and validation analysis; the necessary steps to prepare the calculations; the execution and control of calculations and evaluation of the obtained results; the analysis of the computational uncertainties.

A selected calculation system must be able to describe the physically relevant parameters and the characterizing physical effects of the fissile material arrangement to be analyzed. The calculation system shall be able to determine the safety characteristics of this arrangement (in most cases, but not necessarily limited to, the neutron multiplication factor), as well as the information necessary for the validation of these quantities.

The standard further describes how uncertainty, bias and bias uncertainty for a depletion code as well as for a criticality code can be determined by calculation and analysis of radiochemical assay data and critical benchmark experiments, respectively. It is supported by an explanatory supplement which contains detailed further information, in particular on the treatment of computational uncertainties. Strong focus is given to the potential of sophisticated stochastical methods to evaluate a set of benchmark calculations, to estimate both the systematical bias and its uncertainty, which have to be applied for the application case addressed.

Conclusions

The available criticality safety related DIN standards have been reviewed, revised, and harmonized in terms of terminology, by the competent standards technical committee. This set of standards provides consistent, comprehensive and detailed guidance to assess criticality safety and to ensure sub-criticality at handling, transport and storage facilities of fissile materials outside NPP cores, including final disposal of used fuels. Elaborate guidelines are given for the numerical analysis of criticality safety using computational tools, including the consideration of fuel burnup, and the thorough validation, evaluation and uncertainty estimation of calculation results.

Most of those standards additionally feature non-normative, explanatory annexes which provide a broader technical and scientific support at sufficient level of detail, including e.g. the use of sophisticated stochastical means in uncertainty estimation. In terms of final disposal, the use of probabilistic means is introduced, to keep the overall risk especially for the operational phase at an acceptable level while not unduly compromising the probability of occurrence of a critical excursion in the post-closure phase.

The set of criticality safety related DIN standards has recently been presented to the corresponding ISO working group ISO/TC 85/SC 5/WG 8 “Nuclear criticality safety”. It was concluded, that the DIN system of criticality safety standards could benefit from taking into account concepts from the ISO standards, and vice versa. But since the DIN standards are heavily interlinked (see Figure 1), a simple adoption of standards from the other system will not be effective.

Acknowledgments

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Regardless of the normative work itself, the paper at hand was founded by the German Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety, supported by the Federal Office for Radiation Protection, and endorsed by the members of the DIN Standards Technical Committee “Criticality Safety”.

List of References

[1] German Federal Atomic Energy Act (Gesetz über die friedliche Verwendung der Kernenergie und den Schutz gegen ihre Gefahren, „Atomgesetz“ – AtG), 1 May 2014.

[2] Radiation Protection Ordinance (Verordnung über den Schutz vor Schäden durch ionisierende Strahlen, „Strahlenschutzverordnung“ – StrlSchV), 1 January 2015.

[3] Reactor Safety Commission (Reaktorsicherheitskommission – RSK).

http://www.rskonline.de/English/index.html

[4] Nuclear Waste Management Commission (Entsorgungskommission – ESK). http://www.entsorgungskommission.de/englisch/default.htm

[5] Nuclear Safety Standards Commission (Kerntechnischer Ausschuss – KTA). http://www.kta-gs.de/welcome_engl.htm

[6] German Federal Act on the Transportation of Dangerous Goods (Gefahrgutbeförderungsgesetz – GGBefG) , 1 January 2010.

[7] IAEA SSR-6 “Regulations for the Safe Transport of Radioactive Material, 2012 Edition Specific Safety Requirements” (formerly TS-R-1, 2009), 2012. http://www-pub.iaea.org/MTCD/publications/PDF/Pub1570_web.pdf

[8] German Institute for Standardization (Deutsches Institut für Normung – DIN). http://www.din.de/cmd?level=tpl-home&languageid=en

[9] German Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (BMUB) “Safety Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste”, as at 30 September 2010.

http://www.bmub.bund.de/fileadmin/bmu-import/files/english/pdf/application/pdf/sicherheitsanforderungen_endlagerung_en_bf.pdf

[10] R. Becker, B. Bendick, M. Hennebach, S. Jaag, H.-G. Johann, R. Kilger, B. Klüver, G. Knecht, H. Kühl , J. C. Neuber, I. Reiche, H. Scheib, J. Thiel, S. Tittelbach, M. Treige-Wegener, A. Verst (DIN Working Committee NA 062-07-045 AA), “Draft for a criticality safety standard for final disposal of nuclear fuel“, International Conference on Nuclear Criticality (ICNC) 2011, Edinburgh, United Kingdom, 19-22 September 2011.

ISSUES ABOUT IMPLEMENTATION OF GEOMETRY CONTROL TO MAINTAIN NUCLEAR CRITICALITY SAFETY

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1. Introduction

In accordance with the usual regulations and standards requirements [1], the Nuclear Criticality Safety (NCS) of a unit[1] is achieved by considering one or a combination of parameters of control. The choice of a parameter of control, such as geometry or mass, depends on the process, the fissile content and the need to limit operating constraints. At last, these parameters ensuring the sub-criticality must be identified and controlled.

In many cases, especially when the mass or the concentration of fissile material is too important, the parameter of control may be the geometry of the unit. Then, the unit NCS assessment is achieved by considering the unit dimensions and concludes on the need to control these dimensions in order to verify the compliance between the real geometry and the one considered in the assessment.

2. Limiting dimensions to achieve NCS

a. Parameters of control including a limit on dimensions

The control of the unit's dimensions to establish and maintain the NCS is usually preferred by criticality safety specialists. Indeed, such a control may be achieved by robust passive safety measures and thus requires less operating constraints than other controls (mass, etc.).

Obviously, dimensions of a unit are limited when the parameter of control is the geometry of this unit (e.g. diameter of a cylinder tank or thickness of a slab tank). However, a dimension can be limited to achieve NCS for other parameters such as illustrated hereafter.

Neutron interaction: When a fissile unit is composed of two or more fissile items, the NCS assessment may conclude to limit neutron interactions by requiring a minimal safe distance between items.

Neutron reflection: The dimension of reflectors and its distance to fissile items are relevant in the NCS assessment and may be limited.

Example: When a metal crucible (stainless steel or carbon) is used to mold a fissile material, a maximal safe thickness can be determined in order to limit neutron reflection.

Neutron absorber: In the same way as reflection, the dimension of solid neutron absorbers and their distance to fissile items can be credited in a NCS assessment.

Example: For a borated screen, it could be necessary to determine a minimal safe thickness.

Mass: The quantity of fissile material can be controlled by limiting some dimensions of a unit.

Examples: the respect of a safe mass in a unit can be ensured, for a specific density, by limiting the unit volume, and the number of fissile items, such as drums, can be limited by the number of locations available (i.e. the maximal dimensions of the area).

Note 1: *In this paper, a unit is defined as a part of process or facility composed by fissile item(s) and surrounding materials*.

Moderation: In the same way as mass, the quantity of moderator can be controlled by limiting some dimensions of an item.

Example: the mass of moderator added to a process can be limited by the volume of its container.

Areal density: The respect of safe areal density (in g/cm²) requires a control of the fissile item surface (as well as the fissile mass).

Consequently, the control on dimensions can be required for several parameters contributing to NCS of a unit.

b. Issues associated with control on dimensions

When the NCS is maintained by limiting unit's dimensions, the final requirement related to this approach is to ensure that the dimensions taken into account in the NCS assessment bound the dimensions of the existing unit. This requirement is applicable during the unit lifetime, which includes the design, procurement, commissioning, operations and maintenance phases. To respect this requirement, dimensions relevant for NCS have to be controlled by measurement. This control, usually called verification of the compliance, is not only a comparison between the NCS assessment and the existing unit. Indeed, in accordance with the usual regulations and standards requirements [1], the normal and credible abnormal conditions are taken into account in the NCS assessment, including sufficient safety margins.

Consequently, there are several issues associated with this final requirement:

- to perform a conservative NCS assessment in order to cope with the existing unit;
- to be able to control dimensions relevant for NCS;
- to demonstrate that the NCS assessment bounds the operational unit.

Thus, the main challenges associated with these issues are:

- to account for the verification of the compliance in the NCS assessment, by considering dimensional margins and by identifying dimensions relevant for NCS;
- to define the dimensional limits to be respected during the unit lifetime, in normal and credible abnormal conditions;
- to verify the compliance between the dimensions considered in the NCS assessment and the actual dimensions and to manage the non-compliances.

As these issues are met at different lifetime of the unit (design, commissioning, etc.) and by several people (designers, manufacturers, NCS staff, etc.), it is necessary to properly document how these challenges had been solved. This challenge is addressed by integrating this documentation into the Quality Assurance program. Indeed, the NCS assessment and the implementation of dimensions control is performed under a Quality Assurance system. Moreover, this documentation is expected to be benefic compared to the one usually achieved for the verification of compliance under the Quality Assurance program. In fact, requirements related to the control of dimensions relevant for NCS are safety requirements which are different than other usual controls. For example, sample controls are not sufficient to control these dimensions whereas they may be sufficient for other QA purposes.

This paper focuses and suggests guidance on the main issues related to the implementation of dimensions control. It also presents some examples to meet these challenges and to document properly the dimensions limits to be respected and the verification of compliance.

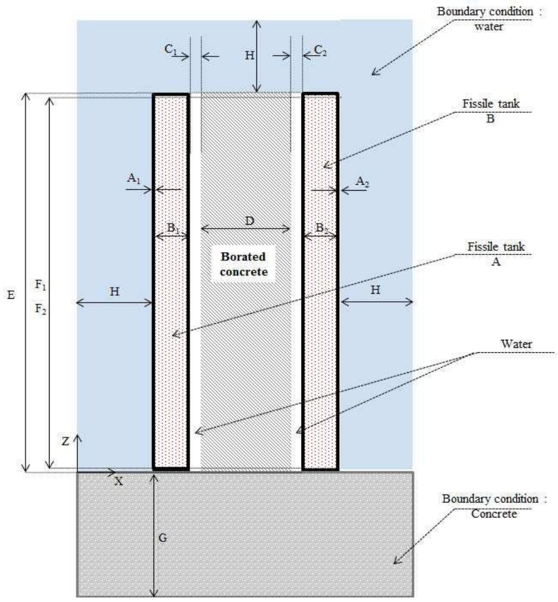

Figure 1 Example – Calculation model

c. Example

In order to illustrate the discussions of this paper, an example of a unit with controlled dimensions for NCS purpose is provided and is completed through the paper.

The unit taken into account is composed by two slab tanks, filled with a fissile solution, separated by borated concrete. The tank material is stainless steel. The boundary conditions of the unit are:

- fully water reflected along X direction and in +Z direction;
- specular reflection along Y direction;
- 100 cm of concrete in -Z direction.

The calculation model is shown on the Figure 1.

3. Perform a conservative NCS assessment

a. Accounting for compliance verification in the NCS assessment

The need to control dimensions and to verify the compliance with the actual dimensions may be taken into account in the NCS assessment. In this way, the risk of detecting non-compliance may be reduced and the link between the NCS assessment and the NCS requirement may be clarified.

Thus, the main challenges are:

- to account for margins[2] and simplification in the NCS assessment;
- to facilitate controls on the actual unit through the NCS assessment;
- to limit the number of dimensions to be controlled.

Taking into account margins in the NCS assessment

The dimensions considered in the NCS assessment could arise from criticality handbooks, pre-design calculation results, design drawings, manufacturers' drawings or from measurements accomplished on an existing unit. These dimensions describing geometry of the unit are:

Note 2: *In this paper, the concept of margin includes margin on dimension and margin on reactivity. At the end, taking into account margins leads to increase the k-eff value.*

- dimensions of pieces of equipment containing fissile material;
- dimensions of materials surrounding fissile items , such as reflectors or neutron absorbers;
- distance between fissile items to another one, to reflectors or to neutron absorbers;
- dimensions and layout of mobile devices.

Accounting for dimensional margins in the NCS assessment on the nominal dimension may allow:

- to bound the real dimensions of the unit during the unit lifetime (including normal and credible abnormal operations);
- to avoid (or limit) non-compliance;
- to cope with possible changes in the geometry due to:
  - normal operations (loading);
  - late update during the design phase;
  - difficulties during commissioning.

It is easier to take into account margins when specific calculations are performed. Indeed, margins could be integrated in the calculation model's value or in modeling simplification (in our example the widths of the tanks are infinite). At the end, the dimensional margins lead to increase the unit k-eff.

Thus, it is recommended to consider the following dimensional margins on nominal dimensions in the NCS assessment:

- manufacturing tolerances, for dimensions values from design or manufacturers drawings;
- measurement uncertainties, for dimensions values from measurement on existing units;
- deformations due to phenomena encountered in normal and credible abnormal operations/conditions (see § 4).

Finally, even if information listed above is not available when assessing the NCS of the unit, accounting for arbitrary margins limits the risk to encounter later difficulties to conclude the assessment. These additional margins could be generic values or could be based on the lessons learned from similar unit.

Easing the dimensions control

The list of dimensions to be controlled can be defined from the calculation model. This approach may allow to clearly linking the NCS assessment to the NCS requirement. So, it could be efficient to take into account in calculations model dimensions controllable. As an example, a distance edge-to-edge should be preferred to a distance center-to-center.

Moreover, in some cases, it could be useful to account for the feasibility of controls in the calculations models. As an example, the units containing highly radioactive fissile materials, such as dissolution liquors containing fission products, can only be controlled before commissioning. Consequently, it is preferable to limit the number of control to achieve and to take margins in the NCS assessment.

Limiting the number of dimensions to be controlled

It is necessary to limit the number of dimensions to be controlled. On the one hand, the dimensions controlled should be restricted to dimensions which are relevant for NCS (i.e. which have an impact on criticality). One the other hand, the limitation of dimensions to be controlled may allow:

- to highlight the real issues related to NCS;
- to reduce errors during verifications (missing a control, etc.);
- to reduce the cost of controls.

So, as it is related to NCS, the methodology considered for this identification should be justified by a criticality specialist. The justification may be guided by:

- sensitivity k-eff calculations on a dimension;
- an analysis of calculations results (estimation of reactivity weight of the different parts of the modeling);
- an expert judgment.

The relevant dimensions are called in this paper **Nuclear Criticality Safety Dimensions (NCSD)**. This distinction from other dimensions means that these dimensions must be ensured during the unit lifetime so as to ensure the sub-criticality.

At last, the list of NCSD may be documented in a specific document. Thus, all the people involved in the design, in the fabrication and in the layout of the unit may be aware of parameters relevant for NCS.

b. Example

For the example previously detailed, the Table 1 presents for each dimension:

- the nominal values and the manufacturing tolerances;
- the values considered in the calculation model;
- the dimensions identified as NCS Dimensions;
- an evaluation of the dimensional margins between the nominal dimension, including manufacturing tolerances, and the calculation model dimensions.

It is supposed in this example that the NCS assessment is performed at the beginning of the design phase. So, it is assumed that the deformations due to normal and abnormal conditions (e.g. pressure, corrosion, earthquake, etc.) are not known yet.

Table 1 Identification of NCS Dimensions

Identification of unit dimension					Calculation value (mm)	Dimensional margin (mm)	Identification of NCS
Dimensions		Name	Nominal value (mm)	Manufacturing tolerances (mm)			NCS Dimension?
Fissile part	Internal thickness	B_1 B_2	65	2	70	3	**YES**
	Internal height	F_1 F_2	3000	10	3500	490	NO
	Internal width	-	3400	10	infinite	-	NO
Stainless steel thickness		A_1 A_2	7	1	5	2	**YES**
Borated concrete	Thickness	D	200	2	202	0	**YES**
	Height	E	3020	10	3500	470	NO
	Width	-	3500	10	infinite	-	NO
Distance borated concrete/tank		C_1 C_2	19.5	5	30 *(sensitivity calculations)*	5.5	**YES**
Surrounding water layer thickness		H	-	-	30	-	NO
Concrete floor thickness		G	-	-	100	-	NO

The height of the fissile shapes (F1 and F2) and of the borated concrete (E) are not defined as NCS Dimensions because a variation on their values would not have a significant impact on reactivity. In the same way, the widths are not a NCS Dimension because infinite widths are modeled. At last, the thicknesses of boundary conditions (H and G) are not defined as NCS Dimensions because they are supposed to be conservative (the unit is fully reflected).

The thicknesses of the fissile tanks (B1 and B2), of the stainless steel (A1 and A2) and of the borated concrete (D) are obviously relevant for NCS.

Moreover, thanks to sensitivity calculations (not shown here), it appears that it is conservative to maximize the distances C1 and C2. Indeed, the neutron reflection provided by water is better than the one provided by the borated concrete. As the sensitivity calculations had been performed for distance from 0 cm to 30 cm, the final calculation value taken into account is equal to 30 cm.

Finally, even if 15 dimensions are taken into account in the calculation model, the number of the so-called NCS Dimensions is equal to 7.

4. Define the NCS Dimensions limits to be respected

For each NCSD, the limit ensuring that the NCS assessment copes with the actual unit geometry has to be defined. Thus, for each NCSD there is a dimension interval allowed to maintain NCS. Finally, if the actual dimensions are included in this interval, the compliance is proven.

Furthermore, in accordance with usual regulations requirements and standards [1], the phenomena affecting the actual dimension are taken into account in the NCS assessment. So, the dimensions limits to be respected may not be equal to dimensions taken into account in the NCS assessment. Consequently, the challenge to address is to define a NCSD limit bounding all phenomena affecting dimensions.

Finally, the NCSD limits should be properly documented in order to aware people involved in operations affecting the unit geometry (e.g. during the design phase or during operations) of dimensions relevant for NCS.

a. Determination and documentation of NCSD limits

Normal and credible abnormal conditions impacting NCSD

The identification of phenomena affecting dimensions and the estimate[3] of their impact on the actual values should be presented in the NCS assessment. Moreover, the need to do this estimate has to be assessed. Indeed, in some abnormal conditions, it is possible to ensure the NCS by other parameters of control (such as a mass limit). Nevertheless, the use of additional parameters of control may lead to define additional NCS requirement (such as controlling the mass).

For examples, phenomena which could impact the NCSD value are:

- in normal conditions: pressure or temperature deformations, corrosion or abrasion deformations, deformation due to loading, etc.
- in abnormal conditions: earthquake, fire, explosion, load drop, etc.

Determination of NCS Dimension limit

For each NCSD, it is necessary to identify if the limit to be respected is an upper or a lower limit. In some specific case, it could be an upper and a lower limit. This evaluation could require performing sensitivity calculations on k-eff estimates.

Note 3. *This estimate could be resort on specific calculations such as mechanical calculations*

Finally, the Nuclear Criticality Safety Dimension limit is defined as follows:

Then, the NCS Dimension limits can be determined by accounting for:

- the dimensions taken into account in the NCS assessment : D_{NCSA} (e.g. value from the calculation model);
- the variation on the NCSD due to phenomena affecting geometry in normal conditions: $\delta_{NC}^{\pm}$
- (positive, δ_{NC}^{+}, or negative value, δ_{NC}^{-});
- the variation on the NCSD due to phenomena affecting geometry in credible abnormal conditions: $\delta_{AC}^{\pm}$ (positive, δ_{AC}^{+}, or negative value, δ_{AC}^{-}).
- if it is a maximal limit to be respected:

$$\textbf{(NCS Dimension limit)}_{\textbf{MAX}} = D_{NCSA} - |\delta_{NC}^{+}| - |\delta_{AC}^{+}|$$

- if it is a minimal limit to be respected:

$$\textbf{(NCS Dimension limit)}_{\textbf{MIN}} = D_{NCSA} + |\delta_{NC}^{-}| + |\delta_{AC}^{-}|$$

Note 4: *the NCSD limit should be reevaluated each time the NCS assessment is updated.*

Documenting NCSD limits

When the NCSD limits are defined, they should be communicated to people involved in the management of the unit geometry, such as when applicable:

- designers (including safety specialists, mechanical specialists, chemical process specialists, etc.);
- manufacturers in charge of unit fabrication and layout;
- inspectors in charge of the pieces of equipment receipt from manufacturers;
- safety staff in charge of the unit during operations (including commissioning and maintenance operations).

The record in a specific documentation of the NCSD limits to be respected makes easier the knowledge of NCS requirements by people listed above. Indeed, depending on the NCS assessment documentation, it may be difficult for people to find quickly the parameters and limits relevant for criticality. This documentation is called in this paper **NCS Dimension Form**.

In many French Nuclear Facilities, this type of documentation had been successfully implemented and formalized (during the design and procurement phases or during the operations).

When this document is provided to people in charge of the unit design, of operating the unit and of controlling the dimensions to be respected, the criticality safety staff missions are to inform them of the NCS issues and to ensure that the NCS requirement are correctly understood.

In addition, after process commissioning, the NCSD limits could be also presented in the operating rules if the NCSD may be easily modified by operators and / or if the number NCSD is low.

Finally, the documentation presenting NCSD limits must be updated when the NCS assessment is modified (including in case of reassessment of normal and abnormal conditions).

b. Example

From the list of NCSD defined in Table 1, the limits to be respected can be determined. Thus, the Table 2 presents for each NCS Dimension, the calculation value taken into account in the NCS assessment, the phenomena which could impact the dimension in normal and abnormal conditions and the NCS Dimension limits to be respected.

Table 2 Determination of NCS Dimensions limits

NCS Dimension	Name	Calculation value (mm)	Pressure / temperature deformation (mm)	Abrasion / corrosion deformation (mm)	Earthquake / Fire	NCS Dimension limit (mm)
Internal fissile thickness	B_1 B_2	70	+ 1	+ 1	0	**MAX : 68**
Stainless steel thickness	A_1 A_2	5	0	- 0.5	0	**MIN : 5.5**
A borated concrete thickness	D	202	-	-	0	**MIN : 202**
Distance borated concrete/tank	C_1 C_2	30	-	-	± 5	**MAX : 25**

It is interesting to note that the stainless steel corrosion thickness is taking into account in the determination of two NCS Dimensions limits (reduction of 0.5 mm on each stainless steel material and increase of 0.5 mm on both sides of the fissile tank). In fact, this phenomenon leads both to decrease the stainless steel thickness and to increase the fissile thickness.

5. Verification of compliance

Once NCS Dimensions limits are set, the verification of the compliance with the real unit geometry has to be proven. The challenges associated with this issue are:

- to perform controls on NCSD;
- to define the periodicity of NCSD controls;
- to verify that each NCSD actual value complies with its limit;
- to manage non-compliances pinpointed.

At last, in order to keep a track, the NCSD controls and the compliance verification should be properly documented and recorded.

a. Control of Nuclear Criticality Safety Dimensions

The controls of NCSD are achieved before process commissioning, after maintenance operations and events (abnormal conditions) affecting the unit geometry. Moreover, the controls are performed during periodic safety review (every 10 years in France) and during periodic controls. Regarding the periodic controls, they may allow to validate the estimation of dimension value deviation due to normal operating conditions (see §4.a). The determination of controls' frequency depends on the safety program, the regulations and on the confidence in the methodology employed.

The two main objectives of the NCSD control are:

- to ensure the applicability of NCS assessment with regard to the actual unit geometry;
- to highlight the dimensional margins of the NCS assessment from the dimensions/design standpoint.

The NCS Dimensions identified in the NCS assessment are usually controlled by measurement. This control can be achieved by a direct measurement or by an indirect one (e.g. by direct measurement of the mold used for the fabrication or of a template). In any cases, in accordance with the Quality Assurance program, the measurement method used for the control has to be validated and a measurement uncertainty has to be associated with each NCS Dimension measured.

The control of a dimension may not require the use of measuring instrument with a high precision. In fact, the choice of the measuring instrument can be based on the dimensional margin between the NCS Dimension limit and the expected real dimension. As an example, if the minimal safe distance between two fissile shapes is equal to 10 cm and the real distance is close to 30 cm, a control by a tape measure is sufficient.

In some cases, the Quality Assurance program related to the unit fabrication may be sufficient to ensure the compliance of the unit with NCS Dimensions when they correspond to basic features (e.g. the use of standardized equipment, pipes with standard sizes, etc.). Indeed, the quality assurance program related to the unit fabrication already ensures the consistency between the actual unit geometry and its general purpose and its intended functioning.

b. Compliance verification and management of non-compliance

Once, measurements are carried out, the verification of the compliance between the value measured (measurement uncertainties included) and the NCS Dimensions limits is done.

Usually, in accordance with the Quality Assurance program, the measured values and the measurement uncertainties are reported in an inspection report by the company in charge of controls (as well as the method of measurement employed for each control). In some cases, it is possible to have several measurements for a single NCS Dimension. For example when:

- pieces of equipment are large and the control of NCS Dimension needs to be achieved at different height;
- there are several identical items.

From the NCS standpoint, it's sufficient to only refer to a single measured value for the verification compliance, provided this single value is the bounding measured value (maximal or minimal value among those recorded).

The comparison, between the measured value (measurement uncertainties included) and the NCS Dimensions limits, should be achieved by a criticality specialist. In this way, the specialist may allow to appreciate dimensional margins between the real geometry and the one considered in the NCS assessment. Moreover, during periodic survey, this verification provides a good overview of geometrical deviations and may help the criticality specialist to evaluate the need for updating the NCS assessment.

Furthermore, non-compliances, when exist, are evaluated by criticality specialists. Several approaches can be chosen to solve non-compliances:

- by updating the NCS assessment (e.g. by performing additional calculations accounting for the real geometry);
- by showing that there are sufficient dimensional margins on other dimensions;
- by showing that there are sufficient margins on other parameters in the assessment;
- by enhancing the measurement techniques;
- by modifying the unit.

Regarding the first point, the NCS assessment may be also updated to account for additional parameters of control. However, this option might lead to define additional parameters to be controlled.

Furthermore, when non-compliances are identified in a design phase or during manufacturing it is recommended to modify, as much as possible, the geometry of the unit or to update the NCS assessment. Indeed, it is not a good option to reduce margins of safety before starting a process.

c. Documentation of the compliance verification

The actual dimensions and the conclusions of the compliance verification may be presented in the NCS Dimensions Form. So, before commissioning the form is completed (with as-built values) and after, through the operating years, the form is updated to account for geometric evolutions (maintenance, ageing effects, and modification) as well as for assessment evolutions (periodic safety review, change in the calculations, etc.). Such a document is in particular very helpful when maintenance operations are realized offsite (by contractors for example).

Moreover, when drafting new NCS Dimensions Forms, the previous NCS Dimensions Form may provide useful lessons learned on the kind of non-compliance raised on similar units.

Finally, the NCS Dimension Form may be consulted during inspection by regulators to audit the list of controls performed.

d. Example

The Table 3 provides an example on how to verify and document the compliance between the real unit and the calculation model for our example. In accordance with information presented above, this verification is achieved by comparing the maximal or minimal NCSD measured value (including measurement uncertainties) to the NCSD limit.

Table 3 Verification of compliance

NCS Dimension	Name	NCS Dimension limit (mm)	Measured value [4] (mm)	Measurement uncertainty (mm)	Conservative actua value (mm)	Compliance
Internal fissile thickness	B_1 B_2	**MAX : 68**	B_1= 61.2	2	63.2	YES
			B_2= 61.5		63.5	YES
Stainless steel thickness	A_1 A_2	**MIN : 5.5**	A_1= 6.1	1	5.1	**NO**
			A_2= 7		6	YES
A borated concrete	D	**MIN : 202**	210	10	200	**NO**
Distance d	C_1 C_2	**MAX : 25**	C_1= 19.5	5	24.5	YES
			C_2= 19.8		24.8	YES

In the above table, there are two non-compliances noted. These non-compliances may be solved as follow:

- Stainless steel thickness (A): for one tank, the minimal as-built value is 0.4 mm lower than the thickness limit. This is not conservative because it leads to decrease neutron absorption by stainless steel. Nevertheless, the conservative actual thickness of the tank#1 (B_1) is lower by 4.8 mm than the limit. This dimensional margin may be judged to compensate the non- compliance on stainless steel thickness (this judgment may be confirmed by sensitivity calculations). Thus, the non-compliance is solved.
- Borated concrete thickness (D): this non-compliance can be solved by changing the measurement instrument in order to have a lower measurement uncertainty. This non- compliance could have been avoided by taking into account more dimensional margins in the calculation model.

Note 5. For each tank a measurement is achieved.

6. Conclusion

In conclusion, this paper focuses on the main issues related to the implementation of geometry control on the NCS assessment and provides some guidance in order to fulfill these challenges.

Furthermore, the discussions presented in this paper can be applied to the control of chemical composition of material. As well as dimensions, chemical elements impacting criticality must be identified and controlled and some margins can be taken into account in calculations in order to cope with difficulties during commissioning (e.g. non homogenous repartition of element) or with density evolutions due to normal and credible abnormal conditions. And, at the end, the control of material composition should be documented and recorded.

Finally, in order to summarize the discussions presented in this paper, the flowchart presented on Figure 3 provides, for one dimension, the issues related to the control of geometry and to the compliance verification.

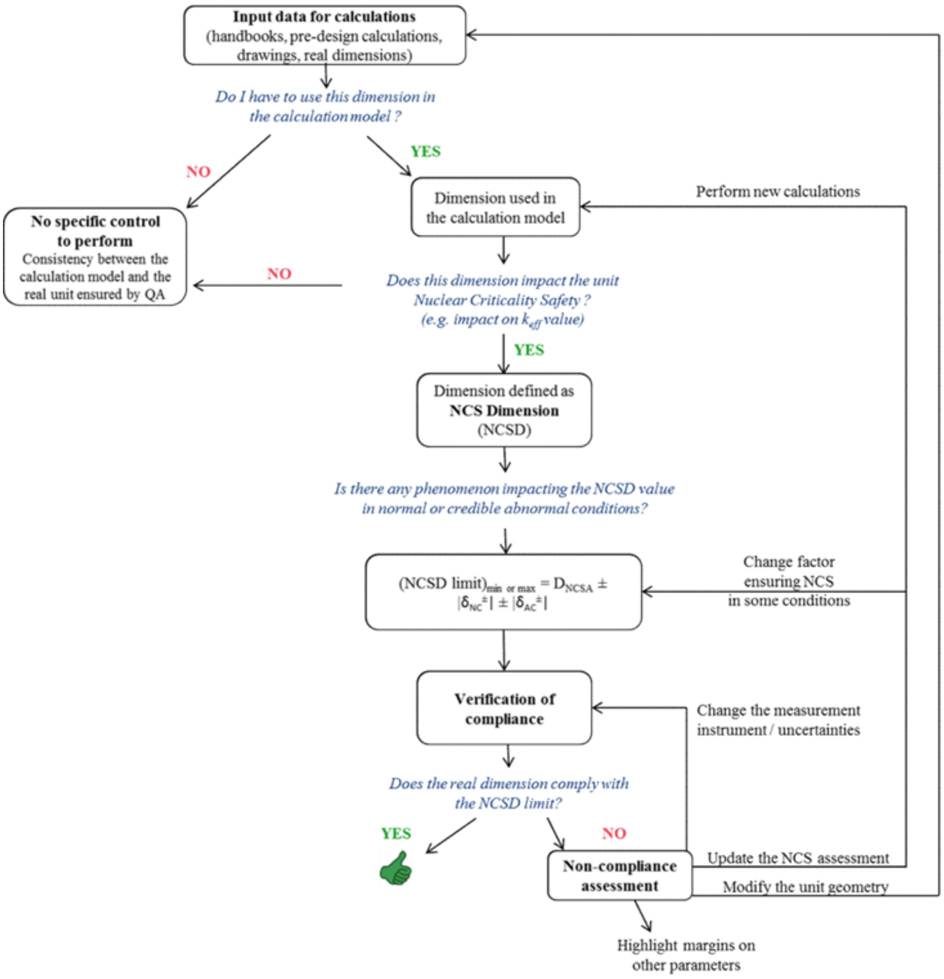

Figure 2 Implementation of a dimension in the NCS assessment

REFERENCE

[1] Usual regulations and standards requirements: ANSI/ANS 8.1 :2014, ISO 1709, French « Décision 2014-DC-0462, Autorité de Sûreté Nucléaire, October 7th 2014 ».

USE OF MAVRIC-SCALE SEQUENCE TO COMPUTE TRANSMISSION FACTORS FOR CRITICALITY ALARM SYSTEM APPLICATIONS

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ABSTRACT

As a result of the implementation of certain design modifications at the Juzbado Fuel Fabrication Facility, it has been necessary to re-calculate the transmission factors to analyze the impact of the new lay-out in the Criticality Alarm System (CAS) detectors distribution. The SCALE package contains the MAVRIC control module that provides a tool for shielding and radiological protection calculations. The MAVRIC sequence has been developed to calculate fluxes and dose rates with a low uncertainty and in a short calculation time. Dose rate calculations using MAVRIC and considering different attenuation thicknesses and materials have been performed for the calculation of the transmission factor. As the first step, the MAVRIC sequence has been validated by comparing its resulting transmission factors against two other different methods:

- Method 1 follows the Regulatory Guide 3.34 to obtain the gamma attenuation factor as a function of concrete thickness. The transmission factor is just its inverse.
- Method 2 is based on the dose half-value layers for concrete as a function of energy spectra obtained from reference data, so the transmission factor is calculated as an average over the energy spectra considered in the CAS design.
- Own method: MAVRIC calculations performed modeling a 20 cm spherical shape source with the energy spectra used in the CAS design, at different positions from the concrete shielding and also for different concrete thickness. We model a punctual detector at a fixed distance to the source position.

Finally, just comparing the transmission factors calculated by these three methods, it is clearly shown that the results obtained with methods 1 and 2 are statistically equivalent to those calculated with MAVRIC. Therefore, it is concluded that MAVRIC is a useful tool to calculate transmission factors for a source spectra in the range 400 to 7100 keV and for concrete shielding, being the transmission factor given by the expression:

$$T = \frac{D_\gamma}{D_{\gamma air}} \times \left(1 - 3 \times \sqrt{\left(\frac{\sigma_\gamma}{D_\gamma}\right)^2 + \left(\frac{\sigma_{\gamma air}}{D_{\gamma air}}\right)^2} \right)$$

where D_γ and $D_{\gamma air}$ are the dose rates calculated by MAVRIC with and without shielding (expressed in rem/h units), and σ_γ y $\sigma_{\gamma air}$ are their uncertainties.

1. BACKGROUND

Juzbado facility is continuously improving the fuel assembly fabrication process and because of that, the fabrication lay out is being changed and new areas added. The regulatory rules compel us to have a

Criticality Alarm System (CAS) covering every area where nuclear material is handled. Due to this requirement, the new building areas shall be continuously monitored by CAS detectors. Juzbado official documents established that each CAS detector shall cover a 36.5 m or 150 m radius area depending of the criticality risk. This distance to be covered is not a straight measurement, but an effective radius that must take into account the radiation attenuation caused by the building construction materials, such as concrete or stainless steel, paying attention to thickness and composition data. Therefore, the effective radius R_{eff} covered by each CAS detector is calculated as:

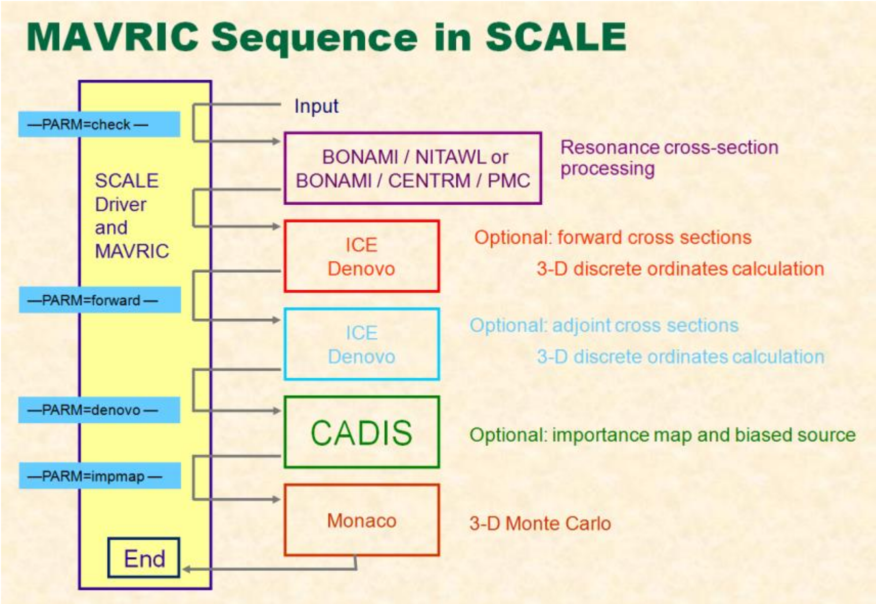
$$R_{eff} = 36{,}5 \times \sqrt{T} \quad \text{or } R_{eff} = 150 \times \sqrt{T} \tag{0}$$

where T is the transmission factor given by the materials present between the detector and the nuclear material. As the new building constructions materials may differ from the rest of the building, it has been necessary to re-calculate the transmission factor.

2. MAVRIC BRIEF DESCRIPTION

The SCALE 6.1 package contains the MAVRIC control module that provides a tool for shielding and radiological protection calculations. The MAVRIC tool has been developed to calculate fluxes and dose rates with a low uncertainty and in a short calculation time.

Figure 1



MAVRIC module contains several functional modules, as shown in Figure 1. Modules BONAMI/NITAWL or BONAMI/CENTRM/PMC calculate the cross section for the defined materials. Then MAVRIC creates a 3-D mesh to calculate the flux variation versus energy and location. Based on this information, MAVRIC makes an "Importance Map", which is used as an entry to perform transport calculations and to obtain an estimated source term distribution. Finally, MAVRIC calculates fluxes and dose rates running the Monaco module and taking into account the Importance Map, the source term distribution and the shielding.

The material composition library includes multiple possibilities, from basic standard materials and solutions to any other arbitrary material. It also allows multiple geometries using the SCALE general geometry package KENO-VI.

3. CODE VALIDATION

The transmission factor T is determined by the ratio of the source radiation intensity I_0 and the intensity I after passing through some material of thickness dl and linear attenuation coefficient μ.

$$T = \frac{I}{I_0} = e^{-\mu dl} \tag{1}$$

3.1 Method 1: Regulatory Guide 3.34

Reference [1] develops a very easy and straight method to calculate the attenuation factor *F* for gamma radiation as a function of concrete thickness *d*. This reference calculates the attenuation due to concrete and applying the NRC Regulatory Guide 3.34 *"Assumptions used for evaluating the potential Radiological consequences of accidental nuclear criticality in a uranium fuel fabrication plant",* and the codes MCNP4A and ANISN. It concludes that:

$$F = 10^{d/30.48}$$

As the attenuation factor is the inverse of transmission factor:

$$T_1 = \frac{1}{10^{d/30.48}} \tag{2}$$

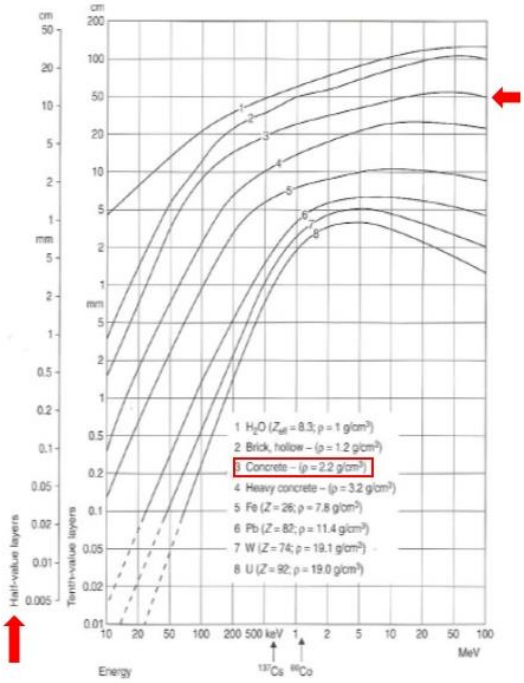
3.2 Method 2: Dose half-value layers for concrete as a function of energy spectra

The half-value layer λ is the material thickness that decreases the intensity of a gamma source to half its value. From expression (1) it can be obtained the lineal attenuation coefficient μ.

$$\mu = \frac{Ln(2)}{\lambda} \tag{3}$$

Figure 2 (see reference [2]) shows the dose half-value layers λ as a function of gamma energy spectrum for 2.2 g/cm^3 density concrete.

Figure 2
Dose half-value layers vs. energy



The gamma energy spectrum considered on the Criticality Alarm System goes from 10 keV to 10 MeV, divided on several intervals of energy E_i. Thus, choosing the corresponding λ values from Figure 2, we are able to calculate the transmission factor T_2:

$$T_2 = \sum_{i=1}^{n}\left(\frac{E_i}{E_T}\right)\times e^{-\left(\frac{\ln(2)\times d}{\lambda_i}\right)} \qquad (4)$$

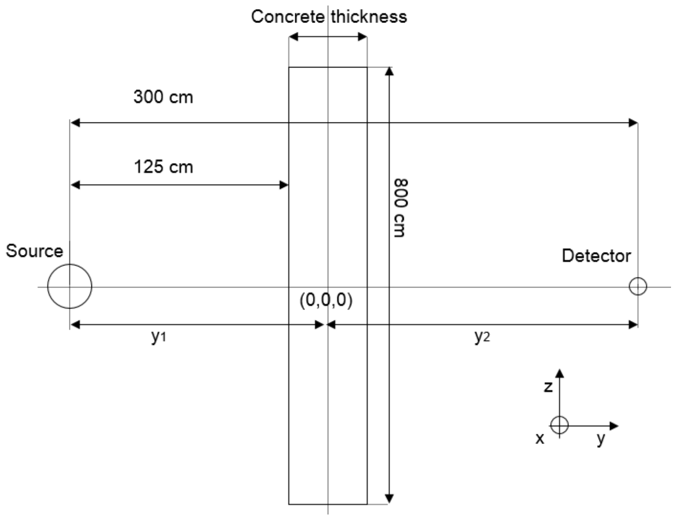
where *d* is the concrete thickness and E_T is the whole energy $E_T = \sum_{i=1}^{n} E_i$.

3.3 Method 3: MAVRIC

MAVRIC allows us to calculate the dose rate measured by a detector. Thus we only have to calculate the gamma dose $D_{\gamma air}$ measured by the detector trough a free path between the source and detector and then, model a concrete wall between them to obtain D_γ (see Figure 3). Transmission factor is then given by the ratio:

$$T_3 = \frac{D_\gamma}{D_{\gamma air}} \qquad \textbf{(5)}$$

Figure 3



Several concrete thicknesses have been analyzed keeping fixed the distances source-detector and source-concrete wall. The source-detector distance is great enough to minimize the source geometrical effects, so it can be considered as a point source. Figure 4 shows part of the MAVRIC input file. The detector is considered as point “pointDetector” and is placed in the “location 1” along the y-axis. We are only interested in gamma dose so we enter “specialdose=9504”, which uses ANSI standard (1977) to calculate the flux-to-dose-rate factors expressed in (rem/h)/(particle/cm^2/s). The source is modeled in the MAVRIC input through the “read sources” card, where the entered data are: the total intensity “strength”, its position, the source itself modeled as a sphere of radius 20 cm, “spectrumDist” (energy spectrum or intensity in counts per second units), and “photonbounds” (gamma radiation energy of each source intensity value).

Figure 4. MAVRIC input

```
read definitions
  location 1
    position 0 160 0
  end location
  response 1
    specialdose=9504
  end response
end definitions
read sources
  src 1
    title="fuente esferica"
    strength=8.11e+12
    origin x=0 y=-140 z=0
    sphere      20
    spectrumDist
        1.865e+10 3.699e+10 3.35e+10 5.018e+10 7.662e+10 1.227e+11 1.554e+11
        2.431e+11 3.552e+11 6.91e+11 1.26e+12 2.664e+12 2.402e+12
    end
    photonbounds
        7100000 6000000 5000000 4500000 4000000 3500000 3000000 2600000 2200000
        1800000 1350000 900000 400000 0
    end
  end src
end sources
read tallies
  pointDetector 1
    locationID=1
    responseID=1
  end pointDetector
end tallies
read parameters
    randomSeed=8655745280030001
    batches=10
    maxMinutes=720
    perBatch=100000
    noFissions
    noSecondaries
end parameters
end data
end
```

Figure 5. MAVRIC output

```
Point Detector Responses
                                   uncollided                                total
 point det   response          value  uncertainty     rel          value  uncertainty     rel
---------- ----------   ------------ ------------ -------   ------------ ------------ -------
         1          1   1.25776E+01  8.50795E-03  0.00068    1.25851E+01  8.52970E-03  0.00068

Total cpu time for this problem was       0.21 minutes.
```

MAVRIC dose rates results $D_{\gamma p}$ and their uncertainties $\sigma_{\gamma p}$ are expressed in rem/h, so if we consider a confidence interval of 99.5%, we can say that the final results in mSv/h are: $D_\gamma = 10 \times D_{\gamma p} \pm 3 \times 10 \times \sigma_{\gamma p}$ mSv/h. The combined uncertainty for the transmission factor T_3 can be calculated as:

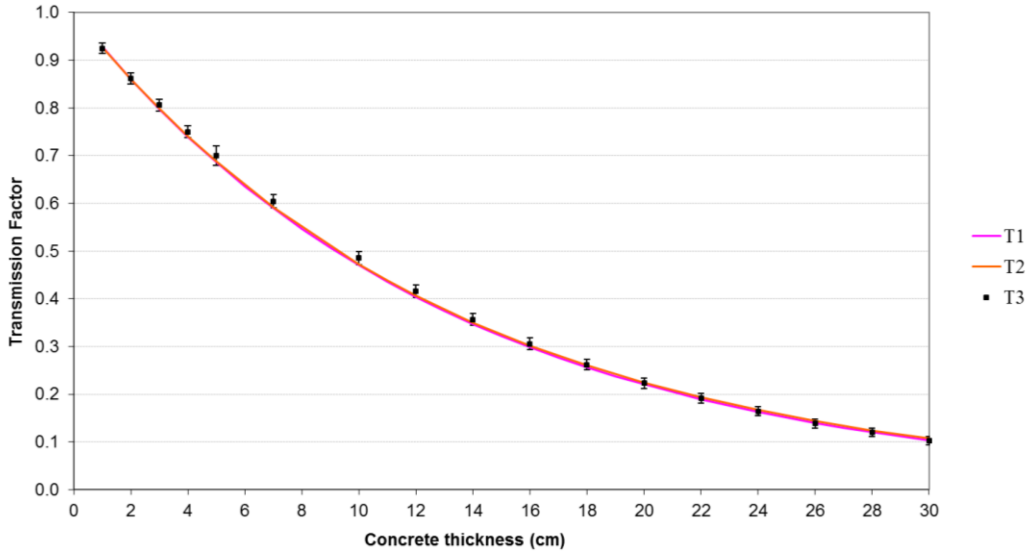
$$\sigma_3 = 3 \times T_3 \times \sqrt{\left(\frac{\sigma_\gamma}{D_\gamma}\right)^2 + \left(\frac{\sigma_{\gamma air}}{D_{\gamma air}}\right)^2} \tag{6}$$

Thus, a pair of values $(T_3, \sigma_3)_i$ for every concrete thickness d_i is obtained.

3.4 Methods' comparison

Figure 6 shows the transmission factors obtained following the three methods. The MAVRIC results T_3 are plotted along with their uncertainty $\pm\sigma_3$.

Figure 6



The values given by methods 1 and 2 are between method 3 values error interval (except at only one thickness). This allows us to conclude that there is no statistical differences between method 3 and the other two methods. Therefore, MAVRIC is an adequate method to calculate transmission factors. A valid and very conservative expression for the transmission factor calculated by MAVRIC is given by its error interval lower value:

$$T = \frac{D_{\gamma}}{D_{\gamma air}} \times \left(1 - 3 \times \sqrt{\left(\frac{\sigma_{\gamma}}{D_{\gamma}}\right)^2 + \left(\frac{\sigma_{\gamma air}}{D_{\gamma air}}\right)^2}\right) \quad (7)$$

APPLICABILITY SCENARIO

Equation (7) can only be used to calculate transmission factors for gamma radiation energy from 400 up to 7100 keV. At a first sight, it would seem that it is only appropriate to 30 cm thick and 2.2 g/cm^3 density concrete. In fact, the transmission factor depends on density, composition and thickness. The lineal attenuation factor depends on cross section σ_{total}, material density ρ and atomic weight A:

$$\mu = \frac{\rho \times \sigma_{total}}{\mathrm{u} \times A} \quad (8)$$

where *u* is the atomic mass unit (1,66x10^{-27} kg). Considering the transmission factor definition given by expression (1), we can obtain different thickness and densities that produces the same transmission factor. If we consider that two different thickness and density shielding produce the same transmission factor $T_a = T_b$, we can see how they relate to each other:

$$T_a = \sum_{i=1}^{13} \left(\frac{E_i}{E_T}\right) \times e^{\left(\frac{\sigma_i}{\mathrm{u} \times A}\right) \times \rho_a \times d_a} = \sum_{i=1}^{13} \left(\frac{E_i}{E_T}\right) \times e^{\left(\frac{\sigma_i}{\mathrm{u} \times A}\right) \times \rho_b \times d_b} = T_b$$

If we assume the same concrete composition, both A and σ_{total} are the same in both sides of the equation, and the gamma energy spectrum would remain unchanged, so it is obtained that:

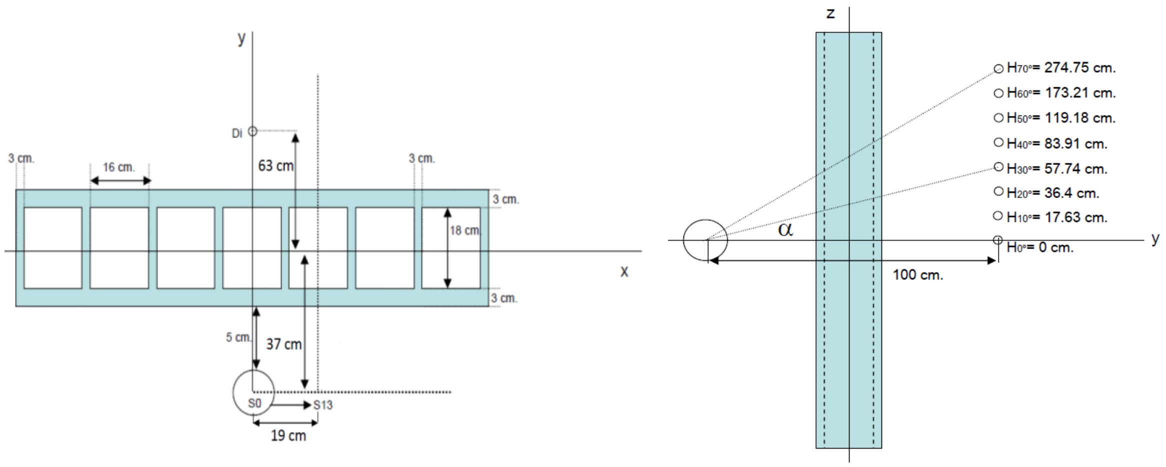
$$d_a = \frac{\rho_b \times d_b}{\rho_a}$$

Therefore, the concrete shielding already analyzed (ρ_a = 2.2 g/cm^3, d_a = 30 cm) produces the same transmission factor that (r_b = 1.0 g/cm^3, d_b = 66 cm) and (r_b = 3.5 g/cm^3, d_b = 18.86 cm).

PRACTICAL SCENARIO

The new facility building is going to be made of concrete blocks of a complex geometry for CAS calculations, due to internal voids as shown in Figure 7. The source S0 is modeled at fixed distance from the concrete wall center, and it is moved along the x-axis to cover the different paths followed by the gamma radiation through the block regions. Eight point detectors D_i are placed at different heights H_i along z-axis to take into consideration how dose rate changes because of different source-detector angles α.

Figure 7



Through MAVRIC calculations and taking into account expression 7, we obtain different transmission factors for every source position and angle. Expression (0) gives us the effective SAC detector coverture radius.

REFERENCES

1. An approximate Method to Calculate 12 Rad Zone, A, Blanchard, D, Biswas and R, Bartholomay, Westinghouse.
2. Health Physics and Radiological Health, Thomas E, Jonson and Brian K,Birky,

Criticality Safety Assessment for As-loaded Spent Fuel Storage and Transportation Casks*

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ABSTRACT

The final safety analysis report (FSAR) or safety analysis report (SAR) for a particular spent nuclear fuel (SNF) cask system documents the bounding models and calculations used to demonstrate that a system meets the regulatory requirements under all normal, off-normal, and accident conditions of SNF storage, and normal and accident conditions of transportation. FSAR/SAR calculations and approved content specifications are intended to be bounding in nature to certify cask systems for a variety of fuel characteristics with simplified SNF loading requirements. Therefore, loaded cask systems tend to have excess and uncredited margins (i.e., the difference between the licensing basis and the as-loaded calculations). These uncredited margins can be quantified by using more detailed canister-specific evaluations that credit the actual as-loaded cask inventory. This paper summarizes an assessment of canister-specific, as- loaded criticality margins for SNF stored in dry casks (a total of 206 as-loaded casks were analyzed) at seven reactor sites including six pressurized water reactor (PWR) sites and one boiling water reactor (BWR) site. The calculated k_{eff} margin typically varies from 0.05 to almost 0.30 Δ k_{eff} for the seven selected reactor sites. The results demonstrate that some loaded casks have significant uncredited safety margins.

Introduction

The regulations for dry cask storage (10 Code of Federal Regulations [CFR] part 72 [1]) and transportation (10 CFR part 71 [2]) require that the spent nuclear fuel (SNF) systems remain subcritical for all normal, off-normal, and accident conditions. The United States Nuclear Regulatory Commission (US NRC) standard review plans (SRPs) [3, 4] specify that the subcriticality of a system should be interpreted as having a neutron multiplication factor, or k_{eff} (also referred to as reactivity in this paper) below 0.95. This k_{eff} should take into account validation bias and bias uncertainty, manufacturing tolerances, and any other applicable biases and uncertainties. SNF cask systems are designed and evaluated for the approved contents defined in the Certificate of Compliance (COC), which specifies bounding (enveloping) fuel characteristics (e.g., fuel type, fuel dimension, initial enrichment, and discharge burnup). The bounding fuel characteristics for a system are developed to estimate the upper limit of k_{eff} (i.e., $k_{eff} < 0.95$), taking into account all the applicable biases and uncertainties. In practice, because of the diversity in the discharged SNF available for loading (e.g., wide variations in SNF assembly burnup values, initial enrichments, and discharge dates), cask systems are typically loaded with assemblies that satisfy the bounding fuel characteristics as defined in the COC with some amount of unquantified and uncredited safety margin.

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This paper summarizes an assessment of canister-specific, as-loaded criticality safety margins for SNF stored in dry casks at seven reactor sites, henceforth referred to as sites A, B, C, D, E, F, and G. While sites A, B, C, D, and E are decommissioned nuclear reactor sites, sites F and G are operating reactor sites. The seven selected sites include six pressurized water reactor (PWR) sites and one boiling water reactor (BWR) site (Site E). Table 1 summarizes the attributes of the analyzed reactor sites.

Table 1: Reactor Site

Reactor site	**A**	**B**	**C**	**D**	**E**	**F**	**G**
Decommissioned	✓	✓	✓	✓	✓		
Operating						✓	✓
PWR	✓	✓	✓	✓		✓	✓
BWR					✓		

In this paper, the inherent criticality safety margin for a loaded cask is determined through comparison with a reference licensing basis calculation. The results demonstrate that significant uncredited criticality margins are present for some loaded casks. The uncredited criticality margin could potentially be used to offset (1) uncertainties associated with postulated effects of system aging beyond the initial cask certification period (i.e., 20 years) and transportation thereafter, and (2) potential increases in SNF system reactivity over a repository performance period (e.g., 10,000 years or more) as the system undergoes degradation and hypothetical changes in geometric configuration. The primary objective is to demonstrate the inherent safety margins in the as-loaded SNF casks and prevent/reduce repackaging of the fuel assemblies into different systems prior to transportation and or disposal. Application of the uncredited margin for disposal criticality evaluation (post-closure criticality) is discussed in a companion paper submitted to this workshop.

The significant amounts of uncredited criticality margin associated with actual commercial SNF cask loading presents a potential opportunity to take an alternative approach to demonstrating subcriticality of SNF storage and transportation casks in compliance with regulatory requirements. As with the approach now used to demonstrate shielding safety of SNF storage systems in accordance with 10 CFR Part 72, the alternative approach could apply a representative criticality analysis during the generic licensing phase. This would be supplemented by site-specific criticality analyses prior to loading based on the as-loaded configurations to demonstrate regulatory compliance. This proposed approach is also discussed in a companion paper submitted to this workshop.

Computer Codes and Analysis Method

A criticality calculation is performed to quantify the k_{eff} of as-loaded SNF systems. Credit taken for the reduction in reactivity resulting from fuel burnup is commonly referred to as burnup credit. Burnup credit criticality safety analysis for SNF in storage and transportation systems requires that isotopic number densities for fuel assemblies be determined by applying assembly- specific irradiation histories. This depletion calculation is followed by a cask criticality evaluation, which uses the isotopic number densities of the fuel from the depletion calculation to determine the system k_{eff}.

A depletion calculation simulates the fuel assembly irradiation history within the reactor to quantify nuclide concentrations at discharge. A decay calculation determines the change in nuclide concentration as a function of time. The SCALE [5] code system is used for the as- loaded criticality assessment and provides the required computer codes and sequences for running depletion and decay calculations using the TRITON sequence and ORIGEN module.

The TRITON two-dimensional (2D) depletion sequence [5] is used to perform depletion calculations that generate cross section libraries for generic assembly/reactor–specific classes and a range of fuel operating conditions. The cross section libraries are then used by ORIGEN to calculate nuclide inventories of SNF assemblies with specific irradiation characteristics. The TRITON 2D depletion calculation sequence employs CENTRM for multigroup cross section processing, NEWT for 2D discrete-ordinates transport calculations, and ORIGEN for depletion and decay calculations. The SCALE depletion and decay sequences have been validated extensively [6, 7].

Conservative irradiation parameters [8] are applied to estimate the upper limit of the neutron multiplication and are used in this paper for criticality evaluations. Depletion modeling parameters are presented in Table 2, and the following two assumptions are used for BWR (Site E) fuel depletion and decay calculations:

- Gadolinium (1 wt% Gd2O3 in 2 rods) is conservatively ignored in the 6 × 6 Site E assemblies. Control blade insertion is assumed during the depletion period.
- Moderator density is assumed to be 0.49 gm/cc for Site E fuel assemblies, which corresponds to 35% core average void fraction.

The SCALE CSAS6 [5] criticality analysis sequence is used to perform criticality calculations for a loaded fuel cask using the KENO-VI Monte Carlo code with the continuous energy ENDF/B-VII.0 cross section library to determine the k_{eff}. KENO-VI has been validated and used extensively for criticality safety evaluations [9].

The uncredited criticality margins are evaluated by employing a comprehensive and integrated data and analysis tool—Used Nuclear Fuel-Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS)—which was developed at Oak Ridge National Laboratory [10] through a collaborative effort among multiple national laboratories and industry participants. UNF-ST&DARDS employs the SCALE analysis sequences and modules discussed previously. UNF-ST&DARDS executes inventory calculations for each unique assembly design (e.g., Westinghouse 17 × 17 optimized fuel assembly [OFA]), accounting for initial enrichment, burnup, and age. It generates explicit criticality models for each fuel assembly and canister with the appropriate canister loading pattern. Note that a prereleased version of SCALE 6.2, which is under development, is used for decay and continuous energy criticality calculations.

Table 2: Depletion Modeling Parameters

Parameter/Reactor Type	B&W[a] PWR	W[b] PWR	CE[c] PWR
Fuel rod mixture	UO2	UO2	UO2
Fuel density (g/cm3)[d]	10.741	10.741	10.741
Specific power (MW/MTU)	30	30	30
Fuel temperature (K)	1144.1	1157	1171.6
Moderator temperature (K)	588.7	598.2	598.55
Moderator density (g/cm3)	0.6905	0.6668	0.6656
Soluble boron concentration (ppm)	1000	1000	1000
Burnable absorber exposure	All assembly guide tubes contain burnable poison rods fully inserted throughout irradiation time	All assembly guide tubes contain Pyrex rods fully inserted throughout irradiation time	None
Type of absorber	Al2O3-B4C	SiO2-B2O3	N/A

[a] B&W = Babcock and Wilcox [c] Combustion Engineering
[b] W = Westinghouse [d] 98% of UO2 theoretical density

Criticality calculations within UNF-ST&DARDS are performed applying 18 node axial burnup profiles, along with 12 actinides and 16 fission product isotopes. Burnup-dependent axial profiles are used for the PWR criticality analyses [11]. A uniform burnup profile is employed for the BWR fuel assemblies at Site E. A preliminary comparison between the bounding BWR burnup profiles provided in Ref. 12 and a uniform profile shows that the uniform profile results in slightly higher reactivity for the Site E fuel assemblies. Additionally, Site E irradiated fuel assemblies are low burnup assemblies (~ 1.3 to 23 gigawatt days per metric ton of uranium [GWd/MTU]). Uniform axial distribution is typically bounding for low burnup assemblies [11]. The isotope set, credited in the criticality calculations, is selected based on the burnup credit isotopes recommended by NUREG/CR-7108 and -7109 [13, 14]. The credited isotopes are listed in Table 3.

Table 3: Isotope Set: 12 Actinides + 16 Fission Products

Actinides					
^{234}U	^{235}U	^{236}U	^{238}U	^{238}Pu	^{239}Pu
^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am	^{243}Am	^{237}Np
Fission products					
^{95}Mo	^{99}Tc	^{101}Ru	^{103}Rh	^{109}Ag	^{133}Cs
^{143}Nd	^{145}Nd	^{147}Sm	^{149}Sm	^{150}Sm	^{151}Sm
^{152}Sm	^{151}Eu	^{153}Eu	^{155}Gd		

Description of Evaluated Sites

The dry storage systems used in Sites A, B, C, D, E, F, and G are briefly described below.

- **Site A** is a decommissioned PWR site. For Site A, 60 as-loaded canister systems from NAC International were analyzed [15]. Each canister can contain up to 24 PWR fuel assemblies. Criticality control in the Site A PWR basket is mainly achieved by a flux trap design. Fuel assemblies inside the canister are maintained in place by the fuel tubes. Neutron absorber sheets are attached on the four sides of the fuel tubes. The gaps between the neutron absorber sheets facing each other constitute the flux trap.
- **Site B** is a decommissioned PWR site. For Site B, 39 as-loaded canisters from NAC International were analyzed [16]. Site B has two fuel baskets: one for a 26 assembly configuration, and one for a 24 assembly configuration. These two baskets are identical except that the top weldment of the 24-assembly configuration consists of 24 fuel tube penetrations. The 24-assembly basket is designed to accommodate higher enriched fuel assemblies than the 26-assembly basket. Criticality control in both baskets is primarily achieved by flux trap design.
- **Site C** is a decommissioned PWR site. For Site C, 20 NUHOMS® systems from Transnuclear [17] that are as-loaded dry-shielded canisters (DSCs) were evaluated. These DSCs can accommodate 24 intact PWR assemblies. Fuel assemblies inside the DSCs are maintained in place by thin-wall guide sleeves. Each guide sleeve is made from stainless steel, with neutron absorber panels attached to each side of the sleeve that faces another assembly. The gaps between the neutron absorber panels facing each other form the flux- trap design.
- **Site D** is a decommissioned PWR site. For Site D, 33 multipurpose canisters (MPCs) — MPC-24E/EF canisters from Holtec International [18] —were analyzed. MPC-24E/EF canisters can

accommodate up to 24 PWR fuel assemblies. Site D baskets are made of stainless steel, and primary criticality control is achieved by flux trap design.

- **Site E** is a decommissioned BWR site. Five as-loaded canisters from Site E were evaluated. Site E uses Holtec's canister for storing their discharged SNF. This system can accommodate 80 fuel assemblies per canister, up to 40 of which may be damaged. The canister uses single MetamicTM neutron absorber panels between storage locations to ensure criticality control [18].
- **Site F** is an operating PWR site. Site F employs the NAC system [15] also used by Site A, and 23 as-loaded canisters from Site F were assessed.
- **Site G** is also an operating PWR site. Site G uses the Holtec International's MPC-32 canisters [18]. From Site G, 26 MPCs were analyzed. The MPC-32 is an all stainless steel canister that can accommodate 32 PWR assemblies and uses an egg-crate basket design with a single neutron absorber panel between adjacent assemblies.

As-Loaded Criticality Analysis

The as-loaded criticality calculations replicate representative conditions documented in the FSARs/SARs to the extent applicable, but with specific as-loaded fuel to determine the inherent uncredited criticality margins. The SRPs provide guidance that the subcriticality of a system is demonstrated by a k_{eff} value below 0.95, taking into account validation bias and bias uncertainty, manufacturing tolerances, and any other applicable biases and uncertainties. However, in performing the as-loaded criticality analyses, a reference criticality model and corresponding reactivity calculation are first established using FSAR/SAR safety case and design basis assembly information without including applicable biases and uncertainties. This model is then applied in calculating a reactivity value based on as-loaded fuel characteristics. The as-loaded reactivity value is then compared to the design basis assembly reactivity value to determine the uncredited criticality safety margin. An overview of the approach is presented in Figure 1. This approach is used in the absence of determining all the applicable biases and uncertainties of the as-loaded casks, as is exercised in the standard licensing analyses. The main sources of uncredited margin (relative to licensing design basis analyses) investigated in this paper include:

- burnup credit for 28 actinide and fission product nuclides previously demonstrated to exhibit significant effect on fuel reactivity, and
- use of actual as-loaded canisters, including modeling of actual assembly-specific attributes including initial enrichments, loading pattern, burnup, and post-irradiation cooling time.

Detailed canister models were developed for Sites A, B, C, D, E, F, and G. Criticality calculations used canister models surrounded by water for optimal reflection. Overpacks typically surround the canisters, but were neglected in the criticality analysis. The criticality models were based on nominal dimensions and centered basket components. Note that the above two simplifications have negligible impact on the uncredited criticality margin, as the same KENO-VI canister model was used for the design basis and as-loaded calculations with design basis and as-loaded fuel characteristics, respectively, as shown in Figure 1. Bounding assembly models (e.g., fresh design basis assembly) as determined in the FSAR/SAR were applied for irregular assemblies (e.g., assemblies with missing fuel rods, damaged fuel assemblies). UNF-ST&DARDS generates explicit criticality models by modeling each fuel assembly in the canister with the appropriate fuel assembly irradiation characteristics identified from canister-specific loading maps. Table 4 shows a representative loading map (assembly location numbering scheme is shown in Figure 2). Table 4 also presents the licensing basis fuel assembly to denote the source of uncredited criticality margin. Figure 2 illustrates the horizontal cross section of the Site C canister used in the criticality margin calculations, and it also shows the detailed modeling approach used for criticality analysis.

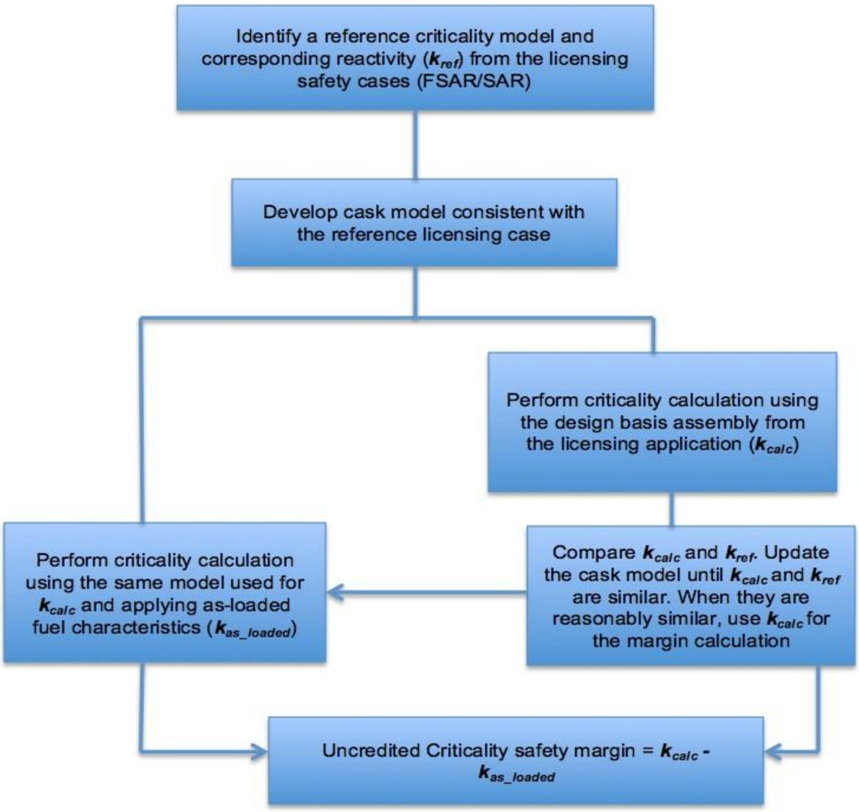

Figure 1: Approach for evaluating the criticality safety margin of the as-loaded casks. Uncredited criticality safety margin is defied as Δ k_{eff} = kcalc – k_{as_loaded} .

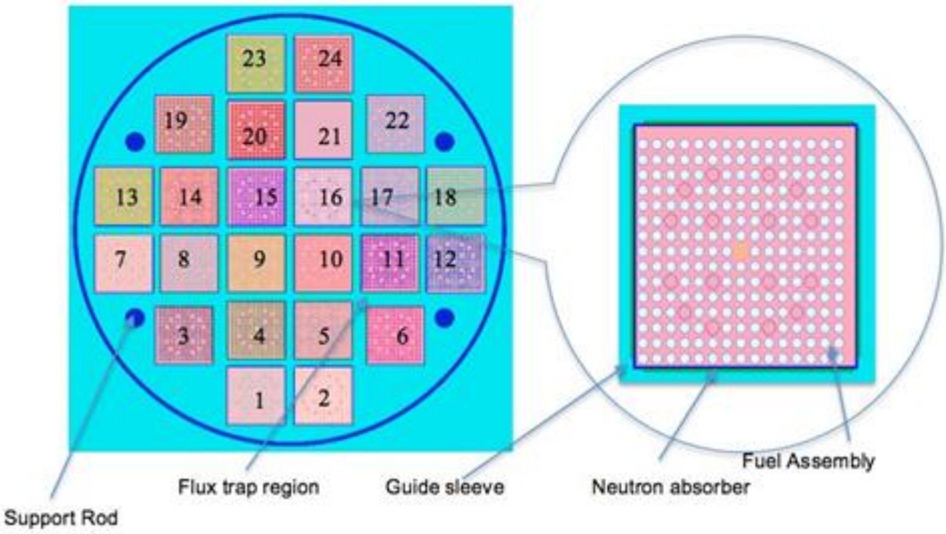

Figure 2: Horizontal cross sectional view of the Site C canister as modeled in KENO-VI.

The calculated as-loaded k_{eff} results, which range from the loading date out to the year 2100, are presented in Figure 3, which shows a small change in cask as-loaded k_{eff} values as a function of time. This change is small mainly because (1) the casks are loaded with assemblies with varying discharge times, and (2) assemblies in the damaged fuel cans (if present in the cask) are modelled as unirradiated. Estimated uncredited criticality margin results for seven selected sites are illustrated in Figure 4, which shows that the uncredited criticality safety margin (Δk_{eff}) typically varies from 0.05 to almost 0.30 Δk_{eff}. However, k_{eff} margin is below 0.05 Δk_{eff} for one of the Site E casks, which contains damaged fuel cans in 38 locations inside the canister with damaged fuel assemblies and or debris. The low uncredited margin values for all sites except Site G are mainly an attribute of the damaged fuel assemblies in the loaded canisters, which are modeled as unirradiated for conservatism. Additionally, a few of the lower criticality margins are due to high enrichment and low burnup assemblies in the canisters. Because burnup credit was already used for licensing the Site G canisters [18], as-loaded criticality analyses applying as-loaded fuel characteristics do not provide large reactivity margin for Site G.

Table 4: Representative Loading Map

Assembly Average Burnup (MWd/MTU)	**Initial Enrichment (wt % 235U)**	**Assembly Location**	**Discharge Year**	**Licensing Basis**
10 000	3.062	1	1987	
28 000	3.143	2	1987	Fresh 3.43 wt% enriched B&W 15 x15 assembly was assumed for all locations
25 397	2.673	3	1978	
25 736	2.676	4	1978	
32 000	3.210	5	1985	
21 000	3.144	6	1987	
20 000	3.190	7	1987	
31 914	2.990	8	1980	
35 360	2.670	9	1980	
38 016	3.200	10	1983	
25 706	2.671	11	1978	
16 998	2.007	12	1977	
29 320	2.998	13	1980	
28 420	2.664	14	1978	
36 545	3.190	15	1983	
35 311	2.993	16	1980	
27 611	2.671	17	1978	
10 000	3.060	18	1987	
10 000	3.056	19	1987	
32 000	3.041	20	1987	
34 000	3.041	21	1985	
28 054	3.188	22	1981	
24 804	3.042	23	1983	
21 000	3.141	24	1987	

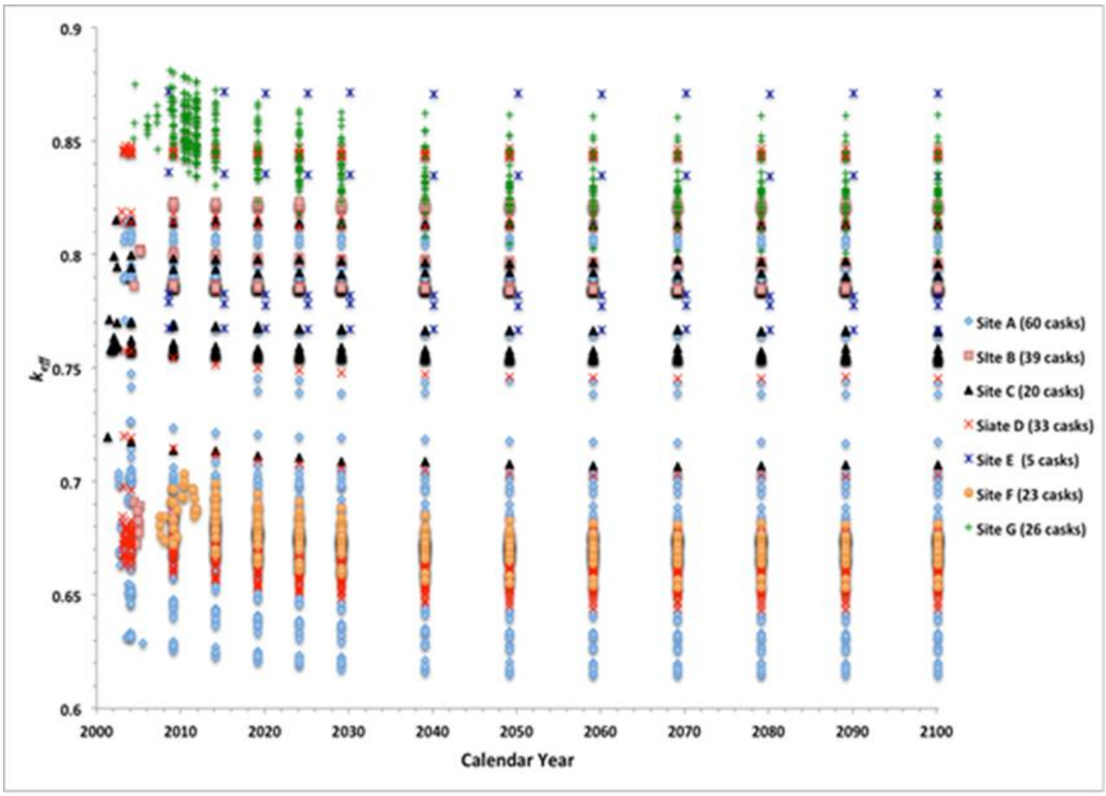

Figure 3: Calculated k_{eff} results for as-loaded casks at Sites A, B, and C as a function of time.

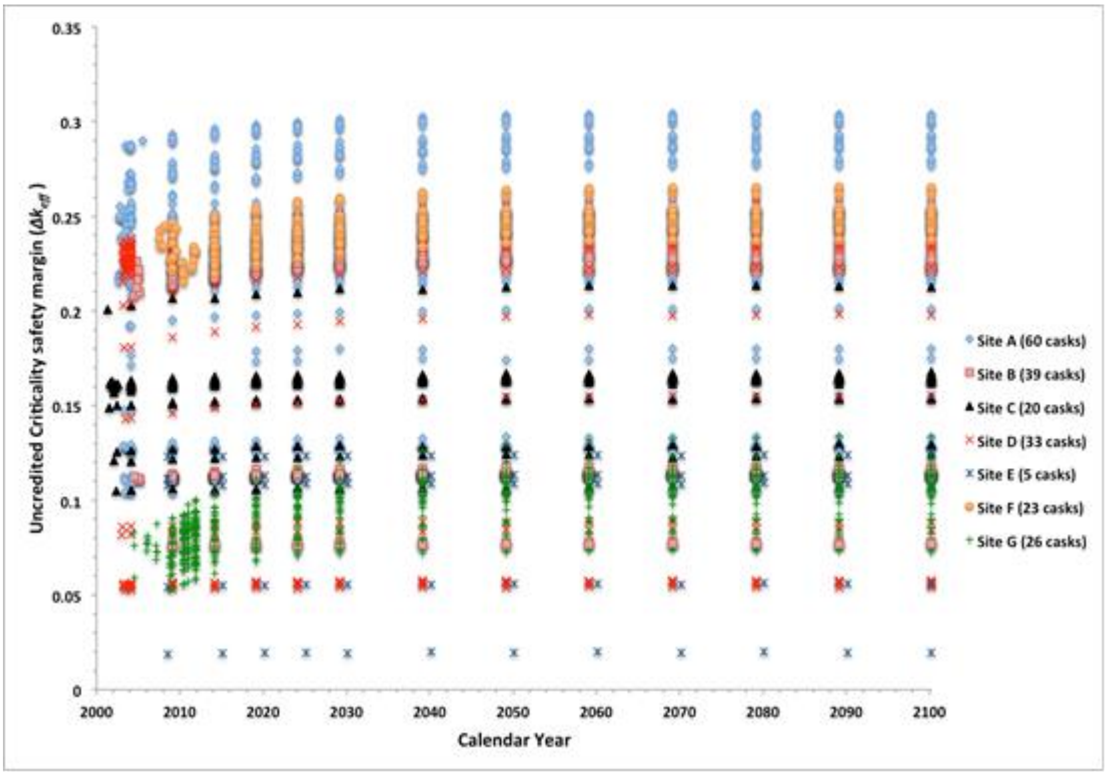

Figure 4: Estimates of available uncredited criticality safety margin at Sites A, B, and C.

Conclusion

Canister-specific as-loaded criticality calculations have been performed for seven reactor sites (total 206 as-loaded casks). This paper demonstrates that most of the as-loaded casks have substantial uncredited safety margins available that could potentially be used to offset postulated safety-related performance losses and uncertainties as systems age. The calculated k_{eff} margin typically varies from 0.05 to almost 0.30 Δ k_{eff}. In this context, it is important to note that Ref. 19 indicates that the maximum increases in k_{eff} for the PWR and BWR cask systems are nearly 4% and 2.4%, respectively, for potentially credible fuel failure configurations that could occur during extended storage and subsequent transportation. Therefore, the inherent criticality safety margins available in the majority of the analyzed casks could easily accommodate potential reactivity increases from fuel failure during extended storage and subsequent transportation. It is important to recognize that the inherent criticality safety margin associated with actual SNF loading configuration is reduced if full (actinides and fission products) burnup credit is used in the licensing application.

Additionally, cask-specific, as-loaded analyses require a large number of calculations using various analysis tools and data. Therefore, a coordinated approach that provides integration between the inventory data and analyses tools is necessary to ensure precision and quality control of calculations. UNF-ST&DARDS has been developed to perform a large volume of as-loaded SNF safety analyses to assess the actual characteristics of loaded casks during long-term storage and subsequent transportation in a variety of dry storage systems with minimal user interaction. The results presented in this report demonstrate that UNF-ST&DARDS can perform the volume of detailed calculations necessary to assess the actual condition of as-loaded dry storage casks based on its integrated data and analysis capabilities.

References

1. Title 10, Code of Federal Regulations, Part 72, 77 FR 39908, Jul. 6, 2012.
2. Title 10, Code of Federal Regulations, Part 71, 77 FR 34204, Jun. 11, 2012.
3. Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility, NUREG-1536, Rev. 1, Office of Nuclear Material Safety and Safeguards, July 2010.
4. Standard Review Plan for Transportation Packages for Spent Nuclear Fuel, NUREG- 1617, Office of Nuclear Material Safety and Safeguards, January 2000.
5. "SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design," Oak Ridge: ORNL/TM-2005/39 Version 6.1, Radiation Safety Information Computational Center at Oak Ridge National Laboratory (2011).
6. G. ILAS et al., "Validation of New Depletion Capabilities and ENDF/B-VII Data Libraries in SCALE," Annals of Nuclear Energy, 46, 43–45 (2012).
7. G. ILAS et al., "Validation of ORIGEN for [Light Water Reactor] LWR Used Fuel Decay Heat Analysis with SCALE," Nuclear Engineering and Design, 273, 58-67 (2014).
8. G. Radulescu et al., A Comprehensive Database and Automation Tool for Used Nuclear Fuel Safety Analyses, ORNL/LTR-2012/469, Oak Ridge National Laboratory, Oak Ridge, TN, June 2012.
9. W. J. MARSHALL et al., "Validation of Criticality Safety Calculations with SCALE 6.2," Proceedings for Nuclear Criticality Safety Division Topical Meeting (NCSD 2013), American Nuclear Society, September 29–October 3, 2013, Wilmington, North Carolina.
10. K. BANERJEE et al., "Integrated Data and Analysis Tool for Used Nuclear Fuel Management," Transaction of American Nuclear Society (ANS) - Winter Meeting, 111, 338–341, November 9–13, 2014, Anaheim, California.

11. J. C. WAGNER et al., "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses," NUREG/CR-6801 (ORNL/TM-2001/273), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tenn. (March 2003).

12. OCRWM, BWR Axial Profile, CAL-DSU-NU-000005 REV 00A, September 2004.

13. G. RADULESCU et al., "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses–Isotopic Composition Predictions," NUREG/CR-7108 (ORNL/TM-2011/514), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tenn. (April 2012).

14. J. M. SCAGLIONE et al., "An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses–Criticality (k_{eff}) Predictions, "NUREG/CR- 7109 (ORNL/TM-2011/514), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tenn. (April 2012).

15. NAC-UMS® Final Safety Analyses Report, Rev. 3, US NRC Docket No. 72-1015, NAC International, January 2004.

16. NAC-STC Safety Analysis Report, Rev. 15, US NRC Docket No. 71-9235, March 2004.

17. Transnuclear, Inc., Multi-Purpose Cask, Rev. 17, Safety Analysis Report NUHOMS®- MP187, Transnuclear, Inc., Fremont, Calif., July 2003.

18. Holtec International, Storage, Transportation, and Repository Cask System Safety Analysis Report, Revision 10, US NRC Docket No. 71-9261, August 2003.

19. W. J. Marshall and J. C. Wagner, Consequences of Fuel Failure on Criticality Safety of Used Nuclear Fuel, ORNL/TM-2012/325, Oak Ridge National Laboratory, Oak Ridge, Tenn., September 2012.

Validation Study for Crediting Chlorine in Criticality Analyses for US Spent Nuclear Fuel Disposition[1]

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ABSTRACT

Spent nuclear fuel (SNF) management practices in the United States rely on dry storage systems that include both canister-based and cask systems. The United States Department of Energy Used Fuel Disposition Campaign is examining the feasibility of direct disposal of dual-purpose (storage and transportation) canisters (DPCs) in a geological repository. One of the major technical challenges for direct disposal is the ability to demonstrate the subcriticality of the DPCs loaded with SNF for the repository performance period (e.g. 10 000 years) as the DPCs and their content undergo degradation over time. Specifically, groundwater ingress into the DPC (i.e., flooding) could allow the system to achieve criticality in scenarios where the neutron absorber plate in the DPC basket has degraded. However, as was shown by Banerjee et al., impurities in the groundwater provide noticeable reactivity reduction for these systems [1]. For certain amounts of impurities in the groundwater, subcriticality can be demonstrated even for DPCs with complete degradation of the neutron absorber plates or a degraded fuel basket configuration. It has been demonstrated that chlorine is the leading impurity with significant neutron absorption in the water that is available in reasonable quantities for the deep geological repository media under consideration. This paper investigates the available integral experiments worldwide that could be used to validate DPC disposal criticality evaluations including credit for chlorine. An in-depth analysis of the available critical experiments and initial validation is presented.

INTRODUCTION

The current spent nuclear fuel (SNF) management strategy in the United States includes reliance on dry storage systems that include both canister-based and cask systems to allow continued operation of the nation's nuclear fleet. Approximately 2 000 MT of SNF is being placed in dry storage per year. Hence, the US Department of Energy Used Fuel Disposition Campaign is examining the feasibility of directly disposing of these dual-purpose (storage and transportation) canisters (DPCs) in a geological repository. Past studies regarding the feasibility of direct disposal have concluded that while possible, demonstrating criticality control over the disposal time period is a challenge. The primary challenge is demonstrating the continued efficacy of the criticality control features as the system degrades over time (e.g. 10 000 years) and groundwater enters the canister. Specifically, groundwater ingress into the DPC (i.e. flooding) could allow the system to achieve criticality in scenarios where the neutron absorber plates between the assemblies in the DPC basket have degraded. However, as was shown by Banerjee et al., impurities in the

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groundwater may have high enough neutron absorption properties that preclude criticality from being achieved [1]. For certain amounts of impurities in the groundwater, subcriticality can be demonstrated even for DPCs with complete degradation of the neutron absorber plates or a degraded fuel basket configuration. It has been demonstrated that the leading impurity with significant neutron absorption in the water that occurred in reasonable quantities for the deep geological repository media under consideration is chlorine [1]. This paper investigates the available integral experiments worldwide that could be used for validation of DPC disposal criticality evaluations including credit for chlorine.

A similar problem was studied by a group from Gesellschaft fuer Anlagen- und Reaktorsicherheit (GRS) mbH in 2013 and presented at ANS NCSD 2013 [2]. However, the geological repository design and the application systems of this study differ from those of the GRS. In summary, the GRS study did not find any evaluated critical experiments that were similar enough to their application to allow them to validate the crediting of chlorine within the German regulatory system.

BACKGROUND

Two specific hypothetical configurations were considered as the application models to be covered by the validation study. Both were a 32 SNF assembly capacity DPC model consisting of stainless steel canister and basket materials with representative 17x17 PWR SNF. Both cases were modeled with 20 gigawatt-days (GWd) per metric ton of uranium (MTU) burnup as representative burnup for SNF. In the first case, the neutron absorber plate material of the DPC was modeled as moderating material (i.e., infiltrated groundwater) to account for degradation and separation of the absorbing material from between fuel assemblies. In the second case, the fuel basket structure and neutron absorber plates were modeled as moderating material to account for additional potential degradation to the canister basket. (Loss of basket materials from between fuel assemblies is considered conservative with respect to criticality evaluations.) An in-depth description and schematics of the two application cases discussed here are available in Ref. 1. The two application models were selected such that the amount of chlorine in the models resulted in a slightly supercritical configuration. The chlorine concentration in the two application models sets a target concentration that would be desirable in the critical experiments used for validation. The ultimate goal of selecting a set of integral experiments is to match the bias of the experiments and applications as close as possible. Table 1 summarizes the two models.

Table 1. Summary of the two application models

Case	No absorber	Degraded basket
Burnup	20 GWd/MTU	20 GWd/MTU
Cl concentration	25 000 ppm (mg/L)	50 000 ppm (mg/L)
k_{eff}	1.00944	1.05222
Cl worth	0.05113 Δk	0.10365 Δk

Chlorine has only two stable isotopes, 75.76% ^{35}Cl and 24.24% ^{37}Cl. The application systems both exhibit a thermal neutron flux. Comparing the thermal capture cross sections of the two isotopes (^{35}Cl: 43.60 b, ^{37}Cl: 0.432 b), it is obvious that only the ^{35}Cl is important for neutron absorption.

The nuclear data library considered in this study is the ENDF/B-VII.1 [3]. The resolved resonance region evaluation for both isotopes of chlorine, which also governs the thermal energy region, was completed in 2003 by an Oak Ridge National Laboratory team led by R. Sayer [4]. The evaluation was subsequently updated in 2007. The goal of the 2003 evaluation was to address several deficiencies in the previous evaluation for chlorine as noted in Ref. 5 in order to support systems where chlorides are present. However, it is important to note that the updated resolved resonance evaluations were never benchmarked on a set of integral experiments. Figure 1 presents the total and elastic scattering cross sections of ^{35}Cl.

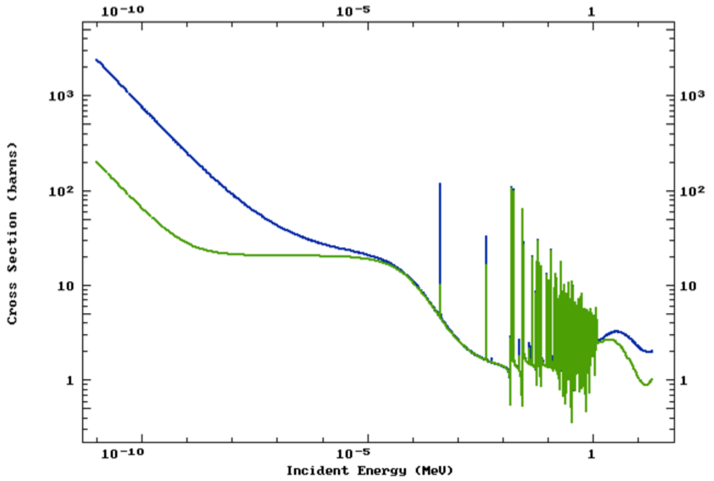

Figure 1. Total (blue) and elastic scattering (green) cross sections for ^{35}Cl

ANALYSIS

Similarity Coefficients

A portion of the computations for this project was done with the SCALE 6.1 code package [6]. In particular, the codes, KENO, TSUNAMI-3D, TSURFER and AMPX were used. A modification of the code SAMINT [7] was used to isolate only the effect of the single chlorine isotope for some of the parameters traditionally computed by TSURFER. As will be discussed later, this is the uncertainty propagated to the k_{eff} of the applications due to the chlorine covariance data and the similarity coefficients of the experiments compared to the applications.

Two sets of covariance data for ^{35}Cl were used in this work. First, the R. Sayer et al. chlorine resolved resonance region evaluations in ENDF/B-VII.1 contained resolved resonance parameter covariance data. These resonance parameter covariance data were processed with the AMPX cross-section processing package (distributed with the SCALE code package beginning with SCALE 6.2) in three different ways:

- 44-group covariance data collapsed with a constant flux suitable for use with a wide range of integral experiments that do not all have a flux spectrum, similar to an approximate light water reactor (LWR) spectrum. This is the standard covariance collapse technique used to calculate the 44-group covariance data available in SCALE.
- 44-group covariance data collapsed with a LWR flux suitable for calculations with integral experiments with flux spectra similar to an approximate LWR spectrum. In particular, this covariance set is useful for propagation of uncertainty calculations for the two applications considered in this study.
- 238-group covariance data calculated with an approximate LWR spectrum. This covariance data set is useful for higher fidelity propagation of uncertainty calculations for the two application systems.

The ^{35}Cl covariance data distributed with the SCALE package do not include the same covariance data as discussed above for ENDF/B-VII.1. The default covariance data for ^{35}Cl are 44-group, low-fidelity covariance data from the Brookhaven - Los Alamos - Oak Ridge (BLO) project [8; 6, Sect. M19]. These low-fidelity covariance data were generated under the BLO project with the objective of providing estimated covariance data for all isotopes in the ENDF database that did not have covariance data. The SCALE covariance data were also used in this research to evaluate the difference in the results obtained with different covariance evaluations. Figure 2 compares three different representations of the relative uncertainty on the total cross section of ^{35}Cl. Note that the BLO uncertainty is slightly lower in the thermal energy region than the ENDF evaluation. While the uncertainty is larger for the SCALE data in the epithermal and resonance region, the sensitivity of the applications and experiments drops off rapidly above the thermal energy region.

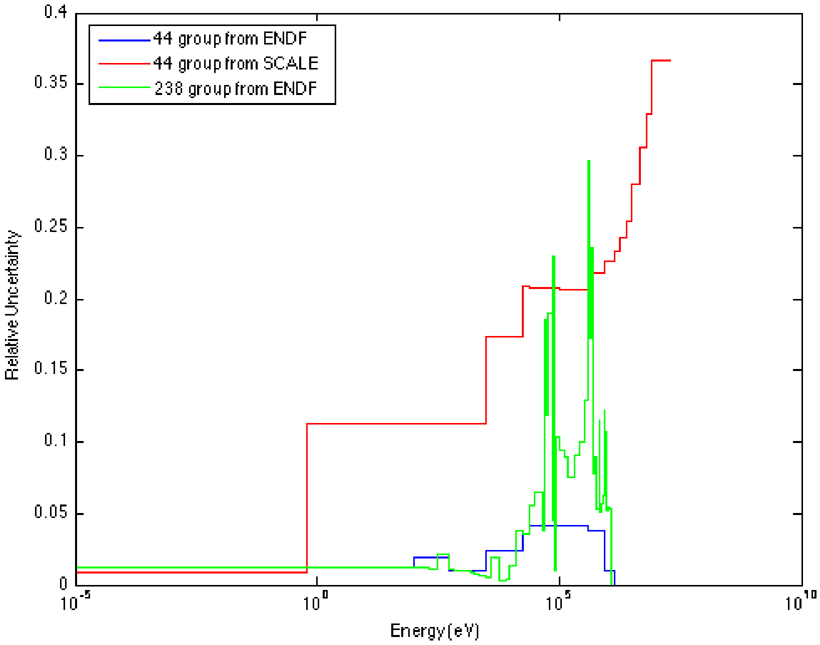

Figure 2. Comparison of the relative uncertainty of the total cross section of ^{35}Cl evaluated three different ways

Six critical configurations that could be helpful in validating the capture cross section of chlorine in the thermal energy region were identified as part of the French MIRTE 2.2 program [9]. Of the six configurations, two contain NaCl solution (conc. = 300 g/l), and four have cruciform PVC separators in the core. However, these are commercial, proprietary experiments and are not freely available.

The *International Handbook of Evaluated Reactor Physics Benchmark Experiments* [10] was consulted, but no configurations with chlorine sensitivities similar to the two applications were identified. Outside of the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (IHECSBE) [11], no other source was found that contained potentially applicable evaluated critical experiments with chlorine sensitivities similar to the applications in this research.

A total of 141 critical configurations containing chlorine were identified in the 2013 edition of IHECSBE. Despite the large number of prospective benchmarks, very few have a similar chlorine sensitivity profile shape and magnitude as the application systems of this study.

The sensitivity profiles of k_{eff} for the different chlorine reactions as a function of neutron energy were calculated for the two application models using TSUNAMI-3D from SCALE 6.1. Figure 3 presents the sensitivity profiles for the total cross section of chlorine for the two application systems, as well as for several of the most similar benchmarks.

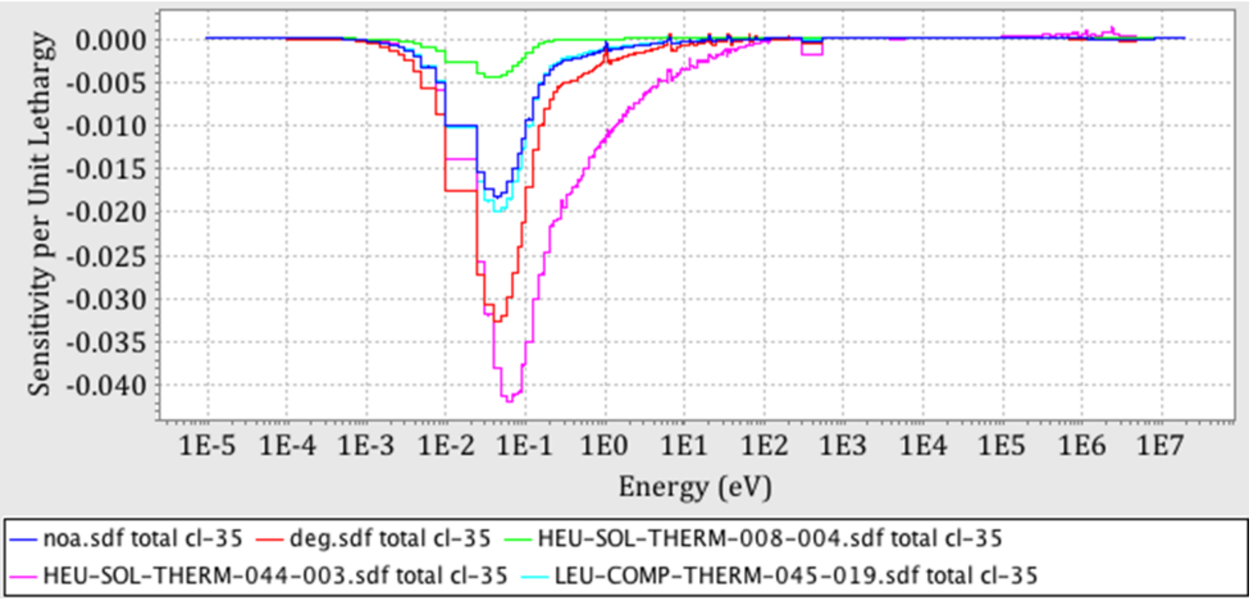

Figure 3. Sensitivity profiles of the k_{eff} for the total cross section of ^{35}Cl as a function of energy. The two application systems are labeled as *noa.sdf* and *deg.sdf*, which represent the no absorber and the degraded fuel basket systems, respectively

The HEU-SOL-THERM (HST)-044-003 system is the only benchmark to have a larger sensitivity for chlorine than the degraded fuel basket application system. Notice, also, that the sensitivity profile of HST-044-003 peaks at a higher energy than the two application systems. While that sensitivity profile has a large magnitude, the shape does not fully resemble that of the two application systems. The LEU-COMP-THERM (LCT)-045-019 benchmarks give an almost perfect match to the no absorber system. Unfortunately, most of the 141 critical benchmarks with chlorine are like HST-008-004 in the sense that they have a very similar shape of the sensitivity profile but a much smaller magnitude. In fact, HST-008-004 is in the top 10 benchmarks when it comes to a quantitative analysis of the similarity between sensitivity profiles.

The chlorine sensitivity profiles for the two application systems were calculated using TSUNAMI-3D. To facilitate a systematic comparison between IHECSBE benchmarks and the two applications in terms of the chlorine, the following three similarity coefficients were defined:

$$C_{ij} = S_i^T C\, S_j /(\delta k_i\, \delta k_j)\,,$$

$$E_{ij} = S_i^T S_j / (|S_i|\, |S_j|)\,, \quad |S_i|=(S_i^T S_i)^{1/2},$$

$$G_{ij} = 1- |S_i-S_j|^2/(|S_i|^2 + |S_j|^2) = S_i^T S_j / (|S|^2),\ |S|^2 = ½(|S_i|^2 + |S_j|^2)\,,$$

where C_{ij}, E_{ij}, and G_{ij} are defined as the similarity coefficients between integral experiments i and j. The variable S_i is the column vector of the sensitivity of experiment i to the multi-group cross sections of a particular chlorine reaction, and the matrix C is the multi-group covariance matrix of that particular reaction. The sensitivities and the covariance matrices were all used in the relative format, and the similarity coefficients were calculated for the total, elastic, and capture cross sections. It is important to note that since the sensitivity variable S is only the sensitivity of the experiment's k_{eff} to chlorine, all three similarity coefficients represent the similarity between the two experiments only in the behavior of the chlorine. The TSURFER code does not report the similarity coefficients for individual nuclides; rather, it reports a single similarity value for all of the nuclear data. For this purpose, the code SAMINT was used to calculate the similarity coefficients only for chlorine.

The typical interpretation of the similarity coefficient C_{ij} is that it is the correlation coefficient between experiments i and j due solely to chlorine nuclear data. The variable δk_i is the uncertainty in k_{eff} due to the uncertainty in the chlorine data. It is used to make C_{ij} a correlation coefficient rather than a covariance. The E_{ij} coefficient is a measure of the similarity of the sensitivity profile shapes for chlorine between the benchmark experiments and the application systems. The G_{ij} coefficient is similar to the E_{ij} coefficient; however, it also accounts for the difference in magnitude between the two sensitivity profiles. Whereas E_{ij} would be equal to unity when one sensitivity profile is a scaled version of another, the G_{ij} similarity coefficient is unity only when the two sensitivity profiles are identical.

The three similarity coefficients discussed above were calculated for each of the 141 benchmarks containing chlorine using the SAMINT code. The results allow for an easy visual analysis of the set of critical benchmarks that could potentially be used to validate the chlorine cross sections. Figure 4 presents the three similarity coefficients calculated for the total, elastic, and capture cross sections of ^{35}Cl against the no absorber application case. All three plots in Figure 4 have the rank of the experiment on the horizontal axis when sorted by the C similarity coefficient for the total cross section.

In Figure 4, a large positive correlation coefficient means that the application system, the no absorber case, is highly correlated to the experiment by the chlorine nuclear data. A value of +1 means complete correlation. On the other hand, a negative correlation coefficient implies that the application and the experiment are anti-correlated through the nuclear data. A value of -1 means complete anti-correlation.

The most important observation from Figure 4 is the small value of the G similarity coefficient for almost all of the 141 benchmarks. This means that there are hardly any critical benchmark experiments that have a sensitivity profile with a magnitude that is similar to the two applications. Even though for the total cross section there are about 100 benchmarks that have very similar sensitivity profile shapes to the applications (indicated by a value of C and E close to unity), less than a dozen of them have large values of G. Table 2 presents all of the available benchmarks with G values larger than 0.4.

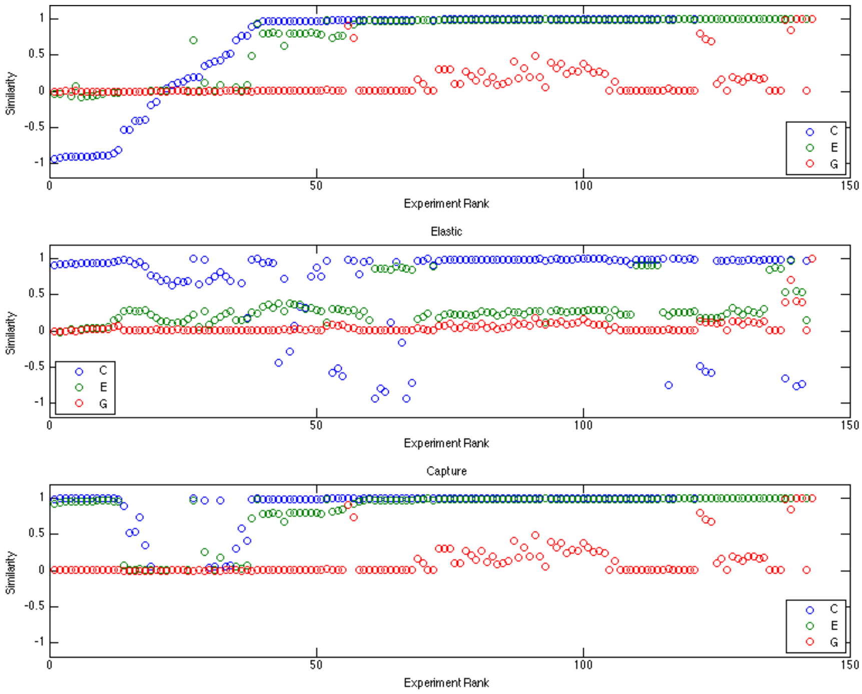

Figure 4. Similarity coefficients C, E, and G for the 141 critical benchmarks containing chlorine. The horizontal axis in all three plots represents the rank of the experiment as sorted by the value of the similarity coefficient C for the total cross section

Table 2. Similarity coefficients for the total cross section for the most applicable benchmark experiments compared to the no absorber case

Experiment	G	C	E	Sensitivity
LCT45-18	0.998	0.999	0.999	0.048
LCT45-19	0.998	0.999	0.999	0.048
LCT45-06	0.989	0.999	0.999	0.052
HST44-02	0.916	0.992	0.917	0.066
UST03-02[a]	0.808	0.999	0.999	0.021
HST44-03	0.740	0.999	0.922	0.135
UST03-04	0.719	0.992	0.999	0.018
UST03-05	0.691	0.999	0.999	0.017
HST08-04	0.488	0.998	0.992	0.010
HST08-12	0.406	0.998	0.991	0.008
LCT45-03	0.401	0.998	0.992	0.008

[a] U233-SOL-THERM (UST)-03-02.

The chlorine similarity coefficients between the two application systems are:

C = 0.99997
E = 0.99983
G = 0.85685

When the similarity coefficients for the benchmark experiments are computed relative to the degraded fuel basket system, the numerical values in Table 2 do change slightly, particularly for *G*. However, the top 11 critical experiments do not change. A plot of the similarity coefficients for the degraded fuel basket system provides similar results to that of Figure 4.

The 11 critical configurations in Table 2 originate from four different experiments: LCT45, HST44, HST08, and UST03. Furthermore, the chlorine content appears as three different materials in the 11 configurations. The chlorine is found in a Plexiglas reflector for the LCT45 and HST08, in PVC rods for HST44, and as a constituent of a paint coating the inside of the solution cylinders in UST03. Based on the chlorine form, it is obvious that none of the experiments have a series of similar configurations where only the chlorine amount changes. All of these factors combine to make validation through traditional trending analysis (regression) difficult. Furthermore, if only the benchmark experiments that have a sensitivity profile for chlorine representative of the application systems are considered, the small sample size results in poor statistics. In this case, neither the normality of the data nor a significantly non-zero trend can be determined.

It is the conclusion of this study that validation through traditional trending analysis is not possible with the current, freely available, evaluated set of critical benchmark experiments. In this case, however, TSURFER analysis is well suited for identifying the level of bias and bias uncertainty based on the available benchmark models.

TSURFER Analysis

TSURFER performs a simultaneous adjustment of the cross-section data for all of the isotopes within the given covariance data using the Generalized Linear Least-Squares approach. TSURFER tries to minimize the cross section changes and the k_{eff} discrepancies for a given set of integral experiments. Since

TSURFER adjusts all of the cross-section data simultaneously for all of the isotopes, a wide range of integral benchmarks should be used. Alternatively, all of the discrepancy in the k_{eff} could be attributed to an error in a small set of isotopes; in reality, many isotopes contribute to the k_{eff} bias of each integral benchmark. Therefore, the entire set of 394 models in SCALE Verified, Archived Library of Inputs and Data (VALID) [12] was used as the background set of integral experiments to establish the appropriate multi-group cross-section changes for all of the isotopes in the two application systems apart from ^{35}Cl. No thermal-neutron-spectrum experiments containing chlorine were part of the VALID library. Two different sets of integral experiments were set up:

- the set of 394 VALID models in addition to the 11 most applicable benchmarks identified in Table 2 and
- the set of 394 VALID models and all of the 141 benchmarks that contained chlorine.

The sensitivity data files (SDFs) for the 141 benchmark models that contained chlorine were distributed with the IHECSBE handbook. These models are not considered as reliable as the VALID models. While the benchmark evaluations in the IHECSBE handbook undergo a rigorous review process, neither the computational model inputs nor the SDF files distributed with the handbook are subjected to a review process as part of the IHECSBE effort. In contrast to the IHECSBE, the SCALE VALID process was set up to ensure high quality model input and SDF files through a review process. However, the 11 most applicable benchmark models identified in Table 2 were hand checked, and the calculated chlorine sensitivity was verified by direct perturbation calculations. The difference between the two sets of the experiments mentioned above is that the first set could be considered reliable but with the scope of the chlorine containing benchmarks limited only to the most applicable ones, and the second should not be considered reliable but encompassing all of the freely-available data. Furthermore, each of the TSURFER runs was repeated with the ENDF/B-VII.1 covariance data for ^{35}Cl and with the BLO approximated covariance data. Table 3 presents the results of the TSURFER analysis. Note that with the following convention of bias, a positive bias for the chlorine is a conservative bias with respect to the safety analysis case. In other words, a positive bias associated with chlorine may indicate that, based on the TSURFER analysis, the applications' calculations are higher because of an error in the chlorine cross sections. This may suggest that the chlorine capture cross section in the current ENDF/B-VII.1 evaluation could be increased slightly.

The propagated chlorine uncertainty was calculated using the SAMINT code. It is evident from Table 3 that the exact numbers for the calculated bias and bias uncertainty depend on which set of benchmark experiments is used in the analysis as well as on which chlorine covariance data are used. However, the same pattern emerges regardless of the set of integral experiments or covariance data. The propagated k_{eff} uncertainty from all of the isotopes for both application systems is around 550 pcm. The ^{35}Cl uncertainty contributes approximately 50 pcm uncertainty to the k_{eff} of the no absorber application case and 100 pcm to the k_{eff} uncertainty of the degraded fuel basket case. In all cases, both the bias from all of the nuclear data and the bias just from the ^{35}Cl are less than the calculated uncertainty. Furthermore, it is clear that the uncertainty in the chlorine cross section can be considered to bound the bias. A similar argument has been previously made for fission product isotopes that had very limited or no critical experiments available [13].

The exact loading pattern of the assemblies' specific for each cask at a nuclear power plant is known. These casks will be referred to "as-loaded". They are usually much less reactive than the two applications chosen for this study, which can be considered extreme cases of the former. For the purposes of demonstration, an as-loaded cask system was analyzed in the same manner as the two application models to determine the bias and uncertainty associated with the chlorine in the groundwater. The DPC was modeled with no absorber (assuming no credit for its presence can be taken after degradation initiates) and without any control component in the guide tubes. The control component was replaced by water, which further increases reactivity. The as-loaded DPC cask was modeled with 4000 ppm of chlorine to be slightly above critical. It was calculated to have a k_{eff} value of 1.00092 +/- 0.00018. The chlorine worth is calculated to be

988 pcm. **Error! Reference source not found.** 4 presents the results, similar to Table 3, for the as-loaded DPC.

For the as-loaded DPC application, Table 4 displays very similar results for all four combinations of the two sets of experiments and the two different covariance libraries used. In summary, the TSURFER analysis suggests that the as-loaded DPC application model calculates a k_{eff} value that is too low by approximately 200–300 pcm. However, almost none of that bias can be attributed to the chlorine cross section. Once again, the total bias that can be attributed to errors in the chlorine cross section is covered by the uncertainty in the chlorine cross section.

Table 3. TSURFER results

	No Absorber	**Degraded Basket**
Initial k_{eff}	1.00940 +/- 0.00544	1.05220 +/- 0.00552
238 group propagated Cl initial uncertainty[a]	0.00058	0.00109
44 group propagated Cl initial uncertainty	0.00056	0.00102
Using all VALID benchmarks and the 11 most applicable chlorine containing benchmarks using ENDF/B-VII.1 covariance data for ^{35}Cl with a flat flux collapse		
Total bias	-0.00127	-0.00066
Final k_{eff}	1.01070 +/- 0.00148	1.05290 +/- 0.00144
^{35}Cl bias	0.00037	0.00070
44 group propagated Cl final uncertainty	0.00052	0.00094
Using all VALID benchmarks and the 11 most applicable chlorine containing benchmarks using the BLO approximate covariance data for ^{35}Cl		
Total bias	-0.00138	-0.00087
Final k_{eff}	1.01080 +/- 0.00146	1.05310 +/- 0.00138
^{35}Cl bias	0.00026	0.00049
44 group propagated Cl final uncertainty	0.00046	0.00087
Using all VALID benchmarks and the 141 chlorine containing benchmarks using ENDF/B-VII.1 covariance data for ^{35}Cl with a flat flux collapse		
Total bias	-0.00016	0.00032
Final k_{eff}	1.00960 +/- 0.00110	1.05190 +/- 0.00141
^{35}Cl bias	0.00021	0.00040
44 group propagated Cl final uncertainty	0.00053	0.00096
Using all VALID benchmarks and the 141 chlorine containing benchmarks using the BLO approximate covariance data for ^{35}Cl		
Total bias	-0.00034	-0.00006
Final k_{eff}	1.00980 +/- 0.00107	1.05230 +/- 0.00135
^{35}Cl bias	0.00002	0.00001
44 group propagated Cl final uncertainty	0.00047	0.00088

[a] One standard deviation is presented as a measure of uncertainty.

CONCLUSIONS

The available literature was surveyed for evaluated critical benchmark experiments that could be used in a validation study to support crediting chlorine as part of criticality analyses for SNF disposal. Two particular DPC models were considered as application models. Both DPC application models were assumed to be flooded by groundwater containing chlorine, with one model having the absorber plate completely deteriorated and the other having the fuel basket and absorber plates completely disintegrated. From the noncommercial resources, only the IHECSBE handbook was found to have evaluated critical experiments with chlorine sensitivities similar to the two application cases. A total of 141 integral experiments were identified to contain chlorine; however, only 11 of these had near enough sensitivity to be considered as a suitable representation of the chlorine in the application systems. Therefore, traditional validation of chlorine through trending analysis was deemed impossible due to the small number of relevant experiments and their diverse nature.

The code TSURFER provided an estimate of the bias uncertainty for the application systems. The bias uncertainty is estimated, at a one-sigma level, to be around 50 pcm for the no absorber application case and around 100 pcm for the degraded fuel basket case.

Table 4. TSURFER results for the as-loaded DPC

Initial k_{eff}	**1.00092 +/- 0.00604**
Propagated Cl initial uncertainty	0.00010
Using all VALID benchmarks and the 11 most applicable chlorine containing benchmarks using ENDF/B-VII.1 covariance data for ^{35}Cl with a flat flux collapse	
Total bias	-0.00283
Final k_{eff}	1.00375 +/- 0.00143
^{35}Cl bias	0.00007
Propagated Cl final uncertainty	0.00010
Using all VALID benchmarks and the 11 most applicable chlorine containing benchmarks using the BLO approximate covariance data for ^{35}Cl	
Total bias	-0.00285
Final k_{eff}	1.00377 +/- 0.00143
^{35}Cl bias	0.00005
Propagated Cl final uncertainty	0.00008
Using all VALID benchmarks and the 141 chlorine containing benchmarks using ENDF/B-VII.1 covariance data for ^{35}Cl with a flat flux collapse	
Total bias	-0.00202
Final k_{eff}	1.00294 +/- 0.00109
^{35}Cl bias	0.00004
Propagated Cl final uncertainty	0.00010
Using all VALID benchmarks and the 141 chlorine containing benchmarks using the BLO approximate covariance data for ^{35}Cl	
Total bias	-0.00204
Final k_{eff}	1.00296 +/- 0.00109
^{35}Cl bias	0.00001
Propagated Cl final uncertainty	0.00008

Recommended next steps would be to automate the procedure described in this report for the validation study for crediting chlorine in criticality analyses for spent nuclear fuel disposition to a general procedure that can be used for any isolated chemical element. Such an automated procedure would be a useful extension to the Used Nuclear Fuel Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS) [14] being developed at Oak Ridge National Laboratory as a collaboration among several national laboratories and industry partners. UNF-ST&DARDS has an automated procedure that assembles as-loaded criticality models for desired DPC configurations. From that point, a general procedure could be constructed that would execute a TSUNAMI-3D run followed by TSUNAMI-IP and TSURFER calculations. The end results would be the uncertainty in k_{eff} associated with the targeted nuclide cross sections and the bias and bias uncertainty according to the TSURFER calculation for the application as a whole and just for the individual chemical element of interest. This information would support future licensing efforts.

REFERENCES

[1] K. Banerjee, J. B. Clarity, and J. M. Scaglione, *Dual Purpose Canister Reactivity and Groundwater Absorption Analysis*, FCRD-UFD-2014-000520, prepared for the U. S. Department of Energy, Office of Used Nuclear Fuel Disposition, by Oak Ridge National Laboratory, June 2014.

[2] R. Kilger, M. Bock, and B. Gmal, “Exclusion of Criticality for a Final Repository in a Saline Host Rock Based on the Neutron Absorbing Properties of ^{35}Cl,” ANS NCSD, Wilmington, NC (2013).

[3] M. B. Chadwick et al., “ENDF/B-VII.1 Nuclear data for science and technology: cross sections, covariances, fission product yields and decay data," *Nuclear Data Sheets* 112.12, 2887-2996 (2011).

[4] R. O. Sayer, K. H. Guber, L. C. Leal, N. M. Larson, and T. Rauscher, *R-Matrix Evaluation of Cl Neutron Cross Sections up to 1.2 MeV*, ORNL/TM-2003/50, Oak Ridge National Laboratory, March 2003.

[5] K. H. Guber et al., *Phys Rev.* C65, 058801 (2002).

[6] *SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design*, ORNL/TM-2005/39, Version 6.1 (2011). Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785.

[7] V. Sobes, L. Leal, G. Arbanas, “Nuclear Data Adjustment with SAMMY Based on Integral Experiments,” ANS Winter Expo, Anaheim, CA (2014). Accepted for publication.

[8] R. C. Little et al., “Low-fidelity Covariance Project,” Workshop on Neutron Cross Section Covariance, Long Island, NY (2008).

[9] N. Leclaire, et al., “MIRTE: An Experimental Program Designed to Test the Reactivity Worth of Several Structural Materials,” *Proc. Int. Conf. Nuclear Criticality and Safety (ICNC2011),* Edinburgh, Scotland (2011).

[10] *International Handbook of Evaluated Reactor Physics Benchmark Experiments*, Report NEA No. 7173 (2014).

[11] B. Briggs, *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, Report NEA No. 7166 (2013).

[12] W. J. Marshall and B. T. Rearden, *The SCALE Verified, Archived Library of Inputs and Data—VALID*,” ANS NCSD, Wilmington, NC (2013).

[13] J. M. Scalione, D. E. Mueller, J. C. Wagner, and W. J. Marshall, “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions,” NUREG/CR-7109 (2011).

[14] J. M. Scaglione, J. L. Peterson, K. Banerjee, K. R. Robb, and R. A. Lefebvre, “Integrated Data and Analysis System for Commercial Used Nuclear Fuel Safety Assessments,” *Proc. Waste Management 2014*, Phoenix, AZ, March 2–6, 2014.

NUCLEAR SAFETY MANAGEMENT AT THE JUZBADO FUEL FABRICATION FACILITY

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ABSTRACT

The main objective of Nuclear Safety Management (NSM) at the Juzbado Fuel Fabrication Facility is to maintain the criticality safety margin and to provide, in case a criticality incident occurs, adequate measures to reduce its consequences on workers, members of the public and the environment. In order to achieve these targets, NSM at the Juzbado Plant relies on well-known and internationally standardized principles:

- Equipment and processes are designed according to previous criticality safety evaluations. These evaluations must be performed under the facility Safety Case conditions, so the probability of occurrence and the consequences of any critical event are minimized.
- Criticality safety evaluations are performed according to the "Double Contingency Principle", which states "Process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible". These safety factors are incorporated depending on the process and equipment considered, but in general can be implemented by two types of controls, Engineering and/or Administrative controls.
- Integrated Safety Analysis as a methodology to identify all the potential accident sequences during the operation of the facility as well as the items upon which the safety to prevent such accidents or mitigate their consequences to an acceptable level relies on (IROFS, Items Relied On For Safety). ISA project plays also an important role in plant modification process, being evaluated in the design stage.
- Specific criticality safety requirements are established on every equipment and process involving nuclear material. These requirements are written on different level documents and updated according to the criticality evaluations results.
- A training program is maintained and tailor-fitted to the extent of the personnel responsibilities concerning nuclear material. This training includes specific requirements regarding nuclear material handling, instructions to prevent a criticality accident and on what to do in case of an accident. Employees, and external personnel, are subjected to a continuous training program on all the safety aspects.
- Routine inspections are performed by the Nuclear Safety department aiming to check that the plant is operating according to Criticality Safety Requirements.

This paper describes the practical approach applied at the Juzbado Plant to implement such principles, along with the safety practices and other specific aspects of operational Nuclear Criticality Safety.

1. ENUSA SAFETY COMMITMENT

ENUSA-Juzbado management believes that safety is the main principle upon all its activities rely on. This commitment is established in the Plant's management documents as a company Chairman's responsibility, and then implemented by the Plant's Director. Therefore, this commitment is taken into account in all the activities involved with the fuel assembly fabrication.

2. EQUIPMENT AND PROCESSES MODIFICATION

The safety evaluation process involving equipment and process modification is performed according to the Spanish Regulatory Body (Consejo de Seguridad Nuclear) requirements, addressed in the safety guide GSG-03.01 "Modifications on fuel assembly fabrication facilities". On application of this safety guide, the evaluation process consists of the following stages:

1. Modification proposal, issued by the user;
2. Assignment of the Design and Review Teams' members;
3. Final design;
4. Spanish Regulatory Body approval (if necessary);
5. Internal approval;
6. Modification clearance.

Nuclear Safety staff participates on stages 2, 3 and 5, so we are going to focus in those stages.

If there is nuclear material involved, Nuclear Safety staff participates in the modification process. The kick-off is given by a Committee that decides if the proposed modification is appropriate from the Safety point of view; if so, the Design and Review Project Teams (DPT and RPT) are built-up. In both teams there is a Nuclear Safety Engineer (NSE), among others, such us a Radiological Protection Engineer, or experts from the operational areas.

Once the design stage starts, the equipment/process design is decided upon the agreement of all the team members, and therefore nuclear safety principles are taken into consideration from the early stages. In fact, this methodology and philosophy improvement avoids not only evaluating a closed design, but also enables that the design is made considering safety principles. The NSE is in charge of performing the criticality evaluations ensuring that the proposed design is "safe", and documenting all the evaluation process.

The design proposed by the DPT is then reviewed by the RPT. The NSE member reviews the evaluation performed by his colleague and assure that the safety principles have been properly applied. He verifies the calculations performed and that their results have been implemented in the proposed design. The RPT sends its comments to the DST leader, who agrees with his team mates and incorporates the comments in the project.

The final project, along with the documentation supporting it, is then approved by the Plant's Safety Committee and then, if needed, sent for approval to the Spanish Regulatory Body. The modification is only allowed to be put in production once ENUSA has this approval. Meanwhile, the equipment is identified as "Under modification. Not use".

3. DOUBLE CONTINGENCY PRINCIPLE

Juzbado Nuclear Safety department is in charge of developing and assessing this Safety Commitment from the criticality point of view. The crucial criteria applied is the Double Contingency Principle that compels us to control at least two independent parameters. This well-known principle, ensures that the nuclear material remains in a subcritical state, thus avoiding the occurrence of the criticality accident. Conservatively, a minimum of three parameters are usually controlled at Juzbado facility (Mass, Geometry and Moderation). The rest parameters are assumed in their most pessimistic values (i.e. enrichment is always considered at 5%, which is the license limit for Juzbado; neutron poisons are not used as a criticality control, etc.). This easies the implementation of NCS controls as there is no specific requirements eventually depending on the product (i.e. different enrichment or gadolinium content).

All the processes and equipment involving nuclear material have to be analysed from the criticality point of view. This analysis is supported by internal documents and ensures subcriticality not only in normal conditions but also under accident conditions. By means of to the Double Contingency Principle, nuclear safety engineers state that even in case that a single control parameter fails, subcriticality is maintained. The internal analysis are then verified by an independent reviewer. In case new processes or equipment are analysed, and if they modify the criticality control parameters, it may be necessary to update the ENUSA official documentation, which requires to apply to the Spanish Regulatory Body (Consejo de Seguridad Nuclear) for approval.

Juzbado is a low enrichment facility, thus the most important control parameter is moderation. Nevertheless, the first approach tends to rely on safe geometry or mass. Safe geometry is applied in case of liquid wastes likely to contain high uranium concentration. In this case, pipes and vessels are designed either with safe diameter or volume. Safe mass is not used very often because it would only allow to handle a small amount of uranium. Therefore, this control can only be applied in those areas where uranium is present in low quantities, as it is the case of the Chemical Laboratory where, by administrative inventory control, a maximum mass of 25 kg of UO_2 is allowed.

When it is not possible to apply mass or geometry safe values, moderation is controlled. As an example, the following general requirements are established:

- It is forbidden to store moderator material in areas or near equipment where moderation control is applied (i.e. during the blending process the additive is prepared just when needed).
- No water pipes pass through areas under internal moderation control (e.g. there are no water pipes passing through the uranium powder storage area).
- Water shall not be used as extinguisher for fire-fighting within internal moderation controlled areas, unless authorized by the Emergency Director (the emergency procedures establish this requirement).

DCP is physically implemented by means of two different kind of controls: engineering and/or administrative. To illustrate how these controls are implemented at Juzbado, a couple of examples follow below:

- **Moderation parameter control improvement**:

The most sensitive process in Juzbado facility, from the moderation control point of view, is the blending process, where a large amount of uranium is mixed with hydrogenated additive. DCP is accomplished by controlling Geometry (blender shape), Mass (amount of uranium poured into the blender) and Moderation (hydrogen-uranium atomic ratio $H/U < 0.41$). In the mainframe of the Juzbado Integrated Safety Analysis (ISA) program, ENUSA has developed a fractionated dosage system that avoids the moderation control failure. This system considers both passive and active engineering controls, first limiting the amount of additives by a pre-set volume container, and second by means of an alveolar valve that spreads the additive addition along the mixing process. This dosage system acts a barrier against the two sequences found in the ISA evaluation: uniform and non-uniform over-moderation. Therefore, the moderation control in the blending process has been changed from just administrative to an active/passive engineering controls supported by administrative practices.

- **Mass parameter control improvement:**

Nuclear Safety organization has launched a modification proposal for all the equipment where mass control relies only on administrative controls. Basically, these equipment are recipients for nuclear scrap material (powder from the pellet grinder, rejected pellets from quality controls, etc.). This scrap is collected in stainless steel cans with the same geometry as the one used in the powder store, so its safety was primarily ensured just as an extension of the latter. These recipients have been analysed as isolated units completely filled with nuclear material and containing any amount of neutron moderator. In fact it was assumed that the internal moderation was given by the minimum critical mass conditions for 5% enrichment. Uranium was modelled as powder or full density pellets. The so modelled isolated units resulted subcritical, which allows to consider mass as a non-controlled parameter. Thus, the criticality controlled parameters for these scrap cans are Geometry and Moderation.

Nevertheless, these cans, once retired from the equipment, have to be placed on a store, where mass needs to be controlled by administrative procedures. Therefore, we have implemented an active engineering control that continuously checks the weight of these scrap collecting cans. This scale has a pre-set weight limit that, when reached, activates an optical/sonorous alarm and automatically stops the equipment and so avoiding it to exceed the mass limit on the powder store. This active engineering control is supported by an administrative requirement that compels the employee to weight the can on another scale (used for material accountancy), and to verify the mass limit before taking it into the powder store.

4. INTEGRATED SAFETY ANALYSIS

The ISA project Juzbado Facility is the program established by ENUSA to systematically peer review all processes, equipment and facilities, as well as staff activities, in order to ensure that all relevant risks that could cause unacceptable consequences have been properly analyzed and that the appropriate safety measures have been implemented. Its scope is as follows:

- Description of the installation process, equipment and structures.
- Identification and systematic analysis of the installation risks. All events, both internal and external, that exceed the safety operation limits should be systematically identified.
- Identification and evaluation of accident sequences and deviations of processes leading to unacceptable consequences, determining the expected probability of such sequences to occur.
- Identification, description and analysis of the devices upon which the safety to prevent such accidents or mitigate their consequences to an acceptable level relies on.
- Identification of management measures and adequate monitoring to ensure reliability and availability of the safety features.

The risk evaluation methodology depends on the node to be evaluated. It could be used any of the standard methods included in the literature (HAZOP, WHAT IF/Checklist, PHA), although our main references are "Guidelines for Hazard Evaluation Procedures (AIChE, 1992)" and NUREG-1513 "Integrated Safety Analysis Guidance Document".

Once an ISA is performed over a process or feature, it is crucial to maintain it updated. Thus, any facility modification shall be assessed by an ISA Reviewer qualified person in order to determine whether it affects to the base ISA analysis. Therefore, ISA project plays also an important role in the plant modification process, being evaluated in the Design Project Team stage (see paragraph 2). At this point, the ISA Reviewer may arrange an ISA group meeting to assess the modification and, in its case, issue a report regarding how the base ISA is modified. This review is also performed if any unusual event which may affect the facility safety occurs.

The severity value (S) assigned to every sequence is given according to Table 1. This is an unmitigated value, so it only takes into account the facility safeguards but not the specific to the sequence itself. Therefore, the severity value assigned corresponds to the worst consequences of the event, regardless the safeguards. The final severity value may be assigned through a Criticality Safety Engineer assessment, which could be based on the Margin of Subcriticality (MoS) and Margin of Safety (MS). This engineering assessment may conclude that the severity should be increased or decrease by 1 from its initial value.

Table 1
Severity (S)

Value	Criticality		Radiological	
	SICP	**OCP**	**Workers**	**Public**
3	Failure of all ICP	Failure all controls	D > 1Sv	D > 250mSv Intake of more than 30mg de U Uranium release exceeding operation limits.
2	Failure of one ICP. Only one ICP remains active	Failure of all controls. Only one control remains active.	50mSv ≤ D ≤ 1Sv	5mSv ≤ D ≤ 250mSv Uranium release below operation limits.
1	Failure of one ICP no challenging the DCP.	Failure of one control no challenging the DCP active.	D < 50mSv	D < 5mSv Uranium release

SICP Several Independent Control Parameters
OCP One control parameter supported by several independent controls.
ICP Independent Control Parameters
DCP Double Contingency Principle
D Effective Dose

The Margin of Subcriticality is defined as MoS = 1- MSM- k_{eff}, where MSM stands for Minimum Subcriticality Margin and is an arbitrary value (usually equal to 0.02), and k_{eff} is the sequence reactivity value on a certain conditions. Therefore, we have an indicator of severity S(A) based on how much the MoS is affected by an accident sequence (MoS(A)) by comparing it to the MoS under normal conditions (MoS(NC)).

$$S(A) = \left| \frac{MoS(A) - MoS(NC)}{MoS(NC)} \right| \leq 1$$

S(A) provides information about how k_{eff} approaches to its critical value after the sequence. S(A) close to 1 means that severity should be increased by 1. Otherwise, if S(A) is close to zero, the severity can be decreased by 1.

The Margin of Safety (MS), for a given criticality control parameter, is the difference between the value of this parameter under normal conditions and the value that results in k_{eff} = 0.98. For instance, the internal moderation parameter is controlled by limiting the Hydrogen – Uranium atomic ratio H/U(NC) < 0.41, so the MS will be calculated as:

$$MS = \frac{\frac{H}{U}(k_{eff} = 0.98) - \frac{H}{U}(NC)}{\frac{H}{U}(NC)}$$

An application example of these criticality safety engineering evaluation is the blending process node, where a large amount of uranium is mixed with hydrogenated additives. The controlled criticality parameters are Mass, Geometry and Moderation. The ISA analysis of this node results on three main sequences regarding moderation control failure, whose k_{eff}, MoS and S(A) values are shown in Table 2:

Table 2

		k_{eff}	**MoS**	**S(A)**
Normal Conditions (**NC**)	Node 3.1 “Blending process”	0.312	0.668	------
Internal moderation control Failure	Sequence 1 “Less U powder”	0.312	0.668	0.000
	Sequence 2 “More pore former (additive)”	0.98	0.00	1.000
	Sequence 3 “Improper mixing process (non-uniform mix)”	0.364	0.616	0.078

Sequences 1 and 3 suppose that Mass and Geometry remain controlled so, considering Table 1, the severity value is S = 1. Regarding sequence 2, moderation and mass failure is considered, so S = 2.

According to MoS and S(A) values for Sequence 1, MoS keeps unchanged compared to its Normal Conditions value, so its severity, unless H/U limit is exceeded, has the lowest value S=1. In fact, this analysis could allow us to reject this sequence as an accident. For sequence 2, S(A) equals to 1 and MoS is 0.00, so its severity can be increased by 1, S=3. Regarding Sequence 3, although S(A) remains below 1, MoS is decreased, so its severity can be increased by 1, so S = 2.

5. SPECIFIC CRITICALITY SAFETY REQUIREMENTS

The conclusions of criticality safety assessments are then written down in the Plant’s documentation. For this purpose the Juzbado facility has different kind of documents, depending both upon the equipment and the process.

- Nuclear Safety Procedures: which apply to Nuclear Safety staff, and cover those activities performed by the Department, not directly related to the fabrication process but which are important to ensure safety.
- For instance, as part of the uranium reception process, internal moderation is controlled even before the uranium powder leaves the shipping facility. The material is released for shipment once it is verified that its moisture weight percentage is below 0.40% and the enrichment is less than 5%. This verification is performed by Nuclear Safety Engineers. Before a fuel assembly batch starts fabrication, its compliance to the applicable criticality safety parameters has to be verified (pitch, pellet and tube diameters, etc.).
- Criticality Safety Sheets (CSS): These apply to the fabrication process staff, and contain instructions involving nuclear material handling. They usually apply to processes rather than equipment.
- For example, in the blending process, there is a CSS relating the administrative practices regarding the moderation control, such as forbidding the storage of hydrogenated additives in the area.

- Criticality Safety Posters: Which also apply to the fabrication process staff, and contain instructions involving nuclear material handling. They usually apply to specific equipment: trolleys used to move pellets from one area to another, small cabinets to storage quality control samples, etc.

6. CRITICALITY SAFETY TRAINING

Training is considered as a basic pillar for operational criticality safety. Staff who handles nuclear material must know not only the risks involved with the operation but also why the safety requirements are as they are. At the Juzbado facility, training is personalized depending on the work the staff performs. As a general approach, there are two training levels:

- **Initial training**:

  This training is delivered according to the Spanish Regulatory Body standards IS-12 and IS-06. Employees are trained upon their arrival at the facility. In case of external personnel, this kind of training is delivered on annual basis. Initial training is split into two different steps.

  - Basic training:
    The first time a person enters the facility, he/she is trained on basic nuclear safety concepts. Of course, the first matter is to let him/her know about the fabrication process. A general presentation is delivered giving an insight on what the uranium is used for and explaining the different stages of the fuel assembly fabrication process. This way, they are able to have an idea and situate themselves in the whole process.

    Next session relates the risks involved on uranium handling, starting on what fission process is and how we can protect ourselves. The fundamentals on the criticality control parameters are explained and also the Double Contingency Principle, focusing in the three most important parameters controlled in the facility (Mass, Geometry and Moderation). Finally, the staff makes a test in order to evaluate whether they have understood the Nuclear Safety fundamentals. Everybody needs to pass this test in order to be qualified to work at Juzbado facility.

  - Specific training:
    Once the basic training test has been passed, employees and external personnel are trained on the specific nuclear safety aspects that their duties require. This is training is usually devoted to the Nuclear Safety Sheets, Procedures and/or Posters that apply to the process and/or equipment where the worker is going to work.

- **Continuous Training**:

  This training is delivered according to the Spanish Regulatory Body standard IS-06. It is addressed to employees and permanent external personnel and is delivered annually. It also develops two different levels:

  - Specific training:
    It is an extension of the initial basic training. Thus, it is not only a review of nuclear safety fundamentals but also specific requirements applying to every single fabrication process stage. These training sessions are arranged and addressed to groups of workers performing the same roles from the nuclear safety requirements point of view (i.e. office staff, ceramic process staff, etc.).

- Update training:
  This is performed in case an employee changes his/her role. So, it is needed to deliver a training session on the nuclear safety requirements that have to be applied. This includes Nuclear Safety Sheets, Procedures and Posters. This training can be also delivered due to important changes in the requirements.

7. NUCLEAR SAFETY INSPECTIONS

Basically, Nuclear Safety staff performs two different kind of inspections in order to ensure and verify that the Nuclear Safety requirements are properly applied. These inspections are performed by qualified staff.

- **Process requirements**: Verification of written requirements on Nuclear Safety Sheets, Procedures and Posters. It is performed at least once a week and its purpose is to verity that the requirements are properly applied in the process areas where uranium is handled (powder, pellets, scrap, wastes, Chemical Laboratory). Inspections are scheduled in order to ensure that all the process areas are covered at least quarterly.

- **Modifications**: As part of the equipment modification process described in paragraph 2, Nuclear Safety Staff verifies that the modification has been done according to the Nuclear Safety evaluation. This inspection shall be performed before using the equipment.

CRITICALITY CONTROL OF FUEL DEBRIS TMI-2 — REVIEW AND FUKUSHIMA EXPECTATION

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It is conceivable that a large amount of fuel debris has been produced in the reactors of Fukushima Daiichi Nuclear Power Station (1FNPS) due to the severe core damage and melting. The amount is far beyond minimum critical mass, and essentially subject to criticality control. Neutron poison, however, cannot be added to coolant water because of its leak from the containment vessels, and the underground water flowing into the coolant water loop. Fortunately, any sign of criticality, such as radioactive xenon (Xe) gas, has not been detected. This presentation will outline the criticality control of fuel debris in the Three Mile Island Nuclear Station, Unit 2 reactor (TMI-2): the estimation of a necessary concentration of boron in the coolant water, and the monitoring and control of boron concentration in the coolant water at the site. Then, criticality control problems of fuel debris during the 1FNPS decommissioning will be described contrasting with the TMI-2 case. Finally, technical expectations and research activities of the fuel debris criticality control for the 1FNPS will be presented. It is possible that the fuel debris will be retrieved under nonborated water at the risk of criticality. In this scenario, the risk control, mitigation measures in case of criticality, would be a key factor in safety of the decommissioning.

I. INTRODUCTION

The great earthquake and tsunami on March 11, 2011, led major failures of core cooling functions in the 1FNPS Unit 1 ~ 3 reactors. It is believed that their cores had been severely damaged and melted, and that a large amount of fuel debris had been produced.[1] It has been already known that both the pressure vessel (PV) and the containment vessel (CV) of each reactor were also damaged. At each unit, coolant water, not borated, is currently being fed into the PV and the CV, flowing through out of them and the reactor building, and being recovered at the turbine building. Radioactive Xe gas in each CV is being measured, which is the only method to monitor criticality condition of the fuel debris. More details relating to criticality in each reactor including locations of the fuel debris have not been confirmed yet although they are being estimated using computer codes for severe accident analysis and criticality analysis.2

The situation of 1FNPS is apparently more serious than the TMI-2 accident that occurred in 1979. The core of TMI-2 was also severely damaged and a significant amount of fuel debris was produced. Its primary system including the reactor vessel was, however, intact, which facilitated holding coolant water. The coolant water was borated and played an indispensable role in keeping the fuel debris subcritical during the retrieval work of fuel debris.

Comparison of those 1FNPS and TMI-2 situations indicates what must be known and what should be done technically in the 1FNPS to establish criticality control of fuel debris. It is, rather, essential that the purpose of the fuel debris retrieval in TMI-2 is to preclude inadvertent criticality.

In this report, criticality control of fuel debris before being retrieved will be discussed.

II. REVIEW OF TMI-2 CASE

Boron Concentration in Water [3,4]

In 1979, just before the accident, boron concentrations in water were 1,050 ppm and 2,300 ppm in the primary system and the refueling water storage tank (RWST), respectively. It is estimated that the concentration increased and became 1,750 ~ 2,300 ppm in the primary system due to the high pressure injection using the borated water in the RWST during the accident.

The concentration in the primary system was raised and maintained at 3,000 ~ 3,500 ppm following criticality evaluations of the core during the year.

In 1983, it was observed and confirmed for the first time that the core had been severely damaged and melted, and that fuel debris had been produced. It was followed by a criticality evaluation considering fuel debris retrieval under water as well.

The evaluation led the determination in 1984 that the lowest boron concentration in the primary system must be 4,350 ppm to secure subcriticality of the fuel debris in the core. The value was derived by the operator, GPU Nuclear Corporation (GPU), and approved by the Nuclear Regulatory Commission (NRC). The concentration in the primary system had been maintained at 4,950 ~ 5,200 ppm since then until completion of the fuel debris retrieval.

The retrieval, started in 1985, completed in 1990 with a little fuel debris remained in the core. It was judged by a criticality evaluation considering disposition of the remaining fuel debris under nonborated water that there is no chance of criticality. It became consequently unnecessary to maintain the boron concentration in the primary system.

Requests of CFRs and Regulatory Guides [3]

The Code of Federal Regulations of the United States, 10 CFR 50 (reactor), 10 CFR 70 (fuel fabrication) and 10 CFR 71 (storage and transportation), required, in those days, criticality control measures of nuclear material to prevent unexpected criticality. The same request was also applied to the criticality control of fuel debris in the TMI-2. Those CFRs, however, did not state a safety margin necessary to prevent criticality.

The NRC considered that a safety margin must be set depending on condition of a facility, and insisted that an operator must demonstrate adequacy of a safety margin in each safety evaluation. Conditions of the TMI-2 core after the accident and the fuel debris retrieval were special, to which existing regulatory guides and precedents of safety reviews could not be suitable. Therefore, the operator was requested to describe every supposed condition during the fuel debris retrieval, uncertainty of a criticality evaluation model, and uncertainty of a criticality analysis code. The operator was made to consider anomalies during the retrieval as well. On the other hand, many of existing regulatory guides were applicable to storage and transportation of fuel debris.

Criticality Evaluation [3,5,6]

The accident occurred after 94-day operation with full power following loading of 177 new fuel assemblies whose total amount was ~ 90 tonnes. Burn-up of each assembly at this time was in the range of 900 ~ 6,000 MWd/t. There were three types of fuel assemblies, which had different initial ^{235}U enrichments, 1.98 wt%, 2.65 wt% and 2.96 wt%, named as "batch 1", "batch 2" and "batch 3", respectively.

A criticality evaluation model was designed, based on the fuel assembly specification and the burn-up condition, as Fig. 1 which safely and conservatively represents every possible geometry change of fuel debris during its retrieval. This model is called as the "licensing model".

The model contains entire fuel inventory, and fuel debris originating from the batch 1 and batch 2 fuel assemblies was modeled as a mixture located at a peripheral region. Fuel debris from the batch 3 fuel assemblies was put at the center of model to maximize its multiplication factor although those assemblies were originally loaded at peripheral positions of the core.

Burn-up credit was taken only in the center region representing the fuel debris from the batch 3 fuel assemblies. The credit counted on depletion of ^{235}U and production of fissile plutonium during the burn. The credit also considered lanthanum, cerium, praseodymium, neodymium, promethium, samarium and europium which are rare earth elements in fission products. The credit rejected fission products soluble in water, gaseous, and possibly volatile under the high-temperature environment during the core melt down. The model does not include materials of fuel assemblies such as cladding and burnable poison, and structural material in the core.

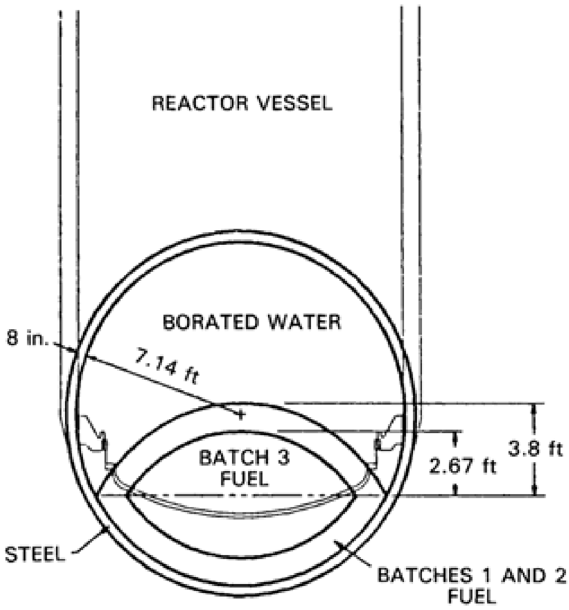

Figure 1 The "licensing model" for criticality evaluation of the TMI-2 fuel debris[6]

Each of the two regions was initially modeled as a homogenous mixture of each fuel debris and borated water whose mixing ratio was determined, according to boron concentration in water, to make its moderation condition optimum. A part of the model was replaced later by a heterogeneous composite which is an array of fuel debris spheres in borated water since observation in the core revealed fuel debris in the form of grains. This is also because multiplication of the heterogeneous composite is higher than the homogeneous mixture, and a radius of the fuel debris spheres was determined to maximize the effect.

Reflectors in the model are iron located under the fuel debris which represents the reactor vessel and borated water over the fuel debris. In this case, iron reflects neutrons more than borated water.

A criticality analysis code KENO V.a was used to compute a multiplication factor of the licensing model, and benchmark analyses were also conducted simultaneously to evaluate an uncertainty of the multiplication factor. Selected were benchmark data based on criticality experiments which had conditions, similar to the fuel debris, that ^{235}U enrichments were 2~ 5 wt%, fuel elements were uranium dioxide, moderator was water, H/U ratio was about 1.76, and boron concentration in the moderator was ~ 5,000 ppm.

The benchmark analyses of multiplication factors concluded that, based on all computations, the uncertainty is 2.5% while most of the results agreed within 1%.

It was determined, following the results, that a criterion value of multiplication factor was 0.99 to judge a system to be subcritical. Criticality of the fuel debris was evaluated by adding the uncertainty 0.025 to a multiplication factor computed by the criticality analysis code, and comparing the sum with the criterion value.

In conclusion, a multiplication factor computed for the licensing model containing borated water of 4 350 ppm was, even with the uncertainty 0.025 included, below 0.99.

It was recognized, even at the time of the evaluation, that uncertainty of the fuel debris condition is a more important factor to determine a safety margin of criticality control rather than the uncertainty of the criticality analysis code. Consequently, excessive conservativeness of the criticality evaluation model shown in Fig. 1 was discussed, and many other models, which might be more realistic, had been proposed. They did not, however, lead to alteration of the lowest boron concentration.

Practice of Criticality Control[3]

Attention was paid for anomalies that could lower the boron concentration, because the criticality control measure of the fuel debris is maintenance of the boron concentration in water of the primary system. Every connection of piping to the primary system which would cause dilution was identified and isolated using double valves, whose conditions were periodically checked. Unnecessary piping was physically disconnected. In addition, each specific work condition, such as the fuel debris retrieval, was controlled to prevent dilution. At the primary system and other locations sharing water, periodic sampling and analysis were conducted to detect any sign of dilution. Locations and frequency of the sampling were carefully determined assuming that coolant water volume may increase 22 m^3 at a rate of 60 L/m. Still more, 10 CFR 70.24 was applied during the fuel debris retrieval, which required criticality monitoring.

During the debris retrieval, water was necessary not for cooling but for shielding to protect workers. Water would have to be fed continuously to the RV if water leaks, water level falls, and the leak cannot be sealed. Even in this case, dilution of the boron concentration could not be allowed. For that purpose, equipment was prepared to recover leaking borated water at the basement of the reactor building, and to return it into the RV.

End of Criticality Control Practice [4]

The fuel debris retrieval completed when GPU had removed fuel debris from the core to the extent reasonably achieved, and when GPU demonstrated and the NRC agreed that inadvertent criticality was precluded. Until meeting the goals, the fuel debris retrieval had been conducted with the following guidelines:

- All fuel will be removed that is reasonably accessible within technically practical methods.
- Sufficient fuel will be removed to ensure the absence of a potential criticality regardless of degree of accessibility and level of difficulty.
- Residual fuel that is not reasonably accessible by practical means and has been determined to have no significant impact on public health and safety may remain.

The retrieval was conducted in the auxiliary and fuel handling buildings (AFHBs), the reactor buildings (RBs), the reactor coolant system (RCS), and the reactor vessel (RV). These locations included places where decontamination works were performed because fuel debris was detected, regardless of the possibility of its quantification, by radiation survey meters, etc.

After the work, remaining fuel debris was surveyed by measurement with instrumentation, visual observation, sample taking and analysis, etc. Mass of the remaining fuel debris was evaluated based on the survey result as follows:

- Auxiliary and fuel handling buildings < 17 kg
- Reactor buildings (excluding the RCS) < 75 kg
- Reactor coolant system (excluding the RV) < 133 kg
- Reactor vessel < 900 kg

Safe fuel mass limit from the viewpoint of criticality was determined as 140 kg. The value was derived from conservative assumptions, with estimation of a condition of remaining fuel debris, based on the knowledge of fuel debris condition gained through its retrieval. The assumptions are, specifically, an averaged composition of fuel debris including burn-up credit, ignoring of impurities, mixing of fuel debris and nonborated water with an optimum moderation condition, and existence of water reflector with a sufficient thickness.

Therefore, it was concluded that there is no chance of criticality in the AFHBs, the RBs and the RCS.

For evaluation of the RV, a criticality evaluation model was so designed as to reproduce in detail the disposition of fuel debris found by the survey, and to have safety margins by increasing quantity of fuel debris, etc. The criticality analysis code computed a multiplication factor of the model filled with nonborated water as 0.983 including the uncertainty, which is below 0.99.

An abnormal condition was also evaluated by assuming that the fuel debris, actually distributed throughout the RV, exists in one place. The assumption is improbable because the remaining fuel debris has firmly adhered to the RV and its distribution can hardly change. Accordingly, this evaluation utilized a model with a realistic fuel debris composition including materials originating from structural components and control rods. An infinite multiplication factor computed by the criticality analysis code was below 0.99 even for the optimum moderation condition.

Finally, it was concluded also in the RV that there is no chance of criticality. The NRC agreed in 1990 with those evaluations and exemption of the criticality monitoring stated in 10 CFR 70.24.

III. KNOWN CONDITIONS IN 1FNPS

Condition of Fuel Debris

Fuel assemblies with the design called "BWR STEP 3" had been loaded in the reactors. Each new fuel assembly contains 6 kinds of UO_2 fuel as shown in Fig. 2 and Table I. The initial ^{235}U enrichment of 4.4 wt% comprises the major- ity of the fuels, whose inventory per assembly is 76.8 kgU. The fuel of 9.6 kgU per assembly has the highest initial enrichment of 4.9 wt%. The initial uranium inventory in total is 170.9 kgU per assembly including fuels of other enrichments and of the UO2-Gd2O3 composite.7

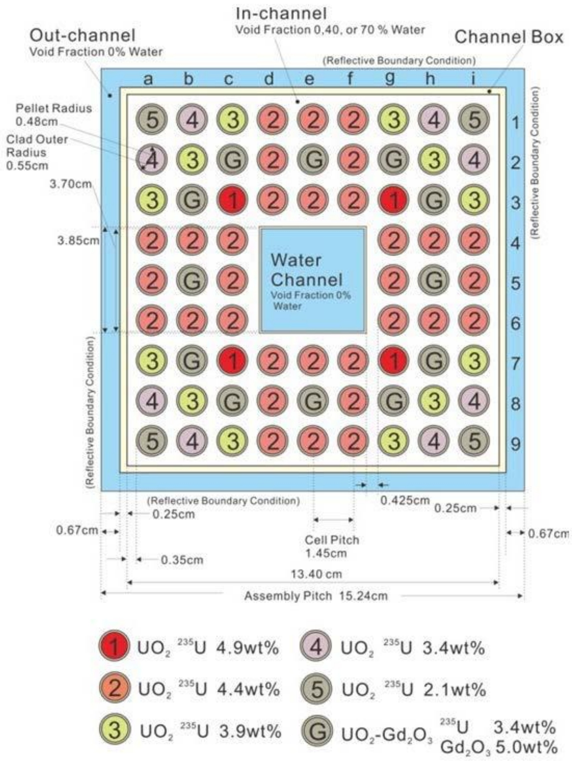

Fig. 2 Benchmark model of the BWR STEP3 fuel assembly

Table I Initial uranium inventory in a BWR STEP3 fuel assembly

^{235}U enrichment	Mass (kgU)
4.9 wt%	9.6
4.4 wt%	76.8
3.9 wt%	28.8
3.4 wt%	19.2
2.1 wt%	9.6
3.4 wt% (with Gd_2O_3)	26.9
Total	170.9

Table II Burn-ups of fuel assemblies in the 1FNPS reactors

Unit 1	Unit 2	Unit 3
5.2 : 64	3.3 :116	4.7 :148*
15.2 : 64	15.8 :116	15.5 :112
24.2 : 80	26.0 :120	28.5 :140
33.3 : 68	35.2 :120	36.2 :112
37.5 : 64	40.6 : 76	40.5 : 36
40.2 : 60	(GWd/t : Number of assemblies)	

* 16 MOX assemblies included.

The Unit 1 reactor in 1FNPS had 400 assemblies, which consisted of 6 batches of burn- up. Each of the Unit 2 and 3 reactors had 548 assemblies of 5 batches. Among these assemblies, 64 in the Unit 1 reactor, and 116 in the Unit 2 reactor had a low burn-up of only 3 ~ 5 GWd/t as listed in Table II. Other assemblies of the same number are older, but still have a burn-up as low as 15 ~ 16 GWd/t. The oldest assemblies have a burn-up of about 40 GWd/t.[8]

The condition of the fuel debris has not yet been identified in any reactor except estimations by severe accident analysis codes. Study on the TMI-2 fuel debris[9], however, suggests that various kinds of fuel debris may also be produced in the 1FNPS reactors, such as hard and loose debris. Especially, loose debris may show a wide variety of composition including structural materials such as Zircaloy and steel. Boron originating from the control rods cannot be expected necessarily to coexist with the fuel debris. It is also possible that the fuel debris in each of the CVs has been generated through the molten core concrete interaction (MCCI). It must be considered that the fuel debris is not uniform and will be found at various locations.

The fuel debris is being cooled with nonborated water although it is highly preferable to add neutron poison and to maintain enough concentration in the water to secure the subcritical condition such as was performed after the TMI-2 accident. Boration is not realistic at present because of the coolant water leakage from each of the CVs and underground water inflow to the coolant water circulation. Boron will be injected only in the event of criticality.[10]

Criticality Characteristics of Fuel Debris

The criticality safety handbook shows the minimum critical masses of homogeneous uranium-water mixtures, 36 kg and 53 kg, respectively for the ^{235}U enrichments of 5 wt% and 4 wt%. Mass control limits that can avoid criticality are also given for heterogeneous UO_2-water composites, that is, 28 kg for the 5 wt% enrichment. Even for the 3 wt% enrichment, its mass limit is still 67 kg[11]. These numbers are small compared to the possible uranium inventory in each fuel assembly with low burnup.

Fuel debris may exist as composites of UO_2 and structural materials such as Zircaloy and steel in each of the PVs. Zircaloy does not greatly affect the criticality characteristics of fuel debris because of its small neutron absorption cross section, but the iron in steel may increase the critical mass of fuel debris because it has strong neutron absorption.

The MCCI product would be a composite of UO_2 and concrete. The major content of concrete is silicon dioxide, which has also a small neutron absorption cross section and neutron moderation capability. The critical mass of UO_2-concrete composite has been evaluated as 400 kg for the fresh UO_2 of 5 wt% ^{235}U enrichment. For the fuel burned up to 12 GWd/t, the critical mass can be as small as 800 kg or 2 000 kg, depending on how the effect of fission products is considered. Only the water bonded in concrete is considered in the evaluation; therefore, the critical masses can be smaller when the MCCI product is submerged in the coolant water.[12] The mass of 2 000 kg is equivalent to 12 fuel assemblies. It is also known that a certain cluster of 16 assemblies in the Unit 2 reactor has an average burn-up of about 14 GWd/t. Thus, this evaluation is not far from reality.

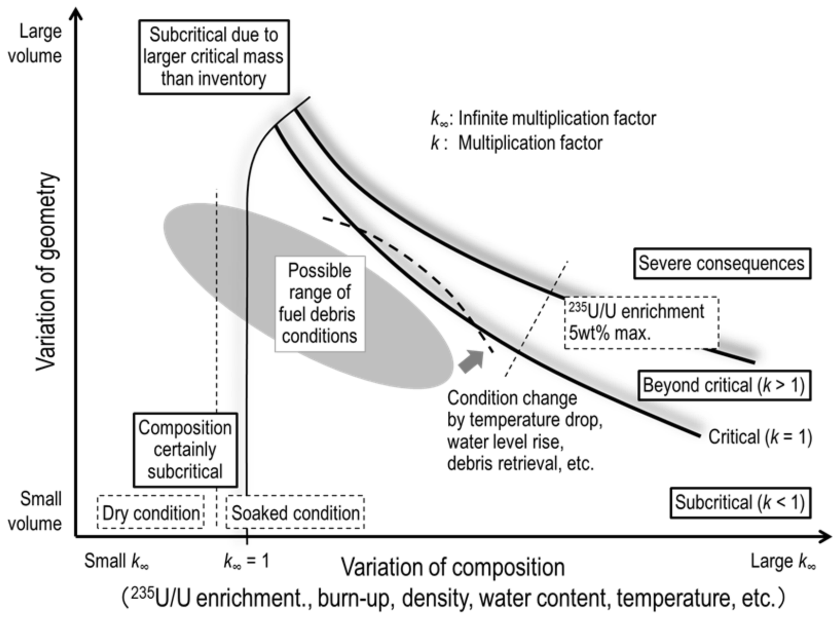

Figure 3. Criticality map of fuel debris.

Before knowing the actual condition of fuel debris, it is possible to compute critical conditions. Such work has been already conducted for many years to produce a handbook or a database for criticality safety. It is easy to extend these standards to wider conditions such as UO_2-steel composite or UO_2-concrete composite. The computation will supply a new set of "criticality maps of fuel debris". These maps will indicate, as shown in Fig. 3, subcritical and critical conditions, and supercritical conditions that would likely bring severe consequences. In the figure, the horizontal line represents variation of composition, and the vertical line represents variation of geometry. Composition on the right has higher reactivity and smaller critical volume. On the left, the composition is certainly subcritical, which can be excluded from the criticality control.

The actual criticality situation will be assessed by placing onto the map the fuel debris condition revealed by observations or sample analyses. It is also necessary to study how the condition can move on this map from expected changes such as temperature drop in the fuel debris or geometry changes by retrieval work of fuel debris, etc.

IV. EXPECTATIONS IN 1FNPS

Prevention of Criticality by Poison or Dry Process

The boration of coolant water was practiced in the TMI-2 and is most preferable. Borated water bounds the criticality characteristics of all debris into a small region indicated as "Boration" in Fig. 4, and keeps the region far from critical condition no matter how much temperature or geometry changes. By securing the lowest boron concentration in water, the subcritical condition can be guaranteed as well. The water issue, however, must be resolved to implement this option. Moreover, structure made of carbon steel or aluminum will act as the water boundary when a CV is filled with water. Then, corrosion of such material by boron must be studied to prevent recurrence of the water issue.

The dry process without using coolant water will be also a certain criticality control method as illustrated in the same figure. There will be, however, other engineering challenges. A CV must be sealed to avoid unexpected intrusion of water. It will be necessary as well to shield radiations and to suppress airborne migration of radioactive materials without water during fuel debris retrieval work.

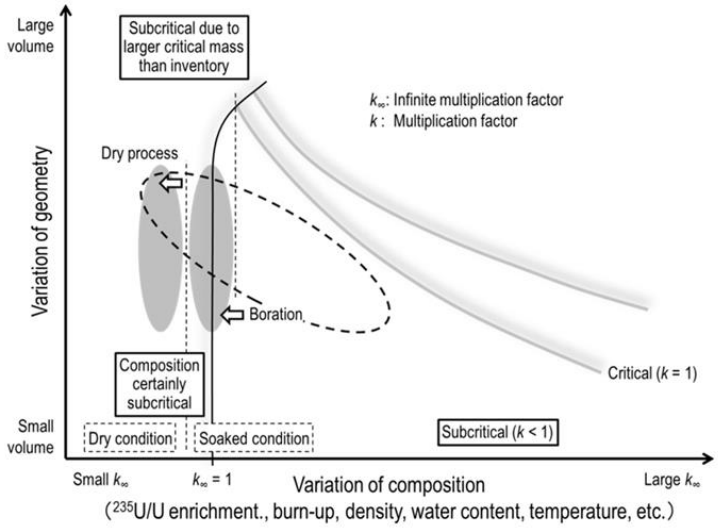

Figure 4. Prevention of criticality by boration or dry process.

Prevention of Criticality by Monitoring

Utilization of borated water may not be feasible unless the water issue is remedied. An alternative may be subcriticality monitoring. It is necessary to detect the signs of approach to the critical condition across the defense line set in the subcritical region in Fig. 5, and an intervention measure must be deployed quickly before the critical condition is reached. Detection may be possible by setting neutron counters near the fuel debris.

There are key natures of the intervention measure to be understood. The injection of neutron poison is the only way, and it will be realistic only if the actual condition of fuel debris is far from critical condition. It will be, however, difficult to make the defense line effective if the buffer zone is small. To retain the effect of intervention even after the event, the neutron poison concentration must be maintained in the coolant water.

Thus, this option does not differ, essentially, from the first option, which is prevention of criticality by poison. Monitoring still makes sense if we integrate it with the first option and use it as an implementation of the "double contingency principle."

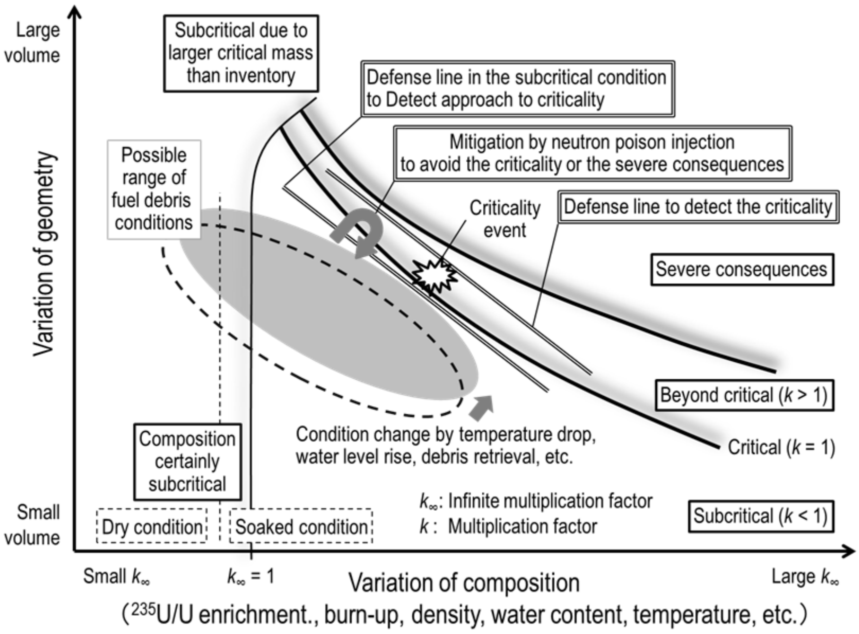

Figure 5. Prevention of criticality and severe consequences by monitoring

Prevention of Severe Consequence

The last option is, in fact, being currently applied. The defense line consists of the Xe gas monitoring and the injection of borated water. The monitoring sensitivity is not sufficient to measure subcriticality but can detect the event beyond critical condition before severe consequences result. The borated water on standby will be injected when the monitoring detects the criticality. A study is underway to improve the monitoring sensitivity to make the detection and intervention quicker and to reduce the risk of this option.

A much bolder idea is also being brought up, which is to consider such quick detection and intervention as a regular reactivity control. A small-scale, controlled chain reaction is permissible in the concept, and the resumption of fuel debris retrieval is allowed after suppressing the criticality. To realize this kind of criticality control, its risk must be fully understood.

Risk Assessment

The risk study is necessary regardless of which option is chosen because the subcritical condition is not secured at present. Even though the fuel debris will not be touched for a while, the temperature in fuel debris may drop gradually in time, which slowly increases reactivity. The risk of "low probability and high consequence events" must be also evaluated. An aftershock of large magnitude may change the fuel debris geometry greatly. The extreme event would be the fall of fuel debris in a PV onto the other in a CV.

Risk of the fuel debris retrieval must be assessed carefully, of course, if it is conducted under nonborated water. Its first step is to understand the actual conditions of fuel debris. Exhaustive observation of the fuel debris should be conducted as early as possible, which enables completing the maps described in the previous sections.

According to each option, engineering work should be performed in parallel to establish design requirements. For the prevention of criticality by borated water, its required lowest concentration must be established. For the prevention of criticality by monitoring, requirements of sensitivity and time response of the monitoring and time response of an intervention measure must be clarified. For the prevention of severe consequences, an allowable limit of fission number must first be set. Then, the time response of detection and intervention must be defined to regulate fission numbers of postulated criticality events within the limit.

Adequacy of those engineering outcomes must be validated to ensure that the criticality control at 1FNPS can practically function. Criticality experiments would be necessary to estimate uncertainties of the criticality map and the lowest boron concentration which are produced by computations because the fuel debris in 1FNPS may have compositions never experienced. The criticality monitoring and intervention measures should be also tested to demonstrate their performance.

V. CONCLUSIONS

After the TMI-2 accident, especially after knowing the core damage, criticality control of fuel debris was established in a classic manner. The control had been practiced through the whole period of the fuel debris retrieval work. The purpose of retrieval itself is to preclude inadvertent criticality.

In the 1FNPS, criticality control of fuel debris has not been established yet. The fuel debris must be put under a secure criticality control, in future, to preclude inadvertent criticality by retrieving it from each of the damaged reactors. It is uncertain, due to the significant water leak of each CV, whether criticality control of the fuel debris in a classic manner can be established for the retrieval work.

It is necessary to develop a new principle to control risk of criticality, in other words, to avoid not criticality but its severe consequences. Technical research and development must be conducted to implement the new principle in the 1FNPS, whose objectives are to quantify the criticality risk and to provide risk control measures whose performance is demonstrated.

ACKNOWLEDGEMENTS

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REFERENCES

1. "Fukushima Nuclear Accident Analysis Report," TOKYO ELECTRIC POWER COMPANY, INC. (2012).
2. K. TONOIKE, H. SONO, M. UMEDA, Y. YAMANE, T. KUGO and K. SUYAMA, "Major Safety and Operational Concerns for Fuel Debris Criticality Control," *Proc. GLOBAL 2013: Int. Nucl. Fuel Cycle Conf.*, Salt Lake City, USA, Sept. 29-Oct. 3, 2013 (2013).
3. W. R. STRATTON, "Review of the State of Criticality of the Three Mile Island Unit 2 Core and Reactor Vessel," DOE/NCT-01, Lawrence Livermore National Laboratory (1987).
4. "Three Mile Island Nuclear Station Unit 2 Defueling Completion Report, Revision 4," GPU Nuclear Corporation (1990),
5. J. R. WOLF, J. L. REMPE, L. A. STICKLER, G. E. KORTH, D. R. DIERCKS, L. A. NEIMARK, D. W. AKERS, B. K. SCHUETZ, T. L. SHEARER, S. A. CHAVEZ, G. L. THINNES, R. J. WITT, M. L. CORRADINI and J. A. KOS, "TMI-2 Vessel Investigation Project Integration Report," NUREG/CR 6197 TMI V(93)EG10 EGG-2734 (1994).

6. D. S. WILLIAMS and J. C. ROMMEL, "An Overview of Nuclear Criticality Safety Analyses Performed to Support Three Mile Island Unit 2 Defueling," *Nucl. Technol.*, **87**, 1134 (1989).

7. K. SUYAMA, Y. UCHIDA, T. ITO and T. MIYAJI, "OECD/NEA Burnup Credit Criticality Benchmark Phase IIIC, Nuclide Composition and Neutron Multiplication Factor of BWR Spent Fuel assembly for Burnup Credit and Criticality control of Damaged Nuclear Fuel," OECD/NEA/WPNCS/EGBUC (2012).

8. K. NISHIHARA, H. IWAMOTO and K. SUYAMA, "Estimation of Fuel Composition in Fukushima-Daiichi Nuclear Power Plant," JAEA-Data/Code 2012-018, Japan Atomic energy Agency (2012). (in Japanese)

9. D. W. AKERS, G. BART, P. BOTTOMLEY, A. BROWN, D. S. COX, P. HOFMANN, S. M. JENSEN, H. KLEYKAMP, A. J. MANLEY, L. A. NEIMARK and M. TROTABAS, "TMI-2 Examination Results from the OECD-CSNI Program," NEA/CSNI/R(91)9, Committee on the Safety of Nuclear Installations, Organization for Economic Cooperation and Development (1992).

10. "Status of Fukushima Daiichi Nuclear Power Station," TOKYO ELECTRIC POWER COMPANY, INC., http://www.tepco.co.jp/en/nu/fukushima-np/index-e.html

11. H. OKUNO, K. SUYAMA, K. TONOIKE, Y. YAMANE, T. YAMAMOTO, Y. MIYOSHI and G. UCHIYAMA, "Second Version of Data Collection Part of Nuclear Criticality Safety Handbook (Contract Research)," JAEA-Data/Code 2009-010, Japan Atomic Energy Agency (2009) (in Japanese).

12. K. IZAWA, Y. UCHIDA, K. OHKUBO, M. TOTSUKA, H. SONO and K. TONOIKE, "Infinite Multiplication Factor of Low-Enriched UO_2-Concrete System," *J. Nucl. Sci. Technol.*, **49**, 1043 (2012).

FRACTIONATED DOSAGE OF HYDROGENATED ADDITIVES IN THE CERAMIC PELLETS FABRICATION PROCESS

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ABSTRACT

The Juzbado Fuel Fabrication Facility has recently completed the implementation phase of the Integrated Safety Analysis (ISA) project. This project aims to identify all the potential accident sequences during the operation of the facility as well as the items upon which the safety to prevent such accidents or mitigate their consequences to an acceptable level relies on (IROFS, Items Relied On For Safety), and to establish management measures to provide acceptable availability and reliability of these safety features. The ISA evaluation carried out on the manufacturing process of green pellets, and particularly on the blending and homogenization nodes where hydrogenated additive is involved, has identified potential accident sequences affecting the control of nuclear material internal moderation, for both uniform and non-uniform over-moderation. Therefore, it has become necessary to implement IROFS. ENUSA has developed a device for the fractionated dosage of hydrogenated additive that acts as a barrier to prevent the occurrence of these sequences. This system acts over the accident uniform over-moderation sequence, first by limiting the additive poured amount by a passive engineering control (pre-set volume container), and second, by limiting the pouring to one single container through an active engineering control. Regarding the sequence of non-uniform over-moderation, the device implements an alveolar valve, providing additive pouring fractionation through a dosage/time preset program.

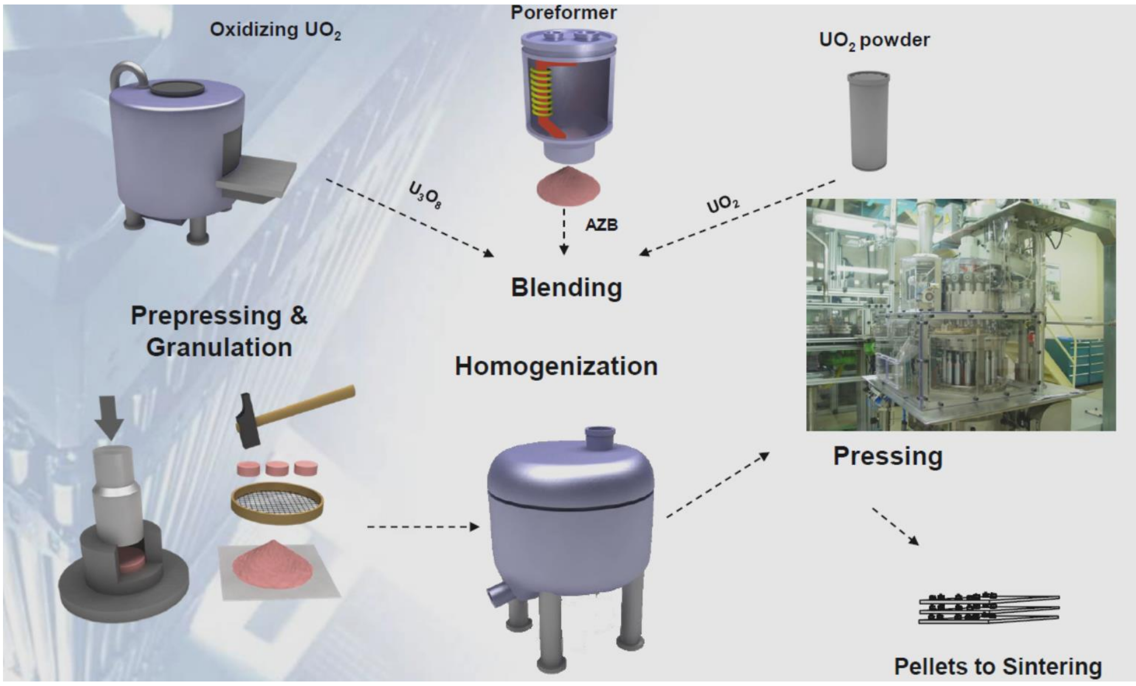
1. PROCESS DESCRIPTION

The ceramic fabrication process begins with the powder preparation and green pellets manufacturing process, which is performed in the PWR, BWR and Gadolinium ceramic areas. This step starts with the uranium dioxide UO_2 reception and its storage in the powder store. The most important criticality control applied on this process is the nuclear material internal moderation, which is controlled even before the uranium powder leaves the shipper facility. It is only allowed to leave the shipper facility if its moisture weight percentage is below 0.40%. This UO_2 powder, U_3O_8 powder (this one is produced at the Juzbado facility by UO_2 oxidation) and a pore forming additive are then mixed and blended. The last one is a hydrogenated additive, so, from the criticality safety point of view, it acts as a neutron moderator. The pore forming additive weight is determined depending on the target pellet density. The blending process is performed in a 600 liter hopper shaped blender, and a uniform blend is obtained by means of a rotating and translating worm gear.

The next stage is the pre-press and coarse graining process. The material coming from the blender is poured in a press machine and fashioned into low density pellets that are easily crumbled by a small grinder. The uranium powder so obtained has higher density than the one coming from the shipper and suitable for the next step of the process.

This grained powder is then poured in a homogenization machine, where it is mixed with a lubricant additive and, by a vibration motion, blended homogeneously. This second additive is made of hydrogen and therefore, it is a neutron moderator.

The homogenized powder is pressed in a high density press machine, obtaining the so called green pellets. Their density is double the grained powder density. Finally, these pellets are sintered, reaching the target density, passed through a grinder to give them the appropriate diameter, checked and loaded into zirconium tubes.



2. CRITICALITY SAFETY CONTROL IN THE PREPARING AND PRESSING URANIUM PROCESS

The criticality safety parameters that are controlled in this process are **geometry**, **mass** and **neutron moderation**. The geometry parameter is controlled by a passive engineering control given by equipment's geometrical characteristics; in case of the blender, this means the hopper height and its upper and lower diameters. Mass and neutron moderation are limited by administrative controls. The maximum hydrogen content in this stage is given by the total hydrogen-uranium atomic ratio $H/U < 0.41$.

Before performing the blending process, a Blending Process Sheet (BPS) is prepared, taking into account the weight of all the ingredients (UO_2, U_3O_8, pore former and lubricant additives). A computer software, called MEDEA, computes the H/U given by each one of these ingredients and checks that the total H/U is below the limit. The BPS can only be released if the H/U limit is accomplished. Before taking the uranium powder drums from the powder store, it shall be checked that they are included in the defined BPS. At the time the pore former is weighted, it shall be verified that it weights as defined by the BPS. Furthermore, before pouring the uranium powder and the pore former in the blender, it shall be verified again that the items are the ones defined in the BPS. In case an item is not included in the BPS or the H/U limit is exceeded, the software shows a warning message and terminates the process.

The criticality safety controls applied in the homogenization process are analogous to the blending process. In this case a so-called Homogenization Process Sheet is defined, including al the material involved in the process.

2.1 NEUTRON MODERATION ACCIDENT SEQUENCE

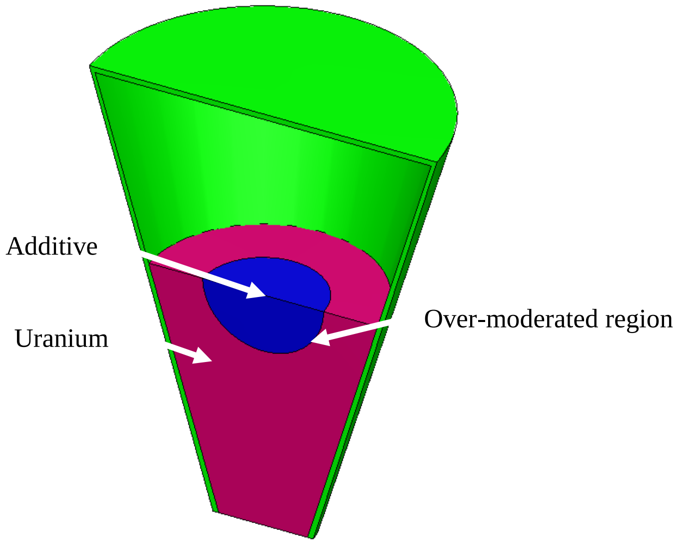
Basically, there are three different pathways leading to a loss of the neutron moderation control, which means exceeding the H/U limit value. These are analyzed in terms of ISA as the following sequences:

- Mixing process improperly done
- More additive
- Less uranium powder

2.1.1 Mixing process improperly done

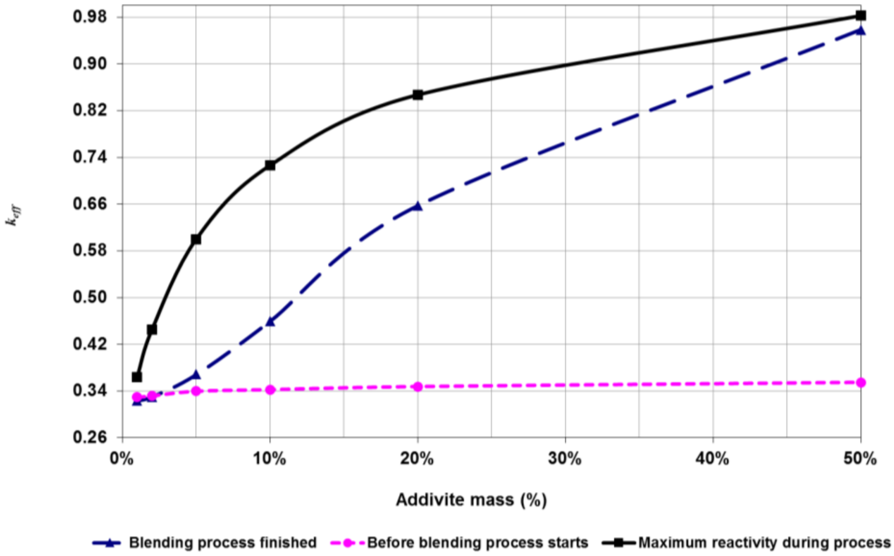
Once all the ingredients are poured in the blender (or in the homogenization equipment), the homogeneous mixture (H/U < 0.41 in all the blender volume) is assured only if the mixing process starts immediately and is properly performed. In this case, an improperly performed mixing process could result in the creation of an over-moderated region where H/U limit is not accomplished (see Figure 2). This situation is called **non- uniform over-moderation**.

Figure 2
Non- uniform over-moderation (blender)



Next figure shows the behavior of k_{eff} during the blending process, starting from the moment all the uranium and additive have been poured into the blender and finishing when the homogeneous mixture is achieved. The over-moderated region (H/U > 0.41) has been modeled as a hemisphere, initially formed only by the additive, which incorporates nuclear material as the blending process advances.

Figure 3
Reactivity - Non- uniform over-moderation (blender)

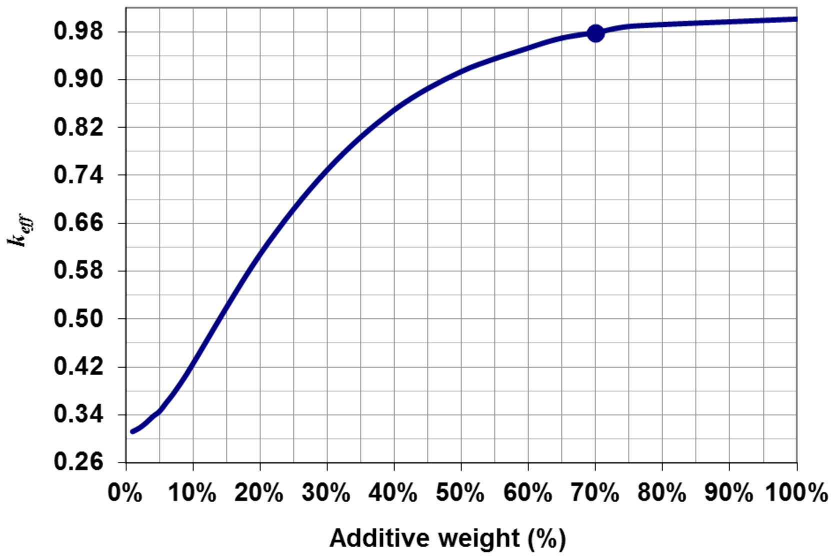


As Figure 3 shows, the reactivity value increases from the curve "Before blending process starts" up to the curve "Maximum reactivity during process" as the process advances, and then retreats to the curve "Blending process finished". This result makes us realized that the fractioned dosage process must ensure that the reactivity value during the process is below the curve "Blending process finished", assuring a large safety margin.

2.1.2 More hydrogenated additive

The second way to lose the neutron moderation control comes from pouring a greater amount of additives than the included in the BPS. This case is named **uniform over-moderation**. Figure 4 shows how the reactivity goes up as the added additive weight increases (expressed in weight percentage). Obviously, the second important factor that must be controlled by the fractionated additive dosage is the additive amount poured into the blender. As can be seen, this sequence could end up in $k_{eff} > 0.98$.

Figure 4
More hydrogenated additive sequence

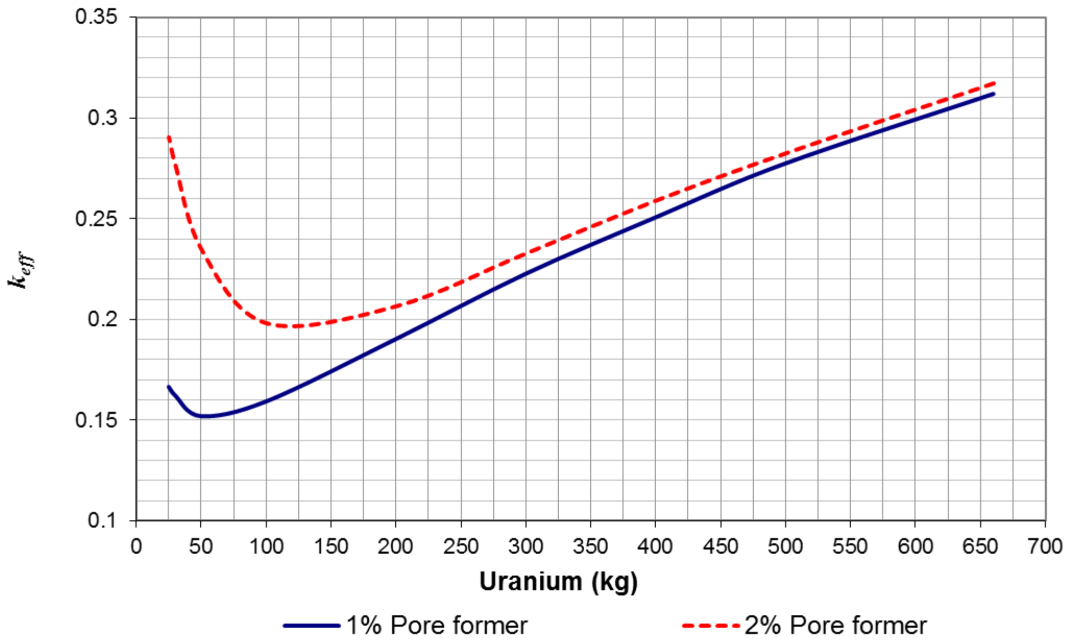


These results claim the maximum severity from the Integrated Safety Analysis point of view, whose methodology compelled us to design and establish IROFs in order to avoid such sequences. Thus, the Juzbado Facility has developed a fractionated dosage device for the hydrogenated additives, not only in the blending process but also in the homogenization process. Section 2.2 is devoted to describe how this fractionated dosage system works. Although focus is set in the blending process because of the higher amount of uranium involved, the functional properties are analogous for the homogenization process.

2.1.3 Less uranium

This sequence could happen in case all the additive were poured into the blender with an uranium mass less than the established in the BPS, so the H/U ratio could increase over the limit value. It is considered that the blending process is properly performed and thus uniform mix conditions are achieved. Figure 5 shows the reactivity values for different uranium masses and also several additive weight percentages. A typical additive weight percentage is 1% of the uranium weight.

Figure 5
Less uranium sequence

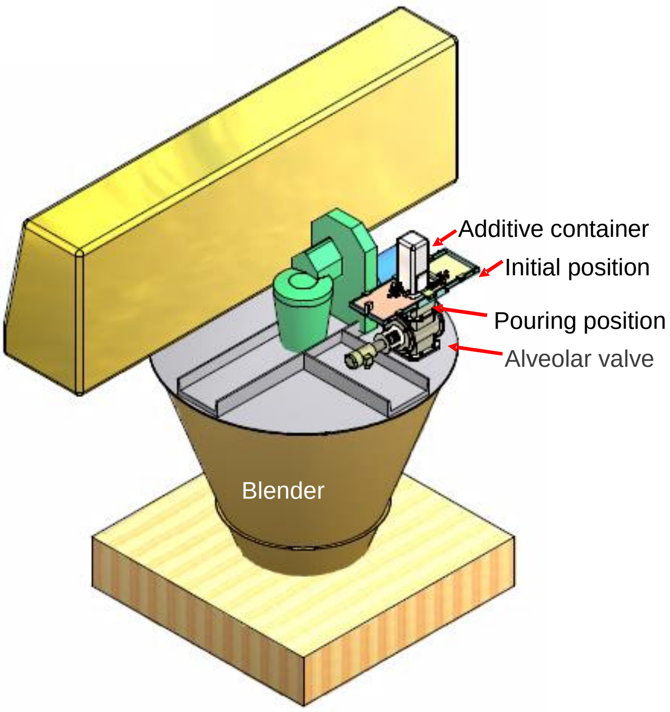


As it can be seen, less uranium than established in the BPS makes k_{eff} to decrease, although the H/U ratio increases and exceeds its limit value. The maximum k_{eff} value is reached in case all the uranium cans have been poured into the blender, so the H/U is less than 0.41. Minimum k_{eff} values are reached at H/U = 2.0. These results show that the less uranium sequence, although exceeding the H/U limit values, does not increase the process reactivity and thus, the safety margin is not affected. Therefore, this sequence has no effect from the criticality safety point of view and, considering the ISA methodology, its severity is acceptable. Thus, there is no need to implement IROFs as barriers against this sequence.

2.3 FRACTIONED DOSAGE OF HYDROGENATED ADDITIVES

The uranium cans are emptied trough the blender upper cabinet. Before the ISA conclusions, all the additive was poured directly into the blender through this cabinet too. Figure 6 shows the designed fractioned dosage equipment. It is placed on the blender upper side, and it is roughly formed by an additive container and an alveolar valve. The equipment designed by ENUSA acts avoiding both non-uniform and uniform over-moderation.

Figure 6



2.4 Uniform over-moderation

The additive container has a preset volume such that even in case of exceeding the additive weight established in the BPS, the moderation of the resulting mixture is below the limit H/U < 0.41. This container is filled with the pore former in a separate area and taken to the blending area just before the process starts: it is not allow to store hydrogenated additives in this area. The additive container is placed on the "Initial position", and remains there until all the uranium cans have been emptied into the blender. After verifying that all the right cans have been poured into the blender, it is allowed to move the additive container to the "Pouring position" and over the alveolar valve. This requirement avoids the additive to be mixed with less uranium than established in the BPS, so H/U value is kept below the limit.

As soon as the blending process starts, the additive container keeps blocked in the pouring position. A material detector at the blender bottom avoids to remove the additive container unless the blender is empty. In case of unexpected blender stop during the process, such as power failure, the additive container remains blocked. These requirements guarantee that the additive container is removed from the blender only if the blending process has been finished properly and only once the blender is empty. Therefore, it is not possible to place more than one additive container on the same blending process.

On the other hand, there is an additive detector between the alveolar valve and the additive container. If it keeps activated once the blending process is finished, which could mean that some additive has not been poured into, it avoids both to finalize the blender and to remove the additive container. In such case it is only possible to re-start the mixing process. This requirement avoids that an additive amount coming from the previous process could be poured in the next process.

2.5 Non-uniform over-moderation

The alveolar valve acts as a stopple, so there is not straight way from the additive container into the blender. The alveolar valve only starts working if the mixing process has already begun. The alveolus volume is very small, so every time the valve spins, the additive amount poured over the uranium is also small, and furthermore, as the blender is in process, the additive is quickly mixed. Therefore, the additive and uranium are always mixed uniformly, avoiding thus the creation of an over-moderated region during the mixing process. As it was said before, if the mixing process is unexpectedly canceled, the alveolar valve stops spinning and no more additive is poured.

Finally, taking into account the calculations performed on k_{eff} behavior during the mixing process, it is found out that the maximum k_{eff} value is reached within the first and fourth minute, depending on whether the process is performed in the homogenization or in the blender machine. Thus, in order to avoid this maximum k_{eff} condition, the additive addition lasts at least six minutes since the mixing process starts.

3. CONCLUSIONS

Considering the high k_{eff} values and the fact that the safety margin for these processes could be challenged, and applying the Integrated Safety Analysis methodology, we are compelled to implement IROFs on the ceramic pellets fabrication process.

This hydrogenated additives dosage process, developed by ENUSA-Juzbado, works as a barrier against the non-uniform and uniform over-moderation, firstly, by limiting the additive mass and, secondly, by spreading it along the mixing process.

Numerical Tests on the Optimum Moderation Criticality Calculations for Uncertainty Factor Analysis

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ABSTRACT

Fissile materials are controlled by limiting mass and separation distance to be maintained in subcritical condition. Changing in fissile material density maintaining mass would affect its volume and separation distance. This paper provides numerical results in effect of criticality safety when material density has changed. The result of test problem shows that criticality safety margin would be reduced by about 9%ΔK when fissile material density changes from 5.7 g/cc to 3.8 g/cc in optimum moderation condition. The test problem in this paper is not realistic, so the effect in real facilities would not be so significant. But this effect would not be negligible and might be significant in certain conditions.

I. Introduction

Nuclear criticality safety of facilities with fissile material should be proved by criticality analysis with proper uncertainties considering hypothetical accident conditions. One of these accidents of dry storage facilities is optimum moderation condition with low density moderator.

To ascertain that the storage facility satisfies safety limit for optimum moderation condition, moderator density of optimum moderation should be determined. If there are system parameters which are very sensitive to criticality, moderator density with optimum moderation might be changed. Thus analysis of criticality uncertainty for sensitive parameter in optimal moderation will be complex and full analysis would be required.

Mass limit and separation distance are usually used to maintain to be subcritical in nuclear material storage facilities. Change of material density leads change of volume, which might affect nuclear criticality significantly. Especially, powder of fissile materials can have various densities due to various compound types or impurities. Criticality calculations with fissile material of various densities have been performed and their results are presented in this paper.

The determination of bias of the result might be one of the important steps of criticality analysis. The bias is determined by comparing with criticality calculation results of “similar” criticality experiments. If same set of criticality experiments is chosen for flooded and optimum moderation cases, flooded and optimum moderation cases should be “similar”. To determine similarity, sensitivity calculation with TSUNAMI code in SCALE6.1 was performed.

II. Test Problem

The test problem in this paper has been constructed to maximize the effect of changing separation distance due to density change, so the test problem is a bit unrealistic. The test problem describes Uranium powder storage facility. Containers with cylinder geometry of Uranium power are assumed to be located in position, where each container has same separation distance of 40cm(center to center). There are infinite arrays of container in radial direction, which consists of eight containers in axial direction as shown in Fig.1. Vacuum boundary conditions in top and bottom of the problem are used.

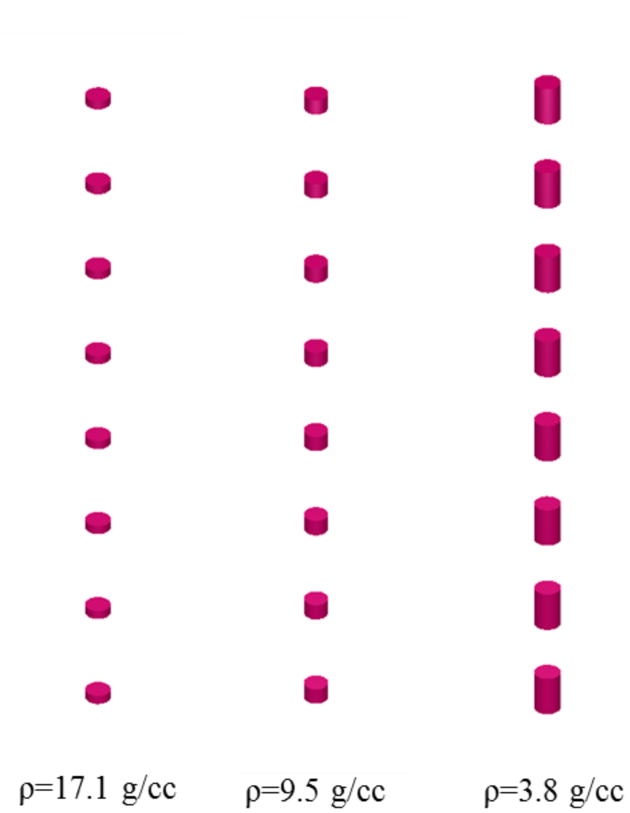

Figure 1. Configuration of the Test Problem with Various Density of Uranium
(reflective B.C for radial directions, vacuum B.C. for top and bottom)

In this problem, changing in material density does not affect separation distance in radial direction, but affect axial separation distance. Only fissile material (20wt% U powder) was considered in criticality calculations.

III. Numerical Results

Criticality calculations were performed by csas5 module in SCALE 6.1 code[1] with ce_v7_endf library which is a continuous energy nuclear data library based on ENDF VII. Uranium density was changed from 17.1g /cc(0.9 x 19.05g/cc, lumped Uranium density) to 3.8g/cc (0.2 x 19.05 g/cc). To find optimum moderation condition, various density(0.01g/cc ~ 0.9g/cc) of water moderator (40 cm x 40cm x 40cm with center of each cylinder container) is assumed. Figure 2 shows the criticality calculation results. Standard deviations of each resulting multiplication factors are less than 0.001.

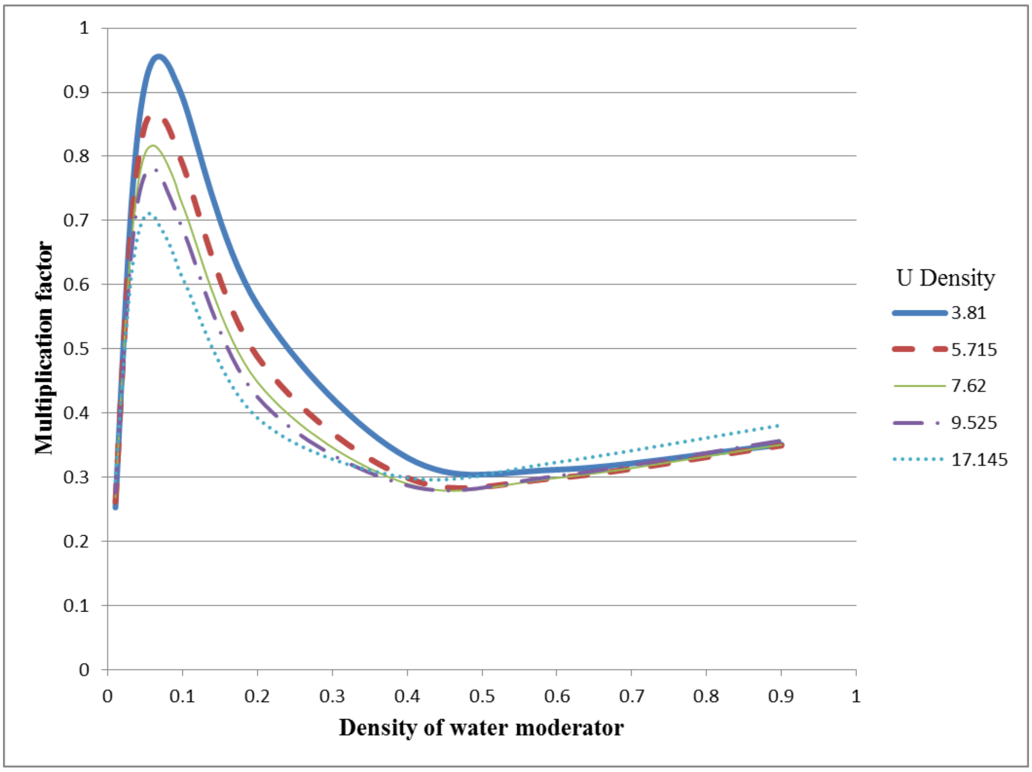

Figure 2. Results of Criticality Calculations

As expected, results of optimum moderation condition show significant difference compared by flooded condition (high moderator density). The multiplication factor difference of optimum moderation results between Uranium with density 3.81g/cc and 5.715 g/cc is about 9% ΔK. It is obvious that the effect of volume change should be considered if this test problem is a real case.

The moderator density with optimum moderation was similar (0.06g/cc~0.07g/cc) for each case, but not same. The multiplication factor difference between moderator density of 0.06g/cc and 0.07g/cc is 0.6~2% ΔK, which is not negligible. Real facilities might have enough safety margin and enough separation distance, which can reduce this effect, but it had better consider volume change effect especially when density of stored fissile material can have various density and subcritical margin is small.

The determination of bias of the result might be one of the important steps of criticality analysis. The bias is determined by comparing with criticality calculation results of "similar" criticality experiments. If we use same set of criticality experiments for cases in the test problem, test cases should be similar with each other. Recently sensitivity calculations are used to determine similarity.[2] In this paper, sensitivity calculations with TSUNAMI module in SCALE6.1 code were performed and The resulting similarity indice of TSUNAMI code, ck and E, of each cases are compared in Table 1.

Table 1 Results of TSUNAMI calculations

Test cases		1	2	3	4	5	6
		U 3.8g/cc H2O 0.07g/cc	U 3.8g/cc H2O 1.0g/cc	U 9.5g/cc H2O 0.06g/cc	U 9.5g/cc H2O 1.0g/cc	U 17.1g/cc H2O 0.06g/cc	U 17.1g/cc H2O 1.0g/cc
1	Ck[a)] E[b)]	1.0 1.0	0.9013 0.8971	0.9752 0.9888	0.8636 0.8406	0.8977 0.9786	0.7991 0.7780
2	Ck E	0.9013 0.8971	1.0 1.0	0.8442 0.9057	0.9454 0.9724	0.7157 08986	0.8492 0.9239
3	Ck E	0.9752 0.9888	0.8442 0.9057	1.0 1.0	0.8669 0.8716	0.9691 0.9933	0.8502 0.8183
4	Ck E	0.8636 0.8406	0.9454 0.9724	0.8669 0.8716	1.0 1.0	0.7780 0.8748	0.9716 0.9833
5	Ck E	0.8977 0.9786	0.7157 0.8986	0.9691 0.9933	0.7780 0.8748	1.0 1.0	0.8089 0.8356
6	Ck E	0.7991 0.7780	0.8492 0.9239	0.8502 0.8183	0.9716 0.9833	0.8089 0.8356	1.0 1.0

a) similarity index produced by using sensitivity profile and covariance
b) similarity index produced by using only sensitivity profile

The resulting ck values and E values show similar behavior that high similarity indices when moderator density is similar. Figure 3 shows neutron energy-wise system absorption behaviors for cases 2(flooded) and 5(optimum moderation), whose c k value shows minimum.

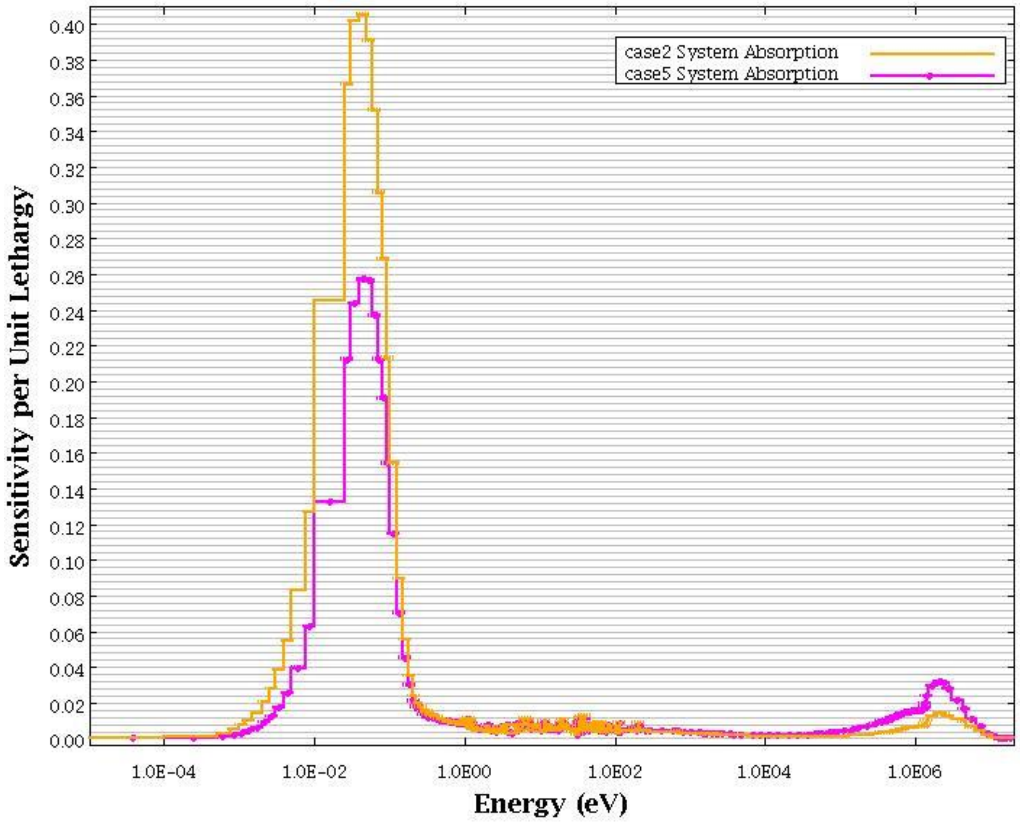

Figure 3. System Absorption Behaviors

Difference in thermal energy due to over moderation is shown in Figure 3, and also sensitivity in moderator also shows different behavior as shown Table 2.

Table 2. Sensitivity Comparisons between Case 2 and 5

	Uranium	H2O
Case 1	0.107	-0.0184
Case 2	0.227	0.3865
Case 3	0.124	-0.0188
Case 4	0.312	0.4193
Case 5	0.191	-0.0614
Case 6	0.379	0.4070

The Sensitivity results for H2O in Table 2 can be predicted from Figure 2. Changing moderator density from optimum moderation condition, multiplication factor decreases and slightly increasing when increasing moderator density from near flooded condition. From Tables 1 and 2, and Figure 3, flooded and optimum moderation conditions are slightly different from each other.

IV. Conclusion

In this paper, criticality and sensitivity calculation results of test problem for fissile material storage facility. Test problem assumes that fissile material volume can be changed as density of fissile material changes if criticality safety is only controlled by mass limit and center-to-center distance. Numerical results show that meaningful difference can exist in optimum moderation condition.

Sensitivity calculations were performed to determine that test cases are similar enough for using same criticality experiment set to evaluate bias. Difference in sensitivity and system absorption has been observed in test cases.

V. References

1. SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design, ORNL/TM-2005/39, Version 6.1, Oak Ridge National Laboratory (2011).
2. **T. Ivanova et al., "**OECD/NEA Expert group on Uncertainty Analysis for Criticality Safety Assessment: Current Activities," *Proceedings of PHYSOR 2010*, Pittsburgh, Pennsylvania, USA, May 9-14, 2010.

Subcriticality Demonstration Options for Direct Disposal of Dual-Purpose Canisters

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Abstract

Addressing the potential for criticality during the repository performance period (10,000 years or more) is one of the major challenges related to direct disposal of spent nuclear fuel (SNF) dual- purpose (storage and transportation) canisters (DPCs) because of the system potentially undergoes degradation in the repository environment. Because nuclear utilities are currently meeting their SNF storage needs on an individual basis by using high-capacity DPCs, the US Department of Energy's Used Fuel Disposition Campaign is assessing the technical feasibility of the direct disposal of DPCs in a geologic repository. The direct disposal of DPCs is an attractive possibility that could reduce and/or eliminate the operational complexity associated with large- scale repackaging of assemblies into different canisters and hence result in less cumulative worker dose before eventual disposal in a geologic repository. This paper investigates three options that could be applied to demonstrate subcriticality for direct disposal of DPCs: (1) taking credit for inherent and uncredited criticality margins associated with actual canister-specific loading configurations; (2) taking credit for different dissolved groundwater species that can provide reactivity reduction by neutron absorption; and (3) application of canister fill materials that could be used in existing and future DPCs to mitigate the potential for post-closure criticality through moderator displacement and/or neutron absorption. The results of this study indicate that demonstrating subcriticality of DPCs over the repository performance period could benefit from detailed canister-specific evaluations and credit for neutron absorbers present in groundwater.

Introduction

Current spent nuclear fuel (SNF) management strategies in the US include reliance on dry storage systems. Utilities are meeting their interim storage needs on an individual basis by using large-capacity dry storage canisters. Canisters designed for storage and transportation are referred to as dual-purpose canisters (DPCs). The US Department of Energy Used Fuel Disposition Campaign is examining the feasibility of direct disposal of DPCs. This paper investigates the feasibility of direct disposal of loaded DPCs from a criticality standpoint by examining attributes that could be credited to maintain subcriticality over a repository performance period (e.g., 10,000 years or more). However, the feasibility determination will also be limited by other considerations, such as thermal and operational constraints that may be site specific. The direct disposal of DPCs is appealing because it could be more cost-effective, minimize the need to repackage assembles into different canisters, and result in less cumulative worker dose during interim storage and handling before eventual disposal in a deep geologic repository. However, direct disposal of the current generation of DPCs poses several engineering challenges, one of which is the potential for criticality over a repository performance period as the system undergoes degradation. The analyses presented here investigate potential degradation scenario end states conducive to criticality and consider scenarios in which water enters a package at some point while in a geologic

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repository. The currently used DPC neutron absorbers are primarily aluminum based and likely do not possess the corrosion-resistant properties necessary to maintain their continued efficacy over disposal time periods in flooded conditions.

DPCs are designed and evaluated for approved contents as defined in their Certificate of Compliance (CoC). The approved content specifications provide bounding (enveloping) fuel characteristics such as fuel type, fuel dimension, initial enrichment, and discharge burnup. The bounding fuel characteristics for a system are developed to establish upper limits of the neutron multiplication factor, k_{eff} (also referred to as reactivity in this paper). In reality, there are many types of discharged SNF, with wide variations in SNF assembly burnup values, initial enrichments, and discharge dates. Therefore, dry storage systems are typically loaded with assemblies that satisfy the bounding fuel characteristics as defined in the CoC with some amount of unquantified and uncredited margin. This uncredited margin is introduced and quantified in a companion paper for DPCs in storage configuration and this paper presents an application of the uncredited margin to offset potential increases in SNF system reactivity over a repository performance period. This paper examines (1) the uncredited margins associated with actual fuel loading for a sample set of existing DPCs and (2) increases in reactivity because of canister flooding and the associated geometrical changes that can occur in the disposal environment as structures, systems, and components degrade. As-loaded (using the canister-specific fuel loading maps) criticality analyses are performed for DPCs loaded at six sites (total 184 DPCs), henceforth referred to as Sites A, B, C, D, E, and F, to evaluate the percentage of DPCs that remain subcritical solely based on the uncredited criticality margin. Additionally, this paper investigates (1) credit for different dissolved groundwater species, such as chlorine, that can provide reactivity reduction by neutron absorption and (2) application of canister fill materials that could be used in existing and future DPCs to mitigate the potential for post-closure criticality through moderator displacement and/or neutron absorption.

This paper presents an updated study of the criticality aspect of DPC direct disposal already discussed in Refs. 1, 2, and 3. This paper features (1) a new site (Site F), which is a boiling water reactor (BWR) site, for as-loaded analysis, and (2) groundwater species and filler material studies for as-loaded DPCs.

Computer Codes and Analysis Method

A criticality calculation is performed to quantify k_{eff} of as-loaded SNF systems. Taking credit for the reduction in reactivity because of fuel burnup is commonly referred to as burnup credit. Burnup credit criticality safety analysis for SNF in storage and transportation systems requires the determination of isotopic number densities for fuel assemblies, commonly known as a depletion calculation. A depletion calculation is followed by a cask criticality evaluation, which uses the isotopic number densities of the fuel that were determined from the depletion calculation to determine the system k_{eff}.

The SCALE [4] code system was used for the as-loaded criticality assessment. SCALE provides the required computer codes and sequences for running depletion and decay calculations using TRITON and ORIGEN depletion and decay modules. The TRITON two-dimensional (2D) depletion sequence [4] was used to perform depletion calculations that generated cross-section libraries for generic assembly/reactor–specific classes and a range of fuel operating conditions, which were subsequently used by ORIGEN to calculate nuclide inventories of SNF assemblies with specific irradiation characteristics.

Conservative irradiation parameters, which are applied to estimate the upper limit of k_{eff}, were used in this paper for criticality evaluations. In addition to the bounding depletion parameters, the two following assumptions were applied for BWR (Site E) fuel depletion and decay calculations:

- Gadolinium (1 wt % Gd_2O_3 in two rods) was conservatively ignored in the 6 × 6 Site E assemblies. Control blade insertion was assumed during the depletion period.

- Moderator density was assumed to be 0.49 gm/cc for Site E fuel assemblies, which corresponds to 35% average core void fraction.

The SCALE CSAS6 [4] criticality analysis sequence was used to perform criticality calculations for a loaded fuel cask using the KENO-VI Monte Carlo code with the continuous energy ENDF/B-VII.0 cross-section library to determine the k_{eff}.

The reactivity of site-specific DPCs was evaluated by employing a comprehensive and integrated data and analysis tool—Used Nuclear Fuel-Storage, Transportation & Disposal Analysis Resource and Data System (UNF-ST&DARDS)—which was developed at Oak Ridge National Laboratory [5] through a collaborative effort among multiple national laboratories and industry participants. UNF-ST&DARDS employs the SCALE depletion and criticality analysis sequences and modules discussed previously. Note that a prereleased version of SCALE 6.2 was used for the analyses presented in this paper.

Criticality calculations within UNF-ST&DARDS are performed applying 18-node axial burnup profiles and 12 actinides and 16 fission product isotopes. Burnup-dependent axial profiles are used for the pressurized water reactor (PWR) criticality analyses [6]. A uniform burnup profile was employed for the BWR fuel assemblies at Site E. Site E irradiated fuel assemblies are low burnup assemblies (~1.3–23 GWd/MTU). For PWR fuel assemblies, Ref. 6 determined that a uniform axial distribution is typically bounding for low burnup assemblies. However, similar conclusion has not yet been drawn for BWR assemblies. The isotope set, credited in the criticality calculations, was selected based on the burnup credit isotopes recommended by NUREG/CR-7108 and -7109 [7].

Degradation Scenarios

An important assumption for criticality analysis is that water enters a breached waste package (DPC inside an overpack) at some point during the repository performance period. Note that if water could be excluded from the repository or from entering a package, the potential for criticality would be negligible. While different geologic settings and material degradation mechanisms might yield a large number of potential scenarios for analysis, two simplified and conservative scenarios were used in this analysis:

- **Loss of Neutron Absorber.** Total loss of basket neutron absorber components from unspecified degradation and material transport processes, with replacement by groundwater. This hypothetical configuration could result if the fuel assemblies and the basket components were more corrosion resistant than the neutron absorber.

- **Basket Degradation.** Loss of the internal basket structure (including the neutron absorber). This hypothetical configuration is potentially relevant for DPC baskets with carbon steel structural components or configurations where the assembly-to-assembly pitch is reduced.

The degradation mechanisms for both neutron absorber and basket structure components over a repository performance period are uncertain. Currently, there is insufficient information available to support the assumption that a neutron absorber's presence would be maintained for the repository performance period in flooded conditions. Note that the corrosion products from the basket materials were not included in any of the criticality analyses because site-specific characteristics would need to be defined to account for the proper chemistry and resultant precipitates.

Analysis with Canister-specific Loading

The safety analysis report (SAR) for a particular DPC-based cask system documents the design basis models and calculations used to demonstrate that the system meets the regulatory requirements (e.g. 10 CFR 71.55 and 71.73) for transportation. SAR calculations and approved content specifications are intended to be bounding and to certify cask systems for a variety of fuel characteristics without placing stringent requirements on where the assemblies are placed in the basket. Therefore, loaded cask systems generally have some amount of uncredited, unquantified margin that may be used to offset the reactivity increase associated with degradation of the neutron absorbing panels over the repository performance period. This study performs k_{eff} evaluations of existing as-loaded canisters and simulates the design basis fuel characteristics documented in the SARs to estimate the uncredited margins compared to the design basis. These calculations assume that the canister is flooded with pure water and neutron absorber materials (panels) are completely degraded and replaced by water. An additional case is studied in which coated carbon steel components are present in the basket with loss of both the neutron absorber and the coated carbon steel components. Stainless steel structural components are assumed to maintain structural integrity through the repository performance period due to their corrosion resistant properties [8]. Canisters deployed at six sites (A, B, C, D, E, and F) are investigated.

UNF-ST&DARDS was used for the as-loaded criticality analyses. UNF-ST&DARDS performs inventory calculations for each unique assembly design (e.g., a Westinghouse 17×17 optimized fuel assembly), accounting for initial enrichment, burnup, and age. It generates explicit criticality models for each fuel assembly and DPC with the appropriate canister loading pattern identified from canister-specific loading maps. The same KENO-VI canister model was used for the design basis and as-loaded calculations with design basis and as-loaded fuel characteristics, respectively. The DPCs used in Sites A B, C, D, E, and F are briefly described below.

- **Site A**: Site A is a decommissioned PWR site. For Site A, 60 as-loaded canister systems from NAC International were analyzed [9]. Each DPC can contain up to 24 PWR fuel assemblies. The basket is of the tube-and-spacer-disk design (Figure 1 illustrates a tube- and-spacer-disk design). Neutron absorber sheets are attached on the four sides of the fuel tube, and the gaps between adjacent assemblies are flux traps. The Site A DPC basket components are stainless steel, so only the loss-of-neutron-absorber degradation scenario was considered.
- **Site B**: Site B is a decommissioned PWR site. For Site B, 39 as-loaded DPCs from NAC International were analyzed [10]. The fuel baskets used at Site B were the 26- and 24- assembly configurations. These two baskets are identical except that the top weldment of the 24-assembly configuration consists of 24 fuel tube penetrations. The 24-assembly basket is designed to accommodate higher enriched fuel assemblies than the 26-assembly basket. Site B baskets are made of stainless steel, so only the loss-of-neutron-absorber degradation scenario was considered.
- **Site C**: Site C is a decommissioned PWR site. For Site C, 20 NUHOMS® systems from Transnuclear [11] that are as-loaded dry-shielded canisters (DSCs that also qualify as DPCs) were evaluated. These DPCs can accommodate 24 intact PWR assemblies. Fuel assemblies inside the DPCs are maintained in place by thin-walled guide sleeves. Each guide sleeve is made from stainless steel with neutron absorber panels attached to each side of the sleeve that faces another assembly. The gaps between the neutron absorber panels facing each other form the flux-trap design. The guide sleeves are arranged inside the canister using radial spacer disks made of coated carbon steel (less corrosion-resistant material [12]) to maintain the flux-trap configuration (Figure 1). In addition to the loss- of-neutron-absorber scenario, the following scenario was considered for Site C because of the use of coated carbon steel spacer disks:

  Degraded Basket—The loss-of-neutron-absorber configuration was extended to include complete degradation of the spacer disks, resulting in a close-packed configuration of collapsed guide sleeves. The degraded disk material was replaced by water.
- **Site D**: Site D is a decommissioned PWR site. 34 multipurpose canisters (MPCs)-24E/EF canisters (DPCs) from Holtec International [13] were analyzed from Site D. The DPC can accommodate up to 24 PWR fuel assemblies. Site D baskets are made of stainless steel, so only the loss-of-neutron-absorber degradation scenario was considered.
- **Site E**: Site E is an operating PWR site. Site E uses the HI-STORM 100 system from Holtec

International [14] with MPC-32 canisters (DPCs). 26 DPCs were analyzed from Site E. The MPC-32 is an all stainless steel canister that can accommodate 32 PWR assemblies and uses an egg-crate basket design with a single neutron absorber panel between adjacent assemblies. Accordingly, loss of neutron absorber was the only degradation scenario considered for the MPC-32. The MPC-32 is licensed for transportation by applying burnup credit for criticality analysis.

- **Site F:** Site F is a decommissioned BWR site. Five as-loaded canisters from Site F were evaluated. Site F uses Holtec's canister [13] for storing their discharged SNF. This system can accommodate 80 fuel assemblies per canister, up to 40 of which may be damaged. Site F baskets are made of stainless steel, so only the loss-of-neutron-absorber degradation scenario was considered.

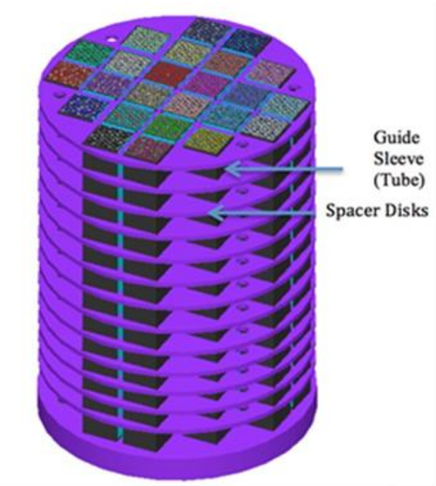

Figure 1: Three-dimensional view of the Site C DPC (full axial length is not shown) as modeled in KENO-VI.

Figure 2 depicts the cross section of Site C DPC for the loss-of-neutron-absorber case as modeled in KENO-VI, while Figure 3 illustrates the cross section of the collapsed basket as modeled in KENO-VI for Site C. These sample figures illustrate the detailed modeling approach used for the criticality analyses.

Figure 4 shows the calculated k_{eff} results for all the analyzed DPCs with the loss-of-neutron-absorber scenario, as a function of calendar year. As mentioned above, the degraded basket analysis was performed only for Site C, as Site C DPCs contain coated carbon steel spacer disks. Figure 5 presents the Site C basket degradation results. For simplicity, computational biases and uncertainties were not developed in this analysis but were simply assumed to be 2% (Δk_{eff}), resulting in a subcritical limit of k_{eff} <0.98. k_{eff} <0.98 is used in this study as a representative acceptance criteria for as-loaded calculations. However, if analyses like these are used to support future disposal licensing, additional validation and assessment of applicable biases will be required. Time-dependent reactivity calculation results are provided for the time range between the calendar years 2001 and 9999 (i.e. approximately 8000 years). Note that after the initial decrease, reactivity increases gradually from approximately 100 years to 10 000 years and beyond due to radioactive decay series and then reaches a second reactivity peak [15]. However, the expected reactivity increase between 8 000 years and the second reactivity peak is not significant (less than 0.005 Δk_{eff}) [15]

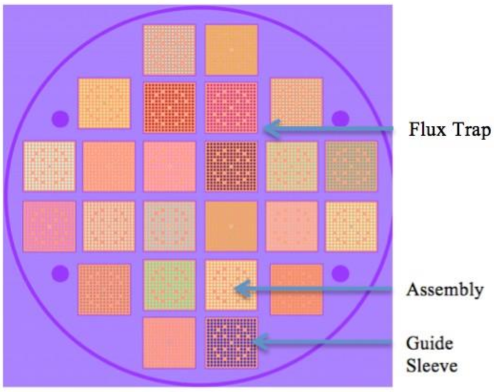

Figure 2: Loss-of-neutron-absorber configuration for Site C as modeled in KENO-VI.

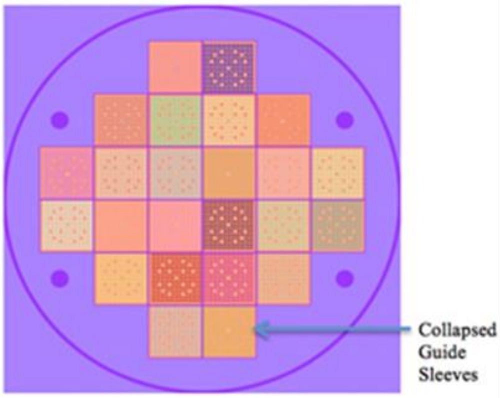

Figure 3: Degraded spacer disks configuration for Site C as modeled in KENO-VI.

Figures 4 and 5 show that *keff* values associated with some of the DPCs are above the representative subcritical limit defined in this paper, especially in later years. Table 1 summarizes the analyses performed for the DPCs stored at the six selected reactor sites. The summary results are presented for the calendar year 9999. Of the 184 DPCs analyzed, all would exceed the subcritical limit defined in this paper (*keff* >0.98) with loss of neutron absorbers if they were loaded with the design-basis fuel used for licensing. Using as-loaded fuel characteristics and burnup credit (28 nuclides), only 25 of the 184 (13.5%) (none from Site C) would exceed the subcritical limit for the loss-of-neutron-absorber scenario. For the Site C DPCs, 18 of the 20 DPCs would exceed the subcritical limit with the basket degradation scenario. Therefore, a total of 43 DPCs (23.4%) would exceed the representative subcriticality limit with a loss-of-neutron- absorber scenario for Sites A, B, D, E, and F and a loss-of-coated-carbon-steel-spacer-disks scenario for Site C.

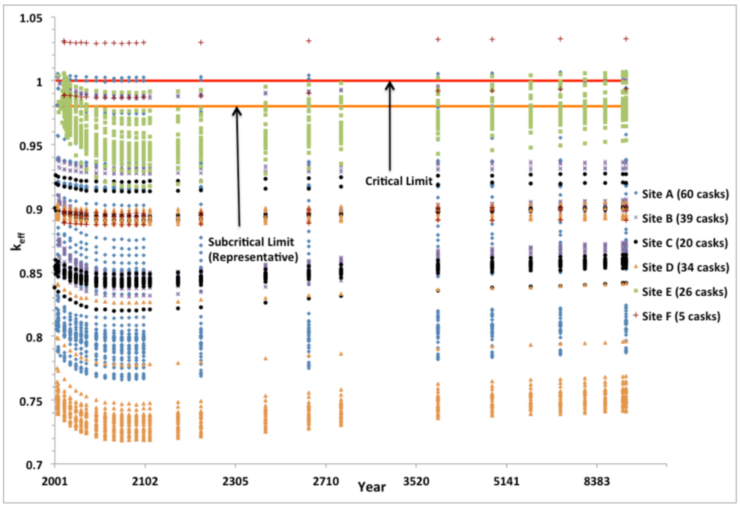

Figure 4: k_{eff} vs calendar year for the loss-of-neutron-absorber case, based on actual loading.

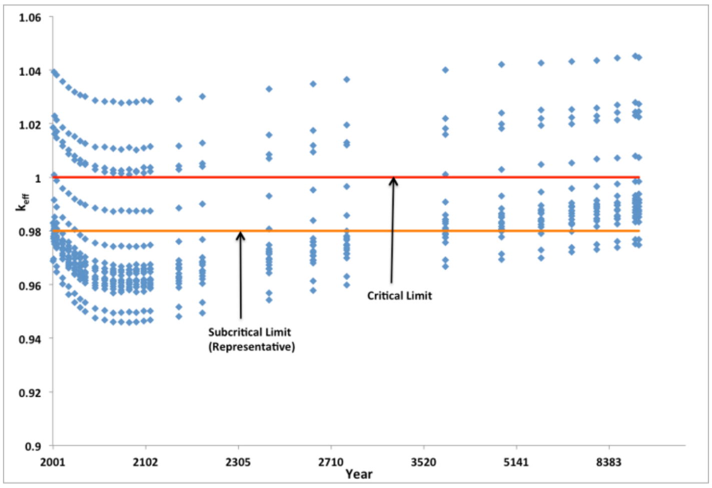

Figure 5: k_{eff} vs calendar year for the Site C basket degradation case, based on actual loading.

Effects of Aqueous Species in Groundwater

As mentioned previously, neutron moderation by flooding with water is needed for a waste package to achieve criticality. However, the groundwater (or pore water) that may potentially flood a breached DPC will contain various dissolved aqueous species. The dissolved aqueous species in the groundwater can (1) act as neutron absorbers (e.g. $_{35}Cl$ and ^{6}Li) and (2) displace moderating elements (e.g. hydrogen). Only the neutron absorption characteristic of the groundwater species is studied in this section.

Currently, various geologic media settings are being considered for repository evaluations; these include crystalline rock, clay/shale, rock salt, and sedimentary rock, among others [16]. Available groundwater data indicate that chlorine (as chloride) is the only naturally abundant neutron-absorbing element in groundwater that can provide significant reduction of reactivity and is available in most of the repository concepts under consideration. However, review of groundwater literature [17, 18] shows that concentrations of dissolved aqueous species vary widely. For example, pore water in Opalinus clay contains about 10 000 mg/L (ppm) of chlorine [17], while the chlorine content of salt brine could be more than 150 000 mg/L [18].

Table 1: Summary of DPC Criticality Analyses in the Calendar Year 9999

Description	Value
Total DPCs analyzed	184
Total DPCs that fail subcriticality with loss of neutron absorber (design-basis loading)	184
Total DPCs that fail subcriticality (as loaded)	
Loss of neutron absorber	25 (13.5%)
Loss of neutron absorber and carbon steel structures	43 (25+18) (23.4%)

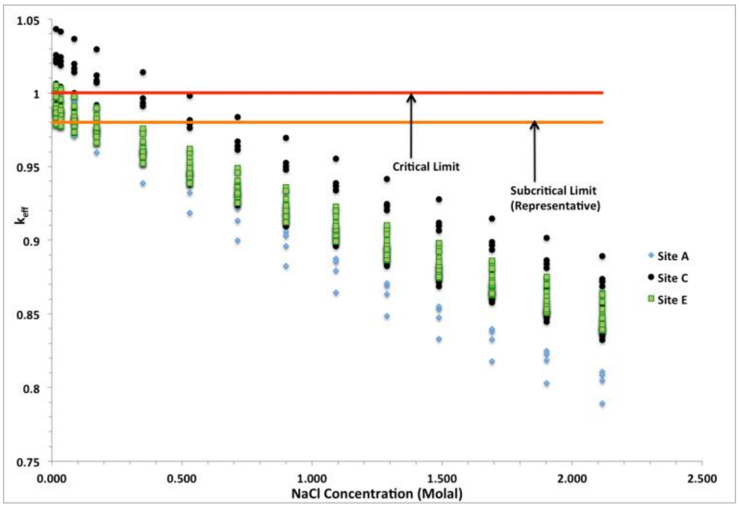

Figure 6: k_{eff} vs NaCl concentration for the loss-of-neutron-absorber (Sites A and E) and degraded basket (Site C) cases, based on actual loading.

Figure 6 presents the reactivity as a function of NaCl concentration in the calendar year 9999 for the DPCs at Sites A and E with loss of neutron absorber and at Site C with a degraded basket. DPCs that yielded k_{eff} 0.98 or greater with fresh water were only analyzed with NaCl solution. Figure 6 indicates that 0.8 molal NaCl solution (~28,000 mg/L Cl) would be sufficient to maintain k_{eff} below 0.98 for all the DPCs at Sites A, C, and E.

Effects of Engineering Filler Materials

An engineering option to mitigate the potential for DPC post-closure criticality is to fill the canister cavity with engineering filler materials that can prevent flooding of the DPC (moderator displacement) during the repository time frame. Filler material can also be selected to provide neutron absorption in addition to its moderator displacement functionality.

This section investigates the moderator displacement aspects of the filler materials. Aluminum was used as a representative filler material that only provides water displacement. Additionally, gibbsite ($Al(OH)_3$), which is a potential corrosion product (mineral) of aluminum in the presence of water [19], was also considered for this study. 58% and 68% volumetric mixtures of filler materials were considered. For example, 58% aluminum was modeled as aluminum slurry, which was a mixture of 58% by volume aluminum powder and 42% by volume water. Figure 7 depicts the cross section of a Site E DPC with the complete loss of neutron absorber as modeled in KENO-VI with filler material (filed up to certain level). It is assumed that the filler material for the loss-of-neutron-absorption scenario uniformly fills all the basket cells

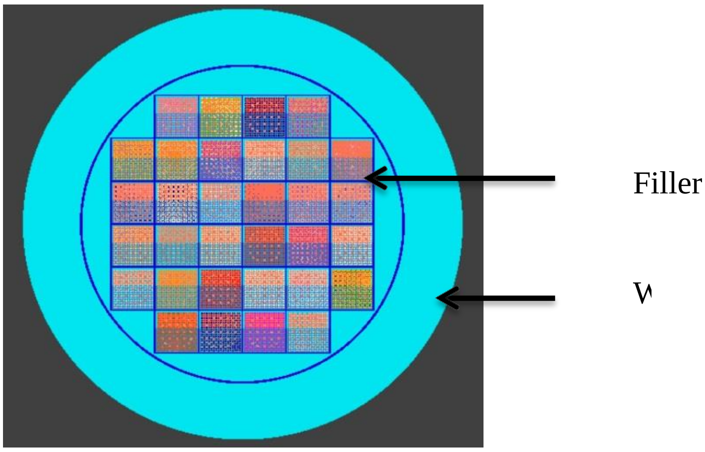

Figure 7: Center plane of the Site E DPC KENO-VI model with complete loss of neutron absorber used for filler materials study.

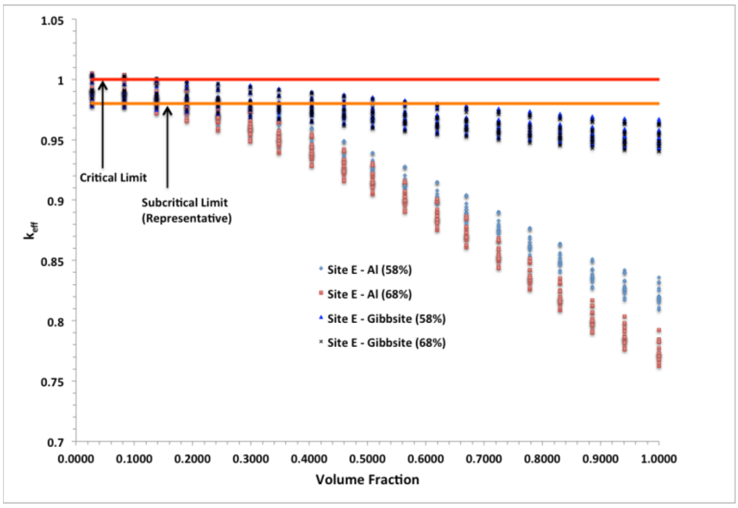

Figure 8: Reactivity as a function of aluminum and gibbsite volume fraction for DPCs at Site E.

Figure 8 presents reactivity as a function of filler material volume fraction for the DPCs at Site E (with loss of neutron absorber) for the calendar year 9999. Only Site E DPCs that yielded k_{eff} or greater with fresh water were analyzed with filler materials. The volume fraction was calculated by dividing the volume of the filler material in a basket cell by the free volume of that basket cell for the complete loss of neutron absorber case. Figure 8 shows that about 30% volume (58% volumetric mixture) is required to be filled (uniformly) by aluminum slurry to maintain k_{eff} below 0.98 for all the DPCs at Site E in the year of 9999. The required filled volume fraction would be slightly lower with a 68% volumetric mixture of aluminum. However, if the aluminum turns into gibbsite over the repository performance period, about 65% volume would be required to be filled. Note that aluminum is only used in this study as a representative material. Gibbsite (potential corrosion product of aluminum over the repository performance period) is included in this study to show that the eventual corrosion product(s) (minerals) of a filler material and its criticality implication must be considered as a filler material selection criterion. Candidate filler materials and filling methods are discussed in Ref. 3.

Conclusion

Direct disposal of SNF currently stored in DPCs is assumed to include a disposal overpack designed to provide support and containment in the specific disposal environment. If one or more of these overpacks failed within the regulatory performance period for disposal, the DPC and its contents could be exposed to groundwater for thousands of years with the possibility of flooding. Neutron absorber materials used in current DPC designs are typically aluminum based and would readily degrade under long-term exposure to groundwater. This paper describes analysis of the potential for criticality. Note that if groundwater can be excluded from waste packages, there is virtually no potential for criticality.

Criticality analyses were performed (using fresh water) for six types of DPCs (total 184 DPCs) located at six sites for two canister degradation scenarios: (1) loss of neutron absorber and (2) basket degradation. Conservatively, degraded materials (basket or absorber) were not credited in the criticality analysis because their locations within the basket were unknown. The main sources of reactivity margin (relative to licensing design basis analyses) investigated in this paper include:

- burnup credit for 28 actinide and fission product nuclides previously demonstrated to exhibit a significant effect on fuel reactivity and
- use of actual as-loaded DPCs, crediting actual assembly design, and reactor depletion conditions.

Additional analyses were performed to calculate the amount of chlorine required to maintain subcriticality (using the representative subcriticality limit) for the as-loaded DPCs at Sites A, C, and E. Site E DPCs were also analyzed with varying volume fractions of filler materials. The analyses performed in this report indicate that demonstrating subcriticality over a repository performance period may be attainable by combining detailed canister-specific analysis with full burnup credit and crediting chlorine in the repository groundwater composition if it is available in high enough concentrations. Preconditioning measures such as adding filler materials to fill the canister void region and displace the moderator could be another option to mitigate post- closure criticality.

References

1. K. Banerjee, J. M. Scaglione, and J. B. Clarity, “Feasibility of Direct Disposal of Dual Purpose Canisters from Criticality Perspective,” Proc. of Institute of Nuclear Materials Management (INMM)—55th Annual meeting, July 20–24, 2014, Atlanta.
2. K. Banerjee, J. M. Scaglione, and J. B. Clarity, “Disposability of Loaded U.S. Dual- Purpose Canisters from a Criticality Standpoint,” Accepted in the International High- Level Radioactive Waste Management Conference, April 12–16, 2015, Charleston, S.C.
3. R. T. Jubin, K. Banerjee, and T. Severynse, “Evaluation of Filler Materials to Control Post-closure Criticality of Dual Purpose Canisters,” Accepted in the International High- Level Radioactive Waste Management Conference, April 12–16, 2015, Charleston, S.C.
4. “SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design,” ORNL/TM-2005/39 Version 6.1, Radiation Safety Information Computational Center at Oak Ridge National Laboratory (2011), Oak Ridge, Tenn.
5. K. Banerjee et al., “Integrated Data and Analysis Tool for Used Nuclear Fuel Management,” Transaction of American Nuclear Society (ANS)—Winter Meeting, 111, 338–41, November 9–13, 2014, Anaheim, Calif.
6. J. C. Wagner et al., “Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses,” NUREG/CR-6801 (ORNL/TM-2001/273), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tenn. (March 2003).
7. G. Radulescu et al., “An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses–Isotopic Composition Predictions,” NUREG/CR-7108 (ORNL/TM-2011/514), prepared for the U.S. Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tenn. (April 2012).

8. Bechtel SAIC company LLC, “Aqueous Corrosion Rates for Waste Package Materials,” ANL-DSD-MD-000001 REV 01, Prepared for US Department of Energy, March 2006.
9. NAC-UMS® Final Safety Analyses Report, Rev. 3, US NRC Docket No. 72-1015, NAC International, January 2004.
10. NAC-STC Safety Analysis Report, Rev. 15, US NRC Docket No. 71-9235, March 2004.
11. Transnuclear, Inc., Multi-Purpose Cask, Rev. 17, Safety Analysis Report NUHOMS®- MP187, Transnuclear, Inc., Fremont, Calif., July 2003.
12. B. Hanson et al., Gap Analysis to Support Extended Storage of Used Nuclear Fuel, Rev. 0, PNNL-20509, Pacific Northwest National Laboratory, Richland, Wash., January 2012.
13. Holtec International, Storage, Transportation, and Repository Cask System Safety Analysis Report, Rev. 10, US NRC Docket No. 71-9261, August 2003.
14. Holtec Final Safety Analysis Report for the HI-STORM 100 Cask System, Rev. 9, February 13 (2010), ADAMS Accession Number ML101400161.
15. J. C. Wagner et al., “Recommendations on the Credit for Cooling Time in PWR Burnup Credit Analyses,” NUREG/CR-6781 (ORNL/TM-2001/272), prepared for the US Nuclear Regulatory Commission by Oak Ridge National Laboratory, Oak Ridge, Tenn. (January 2003).
16. E. Hardin et al., Repository Reference Disposal Concepts and Thermal Management Analysis, FCRD-USED-2012-000219, Rev. 2, US Department of Energy, Used Fuel Disposition R&D Campaign, November 2012.
17. Y. Wang et al., Integrated Tool Development for Used Fuel Disposition Natural System Evaluation–Phase I Report, prepared for the US Department of Energy Used Fuel Disposition, FCRD-UFD-2012-000229, SAND2012-7073P, 2012.
18. J. Winterle et al., Geological Disposal of High-Level Radioactive Waste in Salt Formation, Center for Nuclear Waste Regulatory Analyses, San Antonio, Tex., March 2012.
19. Office of Civilian Radioactive Waste Management Office of Repository Development, Geochemistry Model Validation Report: Material Degradation and Release Model, ANL-EBS-GS-000001 REV 02 (September 2007).

Statistical evaluation of criticality-related events in fuel cycle facilities included in the VIBS database

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ABSTRACT

Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) and the Federal Office for Radiation Protection (BfS) run a database called VIBS (*Vorkommnisse im Brennstoffkreislauf* - Incidents in fuel cycle facilities), where published events in fuel cycle facilities around the world are collected and evaluated as far as possible based on the information available. Until September 2014, 350 criticality-related events have become known. The paper provides a short introduction to the VIBS database in general. In terms of criticality-related events, a dedicated elaboration in terms of type of event, type of facility, type of consequences, and further lessons learned and conclusions was carried out, allowing the evaluation of a comprehensive operating experience and the identification of certain tendencies.

1. INTRODUCTION

Outside nuclear power generating and designated experimental facilities, where fissile material is handled or stored, achieving criticality poses a substantial threat to working staff. Both the competent authorities and the operators of the facilities have a strong interest in continuous optimization of the safety of their facilities. Therefore, all parties involved try to learn from past occurrences in order to avoid them in the future as far as possible and to implement efficient countermeasures. Evaluation of incidents and other operational experiences contributes in first place to the identification of weak points in technical design and process technology and in second place to the development of improvement measures. Within the German approach, a database was established in 1986 to document national and international incidents and lessons learned from the operation of fuel cycle facilities. About 4,700 events were recorded until today.

2. GERMAN PROCEDURE

In Germany, the notification of accidents, incidents and other significant events is addressed in the "Ordinance on the Protection against Damage and Injuries Caused by Ionizing Radiation" (Radiation Protection Ordinance - StrlSchV) [1] and the "Ordinance on the Nuclear Safety Officer and the Reporting of Incidents and other Events" (Nuclear Safety Officer and Reporting Ordinance - AtSMV) [2]. Reporting criteria are specified as far as possible plant-specifically in the AtSMV. This ordinance also contains standardized reporting forms. After the notification has been delivered by the licensee, the competent supervisory *Land* authority analyzes the report with regard to possible safety deficiencies and decides on necessary corrective measures. Usually, external experts from the technical safety organization TÜV (*Technischer Überwachungsverein*) are involved in this context. A copy of the event report is submitted to the Federal Office for Radiation Protection (BfS), which is responsible for the central registration of reportable events in German nuclear facilities. The main data are saved in the fuel cycle incident database called VIBS. *Gesellschaft für Anlagen- und Reaktorsicherheit* (GRS), the nuclear safety organization that supports the Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (BMUB), gets involved in case of events with significant release of radioactivity or potential risk to the public, e. g. events rated INES level 2 or higher, by preparing a more detailed safety evaluation. Moreover, GRS is in charge of recording and evaluating incidents and other significant events in foreign nuclear fuel cycle facilities, because such experiences also provide valuable information for the avoidance of incidents in German facilities. A flow-sheet illustrating the responsibilities and the procedural steps is shown in **Figure 1**. Regarding the completeness and the level of detail of information recorded, a significant difference exists between events in German facilities and abroad. While for events in German facilities the organizations involved have direct and legally defined access to the relevant information, for events in foreign facilities the available information is usually limited to publicly accessible sources, such as the internet, annual reports, conference proceedings, professional journals, etc. Consequently, the VIBS database does not claim completeness of contents for events in foreign facilities. The processing of information in VIBS will be described in chapter 3. The recorded events are documented in form of individual sheets, which are provided subsequently to authorities, experts and operators. Furthermore, GRS prepares annual reports, wherein all events in foreign fuel cycle facilities per calendar year are compiled and statistically evaluated. These reports are also provided to authorities, experts and operators. On the other hand, BfS is responsible for notifying selected events to the international database FINAS (Fuel Incident Notification and Analysis System) [3] of the IAEA and the OECD/NEA.

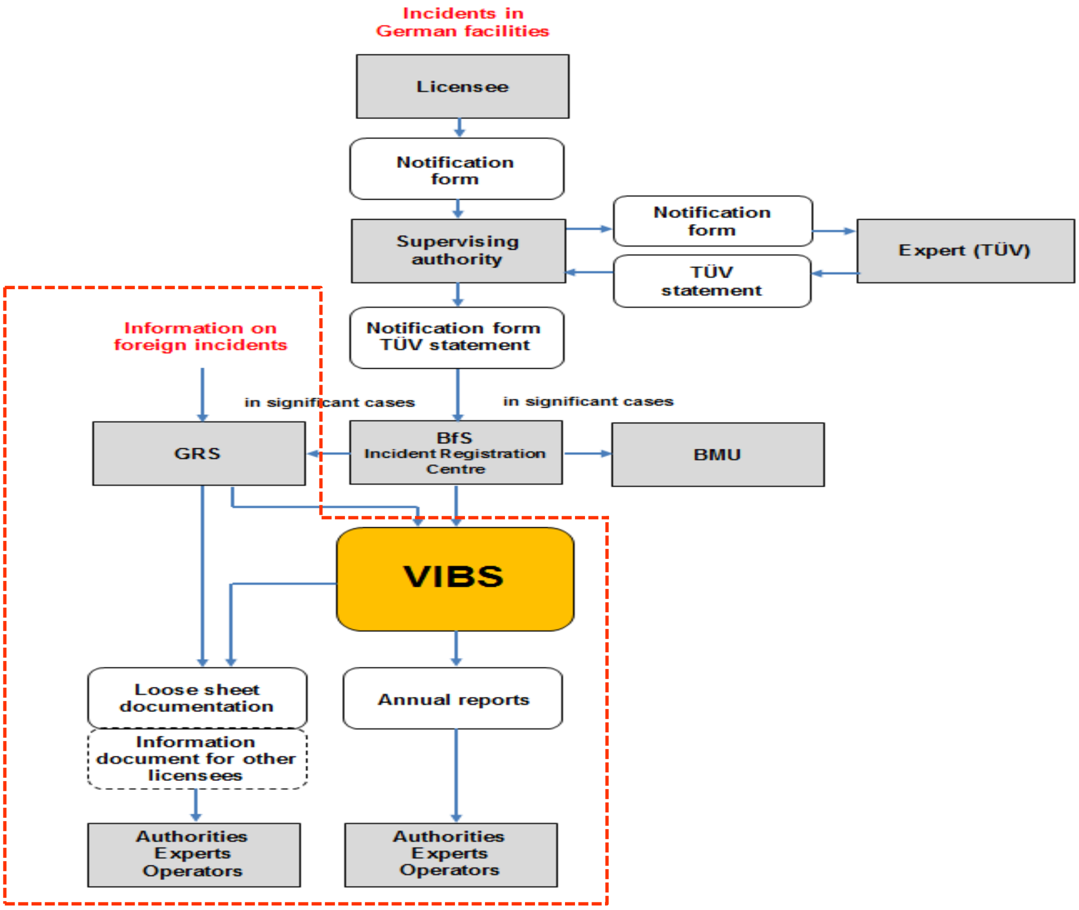

Figure 1: Reporting procedure for incidents in German and foreign (red box) nuclear fuel cycle facilities

3. DATABASE VIBS

In 1986, the ORACLE database VIBS was established for the coverage of German and foreign incidents in nuclear fuel cycle facilities. The input screen for general event information is shown in **Figure 2**. The database is organized in such a manner that, for each registered event, information is stored about the facility involved, its licensee, date and type of event (e. g. criticality, explosion, fire, releases), followed by a running text with a brief description of the event including its chronology, causes and performed countermeasures. As far as possible, information on involved systems, components or devices and on the radiological consequences is recorded as well. If necessary, a preliminary rating of the INES level of the event is carried out, but only in those cases where no rating has been provided by the respective licensee or the competent safety authority. As a result of this procedure, a loose sheet is produced, which includes all the above-mentioned information concerning the event.

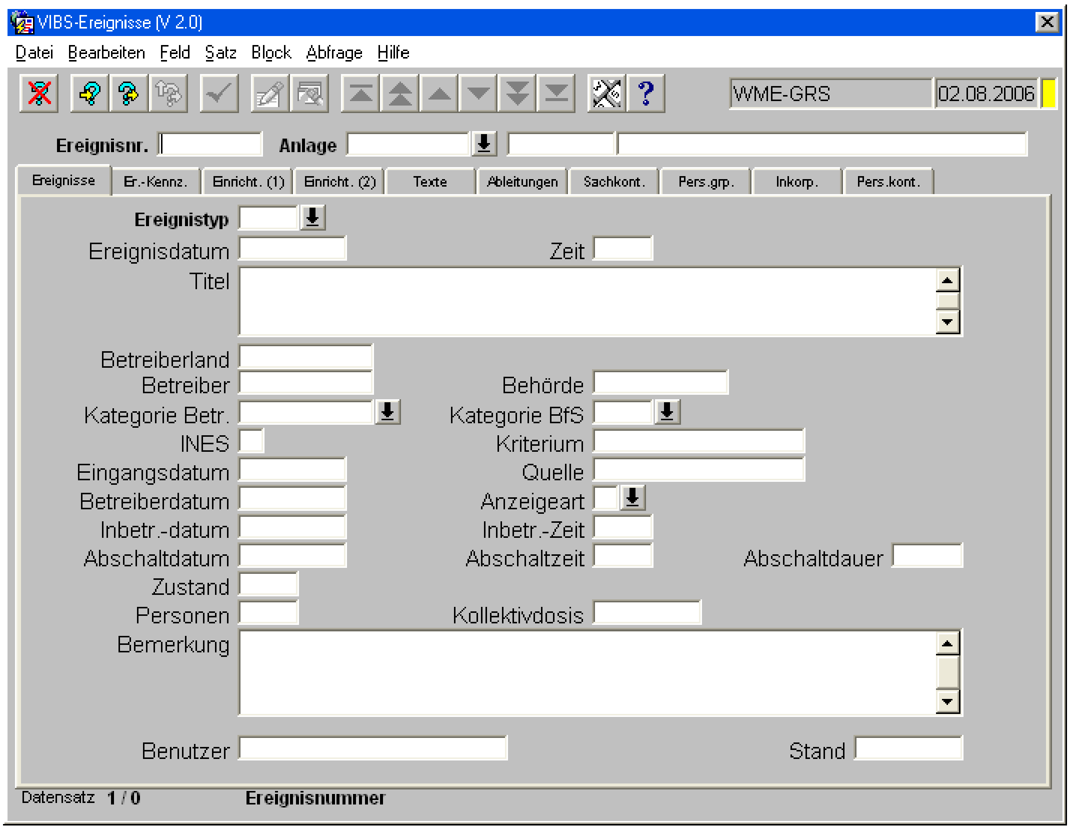

Figure 2: Input screen of the database VIBS

Currently 4,694 events are recorded in the VIBS database, of which 1,542 have occurred in German facilities and 3,152 in foreign facilities. Entry of key words and key numbers by the user allows a simple data evaluation via a full text search function. The potential of VIBS will be demonstrated in the following chapter, using criticality safety related incidents as an example.

4. CRITICALITY-RELATED EVENTS

4.1 DATA BASE

Till the end of September 2014, 350 criticality safety related incidents in fuel cycle facilities and research institutions became publicly known. In 326 of them, including 5 events in German facilities, safety precautions against criticality were affected or there have been lessons learned regarding criticality safety. In 24 cases, a criticality excursion occurred.

The 326 events without an excursion are classified in nine categories as follows:

a) Deficiencies in administrative control of fissile material (enrichment level, material type) (37 events)
b) Deficiencies in or omission of mass control (76 events)
c) Non-compliance of geometric limitations of canisters or components, or problems with neutron absorbing materials (33 events)
d) Deficiencies in or violation of moderation restrictions (50 events)
e) Leakage or misrouting of solutions containing fissile material (18 events)
f) Unintended accumulation of fissile material (30 events)
g) Increase of neutron interaction due to violation of safety precautions regarding distances or canister amount limitations (29 cases)
h) Deficiencies in clearance and authorization of activities, or in the correction of malfunctions / failures (20 events)
i) Impairments or failure of criticality safety related systems (33 events)

The events occurred in following facility types of the nuclear fuel cycle:

- FA - Fuel assembly fabrication
- EN - Enrichment
- RP - Reprocessing
- RF - Research facility
- PF - Radioisotope production facility
- OT - Others (e. g. Spent fuel storage)

Table 1 shows the occurrence of events in the different types of fuel cycle facilities.

Table 1: Number of categorized events without excursions in specific fuel cycle facilities

Cat.	FA	EN	RP	RF	PF	OT	Sum	%
a)	18	9	2	6	1	1	37	11,35
b)	47	8	8	11	1	1	76	23,31
c)	29	3	1	0	0	0	33	10,12
d)	34	15	1	0	0	0	50	15,34
e)	13	1	4	0	0	0	18	5,52
f)	25	5	0	0	0	0	30	9,20
g)	17	6	2	1	1	2	29	8,90
h)	11	8	1	0	0	0	20	6,13
i)	25	3	1	1	2	1	33	10,12
Sum	219	58	20	19	5	5	326	100,00
%	67,18	17,79	6,13	5,83	1,53	1,53	100,00	

INES rating of those events led in most cases to the lowest level 0 (86 events) and level 1 (230 events), whereby ratings in some cases have to be treated as a tentative appraisal due to sometimes missing or incomplete in-depth information on the event. The remaining 10 events were rated as INES level 2. **Table 2** shows the INES rating of the events for the specific facility types.

Table 2: INES rating of 326 events without excursion listed by rating and facility type

INES	FA	EN	RP	RF	PF	OT	Sum	%
0	62	17	1	2	3	1	86	26,38
1	148	41	18	17	1	4	230	70,55
Sum	219	58	20	19	5	5	326	100,00
%	67,18	17,79	6,13	5,84	1,53	1,53	100,00	

Following same categorizations and facility types, an overview of the 24 events leading to criticality excursions is provided in **Table 3**.

Table 3: Number of categorized events with criticality excursions in specific fuel cycle facilities

Cat.	FA	EN	RP	RF	OT	Sum	%
a)	0	0	4	1	0	5	20,83
b) **c)**	1	0	3	0	1	5	20,83
e) **f)**	2	0	2	1	1	6	25,00
Sum	4	1	13	4	2	24	100,00
%	16,67	4,17	54,17	16,67	8,33	100,00	

The criticality-related events with excursions which are documented in the VIBS database were rated with INES level 3 in 17 cases and INES level 4 in 7 cases.

It can be gathered from **Table 2** that around 85 % of the events without excursion happened in fuel assembly fabrication and in enrichment plants. Nearly a quarter of those events can be assigned to the category *b* (mass control fissile material), followed by *d* (moderation control) with approx. 15 %. Categories *a* (acceptance fissile material), *c* (geometric limitations), i (loss of criticality related systems), *f* (accumulation of fissile material) and *g* (neutron interaction) have a share of 10 ± 1 %, whereas *h* (authorization of actions) and *e* (misrouting of solution) have the lowest share of around 6 % and 5,5 %, respectively. Despite the much lower number of events with excursions, its plant type related occurrence implies a higher likelihood in reprocessing facilities with 54 %, followed by fuel assembly fabrication and research facilities with 17 % each. The 24 events with excursion happened due to categories *c* (geometric limitations) and *e* (misrouting of solution) with six events each, followed by *a* (acceptance fissile material) and *b* (mass control fissile material) with five events each. Two events were assigned to category *f* (accumulation of fissile material).

4.2 LESSONS LEARNED

An in-depth analysis of the available data shows that control of mass limitations as well as adherence to controlled moderation is of particular interest for maintaining criticality safety in fuel cycle facilities. Measurements based on probe sampling are only as reliable as the personnel performing them. Moreover, accumulation of fissile material in areas where it is not supposed to be (as for instance a vacuum cleaner bag), deficiencies in verifying acceptance criteria, and inappropriate storage of fissile material constitute significant adverse effects for criticality safety.

Two processes in fuel assembly fabrication proved a notable vulnerability in terms of criticality safety precautions. First emphasis was related to waste water systems, which were attributable to prior incidents, susceptible to infiltration of uranium, and consequently significant criticality relevant accumulation of uranium in unfavorable areas along the system. Second emphasis was related to dissolution of scrap uranium or non-conforming return material. In both processes, the content and the amount of fissile material have to be monitored. The monitoring itself, as well as the organizational provisions for the inspection of the functionality of the measuring equipment frequently turned out to be susceptible to human errors. Furthermore, existence or usage of geometrically unsafe canisters, boxes or by design neglected systems/components represent additional threats to criticality safety. Another important finding was common water intrusion into facilities due to faulty water proofing of the roofs. In most cases there have been smaller leakages. Nevertheless, criticality safety could be affected due to the relatively small volumes handled. Therefore, for buildings with areas of controlled moderation, a periodic and carefully documented inspection of the building condition is considered to be an indispensable precautionary measure.

In 316 of the 326 events without criticality excursions, further safety provisions were effective with the result that criticality safety was affected but still ensured. This can be seen as a confirmation of the internationally applied criticality safety concept on the basis of the double contingency principle. Nevertheless, ten events (INES 2) lead to situations where predetermined safety limits either were exceeded or none of the designated safety provisions were maintained in a controlled manner. In these cases, the fact that no criticality accident occurred is only attributable to the given circumstances, which were not in a controlled state at given times.

5. CONCLUSIONS

In Germany, the VIBS database with its current 4,700 entries is an established tool since 1986, which allows authorities, experts and operators to keep track of incidents in national and foreign fuel cycle facilities. The database is providing information not only on the incidents itself but also on their causes, affected systems, structures and components, and the respective countermeasures. The structure of the database enables the user to perform specific evaluations, which can be used to identify certain tendencies. Such evaluation was performed for 350 criticality safety related events in fuel cycle facilities around the world. A distinction was made between the events regarding the presence of a criticality excursion. Based on the available data, it could be shown that 85 % of the events without an excursion happened in fuel assembly fabrication and enrichment plants, while excursions mainly occurred in reprocessing facilities. Most events were caused by deficiencies in mass control and moderation restrictions. 316 of the 350 events were rated INES 1 or 0, which means that further safety provisions were still effective as intended in the internationally applied criticality safety concept.

ACKNOWLEDGEMENTS

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REFERENCES

[1] Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung - StrlSchV) vom 20. Juli 2001 (BGBl. I 2001, Nr. 38, S. 1714), zuletzt geändert durch Artikel 5 der Verordnung vom 11. Dezember 2014 (BGBl. I 2014, Nr. 58, S. 2010),
Ordinance on the Protection against Damage and Injuries Caused by Ionizing Radiation (Radiation Protection Ordinance) of 20 July 2014, last amendment of 11 December 2014.
http://www.bfs.de/de/bfs/recht/rsh/volltext/A1_Englisch/1A_12_14_StrlSchV.pdf

[2] Verordnung über den kerntechnischen Sicherheitsbeauftragten und über die Meldungen von Störfällen und sonstigen Ereignissen (Atomrechtliche Sicherheitsbeauftragten- und Meldeverordnung - AtSMV) vom 14. Oktober 1992 (BGBl. I 1992, Nr. 48, S. 1766), zuletzt geändert durch Verordnung vom 8. Juni 2010 (BGBl. I 2010, Nr. 31, S. 755),
Ordinance on the Nuclear Safety Officer and the Reporting of Incidents and other Events (Nuclear Safety Officer and Reporting Ordinance) of 14 October 1992, last amendment of 8 June 2010.
http://www.bfs.de/de/bfs/recht/rsh/volltext/A1_Englisch/A1_10_10.pdf

[3] IAEA/NEA Fuel Incident Notification and Analysis System (FINAS) Guidelines. IAEA Services Series No. 14, September 2006, http://www-pub.iaea.org/MTCD/publications/PDF/svs_014_web.pdf

[4] W. Mester, B. Gmal, and S. Geupel: German approach to the compilation and evaluation of abnormal occurrences in nuclear fuel cycle facilities. EUROSAFE Forum 2011, Paris, 7–8 November 2011, Workshop “Operating experience feedback on nuclear installations other than NPP”. http://www.eurosafe-forum.org/2011-oef-workshop

NUCLEAR SAFETY MANAGEMENT AT THE JUZBADO FUEL FABRICATION FACILITY

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ABSTRACT

The main objective of Nuclear Safety Management (NSM) at the Juzbado Fuel Fabrication Facility is to maintain the criticality safety margin and to provide, in case a criticality incident occurs, adequate measures to reduce its consequences on workers, members of the public and the environment. In order to achieve these targets, NSM at the Juzbado Plant relies on well-known and internationally standardized principles:

- *Equipment and processes are designed according to previous criticality safety evaluations. These evaluations must be performed under the facility Safety Case conditions, so the probability of occurrence and the consequences of any critical event are minimized.*
- *Criticality safety evaluations are performed according to the "Double Contingency Principle", which states "Process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible". These safety factors are incorporated depending on the process and equipment considered, but in general can be implemented by two types of controls, Engineering and/or Administrative controls.*
- *Integrated Safety Analysis as a methodology to identify all the potential accident sequences during the operation of the facility as well as the items upon which the safety to prevent such accidents or mitigate their consequences to an acceptable level relies on (IROFS, Items Relied On For Safety). ISA project plays also an important role in plant modification process, being evaluated in the design stage.*
- *Specific criticality safety requirements are established on every equipment and process involving nuclear material. These requirements are written on different level documents and updated according to the criticality evaluations results.*
- *A training program is maintained and tailor-fitted to the extent of the personnel responsibilities concerning nuclear material. This training includes specific requirements regarding nuclear material handling, instructions to prevent a criticality accident and on what to do in case of an accident. Employees, and external personnel, are subjected to a continuous training program on all the safety aspects.*
- *Routine inspections are performed by the Nuclear Safety department aiming to check that the plant is operating according to Criticality Safety Requirements.*

This paper describes the practical approach applied at the Juzbado Plant to implement such principles, along with the safety practices and other specific aspects of operational Nuclear Criticality Safety.

1. ENUSA SAFETY COMMITMENT

ENUSA-Juzbado management believes that safety is the main principle upon all its activities rely on. This commitment is established in the Plant's management documents as a company Chairman's responsibility, and then implemented by the Plant's Director. Therefore, this commitment is taken into account in all the activities involved with the fuel assembly fabrication.

2. EQUIPMENT AND PROCESSES MODIFICATION

The safety evaluation process involving equipment and process modification is performed according to the Spanish Regulatory Body (Consejo de Seguridad Nuclear) requirements, addressed in the safety guide GSG-03.01 "Modifications on fuel assembly fabrication facilities". On application of this safety guide, the evaluation process consists of the following stages:

1. Modification proposal, issued by the user;
2. Assignment of the Design and Review Teams' members;
3. Final design;
4. Spanish Regulatory Body approval (if necessary);
5. Internal approval;
6. Modification clearance.

Nuclear Safety staff participates on stages 2, 3 and 5, so we are going to focus in those stages.

If there is nuclear material involved, Nuclear Safety staff participates in the modification process. The kick-off is given by a Committee that decides if the proposed modification is appropriate from the Safety point of view; if so, the Design and Review Project Teams (DPT and RPT) are built-up. In both teams there is a Nuclear Safety Engineer (NSE), among others, such us a Radiological Protection Engineer, or experts from the operational areas.

Once the design stage starts, the equipment/process design is decided upon the agreement of all the team members, and therefore nuclear safety principles are taken into consideration from the early stages. In fact, this methodology and philosophy improvement avoids not only evaluating a closed design, but also enables that the design is made considering safety principles. The NSE is in charge of performing the criticality evaluations ensuring that the proposed design is "safe", and documenting all the evaluation process.

The design proposed by the DPT is then reviewed by the RPT. The NSE member reviews the evaluation performed by his colleague and assure that the safety principles have been properly applied. He verifies the calculations performed and that their results have been implemented in the proposed design. The RPT sends its comments to the DST leader, who agrees with his team mates and incorporates the comments in the project.

The final project, along with the documentation supporting it, is then approved by the Plant's Safety Committee and then, if needed, sent for approval to the Spanish Regulatory Body. The modification is only allowed to be put in production once ENUSA has this approval. Meanwhile, the equipment is identified as "Under modification. Not use".

3. DOUBLE CONTINGENCY PRINCIPLE

Juzbado Nuclear Safety department is in charge of developing and assessing this Safety Commitment from the criticality point of view. The crucial criteria applied is the Double Contingency Principle that compels us to control at least two independent parameters. This well-known principle, ensures that the nuclear material remains in a subcritical state, thus avoiding the occurrence of the criticality accident. Conservatively, a minimum of three parameters are usually controlled at Juzbado facility (Mass, Geometry and Moderation). The rest parameters are assumed in their most pessimistic values (i.e. enrichment is always considered at 5%, which is the license limit for Juzbado; neutron poisons are not used as a criticality control, etc.). This easies the implementation of NCS controls as there is no specific requirements eventually depending on the product (i.e. different enrichment or gadolinium content).

All the processes and equipment involving nuclear material have to be analysed from the criticality point of view. This analysis is supported by internal documents and ensures subcriticality not only in normal conditions but also under accident conditions. By means of to the Double Contingency Principle, nuclear safety engineers state that even in case that a single control parameter fails, subcriticality is maintained. The internal analysis are then verified by an independent reviewer. In case new processes or equipment are analysed, and if they modify the criticality control parameters, it may be necessary to update the ENUSA official documentation, which requires to apply to the Spanish Regulatory Body (Consejo de Seguridad Nuclear) for approval.

Juzbado is a low enrichment facility, thus the most important control parameter is moderation. Nevertheless, the first approach tends to rely on safe geometry or mass. Safe geometry is applied in case of liquid wastes likely to contain high uranium concentration. In this case, pipes and vessels are designed either with safe diameter or volume. Safe mass is not used very often because it would only allow to handle a small amount of uranium. Therefore, this control can only be applied in those areas where uranium is present in low quantities, as it is the case of the Chemical Laboratory where, by administrative inventory control, a maximum mass of 25 kg of UO_2 is allowed.

When it is not possible to apply mass or geometry safe values, moderation is controlled. As an example, the following general requirements are established:

- It is forbidden to store moderator material in areas or near equipment where moderation control is applied (i.e. during the blending process the additive is prepared just when needed).
- No water pipes pass through areas under internal moderation control (e.g. there are no water pipes passing through the uranium powder storage area).
- Water shall not be used as extinguisher for fire-fighting within internal moderation controlled areas, unless authorized by the Emergency Director (the emergency procedures establish this requirement).

DCP is physically implemented by means of two different kind of controls: engineering and/or administrative. To illustrate how these controls are implemented at Juzbado, a couple of examples follow below:

- **Moderation parameter control improvement**:

  The most sensitive process in Juzbado facility, from the moderation control point of view, is the blending process, where a large amount of uranium is mixed with hydrogenated additive. DCP is accomplished by controlling Geometry (blender shape), Mass (amount of uranium poured into the blender) and Moderation (hydrogen-uranium atomic ratio $H/U < 0.41$). In the mainframe of the Juzbado Integrated Safety Analysis (ISA) program, ENUSA has developed a fractionated dosage system that avoids the moderation control failure. This system considers both passive and active engineering controls, first limiting the amount of additives by a pre-set volume container, and second by means of an alveolar valve that spreads the additive addition along the mixing process. This dosage system acts a barrier against the two sequences found in the ISA evaluation: uniform and non-uniform over-moderation. Therefore, the moderation control in the blending process has been changed from just administrative to an active/passive engineering controls supported by administrative practices.

- **Mass parameter control improvement**

  Nuclear Safety organization has launched a modification proposal for all the equipment where mass control relies only on administrative controls. Basically, these equipment are recipients for nuclear scrap material (powder from the pellet grinder, rejected pellets from quality controls, etc.). This scrap is collected in stainless steel cans with the same geometry as the one used in the powder store, so its safety was primarily ensured just as an extension of the latter. These recipients have been analysed as isolated units completely filled with nuclear material and containing any amount of neutron moderator. In fact it was assumed that the internal moderation was given by the minimum critical mass conditions for 5% enrichment. Uranium was modelled as powder or full density pellets. The so modelled isolated units resulted subcritical, which allows to consider mass as a non-controlled parameter. Thus, the criticality controlled parameters for these scrap cans are Geometry and Moderation.

  Nevertheless, these cans, once retired from the equipment, have to be placed on a store, where mass needs to be controlled by administrative procedures. Therefore, we have implemented an active engineering control that continuously checks the weight of these scrap collecting cans. This scale has a pre-set weight limit that, when reached, activates an optical/sonorous alarm and automatically stops the equipment and so avoiding it to exceed the mass limit on the powder store. This active engineering control is supported by an administrative requirement that compels the employee to weight the can on another scale (used for material accountancy), and to verify the mass limit before taking it into the powder store.

4. INTEGRATED SAFETY ANALYSIS

The ISA project Juzbado Facility is the program established by ENUSA to systematically peer review all processes, equipment and facilities, as well as staff activities, in order to ensure that all relevant risks that could cause unacceptable consequences have been properly analyzed and that the appropriate safety measures have been implemented. Its scope is as follows:

- Description of the installation process, equipment and structures.
- Identification and systematic analysis of the installation risks. All events, both internal and external, that exceed the safety operation limits should be systematically identified.
- Identification and evaluation of accident sequences and deviations of processes leading to unacceptable consequences, determining the expected probability of such sequences to occur.
- Identification, description and analysis of the devices upon which the safety to prevent such accidents or mitigate their consequences to an acceptable level relies on.
- Identification of management measures and adequate monitoring to ensure reliability and availability of the safety features.

The risk evaluation methodology depends on the node to be evaluated. It could be used any of the standard methods included in the literature (HAZOP, WHAT IF/Checklist, PHA), although our main references are “Guidelines for Hazard Evaluation Procedures (AIChE, 1992)” and NUREG-1513 “Integrated Safety Analysis Guidance Document”.

Once an ISA is performed over a process or feature, it is crucial to maintain it updated. Thus, any facility modification shall be assessed by an ISA Reviewer qualified person in order to determine whether it affects to the base ISA analysis. Therefore, ISA project plays also an important role in the plant modification process, being evaluated in the Design Project Team stage (see paragraph 2). At this point, the ISA Reviewer may arrange an ISA group meeting to assess the modification and, in its case, issue a report regarding how the base ISA is modified. This review is also performed if any unusual event which may affect the facility safety occurs.

The severity value (S) assigned to every sequence is given according to Table 1. This is an unmitigated value, so it only takes into account the facility safeguards but not the specific to the sequence itself. Therefore, the severity value assigned corresponds to the worst consequences of the event, regardless the safeguards. The final severity value may be assigned through a Criticality Safety Engineer assessment, which could be based on the Margin of Subcriticality (MoS) and Margin of Safety (MS). This engineering assessment may conclude that the severity should be increased or decrease by 1 from its initial value.

Table 1
Severity (S)

Value	**Criticality**		**Radiological**	
	SICP	**OCP**	**Workers**	**Public**
3	Failure of all ICP	Failure all controls	D > 1Sv	D > 250mSv Intake of more than 30mg de U Uranium release exceeding operation limits.
2	Failure of one ICP. Only one ICP remains active	Failure of all controls. Only one control remains active.	50mSv ≤ D ≤ 1Sv	5mSv ≤ D ≤ 250mSv Uranium release below operation limits.
1	Failure of one ICP no challenging the DCP.	Failure of one control no challenging the DCP active.	D < 50mSv	D < 5mSv Uranium release

SICP Several Independent Control Parameters
OCP One control parameter supported by several independent controls.
ICP Independent Control Parameters
DCP Double Contingency Principle
D Effective Dose

The Margin of Subcriticality is defined as MoS = 1- MSM- k_{eff}, where MSM stands for Minimum Subcriticality Margin and is an arbitrary value (usually equal to 0.02), and k_{eff} is the sequence reactivity value on a certain conditions. Therefore, we have an indicator of severity S(A) based on how much the MoS is affected by an accident sequence (MoS(A)) by comparing it to the MoS under normal conditions (MoS(NC)).

$$S(A) = \left| \frac{MoS(A) - MoS(NC)}{MoS(NC)} \right| \leq 1$$

S(A) provides information about how k_{eff} approaches to its critical value after the sequence. S(A) close to 1 means that severity should be increased by 1. Otherwise, if S(A) is close to zero, the severity can be decreased by 1.

The Margin of Safety (MS), for a given criticality control parameter, is the difference between the value of this parameter under normal conditions and the value that results in k_{eff} = 0.98. For instance, the internal moderation parameter is controlled by limiting the Hydrogen – Uranium atomic ratio H/U(NC) < 0.41, so the MS will be calculated as:

$$MS = \frac{\frac{H}{U}(k_{eff} = 0.98) - \frac{H}{U}(NC)}{\frac{H}{U}(NC)}$$

An application example of these criticality safety engineering evaluation is the blending process node, where a large amount of uranium is mixed with hydrogenated additives. The controlled criticality parameters are Mass, Geometry and Moderation. The ISA analysis of this node results on three main sequences regarding moderation control failure, whose k_{eff}, MoS and S(A) values are shown in Table 2:

Table 2

		k_{eff}	**MoS**	**S(A)**
Normal Conditions (**NC**)	Node 3.1 "Blending process"	0.312	0.668	------
Internal moderation control Failure	Sequence 1 "Less U powder"	0.312	0.668	0.000
	Sequence 2 "More pore former (additive)"	0.98	0.00	1.000
	Sequence 3 "Improper mixing process (non-uniform mix)"	0.364	0.616	0.078

Sequences 1 and 3 suppose that Mass and Geometry remain controlled so, considering Table 1, the severity value is S = 1. Regarding sequence 2, moderation and mass failure is considered, so S = 2.

According to MoS and S(A) values for Sequence 1, MoS keeps unchanged compared to its Normal Conditions value, so its severity, unless H/U limit is exceeded, has the lowest value S=1. In fact, this analysis could allow us to reject this sequence as an accident. For sequence 2, S(A) equals to 1 and MoS is 0.00, so its severity can be increased by 1, S=3. Regarding Sequence 3, although S(A) remains below 1, MoS is decreased, so its severity can be increased by 1, so S = 2.

5. SPECIFIC CRITICALITY SAFETY REQUIREMENTS

The conclusions of criticality safety assessments are then written down in the Plant's documentation. For this purpose the Juzbado facility has different kind of documents, depending both upon the equipment and the process.

- Nuclear Safety Procedures: which apply to Nuclear Safety staff, and cover those activities performed by the Department, not directly related to the fabrication process but which are important to ensure safety.
- For instance, as part of the uranium reception process, internal moderation is controlled even before the uranium powder leaves the shipping facility. The material is released for shipment once it is verified that its moisture weight percentage is below 0.40% and the enrichment is less than 5%. This verification is performed by Nuclear Safety Engineers. Before a fuel assembly batch starts fabrication, its compliance to the applicable criticality safety parameters has to be verified (pitch, pellet and tube diameters, etc.).
- Criticality Safety Sheets (CSS): These apply to the fabrication process staff, and contain instructions involving nuclear material handling. They usually apply to processes rather than equipment.

- For example, in the blending process, there is a CSS relating the administrative practices regarding the moderation control, such as forbidding the storage of hydrogenated additives in the area.
- Criticality Safety Posters: Which also apply to the fabrication process staff, and contain instructions involving nuclear material handling. They usually apply to specific equipment: trolleys used to move pellets from one area to another, small cabinets to storage quality control samples, etc.

6. CRITICALITY SAFETY TRAINING

Training is considered as a basic pillar for operational criticality safety. Staff who handles nuclear material must know not only the risks involved with the operation but also why the safety requirements are as they are. At the Juzbado facility, training is personalized depending on the work the staff performs. As a general approach, there are two training levels:

- **Initial training**:

  This training is delivered according to the Spanish Regulatory Body standards IS-12 and IS-06. Employees are trained upon their arrival at the facility. In case of external personnel, this kind of training is delivered on annual basis. Initial training is split into two different steps.

  - Basic training:
    The first time a person enters the facility, he/she is trained on basic nuclear safety concepts. Of course, the first matter is to let him/her know about the fabrication process. A general presentation is delivered giving an insight on what the uranium is used for and explaining the different stages of the fuel assembly fabrication process. This way, they are able to have an idea and situate themselves in the whole process.

    Next session relates the risks involved on uranium handling, starting on what fission process is and how we can protect ourselves. The fundamentals on the criticality control parameters are explained and also the Double Contingency Principle, focusing in the three most important parameters controlled in the facility (Mass, Geometry and Moderation). Finally, the staff makes a test in order to evaluate whether they have understood the Nuclear Safety fundamentals. Everybody needs to pass this test in order to be qualified to work at Juzbado facility.

  - Specific training:
    Once the basic training test has been passed, employees and external personnel are trained on the specific nuclear safety aspects that their duties require. This is training is usually devoted to the Nuclear Safety Sheets, Procedures and/or Posters that apply to the process and/or equipment where the worker is going to work.

- **Continuous Training**:

  This training is delivered according to the Spanish Regulatory Body standard IS-06. It is addressed to employees and permanent external personnel and is delivered annually. It also develops two different levels:

  - Specific training:
    It is an extension of the initial basic training. Thus, it is not only a review of nuclear safety fundamentals but also specific requirements applying to every single fabrication process stage. These training sessions are arranged and addressed to groups of workers performing the same

roles from the nuclear safety requirements point of view (i.e. office staff, ceramic process staff, etc.).

- Update training:
  This is performed in case an employee changes his/her role. So, it is needed to deliver a training session on the nuclear safety requirements that have to be applied. This includes Nuclear Safety Sheets, Procedures and Posters. This training can be also delivered due to important changes in the requirements.

7. NUCLEAR SAFETY INSPECTIONS

Basically, Nuclear Safety staff performs two different kind of inspections in order to ensure and verify that the Nuclear Safety requirements are properly applied. These inspections are performed by qualified staff.

- **Process requirements**: Verification of written requirements on Nuclear Safety Sheets, Procedures and Posters. It is performed at least once a week and its purpose is to verity that the requirements are properly applied in the process areas where uranium is handled (powder, pellets, scrap, wastes, Chemical Laboratory). Inspections are scheduled in order to ensure that all the process areas are covered at least quarterly.

- **Modifications**: As part of the equipment modification process described in paragraph 2, Nuclear Safety Staff verifies that the modification has been done according to the Nuclear Safety evaluation. This inspection shall be performed before using the equipment.

A COMPARISON OF RECENT EVENTS IN THE USA WITH HISTORICAL CRITICALITY ACCIDENTS

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ABSTRACT

Several important criticality safety related events have occurred in US NRC regulated Fuel Cycle Facilities since the last criticality accident of 1999 in Tokai-mura, Japan. Although criticalities did not occur in these events, they do constitute breakdowns in the systems of controls established to prevent criticality. These events, their primary causes, and the lessons learned are described in order to raise awareness with the expectation that doing so will reduce the occurrence of similar events in the future. In addition, these events are compared to similar historical criticality accidents to show the commonality of root causes.

These important events and historical accidents will be used to demonstrate how the breakdown in basic safety principles and the failure to maintain existing control systems continue to contribute to accidents and events. These lessons learned from historical criticality accidents remain relevant today. The failure to properly analyze possible upsets continues to be a leading cause of criticality accidents and significant events. This results in an incomplete system of controls, possible upsets being dismissed as incredible, or not even considered at all. Another important cause is the failure to maintain existing safety controls and enforce procedures. This shows that complacency in criticality safety and failure to apply lessons learned from previous events can lead to a repeat of circumstances which increase the likelihood of a criticality.

I. INTRODUCTION

This paper will compare three of the most significant criticality safety-related events that have occurred in US NRC-regulated Fuel Cycle Facilities since the last criticality accident of 1999 in Tokai-mura, Japan with historical criticality accidents. The historical criticality accidents have been well studied. However, it is the experience of the authors that events with similar root causes and characteristics continue to occur. It is important to learn from past experience in order to prevent future accidents from occurring. To quote the famous philosopher George Santayana, “Those who cannot remember the past are condemned to repeat it.”

This paper will demonstrate how the breakdown of basic safety principles and the failure to maintain existing control systems continue to contribute to events today. In the authors’ experience the most important factor in recent events has been the failure to properly analyze all possible upsets. In some cases an operation or credible upset was not considered by the criticality safety analyst. In other cases credible upsets were considered, but were incorrectly considered incredible. The mistake that is most commonly made by analysts is not incorrectly calculating k_{eff} or determining that a configuration is subcritical. Instead the most common type of mistake is to incorrectly determine what configurations can occur. This is at the heart of all approaches to safety in all fields. First you consider what negative outcomes can occur. Then you determine what steps can be taken to prevent, reduce the likelihood of, or reduce the consequences of these negative outcomes. This principle is enshrined in ANS-8.1-2014, Section 4.1.2, as

“Before a new operation with fissile material is begun, or before an existing operation is changed, it shall be determined that the entire process is subcritical under both normal and credible abnormal conditions.” To do this it is obviously necessary to understand what the “credible abnormal conditions” are. Only then can the analyst determine what controls are needed to ensure that “the entire process is subcritical”.

Another key cause of accidents and events is the failure to maintain existing safety controls and enforce procedures. The best analysis possible and best possible controls and procedures will not prevent accidents if these controls are not maintained and procedures are not followed.

II. CASE STUDIES

II.A Case Study #1: The Westinghouse Incinerator Event, 2004

A US NRC-licensed fuel facility operated an incinerator to burn uranium bearing dry waste (References 4 and 5). This incinerator contained an unfavorable geometry upper chamber above the combustion chamber. The upper chamber was in turn connected to a scrubber through crossover piping.

During the initial evaluation of the incinerator, licensee engineers concluded that criticality was not credible outside the primary combustion chamber. Based on the input waste stream, licensee nuclear criticality safety (NCS) engineers determined that most of the ash coming from the incinerator would accumulate in the primary combustion chamber and that ash resulting from incineration would never exceed a concentration of 21.6 weight percent uranium (wt% U), which is always subcritical in infinite media at optimal moderation. However, the uranium content of the ash was unknown. The licensee took samples of ash in the upper chamber and used the results as the basis for an assumed maximum uranium content of ash that would accumulate in the upper chamber. Ash with higher uranium content was assumed to be too heavy to collect in the upper chamber. To ensure that the 21.6 wt% U value was not exceeded NCS engineers imposed a requirement that the radiological safety organization perform surveys of the upper chamber. These surveys were never performed, but surveys, for material control & accounting were performed. However, the results of these surveys were never communicated back to the NCS organization.

The maximum uranium content assumption was not physically bounding. The incinerator underwent several design changes to improve its performance. These changes were reviewed by the licensee’s NCS function but the maximum uranium content assumption was not challenged. The licensee eventually discovered significant accumulations of ash with uranium concentrations in excess of 21.6 wt% U. Although these accumulations were dry, potential sources of water were available through the wet scrubber system and moderation was not controlled. At the time of discovery there was about 400 kg [880 lb] of ash, with about 27 wt% uranium content.

This event was similar to a number of historical accidents that involved an accumulation in an unfavorable geometry. However, unlike any of the previous accidents, this event involved uranium powder instead of solution, which greatly contributed to preventing a criticality event. Despite the fact that low-enriched uranium (LEU) was involved, more than a critical mass of material still accumulated.

Figure 1 displays the causal factors the authors believe contributed to this event and similar historical accidents. In these cases operational changes and procedural violations resulted in greater than anticipated accumulations in unmonitored, unfavorable geometry areas where controls preventing accumulations were not available and reliable. In many of these cases the upset condition was improperly analyzed. The Westinghouse event was very similar to the 1961 accident at Siberian Chemical, in that a large accumulation of material occurred in a location were little was expected. This is a characteristic of many of the following events and accidents.

Causal Factors	The Incinerator Event [4 & 5]	Siberian Chemical Combine, 14 July 1961 [3]	Siberian Chemical Combine, 2 Dec 1963 [3]	Electrostal, 3 Nov 1965 [3]	Windscale Works, 24 Aug 1970 [3]	Novosibirsk Chemical, 15 May 1997 [3]
Upset not analyzed or improperly analyzed	■		■		■	■
Upset dismissed as incredible	■					■
Process poorly understood or overly complex	■		■		■	
New or unusual process evolution		■		■		
Heavy reliance on administrative control		■		■		
Poor communication	■					
Involved unfavorable geometry	■	■	■	■	■	■
Equipment failure		■				■
Poor configuration management/change control	■	■		■		■
Improperly secured out-of-service equipment						
Inadequate procedure/poor human factors						
No procedures/Verbal instructions						
Procedure not followed/Workaround	■	■		■		
Improper maintenance/post-maintenance testing				■		
Material in unexpected location/accumulation	■	■		■	■	■

Figure 1. Causal factors for the incinerator event and similar historical accidents

After this event the licensee shutdown the incinerator while it re-analyzed the process and imposed a new suite of controls based on mass control. It also instituted better communication between safety disciplines to ensure the criticality safety organization would have access to information concerning fissile material accumulations. The licensee has since re-started this process under the new suite of controls and is now operating normally.

This event had no real impact on the NRC's regulatory scheme as the NRC and its licensees were still adjusting to the creation of Title 10 Code of Federal Regulations Chapter 70 Subpart H a few years before. The initial analysis of the incinerator event had been performed before Subpart H was written. Subpart H requires the use of a more formalized approach to identifying accident sequences and declaring controls than the regulation had previously required.

II.B Case Study #2: The Nuclear Fuel Services Solution Spill, 2006

This US NRC-licensed fuel facility operates both high-enriched uranium (HEU) processes as well as LEU processes. It had constructed a new facility to convert HEU compounds into concentrated uranyl nitrate solution, and process the solution through a solvent extraction system for final purification (Reference 6). Delays in the construction of the new facility resulted in schedule pressure on the licensee and the installation of non-essential equipment not being completed. One such piece of equipment was a tray dissolver filter glovebox that was tagged as out-of-service but not isolated from the main HEU solution transfer line.

During a pre-operational review to compare installed piping and valves to design drawings, an engineer mistook a diverter valve for a block valve and the associated as-built drawing was changed to reflect this error. The incorrect drawing gave the impression that the glovebox was isolated from other process lines.

The licensee's startup procedures included hydrostatic testing of process equipment with natural uranyl nitrate solution. On several occasions after system startup, operators observed and reported solution in the glovebox. But the solution was assumed to be from the hydrostatic testing and was not sampled.

Later, the licensee decided to move the glovebox and the equipment it contained to a new location in the facility. Uranyl nitrate solution was found in the filters and operators drained the system without a specific work procedure and thus did not sample the solution in the filters, restore the original valve line-up, or fully re-tighten the filter cover bolts. The next day a large routine HEU solution transfer took place through the transfer line the glovebox was connected to. Approximately 37 liters [9.8 gallons] of concentrated HEU solution spilled into the filter glovebox, overflowed out of the glovebox drains, and spilled onto the floor. Operators observed the spill as it passed under a door, investigated, observed solution spraying from the tray dissolver filters, and took corrective actions that terminated the event. While investigating the spill the licensee discovered a previously unknown elevator pit near the spill flow path.

This event was similar to a number of the historical accidents that involved inadvertent transfers of solution to an unfavorable geometry. Both the glovebox and the elevator pit were unfavorable geometry. The drains on the glove box were not being maintained as controls, and could have been blocked; however, in this case they maintained the solution to a safe slab height as designed.

Figure 2 displays the causal factors the authors believe contributed to this event and similar historical accidents. In these cases solution was inadvertently transferred to unfavorable geometry vessels. This event was similar to the 1958 accident in Oak Ridge, in that out-of-service components, valve failures, procedural issues, and failing to recognize signs of leaking solution combined to result in an accident.

Causal Factors	The Solution Spill [6]	Oak Ridge, 16 June 1958 [3]	Idaho Chemical, 25 January 1961 [3]	Hanford, 7 April 1962 [3 & 7]	Mayak, 10 Dec 1968 [3]
Upset not analyzed or improperly analyzed					
Upset dismissed as incredible					
Process poorly understood or overly complex					
New or unusual process evolution					
Heavy reliance on administrative control					
Poor communication					
Involved unfavorable geometry					
Equipment failure					
Poor configuration management/change control					
Improperly secured out-of-service equipment					
Inadequate procedure/poor human factors					
No procedures/Verbal instructions					
Procedure not followed/Workaround					
Improper maintenance/post-maintenance testing					
Material in unexpected location/accumulation					

Figure 2. Causal factors for the Solution Spill and similar historical accidents

As a result of this and other events at this facility, the NRC issued a number of Orders shutting down the facility until its safety programs were overhauled. After the licensee completed the required changes to its facility and programs, the NRC allowed it to resume operations with fissile material. The glovebox has since been moved, and the pit in the floor fixed.

The NRC later issued Information Notice (IN) 2007-32 (Reference [6]) to inform licensees of the criticality hazard associated with improperly secured out-of-service equipment.

II.C Case Study #3: The B&W Vacuum Cleaner Event, 2007

Another fuel cycle licensee used raschig ring filled vacuum cleaners to collect floor-mopping solution in areas that processed concentrated HEU solutions (References 8 and 9). The raschig ring-filled vacuum cleaners were occasionally moved to other areas of the facility under instructions in a radiological control procedure. The licensee's radiological control procedures were not reviewed by the criticality safety organization because these procedures were not used to conduct process operations. The procedure used to transport the vacuum cleaners required the vacuum cleaners to be double-bagged in order to control the spread of contamination. The licensee's container control requirements prevented the introduction of plastic bags into areas with HEU solution, but these requirements did not apply in the areas the vacuum was being transported through.

In this event a raschig ring filled vacuum cleaner, nearly full of HEU solution at low uranium concentration, was double bagged in plastic and was being moved per procedure. Because the vacuum cleaner did not have lifting brackets and the licensee had not established specific procedural requirements for transporting vacuum cleaners the move was attempted without proper safety precautions (moving unsecured on a forklift). During this move, the vacuum cleaner was dropped and most of the fissile solution poured into the space between inner and outer plastic bags, separate from the raschig rings, and some also spilled to the floor. The solution in the vacuum during the event was low concentration, as was normal for the operation of the vacuum cleaners. However, the licensee's NCS analysis, procedures, and controls allowed operators to use these vacuum cleaners on high uranium concentration solutions and they had been used that way on occasion. The licensee had not anticipated that solution could become separated from the raschig rings, except in a spill to the floor which was assumed to result in a favorable geometry slab. This upset was unanalyzed because the criticality safety organization did not know that full vacuums would be transported double-bagged in unfavorable geometry bags per the radiological control procedure.

Figure 3 displays the causal factors the authors believe contributed to this event and similar historical accidents. In these cases manual handling operations were conducted outside of approved procedures, or without criticality safety oversight. The licensee's failure to have the criticality safety organization review radiological procedures was the key failure that led to this event. It is noteworthy that the operation was being conducted per an approved procedure. The lack of communication between safety disciplines was another key factor. In this respect it is similar to a number of other events and historical accidents.

Causal Factors	The Vacuum Cleaner Event [8 & 9]	United Nuclear Fuels, 24 July 1964 [3]	Mayak, 2 Jan 1958 [3]	Mayak, 10 Dec 1968 [3]	JCO Fuel Fabrication Plant, 30 Sept 1999 [3]
Upset not analyzed or improperly analyzed	■	■	■	■	
Upset dismissed as incredible					■
Process poorly understood or overly complex	■	■	■	■	■
New or unusual process evolution		■	■	■	■
Heavy reliance on administrative control		■			■
Poor communication	■				
Involved unfavorable geometry	■	■	■	■	■
Equipment failure				■	
Poor configuration management/change control	■	■	■	■	■
Improperly secured out-of-service equipment					
Inadequate procedure/poor human factors	■	■	■		■
No procedures/Verbal instructions	■	■	■	■	
Procedure not followed/Workaround		■	■	■	■
Improper maintenance/post-maintenance testing					
Material in unexpected location/accumulation	■			■	

Figure 3. Causal factors for the Vacuum Cleaner Event and similar historical accidents

As a result of this and other events involving these vacuum cleaners, the licensee has moved away from using them. The licensee has also stopped using raschig rings and has increased its use of favorable geometry equipment, including procuring a number of favorable geometry vacuum cleaners.

This event had no real impact on the NRC's regulatory scheme or inspection program as operations in accordance with the regulations and facility license would have prevented this event. The event did contribute to the issuance of IN 2008-14 (Reference [8]) warning licensees of events caused by conducting operations that have not been reviewed by the criticality safety organization and placing an increased inspection emphasis on ensuring that operations are conducted according to procedures that are approved by the criticality safety organization.

III. OVERALL LESSONS LEARNED: FAILURE TO LEARN 'LESSONS LEARNED'

Figure 4 displays the causal factors the authors believe contributed to these events and a summary of the causal factors for the historical accidents identified in the tables above.

Causal Factors	The Incinerator Event [4 & 5]	The Solution Spill [6]	The Vacuum Cleaner Event [8 & 9]	The Historical Accidents
Upset not analyzed or improperly analyzed				8
Upset dismissed as incredible				2
Process poorly understood or overly complex				6
New or unusual process evolution				9
Heavy reliance on administrative control				4
Poor communication				3
Involved unfavorable geometry				12
Equipment failure				4
Poor configuration management/change control				8
Improperly secured out-of-service equipment				1
Inadequate procedure/poor human factors				5
No procedures/Verbal instructions				3
Procedure not followed/Workaround				7
Improper maintenance/post-maintenance testing				3
Material in unexpected location/accumulation				8

Figure 4. Causal factors for the Recent Events

In the authors' experience the most important factor in these recent events has been the failure to properly analyze all possible upsets. Not understanding the process has been one cause of the failure to analyze possible upsets. Reference [3] advises in its lessons learned section that, "The processes should be familiar and well understood so that abnormal conditions can be recognized." As the process becomes more complex, it becomes more difficult for the analyst to address all possible upsets. Reference [3] calls out one such complex process and advises that, "Operations involving both organic and aqueous solutions require extra diligence in understanding possible upset conditions if mixing of the phases is credible."

In the case studies, poor communication was a primary reason for both improperly analyzed upsets and improper implementation. In case study #1 the criticality safety organization was unaware of information that would have allowed them to correct the erroneous assumption in the incinerator analysis and impose additional controls. In case study #2 an erroneous drawing indicated that the glovebox was isolated from fissile material and the significance of operator reports of fissile material in the glovebox was not recognized. In case study #3 the criticality safety organization was unaware of activities being performed under a radiological control procedure. Reference [8] provides more examples of events that occurred due to operations with fissile material being conducted under procedures that the criticality safety organization had not reviewed. To help prevent future criticality accidents, management should ensure that there is good communication between different safety and operations organizations.

The importance of favorable geometry in maintaining criticality safety has long been recognized and remains one of the best ways to ensure criticality safety. While any criticality accident would by definition involve an unfavorable geometry, the specific geometries involved in these case studies are of interest. Case study #1 involved an unfavorable geometry area which was found to contain unsafe amounts of fissile material bearing ash contrary to safety assumptions. This demonstrates the importance of ensuring that upsets do not result in fissile material in undetected locations. Case study #2 involved an unfavorable geometry pit in supposedly flat floor; which demonstrates the importance of ensuring that favorable geometry equipment are truly favorable geometry. Case study #3 involved a raschig ring filled unfavorable geometry vacuum cleaner and plastic bags which inadvertently became containers after the vacuum cleaner spilled. Plastic bags and sheets are a particularly difficult type of container to control because it is so easy to bring them into volume controlled areas, and because they are widely used for contamination control and other legitimate purposes. It is also noteworthy that during the spill the vacuum cleaner's raschig rings became separated from the solution. Reference [3] advises in its lessons learned section that, "Unfavorable geometry vessels should be avoided in areas where high–concentration solutions might be present." This advice remains valid, and events have shown that care should also be taken with items that are not intended as vessels that could hold fissile material (e.g. ducts, pits, plastic wrap). All these case studies would be trivial events if unfavorable geometries had not been involved.

In all the cases discussed above fissile material ended up in a location where the criticality safety organization was not aware of it. In case studies #2 and #3 this was due to procedures or processes that the criticality safety organization did not review or were unaware of. Reference [3] also advises that, "Criticality control should be part of an integrated program that includes fissile material accountability." More broadly stated it is obvious that the discovery or indication of fissile material in unexpected locations should be investigated by the criticality safety organization, even if the material is in out-of-service components.

Procedural issues have contributed to nearly every event, both recent and historical ones. In some cases there were no procedures for an operation, in others the procedures were inadequate, and in others personnel failed to follow procedures. For example, in case study #2, the glovebox filter was drained without using a specific work procedure which would have required the operators to sample the solution and alerted them to the presence of HEU and the associated criticality hazard. Reference [3] advises that, "Important instructions, information, and procedural changes should always be in writing." In order to prevent accidents, operations with fissile material need to be governed by written procedures that have been reviewed by criticality safety personnel. Even so, procedures and their implementation by humans will not be perfect and good communication between the operators and the criticality safety organization is essential to ensure that the procedures are carried out as intended, and to resolve any confusion or inadequacies that are identified. To this end Reference [3] warns that, "It is one thing to have written procedures that are intended to be followed in order to provide for safe operations. It is another that these procedures are understood and being followed as intended." The NRC noted several cases in Reference [8] where a specific type of procedural issue, operations with procedures that were not reviewed by the criticality safety organization, resulted in an event. As mentioned above, this can be addressed by ensuring good communications, and by emphasizing to operators that all operations with fissile material need to be conducted per a procedure that has been reviewed by the criticality safety organization.

Configuration control, including its subtopic of properly securing out-of-service equipment, and the related topics of maintenance and post-maintenance testing have contributed to a number of historical accidents and recent events. In case study #3, the NCS organization's failure to review the procedures used to transport the vacuum cleaners, is a configuration control failure because review of procedures is part of configuration control. As a result of events in NRC regulated facilities an increased emphasis has been placed on these areas.

IV. CONCLUSION

This paper has reviewed a few of the more significant events that have occurred in commercial US nuclear fuel cycle facilities. Based on the characteristics and causal factors for these events, the authors conclude that breakdowns in the systems of controls established to prevent criticality and the failure to properly analyze possible upsets continue to contribute to significant events, despite the fact that these events are relatively similar to the historical criticality accidents. To avoid future criticality accidents careful attention must be paid to ensuring that all operations with fissile material are reviewed by criticality safety organization, that possible upsets are thoroughly analyzed, and that the system of controls are implemented as intended.

References

1. American Nuclear Society, Nuclear Criticality Safety Division Newsletter, “Thoughts on the History of Criticality Control Practices,” by David R. Smith, 1999.
2. ANSI/ANS-8.1-2014, “Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors.”
3. LA-13638, “A Review of Criticality Accidents,” dated May 2000.
4. “NRC INSPECTION REPORT NO. 70-1151/2004-001,” dated May 13, 2004. (ML041350148)
5. IN 2004-14, “Use of Less Than Optimal Bounding Assumptions in Criticality Safety Analysis at Fuel Cycle Facilities,” dated July 19, 2004. (ML041760122)
6. IN 2007-32, “Out-Of-Service Equipment Connected to In-Service Process Line Results in Fissile Solution Spill at Fuel Cycle Facility,” dated October 15, 2007. (ML072530077)
7. ID-18431, “Summary Report of Accidental Nuclear Excursion Recuplex Operation 234-5 Facility,” dated April 15, 1963.
8. IN 2008-14, “Criticality Safety-Related Events Resulting from Fissile Material Operations Under Procedures Not Reviewed by Criticality Safety Staff,” dated July 29, 2008. (ML081890131)
9. “NRC INSPECTION REPORT 70-27/2007-006 AND NOTICE OF VIOLATION,” dated October 19, 2007. (ML072920120).

International Workshop on
Operational and Regulatory Aspects of Criticality Safety

From geometrically safe configuration to Burn-up credit

Fredrik Johansson

ABSTRACT

SKB (Swedish Nuclear Fuel and Waste Company) is responsible for the back-end of all Swedish spent nuclear fuel from the Swedish nuclear program. All spent fuel is stored in pools at the intermediate storage (Clab), which was built in 1985. At the start the limiting enrichment was 3.3 % U-235 and all fuel was stored in regular steel canisters.

During almost 30 years of operation a lot has changed, e.g.

- Enrichment levels have been increased to maximum 4.6% U-235.
- Compact canisters, with borated steel, have been introduced.
- Gd-credit has been introduced.
- Use of blocked positions in the storage canisters for the most reactive fuel types.
- An application to use Burn-Up credit has been submitted to the authorities.

This has taken Clab from the original situation there safety was ensured by geometrically safe configurations to a situation there safety is ensured by neutron absorbers and administrative procedure. However this change happened gradually during many decades and there was a lack of awareness in the organisation that the situation was actually fundamentally new.

Recent development has changed this situation, e.g.

- The criticality analysis has been redone with more rigorously tested and validated codes.
- Gap analysis for criticality safety has been performed against modern international guidelines.
- Update of procedures, instructions and internal educations are ongoing.

This paper will discuss these changes and the findings concluded from redoing old analysis.

1 SKB - THE COMPANY AND ITS´ RESPONSIBILITY

SKB is co-owned by the owners of the Swedish NPPs and is responsible for the back-end of all Swedish spent nuclear fuel and radioactive waste from the Swedish nuclear program. The responsibility starts when the spent fuel leaves the NPP and ends at the final repository in disposal copper canisters 500 meters beneath the surface at the Forsmark site, see Figure 1-1.

This means that SKB is responsible for the criticality safety at the intermediate pool storage (Clab), the encapsulation plant with storage capacity (Clink), the Spent Fuel Repository and the transportation between these sites. Clab is operational since 1985. In March 2011 SKB applied for a permit to build the future facility Clink and the Spent Fuel Repository. It is currently being reviewed by the Environmental Court and the Swedish Radiation Safety Authority (SSM).

The Swedish nuclear fleet consists at present day of 10 commercial NPPs, 7 ABB Atom BWRs and 3 Westinghouse PWRs. Two ABB Atom BWRs, situated at the Barsebäck site, were shut down more than 10 years ago for political reasons. Previously, one pressurized heavy water reactor at Ågesta and a research reactor at the Studsvik site have also been shut down.

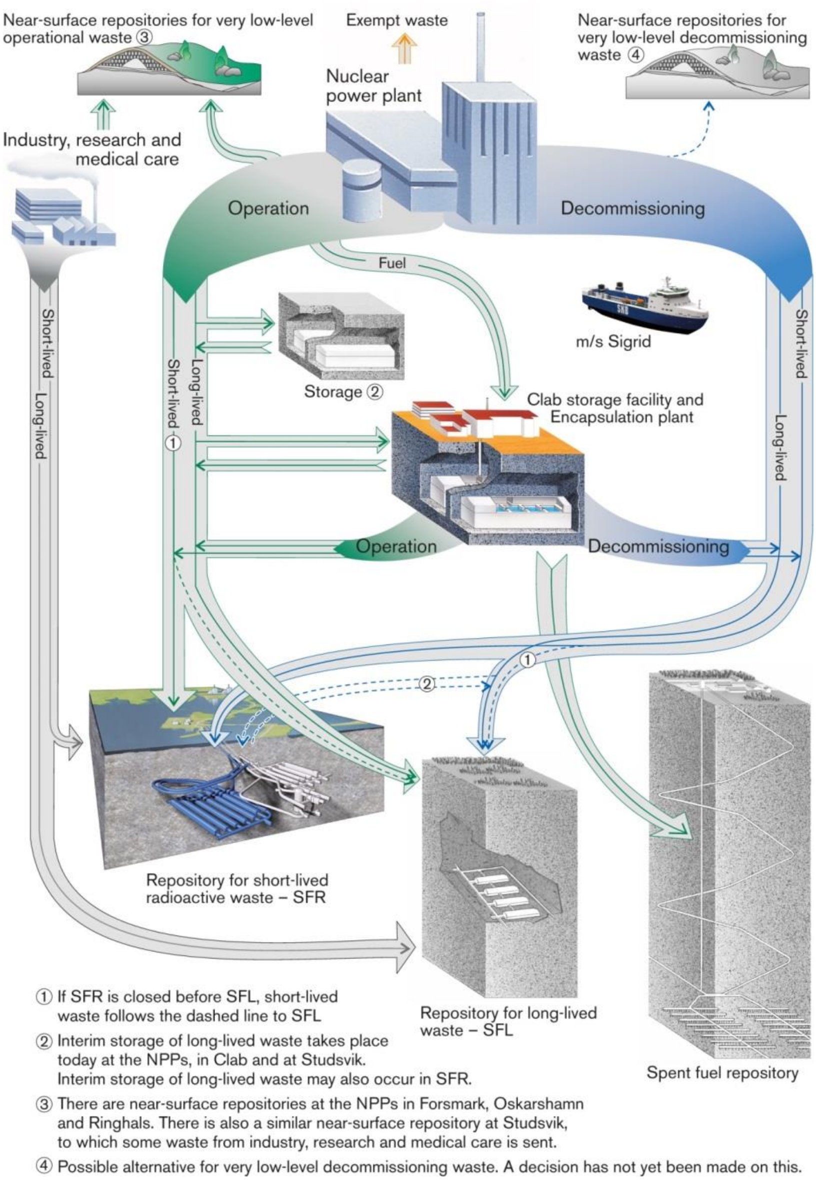


Figure 1-1. The Swedish system for radioactive waste and spent nuclear fuel. [1]

2 ACCEPTANCE CRITERIA FOR CRITICALITY

In the Swedish regulations (SSMFS 2008:1[2] and SSMFS 2008:17[3]) there are no specific requirements on how to perform criticality analysis. The Swedish requirements have instead been established by the analysis that SSM has accepted. The basic acceptance criteria are:

- $k_{eff} < 0.95$ for events in category H1/H2 (Plant condition 1-3 [4], events expected to happen during the lifetime of the plant.)
- $k_{eff} < 0.98$ for events in category H3/H4 (Plant condition 4-5 [4], accidents not expected to happen during the lifetime of the plant.)

The double contingency principle has to be satisfied meaning the combination of two improbable and independent events shall not cause criticality. Acceptance criteria will be based on the combined probability of the events. (This version of the double contingency principle is stricter than the version accepted by NRC [5] who states that criticality can be allowed combining two independent and unlikely changes in process conditions.)

In the case of Clab the original selection of analysed events were performed during design of the facility. The list of events was later complemented at a few occasions when modifications of fuel related systems were made.

New fuel types have to be approved by SSM before they are loaded into the nuclear reactors. In order to ensure that the operators can take care of the fuel both in operation, intermediate and final storage, also an approval from SKB is required.

3 CLAB – 30 YEARS OF OPERATION - THE HISTORY OF CRITICALITY SAFETY ANALYSIS

All spent fuel in Sweden is, after a cooling period, sent to the intermediate storage (Clab) where it is stored in large underground fuel pools. Clab was taken in operation 1985. At the start of operation the limiting enrichment was 3.3 % U-235 and all fuel were stored in normal canisters for 16 BWR or 5 PWR assemblies, see figure 3.1.

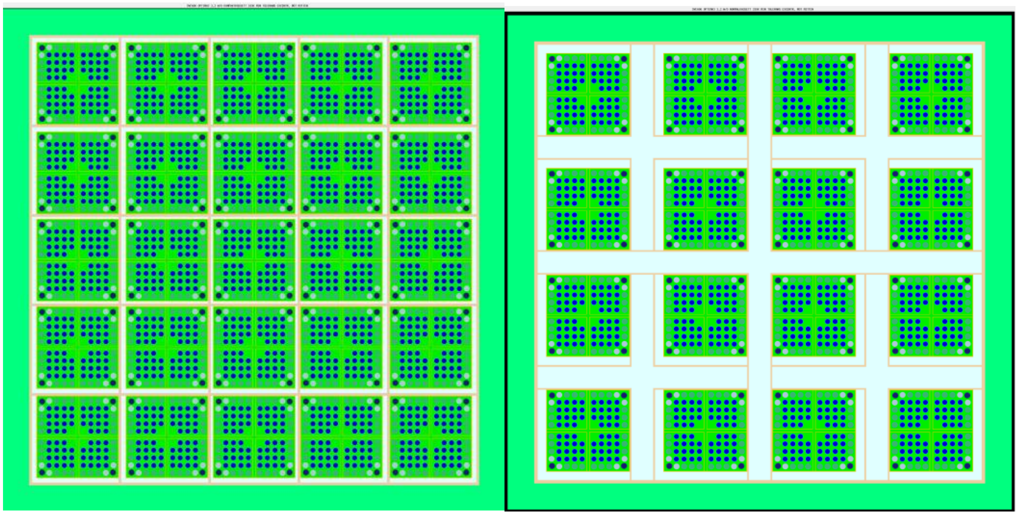


Figure 3-1. To the left BWR Compact canister and to the right BWR Normal canister.

In the original criticality safety analysis there were good margins to the criticality acceptance criteria (k_{eff}=0.95) for all events. Even in highly unlikely incidents of the type: all fuel in a canister are broken and located at the bottom of the canister in an optimal mix between water and uranium, there was margin to the criteria. So, in regard to criticality safety the facility was literally "fool proof".

During the 30 years of operation a lot of things have changed, e.g. development of more reactive fuel types, higher enrichment levels and higher demands on storage capacity. In chronological order:

- Enrichment levels were raised to 3.6% U-235 BWR and 3.75% PWR. New more reactive fuel types were introduced.
- The storage capacity was extended, from 3000tU to 5000tU by introduction of Compact canisters made of borated steel for 25 BWR or 9 PWR assemblies. The enrichment level was raised to 4.2% for PWR.
- Gd-credit was introduced and enrichment levels for BWR were increased to 4.2 % U-^{235}. Administrative routines were introduced to control the level of Gd.
- The facility was extended with a new pool and the storage capacity was increased to 8000 tU. Presently a study on increasing the capacity to 11000tU is ongoing.

- New fuel exceeding the reference limits caused introduction of blocked positions in the storage racks”. Administrative routines were introduced to control the correct positions of blocking devices.
- The use of blocked positions in the compact canister was introduced for fuel enrichments between 4.2 % - 4.6 % U-235 for PWR. The use of normal canisters for PWR was restricted.
- An application to use Burn-Up credit was submitted to the authorities for the final repository and the encapsulation plant with storage capacity.

All these changes, except the last bullet which currently is under review, have been approved by SSM and has formed the new praxis for how to perform criticality analysis (even though they are not transformed into formal regulation). In the overall assessment of these changes SSM has also considered that SKB has improved QA-procedures for how to approve new fuel and fuel types from the NPPs. The Safety Analysis Report (SAR) has been updated on all these occasions. However the original analysis has never been completely redone, i.e. it is now composed by different analysis done with different methodologies, origin from different decades. These calculations have been made with different criticality codes, based on different validations.

And most important:

The old “fool proof” facility is no more. Criticality safety is now ensured by different types of neutron absorbers, blocking plugs and administrative routines. This is a completely different situation from 1985. But the changes have been introduced gradually, and gradual changes do not automatically alarm an organization.

Even though we should be aware that a lot of accidents happen when people are performing according to old routines unaware that the conditions have changed, i.e. Tokai-Mura.

4 THE BOILING FROG

“*The **boiling frog** story is a widespread anecdote describing a frog slowly being boiled alive. The premise is that if a frog is placed in boiling water, it will jump out, but if it is placed in cold water that is slowly heated, it will not perceive the danger and will be cooked to death. The story is often used as a metaphor for the inability or unwillingness of people to react to significant changes that occur gradually*.”

From Wikipedia, the free encyclopedia

As mentioned in the chapter above Clab is presently in a situation there safety is not guaranteed by geometrical safe configuration. But the travel away from that state has been gradual during 30 years of operation. Referring to the poor frog, we have slowly increased the temperature of the water and we have not noticed the difference.

As an examples of this in the past SKB relied heavily on consultants to do the criticality safety work. Procedures and instructions concerning criticality safety were few. The criticality analysis had never been completely revised.

This is not to say that criticality safety was ignored. Changes in the facility could not be done without going through a rigorous review process there safety in different perspective was of highest concern. It is just that criticality safety was not specifically mentioned. And if something is not specifically mentioned the risk increases that it is overlooked, especially since it is not always evident for a non-specialist what changes might affect the criticality.

5 TIME FOR CHANGE

5.1 The wake up calls

In March 2011 SKB applied to the Swedish Environmental Court and to SSM for permit to build a final repository for spent Fuel and Clink. The application contained a criticality safety analysis for the canisters to be used in Clab and Clink and the copper canister to be used in the final repository for spent fuel. In 2012 SSM requested several updates to the application. One of SSM's objections concerned the validation of the codes and methods used in the SKB application. SSM especially emphasized that SKB must better motivate the selection of critical experiments used in the validation suite.

In May 2013 SSM imposed on Clab to update the Safety Analysis Report (SAR) with further detail on the event classification. SSM noted that it follows from SSMFS 2008:1 that the safety analysis of a facility shall be based on a systematic performed inventory of events. The old Clab inventory of events made no distinction between H3 and H4 events and treated H5 (probabilities lower than 10^{-6}) events as a rest risk. After the Fukushima there are new requirements from SSM to include also events beyond design.

5.2 The shape up

For parts of the organisation it was now evident that it was not enough to put another additional criticality analysis on top of the existing ones. It was necessary to do a complete make-over of the criticality analysis methodology, and even more it was also necessary to take a wider look on the whole area of criticality safety.

An improvement project was formed with the object to raise the safety level and meet the demands of SSM. Following has already been done,

- New methodology for criticality analysis, including
  - more thought through and consistent treatment of uncertainties and
  - a specification and update of which requirements and guidelines to use.
- New validation of criticality codes (Scale 6.1) [6] based on ANS/ANSI 8-24 [7] and ANSI 2008-7 [8]. Selections of experiments using the Tsunami tool (part of the Scale code package) to select critical experiments with same neutron physical properties as the safety cases. The selection was complemented by additional experiments to cover all necessary materials and physical properties. Consideration taken to the fact that many criticality experiments are correlated.
- New criticality analysis for Clink and the final repository, including BU-credit. Enhanced quality control of all input data.
- New Gd-credit analysis for all facilities, taking into consideration the new enrichment, fuel types and BA-levels (Gd).
- Gap analysis against IAEA SSG-27 " Criticality Safety in the handling of fissile material" [9]
- More personnel employed dedicated for fuel projects in general and criticality in particular. Focus on competence development and increased knowledge in criticality codes and criticality analysis.

Ongoing activities are

- Update procedures and instructions to match IAEA SSG-27.
- Internal education in "Criticality Safety".
- Update of Clab criticality safety analysis based on new event inventory.

6 FINDINGS IN REDOING CRITICALITY ANALYSIS

As mentioned in the chapter above the criticality analysis and BA-credit analysis were redone based on the new methodology. Lessons learned from that work are summarised below:

- Some assumptions made in old analysis are not considered appropriate today. E.g. when the compact canister was introduced conservative use of the tolerances were made for the material composition but for the normal canister nominal values were used.
- Historical data needed to make the criticality analysis can be very hard to find. It is important to secure input data in an orderly fashion.
- Old analysis is often done with simplification not necessary today. Accuracy can be improved by new more precise modelling.
- The safety concern using old analysis is not in the quality of old analysis itself. It is the organisations capability to properly treat old analysis in a (slowly) changing world. Competent consultants can make excellent analysis but are not in control of how they are used in the organisation, especially in the long run when the assumptions they did are changing. E.g. the original SKB Gd-credit study from 1994 was done with 8x8 fuel as the reference. It was shown that all fuel in Clab was less reactive for all burn-ups with a certain amount of Gd content. That included also one variant of 10x10 fuel. The minimal Gd-content used in the analysis was transformed into limits in the Clab SAR. This limit didn’t change when new more reactive 10x10 fuel types later were introduced. The procedure of approving new fuel types did only check reactivity for fresh fuel and had no check if the assumptions in the Gd-study still were valid. Looking at it strict they were not in this case.
- Going through the event inventory and doing critical calculations for different assumptions you will start to think about if your prerequisites for the calculations are right. The chance is great that you will start to question the event list and you will understand that your calculation is of second to none importance if the plant operator does not perform the way your calculation requisites are set up. So every decade you should redo all your analysis, preferably without employees who were present the last time you did the studies. It is only when you have to do it yourself you will start to question the basic assumption in the study. Rigorously quality controls and review processes can never replace having in-house competence.

7 FINDINGS IN DRIVING CHANGE IN CRITICALITY SAFETY AREA.

Lessons learned from ongoing work in this field is summarised below:

- Redoing the criticality analysis is the easy part in improving criticality safety. The difficult part is to create a sense of urgency for change and improvement in an area which has never experienced any problem. It is very easy that you get objections of the type “We have been doing like this for thirty years and it has always worked fine”. In the mission to create a sense of urgency the benefit of internationally recognised standard and guidelines is most valuable. If you can pinpoint areas there the company do not fulfil the ANS/IAEA guidelines you are more likely to get attention and succeed with you improvement work.
- In the criticality safety analysis a lot of assumptions are made. A lot of the facility is modelled in different detail. In many cases uncertainties in these assumptions are treated rigorously using tolerance data from fuel vendors and with very high precision. Criticality safety specialists are sweating over different uncertainties to win parts of a percentage on reactivity. But in reality it is very unlikely that a criticality incident will occur due to errors in the calculation. It is much more likely to happen due to some unforeseen event or lack/violation of safety procedures. This should be considered when prioritising.
- Both the SKB and SSM shape up was initiated by the new application to build a spent fuel repository. The application forced SKB to raise the standard of its criticality safety reports to meet the latest international standards and guidelines. SSM also started to look at the old facility (Clab)

with the same eyes they had reviewed the new application for the spent fuel repository. In a world with an ageing nuclear fleet and scarcity of new builds the risk of stagnation is evident. Both operators and authorities should be aware of that risk.

8 REFERENCES

[1] TR-10-13 SKB RD&D Programme 2013

[2] SSMFS 2008:1.The Swedish Radiation Safety Authority's Regulations concerning Safety in Nuclear Facilities.

[3] SSMFS 2008:17. The Swedish Radiation Safety Authority's Regulations concerning the Design and Construction of Nuclear Power Reactors

[4] ANSI/ANS-52.1-1983; Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants

[5] NUREG-1520, rev1. Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility

[6] SCALE: A comprehensive modeling and simulation suite for nuclear safety analysis and design. ORNL/TM-2005/39, Version 6.1, Oak Ridge National Laboratory, Oak Ridge, Tennessee, June 2011

[7] ANSI/ANS-8.24-2007: Validation of neutron transport methods for nuclear criticality safety calculations. New York: American Nuclear Society.

[8] ANSI/ANS-8.27-2008: Burnup credit for LWR fuel. New York: American Nuclear Society.

[9] IAEA SSG-27: Safety guide "Criticality Safety in the handling of fissile material"

Major lessons learned by the global analysis of significant events related to criticality safety declared in France between 2005 and 2014

J.P. Daubard, V. Lhomme, IRSN

Abstract

One of the main missions of the IRSN is to provide expert counsel to the government agencies responsible for nuclear safety and radioprotection regarding nuclear and radiological risks associated with both civil and military activities in France. In this framework, the IRSN conducts ongoing technical surveillance of the safety status of each installation, and in particular, analyzes all reports concerning significant events which are transmitted by licensees to the nuclear safety authorities. To supplement this case by case analysis, the IRSN also performs an overall analysis of all events in order to identify global “lessons learned” and to draw generic conclusions.

The paper presents the main tendencies and major conclusions drawn from the global analysis conducted by the IRSN of events related to criticality which have occurred in the past ten years in civil nuclear installations other than reactors. The article underlines the importance of organizational and human failures as the root cause of the majority of these events. It also stress the necessity of performing a deep analysis of these causal factors in order to identify the pertinent corrective actions to be taken to improve the long-term safety of the installations. The article is illustrated by three events classified as level 2 on the INES scale, which are particularly representative of the lessons learned by the global analysis.

1. General review of control and assessment of nuclear safety and radioprotection of basic civil nuclear installations in France

In France, nuclear safety and radioprotection of civil nuclear activities is controlled, on behalf of the French Government, by the French Nuclear Safety Authority (ASN), which has been an independent administrative authority since law no. 2006-686 of 13 June 2006 relative to nuclear transparency and safety (the TSN law) came into force. Within this framework, ASN's mission is to contribute to the establishment of regulations, monitor compliance with the rules and requirements to which French civil basic nuclear installations must conform, inform the public and assist the government in the event of a radiological emergency situation. ASN relies on technical assessments provided, in particular, by the French Institute for Radiological Protection and Nuclear Safety (IRSN), to carry out its missions.

IRSN's main missions

The IRSN is the leading French public service authority in nuclear and radiological risk assessment and research. Created in May 2001, IRSN is an independent public authority with industrial and commercial activities, which comes under the joint authority of the Ministries of Defence, the Environment, Industry, Research, and Health.

One of IRSN's main missions is to contribute to the assessment of nuclear and radiological risks associated with civil and defence activities and installations in France, in support of the relevant public authorities in the field of nuclear safety and radioprotection (ASN for the civil sector and ASND for defence-related activities). At the request of these authorities, IRSN examines and assesses the various safety reports submitted by the operators of basic nuclear installations at the various regulatory stages in the life of installations, as well as in the event of changes that affect them, or following significant events affecting their safety.

In this context, IRSN specifically monitors the safety of basic nuclear installations in order to obtain the most accurate knowledge possible about these installations (including any changes) and their feedback.

Capitalising on the knowledge of these installations enables the safety assessment of basic nuclear installations to better reflect the risks they pose. The purpose of the safety assessment of basic nuclear installations is to provide the authorities (specifically ASN) with timely information that enables them to make an assessment and take decisions regarding the installations in question. In this context, IRSN ensures ongoing technical safety monitoring of each installation (including any changes) and its feedback. The feedback particularly concerns the analysis of significant events reported to the safety authorities by plant operators, based on the event reports they submit. To supplement this case-by-case analysis, IRSN examines all the events in order to learn more general lessons from this feedback and especially to draw generic conclusions. This examination also aims to highlight the improvements noted and the areas where plant operators still need to make progress.

Information regarding the reporting of events to ASN and their analysis by IRSN

The safety of basic nuclear installations requires constant vigilance by everyone concerned and particularly by the plant operators who are primarily responsible for their safety; it must be subject to ongoing improvements. In this context, the French regulations applicable to basic nuclear installations require plant operators to implement an appropriate system to detect, manage and deal with any disparities or anomalies arising in their plants. This system aims to detect disparities of minor importance, which do not necessarily require an individual in-depth analysis, but which may be of interest insofar as their repetitive nature might indicate a problem requiring such an analysis. The objective is to detect "early warning signs" that might be precursors to more serious events.

These regulations also require all basic nuclear installation operators to report to ASN any events that they consider to be significant. That is why ASN has defined the criteria for reporting significant events. They are presented in a guide published on 21 October 2005. As the principles of detecting and dealing with anomalies and incidents can be transposed from safety to radioprotection and environmental protection, the reporting criteria can apply to significant safety events, significant radioprotection and environmental events and significant transport events.

When an event that meets one of the criteria established by ASN occurs, the plant operator is required to report the event to ASN within a maximum of two days of its detection. In addition, he must send his analysis of the event in a significant event report within a maximum of two months of its occurrence. If this document is not final, the plant operator is required to send an updated significant event report at a later date. The documents sent to ASN must also be sent to IRSN. As a general rule, ASN performs an initial systematic analysis of each significant event reported by plant operators and determines the classification of these events on the International Nuclear Event Scale (INES). According to the significance of the event, ASN may have to quickly gather more precise information from the plant operator concerned, for example as part of a "reactive" inspection, or ask IRSN to carry out an in-depth technical analysis. Depending on the information gathered, and with or without IRSN's assessment, ASN may have to request that additional measures to those taken or envisaged by the plant operators be put in place to avoid a recurrence of the event concerned.

2. IRSN's approach to analysing and processing events reported to ASN

For many years, IRSN has been focusing considerable means and resources on capitalising on operating feedback from basic nuclear installations. For basic nuclear installations other than power reactors, such capitalisation mainly depends on the use of information obtained from databases regarding events occurring in French facilities ("SAPIDE LUDD" database managed by IRSN) and foreign facilities (FINAS database managed jointly by the IAEA and the OECD/NEA, as well as Internet monitoring). More specifically, the SAPIDE LUDD database was designed to archive and easily access organised information on significant events occurring in basic nuclear installations other than power reactors. The "SAPIDE LUDD" database currently contains more than 7,000 completed "event" records, the oldest of which relate to events that occurred in the sixties. Apart from the events-related information archiving function, this database was developed as a tool for analysing feedback associated with these events. That is why the database includes a standard classification and indicator system designed to classify each event, as well as a search tool for the assistance of logic operators.

This database is used by the Institute partly to improve the quality of assessments conducted in support of the safety authorities and partly to perform global analyses of the type of events occurring in the facilities. The global analysis of significant events reported by plant operators allows IRSN to:

- learn global lessons about the safety and radioprotection of basic nuclear installations other than PWRs, by highlighting the strengths and weaknesses of controlling the risks associated with the operation and use of these facilities;
- highlight good practices implemented by plant operators to prevent events recurring and improve the safety and radioprotection of basic nuclear installations.

In this respect, it should be emphasised that since 2009, IRSN has published a two-yearly public report drawing on the lessons learned from the global analysis of significant events occurring in basic nuclear installations other than PWRs and reported to ASN; the latest report of this type, which covers events that occurred in 2011 and 2012, is available in English for consultation on the IRSN website Publications and reports: Nuclear safety.

3. Overview of civil laboratory and plant facilities

The classification criteria of nuclear facilities as basic nuclear installations are defined in decree no. 2007-830 of 11 May 2007 in pursuance of the above-mentioned TSN law. By the end of 2014, France had 124 basic nuclear installations, including the fleet of pressurised water nuclear reactors (PWR) operated by EDF, 10 research reactors in operation and 72 nuclear installations such as "laboratories, plants, decommissioned facilities, facilities being dismantled and waste processing, storage and disposal facilities" (hereinafter referred to as LUDD). LUDD-type basic nuclear installations vary greatly (in terms of activity and risk) and are run by different operators (AREVA, CEA, EDF, ANDRA, IONISOS, etc.). LUDD-type basic nuclear installations fall into five broad categories:

- nuclear fuel-cycle facilities;
- research facilities and associated support facilities;
- decommissioned facilities or facilities being dismantled;
- non-nuclear fuel-cycle industrial facilities;
- radioactive waste disposal facilities.

The reporting of significant safety events related to criticality risks involves only the first three categories of basic nuclear installations mentioned above. A brief presentation of these three categories is given below.

NUCLEAR FUEL-CYCLE FACILITIES

This category mainly includes the thirteen basic nuclear installations operated by the AREVA group where the nuclear fuel is prepared for use in nuclear reactors and where the spent fuel is reprocessed after use. They are as follows:

- uranium enrichment plants (Georges Besse I and II) located on the Tricastin site; the Georges Besse I plant was shut down in 2012 and is now being prepared for dismantling;
- the TU5 and COMURHEX plants where uranium obtained from spent fuel reprocessing is converted, located on AREVA's Pierrelatte site;
- FBFC plants where fuel assemblies for PWRs and fuel for research reactors are produced, located on the Romans-sur-Isère site;
- MELOX MOX fuel fabrication facility, located on the Marcoule site;
- spent fuel reprocessing plants (UP3-A and UP2-800) currently in operation, located on the La Hague site and the old UP2-400 plant that has been shut down and is now being prepared for dismantling (with the exception of the HAO facility that has already been subject to a final shutdown and dismantling order (MAD DEM)).

RESEARCH AND ASSOCIATED SUPPORT FACILITIES

This facility category includes the research laboratories operated by the CEA at the Cadarache site (LECA, STAR, CHICADE and LEFCA laboratories), Marcoule site (ATALANTE) and Saclay site (LECI), as well as the support facilities, some of which are dedicated to waste and radioactive liquid effluent management and others to the storage of fissile materials or irradiated fuels. A large number of these basic nuclear installations use or store fissile materials.

FACILITIES DEFINITIVELY SHUT DOWN OR BEING DISMANTLED

This facility category includes LUDD-type facilities that have been subject to a final shutdown and dismantling authorisation order : they are located at the CEA Fontenay-aux-Roses, Saclay, Grenoble and Cadarache sits and on the SICN Veurey-Voroize and EDF Saint-Laurent-des-Eaux sites. The basic nuclear installations operated by the CEA and specifically those on the Cadarache site (ATPu, LPC) are the facilities the most subject to criticality risks.

4. Main lessons learned from the global analysis of "criticality" events

4.1 Introduction

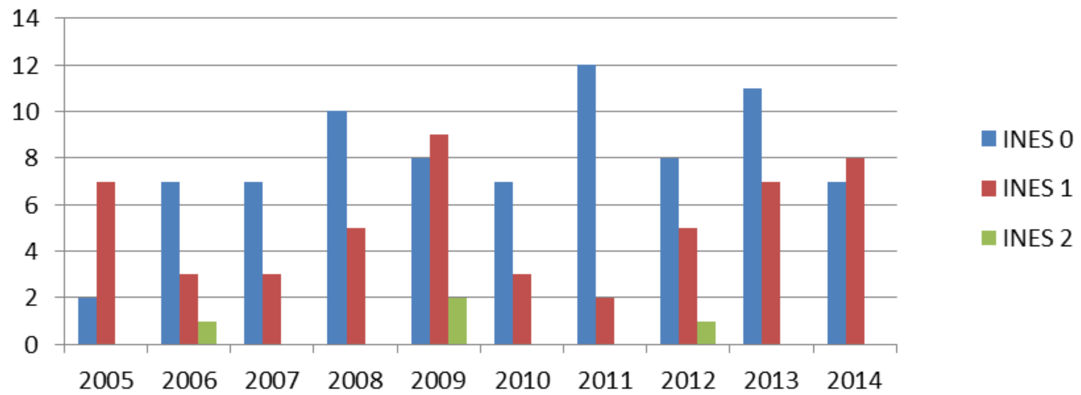
Between 2005 and 2014, i.e. during the past ten years, 135 significant events related to criticality risks (subsequently called "criticality" events) were reported to ASN by the operators of LUDD basic nuclear installations. These events were all reported under significant safety-related event report criterion 3 of the above-mentioned 2005 ASN guide. This criterion applies to any "Event causing a breach of one or more safety limits defined in the safety reference document or the decree authorising the construction of the facility". This category specifically includes cases of non-compliance with the facilities' safety reference document (technical requirements, chapters of the General Operating Rules relating to operating, safety, criticality and radioprotection instructions and to periodic testing).

The following global analysis of "criticality" events occurring between 2005 and 2014 shows the generic lessons learned from the failures observed. However, before presenting the results of this analysis, IRSN wishes to emphasise that the work carried out depends to a great extent on the depth of the analyses of the events reported by plant operators in the significant event reports submitted to ASN. In fact, except for a small number of events for which additional information is available (for example following ASN inspections or IRSN assessments), these reports are IRSN's only source of information.

In this respect, IRSN has observed disparities, sometimes significant, in the content of the analyses given in the plant operators' reports, although certain basic nuclear installations have shown a trend towards improvement over several years. In a number of cases, the reports simply identify the "root" causes and often look no further than equipment failure or human error, without attempting to identify the more fundamental or "deep" causes (including causes of an organisational nature). The inadequacy or absence of a precise identification of "root" causes in some event reports does not always ensure a proper understanding of the different types of failures (technical, organisational or human) that caused the events and, consequently, the generic or recurrent aspects. The inadequacies are a factor that limits the identification of the generic lessons learned from the global analysis of the events.

4.2 Global trends in the numbers of reported events

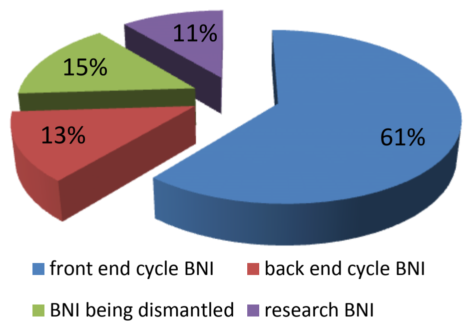
As mentioned above, over the last ten years (2005 to 2014), 135 significant "criticality" events were reported to ASN by the operators of LUDD basic nuclear installations. The increase in the annual number of "criticality" events recorded between 2008 and 2014 compared with those recorded between 2005 and 2007 is generally consistent with the increase recorded for all the significant "safety" events reported to ASN. For several years, the number of significant "criticality" events reported to ASN has generally been stable, in the region of 15 events per year. Fifty-two of these "criticality" events were classified as level 1 on the INES scale and four were classified as level 2. It should be emphasised that between 2005 and 2014, only five events relating to LUDD basic nuclear installations were classified as level 2 on the INES scale in France, including the four events related to criticality risks that occurred in 2006, 2009 and 2012.



Number per year of significant "safety" events related to criticality risks for each INES level

4.3 Breakdown of "criticality" events reported for each type of facility

The analysis of the distribution of "criticality" events according to the type of facilities shows that **approximately 75 % of these events took place in nuclear fuel-cycle facilities** (front end or back end) that were in operation, or during the work phase prior to dismantling; several events involve the old Georges Besse I gaseous-diffusion enrichment plant, which has been shut down since 2012.



This high representation of nuclear fuel-cycle facilities (front and back end) has been constant over the past ten years. The events involving these facilities can be broken down as follows:

- Approximately 52% for UOX or MOX fuel manufacturing facilities (FBFC and MELOX plants);
- Approximately 27% for enrichment plants, the majority of which involve the Georges Besse I gaseous-diffusion enrichment plant and some of which involve the Georges Besse II centrifuge enrichment plant (commissioned in 2012);
- Approximately 18% for the spent fuel reprocessing plants;
- Approximately 3% for the other nuclear fuel-cycle facilities.

The last 25 % of "criticality" events is divided between:

- basic nuclear installations in the MAD–DEM phase (21 events), including the old MOX fuel fabrication plants located on the Cadarache CEA site (ATPu plants and chemical purification laboratory (LPC);
- basic nuclear research installations, including laboratories using new or irradiated fuels or storage facilities for fissile material or irradiated fuels or waste management and radioactive liquid effluents. The number of events related to criticality risks reported by the CEA has been very low for several years.

4.4 Type of events related to criticality risks

4.4.1 Events related to fissile material mass failures

Approximately 35% of "criticality" events involve failures related to the control of fissile material masses. This category of events includes:

- predominantly (approximately 80% of events) excessive fissile material mass or malfunctioning of systems used to control this mass;
- some slow accumulation of fissile material not related to a failure in the mass control systems used (accumulation related to a phenomenon of change in the physical condition of the material (precipitate) for example);
- some shortcomings in estimating the masses of fissile materials when not specifically measured, for example by weighing.

The significance of this category of events is certainly related to the fact that control by limiting the mass of fissile materials is one of the control modes most frequently used in LUDD basic nuclear installations and the most subject to human deficiencies.

A significant proportion of these events led to a mass limit specified in the safety documents for the facilities in question being exceeded. However, none of the events where a mass limit was exceeded jeopardised the sub-criticality of the equipment concerned, given the large margins used for specifying the mass limits given in the safety documents, based on the analysis of various abnormal situations.

With respect to feedback, two types of events should be specifically highlighted. They are described below.

4.4.1.1 Failures in estimating the mass of fissile material accumulated at workstations

Several events showed that the mass of fissile material accumulated at workstations or in other associated equipment (ventilation ducts, etc.) was incorrectly estimated.

The event of 6 October 2009 that occurred at the CEA/Cadarache ATPu facility, which is described in Appendix 1, illustrates this problem. This event led to the discovery of a larger than expected quantity of fissile materials held in glove boxes. It was classified by ASN as level 2 on the INES scale, owing to the many failures noted. Other significant events led to the following discoveries:

- an accumulation of plutonium in an exchanger installed after a filtration device on a glove-box air-cooling loop, when such an accumulation had not been envisaged for the fissile material balances for this equipment;
- in a metal-clad containment, a uranium mass greater than that defined in the safety documents, which had accumulated beneath a work surface; this accumulation was the result of the periodic cleaning being less thorough than expected, due mainly to the fact that the design of the installation did not facilitate this type of cleaning (cross-pieces underneath the work surface preventing the remote manipulation arms accessing part of the floor) and insufficient inspections to ensure that the area had been thoroughly cleaned;
- uranium oxide accumulated in ventilation ducts in the FBFC plant.

The analysis of these events led IRSN to question the practical application of the principles implemented by the licensees to monitor the accumulation of fissile materials, in particular with regard to "resetting the material balance", in terms of criticality, after periodic cleaning of the working enclosures (glove boxes, metal-clad containment, etc.). For IRSN, this feedback underlined the importance of licensees ensuring that the measures they had put in place would guarantee effective compliance with the mass limits of fissile materials at workstations for which a risk of fissile materials accumulation had been identified. These measures should make it possible to fully assess:

- the quantities of fissile material at currently operating workstations, given the uncertainties regarding the masses of fissile materials introduced into and removed from each workstation,
- the residual mass of fissile materials at workstations to be retained after periodic cleaning operations, allowing the fissile materials balance to be reset.

In addition, this feedback served to remind plant operators of the importance of taking great care, during criticality risk analyses, to identify all malfunctions that might lead to an accumulation of fissile materials in equipment associated with process equipment. In fact, several events revealed the inadequacy of the measures adopted (filters, etc.) to prevent materials from being carried into the ventilation ducts or exchangers.

Following IRSN's assessments, in 2009 and 2010, ASN asked all French basic nuclear installation operators to take the feedback from these events into account, particularly the event that occurred at the ATPu. In response to this request, the licensees concerned analysed the adequacy of the principles and practices implemented with regard to any accumulation of materials in their facilities. In certain cases, this led to improvements such as formalising good practices or regularly performing appropriate checks to detect any accumulations being put in place. For IRSN, the additional measures adopted by licensees are headed in the right direction. However, IRSN stresses that licensees should be extremely vigilant with regard to compliance with the provisions for managing and monitoring fissile materials in basic nuclear facilities, both in terms of operating practices (regular cleaning of units) and technical provisions (monitoring, measurements), in order to control the risk of fissile material being accumulated in their plant units.

Moreover, IRSN considers that this feedback should also be used in the design of new workstations for which criticality risks are managed by controlling the masses of the fissile materials present. It is particularly important, as far as possible, to prevent any accumulation of fissile materials in areas that would be difficult to clean or where it would be impossible or difficult to check whether this materials were present. This requires designing equipment that can easily be disassembled to allow visual inspections of its internal parts. It can also involve taking steps to make it easier to take nuclear measurements to identify fissile material accumulation points.

4.4.1.2 Exceeded fissile materials mass limit in waste drums

Several events have shown disparities in the authorised characteristics (mass, isotopic composition, etc.) of fissile materials in waste drums. These events involve "old" waste drums, i.e. drums that had been conditioned for many years. These observations were made when the drums were removed from storage and transported for reconditioning. IRSN's analysis shows that the main failures responsible for the significant discrepancies in the masses of fissile material contained in these drums are due to:

- waste drum filling errors and lack of double checks;
- mass estimation methods inappropriate to the type of material;
- errors in the isotopic composition of the fissile materials (mainly plutonium) used to interpret the measurements;
- lack of consideration of measurement uncertainties, or incorrectly calibrated measurement stations.

Many French or foreign nuclear installations may be affected by errors in the authorised characteristics (mass, isotopic composition, etc.) of fissile materials contained in "old" waste drums. In this respect, feedback from these events indicates that greater care is required when recovering or destocking "old" waste drums. This is due to possible inadequacies in the measures taken to determine the data of the fissile materials in the drums when the installations were in operation. Under such conditions, IRSN believes that it would be good practice for plant operators to systematically measure the mass of fissile material before recovering or destocking an "old" waste drum.

In addition, feedback from these events underlines the importance of plant operators taking appropriate measures to prevent errors in the isotopic composition of the fissile materials occurring during measurement, take measurement uncertainties into account and regularly calibrate the equipment.

4.4.2 Events corresponding to failures in controlling fissile material

10% of "criticality" events reported between 2005 and 2014 are due to failures in controlling fissile materials or the equipment that contains it. These events are very varied: poor management of radioactive sources containing fissile materials, not controlling the ^{235}U content when receiving uranium solutions, etc. In the vast majority of cases, these events are the result of organisational or human failures.

4.4.3 Events related to the control mode by limiting the moderation of fissile materials

Approximately 17% of the "criticality" events reported during the period 2005 to 2014 are associated with the mode of control by limiting the moderation of fissile materials. These events are related to the presence of moderating materials under unforeseen conditions (limit exceeded, labelling error, etc.). A very significant number of these events involved the FBFC fuel manufacturing plants and the Georges Besse I enrichment plant. Although the events at the Georges Besse I plant involve leaks in process heat exchangers related to equipment failure (specifically corrosion of heat exchangers), the other events (mainly involving the FBFC plant) are predominantly due to organisational or human failures. More specifically, feedback from these events shows, in particular, that attention must be paid:

- to the robustness of the organisational measures taken to guard against water accidentally entering equipment containing fissile material (lock-out device, tag, etc.);
- to human interventions on equipment for which the mode of control by limiting the moderation of fissile material is used.

4.4.4 Events related to the mode of control by geometry

Approximately 17% of the "criticality" events reported during the period 2005 to 2014 are associated with the mode of control by geometry. A significant proportion of these events relate to failure to comply with requirements associated with the storage conditions for fissile materials, such as positioning errors for equipment containing fissile materials, failure to comply with the minimum distances required between equipment and workstations, disparities in the dimensions of safe geometry equipment. A few events of varying origin should also be highlighted (overflows, leaks, non-scheduled dismantling of equipment, etc.).

The event that occurred at the FBFC plant on 24 September 2012, which is described in Appendix 2, illustrates non-compliance with the "standard" requirements associated with the storage of fissile materials; this event was classified by ASN as level 2 on the INES scale. In fact, these events were generally caused by organisational and human failures, such as incomplete or poorly explained operating documents, ignorance of the requirements on the part of the operators, or control failures. However, the analyses of the events submitted in the reports do not generally identify the root causes of these failures.

In addition, several events revealed that the actual dimensions of equipment for which the control mode by geometry was used (known as "safe geometry equipment") show deviations from the dimensions specified in the safety documents (for storage equipment, ventilation ducts, etc.). Feedback from these events highlights the need for licensees' safety reviews to include periodic checks for compliance with the requirements specified for equipment considered to be "safe geometry" equipment. In this respect, it is important to remember that the applicable French regulations require that all basic nuclear installations undergo a safety review every ten years. The safety review includes two major parts:

- a compliance review of the facility and specifically of the main safety-related equipment, to check that upgrades (changes, etc.) to the facility have not led to deviations from the requirements specified at the design stage;
- a safety reassessment of the installation.

4.4.5 Events related to criticality detection and alarm systems

Approximately 7% of events reported during the period 2005 to 2014 involve the criticality detection and alarm systems installed in certain plants. Most of these events occurred in CEA facilities and in installations on the spent fuel processing site at La Hague.

They were mainly various types of false alarms, operating faults following interventions (such as periodic inspections or tests, or maintenance work) on equipment in these systems, or false alarms due to failure to carry out the checks specified in the safety documents. Feedback from these events specifically highlights the importance of tests and interventions on equipment being very carefully prepared and monitored, in particular to ensure that the equipment is working properly after interventions have been completed.

4.5 Global analysis of "criticality" events

This section aims to identify general trends in the causes of "criticality" events reported to ASN. Regarding the differences of the analyses submitted by plant operators in significant event reports, it is not possible to identify the precise lessons learned. More specifically, it is often not possible to discover the "root" causes of equipment failures identified (a maintenance fault? a design fault? a badly prepared/executed intervention??) or the "root" causes of human errors (poor preparation? poor work organisation? errors made during an intervention?).

Whatever the reason, the analysis performed shows that technical failure is the main cause of approximately 25 to 30% of events. According to the analyses submitted, it appears that a small proportion of failures are due to design faults. The types of equipment affected by equipment failure are very varied: leaks in heat exchangers used in the uranium gaseous-diffusion enrichment process due to corrosion mechanisms, failures in software used to monitor the masses of fissile materials due to inadequate consideration of particular situations outside the normal process procedure (several events of this type were reported at the MELOX facility), measurement equipment failures, etc.

Most of the "criticality" events reported to ASN are caused by organisational or human failures. No changes in the causes of these events could be identified due to the highly uneven content of the analyses submitted in the significant events reports. The main organisational or human failures identified by plant operators are:

- inadequate or insufficient operating documents;
- non-compliance with procedures or instructions;
- operator errors related, for example, to confusion in identifying equipment;
- failures in personnel training;
- team organisational failures (interface faults, failure of personnel to communicate with one another, etc.).

The global analysis conducted by IRSN identified the fact that a significant proportion of "criticality" events occurred in situations outside the scope of "normal" operations (following changes in procedures, during interventions or degraded situations as a result of equipment failures, specific operations leading to the establishment of a special procedure). The event that occurred at the ATPu on 6 November 2006, described in Appendix 3, perfectly illustrates this type of event. For IRSN, this feedback shows that plant operators must pay attention to the management of these particular situations. Special attention must be paid to the organisational measures that govern operations and to the conditions for implementing them to make them less susceptible to failures; attention must also be paid to the quality of operating documents, the provisions for supporting operators in their knowledge of the procedures and to regularly checking this knowledge.

More generally, IRSN believes that this feedback shows that a significant proportion of the failures identified by plant operators (documentation errors, poor management of "abnormal" situations) are due to a failure to analyse the actual working situations of operators working in the facilities and the conditions under which they occur (context, work organisation, etc.). This can in fact lead to the underestimation or non-identification of the difficulties of performing certain operations. It can result in the definition of inappropriate technical and organisational measures relating to criticality risk management (unsuitable man-machine interface, storage facilities, incomplete or inadequate procedures, etc.) or measures that do not allow to deal with situations other than those specified within the context of normal plant operation to be easily managed.

However, for IRSN, only a socio-technical approach can allow to define a relevant set of criticality safety rules favouring efficient and safe human activities. This approach must be based on a risk analysis, combining technical, organisational and human aspects, in order to define appropriate measures for controlling the risks encountered in nuclear facilities. These analyses must be based on an in-depth knowledge of staff operating practices required for operation and for risk control, as well as a detailed knowledge of the specific context in which these actions are performed.

Appendix 1

EVENT OF 6 OCTOBER 2009 AT THE CEA-CADARACHE PLUTONIUM TECHNOLOGY FACILITY

The plutonium technology workshop (ATPu) on the CEA-Cadarache site manufactured uranium and plutonium oxide based fuels for fast neutron and light water reactors between 1962 and 2003. In 2003, the plant operator stopped commercial production from the facility. Between September 2003 and June 2008, the ATPu reconditioned and dispatched the manufacturing scraps still at the facility to the AREVA plant in La Hague. The final shutdown and dismantling order for the facility was signed in March 2009, after expert assessment by IRSN of the safety documents submitted to ASN.

On 6 October 2009, the plant operator reported a significant event to ASN involving the gradual discovery that the masses of fissile materials retained were significantly larger than expected during glove box dismantlement operations.

With regard to the glove box decontamination operations performed under the final shutdown and dismantling order, criticality risk prevention is based on the limitation of the mass and moderation of fissile materials. The maximum mass of fissile materials adopted, which is common to all monitored stations, is an estimated envelope value for the residual mass at the station with the highest mass according to the “material retained” account. The annual inventory of May 2008 was based on data from a program and indicated a total retained mass of plutonium of approximately 8kg across all the glove boxes

In June 2009, the review performed by the operator showed that the mass of plutonium recovered since the last inventory was significantly higher than expected. In October 2009, another review showed that the total mass of plutonium recovered during dismantlement performed up until this date was of the order of 22kg. Given the mass of plutonium that was still estimated to be retained in the glove boxes, the plant operator estimated that the mass of plutonium in the glove boxes could be as high as 39kg.

This review led the plant operator to report a significant event to ASN on 6 October 2009. ASN classified this event as level 2 on the INES scale, because this underestimate had gone undetected throughout the facility operation period and the event was reported to ASN very late. On 14 October 2009, ASN suspended dismantling operations in the facility and required prior consent for the resumption of work. IRSN submitted an opinion on this event to ASN a few days after the event was reported. This opinion and an information notice can be viewed on the IRSN website (www.irsn.fr).

Investigations carried out by the plant operator showed that the cause of the gradual build-up of fissile materials in the glove boxes was associated with the fact that operation of the ATPu facility led to the spread of fissile materials during the numerous operations to dock and undock the vessels called “jars” and pour out materials that had not been fully recovered. Some glove boxes were designed such that they created retention areas that could not be accessed without complete disassembly where materials built up gradually and could not be detected via visual inspection. In addition, the quantification of residual materials in the glove boxes using radiological inspections and dose rate measurements was made difficult by obstructions in the glove boxes and the large quantities of fissile materials present. As a result, the full amount of disseminated materials could not be recovered during cleaning operations carried out in the operating phase.

In addition, the incorrect assessment of the masses of plutonium retained was associated with uncertainties about the masses of plutonium attributed to incoming and outgoing products (weighing of powder containers and pellet boxes, measurement of plutonium in the waste removed, etc.). In particular, when a deviation in the mass balance was observed after cleaning of a glove box and it could not be associated with a specific event, the plant operator attributed it to the significant uncertainties associated with waste container measurements. This mass was therefore allocated to the waste via an “adjustment account”, while part of it probably corresponded to actual retention of material in the glove boxes.

During 2010 and 2011, ASN gradually authorised the plant operator to restart dismantling activities on the basis of new safety files assessed by IRSN. The operator classified glove boxes into five categories for these files in accordance with the method used to estimate the masses of residual fissile materials. For each glove box category, material recovery provisions were defined that took into account the new estimates.

For IRSN, this event underlines the importance of good glove box design, with criticality risk management that involves monitoring the fissile material masses present. In particular, it is important that this equipment should be designed as much as possible to avoid areas that cannot be accessed or cleaned, where fissile materials could build up. In any event, if it is not possible to eliminate such areas, measures should be planned to assess any retention or identify areas for which more thorough cleaning is required. This event also underlines how important it is that plant operators check the robustness of provisions in place for making an “envelope” estimate of masses of fissile materials present in glove boxes during operation and especially after periodic cleaning.

Given the potentially generic nature of the event, in October 2009 ASN asked basic nuclear facility operators to take into account the corresponding operating feedback. ASN in particular required that they perform a complete review of the residual masses of fissile materials present at facility work stations, whether in operation or dismantling phase. More specifically, ASN asked plant operators to specify methods for monitoring any build-up of fissile materials, any measures that they have taken or planned for the safe recovery of residual materials at the work stations in quantities above those estimated and the measures taken or planned to prevent the uncontrolled build-up of residual materials at work stations.

Appendix 2

EVENT ON 24 SEPTEMBER 2012 IN THE PWR FUEL FABRICATION PLANT IN ROMANS-SUR-ISÈRE

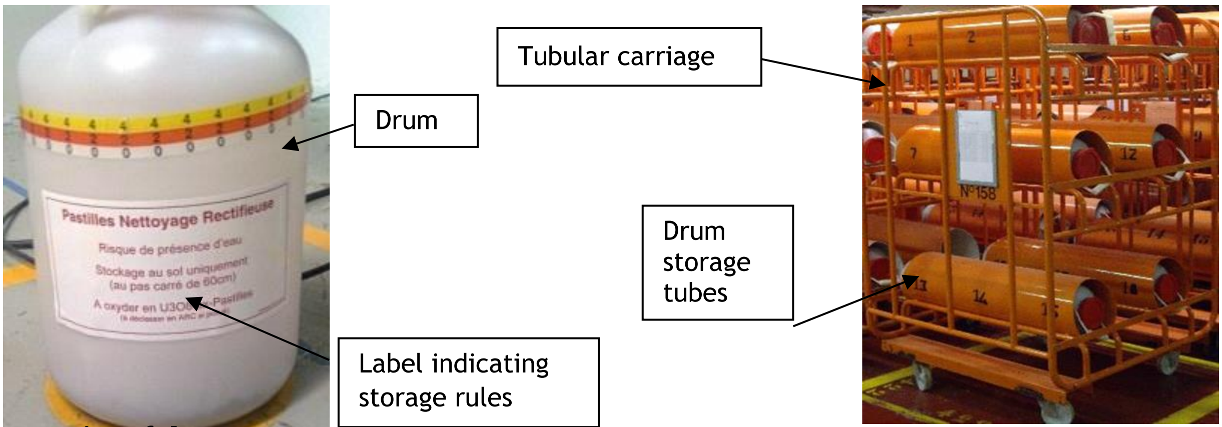
At the Romans-sur-Isère site, the AREVA subsidiary FBFC operates two basic nuclear facilities: a fuel element fabrication plant for research reactors and a nuclear fuel assembly fabrication plant for pressurised water reactors (PWRs). The fuel assembly fabrication plant for PWRs consists of several buildings in which the different manufacturing operations are carried out. In building AP2, one of the fuel fabrication steps involves grinding the sintered UO2 pellets underwater. This operation adjusts the diameter of the pellets by passing them between two grinding wheels to obtain the required diameter for introduction into the fuel rod cladding.

The fuel fabrication operations produce:

- manufacturing scrap in the form of entire pellets or pellet pieces, considered dry products, i.e. containing a very low level of moderating material (water). This scrap is packaged in 10 liter drums;
- sludge resulting from centrifugation of grinding process water, considered a wet product. This sludge is packaged in nacelles;
- grinding wheel cleaning scrap (pellets, chips, dust, sludge, etc.), considered as a wet product. This scrap is packaged in 10 liter drums.

All this scrap is calcinated in a furnace where uranium is oxidised as U3O8, which is recycled in the pellet fabrication process. The plant operator has two calcination furnaces: one in building AP2, the other in building R1.

To prevent the risks of criticality, operating rules define the conditions for using the drums. Depending on the type of fissile material in a drum (powder or pellet, dry or wet product), these rules set the maximum uranium oxide mass per drum as well as the conditions for identifying (dry or wet products), transporting or storing the drums. The rules are stricter for wet products to the extent that they contain moderating material. The manufacturing scrap is transferred between buildings AP2 and R1 in 10 liter drums, which are transported either individually (wet products) or in a "tubular" carriage (up to 18 drums of drum products per carriage). These operations are performed by operators.



Detection of the events

On 17 September 2012, the plant operator discovered in building AP2 a tubular transfer carriage with three drums labelled as containing wet products, whereas these carriages should only transfer dry products. The drums were removed from the carriage. The plant operator's investigations did not bring any other anomalies to light. The event was reported by the plant operator to the ASN, which classified it as level 1 on the INES scale.

On 24 September 2012, in building R1, the plant operator discovered wet products upon opening a drum labelled as containing dry products. The plant operator suspended production in the pellet grinding units as well as the transfer of drums and initiated a check of all drums liable to contain wet products. This check revealed six other anomalies relative to drum management rules. The plant operator reported this event to the ASN and proposed it be classified as level 1 on the INES scale.

The ASN organised a "rapid-response" inspection on 28 September 2012, which revealed that several drums were non-compliant with the criticality risk prevention rules. This led to reclassifying the 24 September event as level 2 on the INES scale. The ASN also notified the plant operator that it was required to improve the management of drums containing fissile material from the grinding process.

Lessons learned from the events

Based on the analysis performed, the plant operator concluded that the events were linked to insufficient consideration given to human actions in the criticality risk analysis carried out when the facility was designed. The analysis of the events revealed organisational and human failures (poorly explained or incomplete operating procedures, no checking of drum labelling and storage, insufficient knowledge of criticality risk control rules by the operators, particularly with regard to the storage conditions of wet product drums in tubular carriages). Furthermore, following shutdown of one of the old calcination furnaces in building AP2 and its replacement by the furnace in building R1, the material flows within the facility were modified without sufficient analysis of the consequences of this change, specifically in terms of additional operating requirements for the personnel.

The measures immediately implemented by the plant operator after the above-mentioned events improved wet product drum identification as well as the rules for drum use and storage. Checks of drum use were also put in place. In addition, the plant operator proposed using a new type of drum for wet products, whose shape would prevent loading in the tubular carriages for transferring dry products, as well as a new carriage specifically for wet product drums.

The IRSN estimated that the proposed measures were suitable overall for preventing the recurrence of this type of event, but made several recommendations to the ASN with the aim that the plant operator improve the safety documentation for its facility. According to the IRSN, the plant operator should also extend the feedback from these events to the drums of materials from equipment other than the grinding wheels.

Appendix3

EVENT OF 6 NOVEMBER 2006 IN THE ATPu FACILITY AT THE CADARACHE SITE

The event on 6 November 2006 relates to the conditioning of MOX fuel manufacturing scraps so that they can be evacuated from the facility. The different equipment required for this operation (grinder, mixer, etc.) is installed in separate glove boxes. The scraps are conditioned in containers called "jars"; transfers take place on a conveyor belt. Before each jar is moved, the fissile material it contains must be weighed. This weighing ensures compliance with fissile material weight limits in different equipment, limits defined to prevent criticality risks. A software application is used to manage these weights.

On 6 November 2006, a batch of scrap was inserted into the grinder which still contained part of the previous batch. The fissile material in the grinder therefore weighed more than the maximum permitted limit for this equipment. It is important to note that the sub-criticality of the grinder would not have been compromised even if it had accidentally contained double of the permitted weight limit of fissile materials. The weight limit for fissile materials in the grinder was based on a study in which various potentially abnormal situations were examined, in particular double batching of the grinder. This event was rated 2 on the INES scale.

Following his investigations, the plant operator determined that the event was due to a breakdown in the scales used to weight the jar receiving the fissile materials after grinding, the Shift Manager decided to change the operating procedure and replace these scales at the grinding station outlet with scales at the next "mixer" station. To transfer the jar to the "mixer" station, an operator input into the computer application mentioned above a weight of fissile materials for the jar (a weight which should have been given by the faulty scales); this "fictitious" weight was numerically equal to the weight of the scrap inserted into the grinder. The computer application automatically removed this "fictitious" weight from the "grinder" station and allocated a nil residual weight of fissile material to it. Fissile material weights were supposed to be updated in the computer application after the actual weighing of the suspect jar at the "mixer" station.

However, due to an unidentified dysfunction, the grinder was not emptied correctly into the jar; part of the fissile materials remained in the grinder. Other dysfunctions (conveyor breakdown delaying the transfer of the jar and incomplete transmission of information between operating teams especially) resulted in the jar not being weighed at the “mixer” station and therefore the fissile materials contained in the jar and the residual weight in the grinder were not rectified in the computer application before a new batch of scrap was inserted into the grinder. The error was detected when the previous jar was weighed at the "mixer" station.

The plant operator considered that human and organisational factors were the major causes of this event. He indicated that the event was linked to insufficient safety culture by the personnel; the operating procedure had been altered without analysing safety and without applying the change management procedure provided for in such circumstances; organisation shortcomings (checking operations, transfer of information between teams, etc.) were also highlighted. The plant operator launched a major action plan to remedy the shortcomings identified (strengthening teams responsible for safety, improving operating provisions, boosting training and the safety culture, etc.). IRSN considers that this event illustrates how important it is for plant operators to make provision for controlling the risks of modifying operating conditions as well as the human and organisational factors in their facilities, especially for the management of operating contingencies.

Criticality risk management: why analysis of operating practices maters?

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Abstract

The management of criticality risk in Fuel Cycle Facilities (FCFs) relies mainly on a set of prescriptions and requirements established by the licensees for achieving safety objectives. This paper intends to show that, beyond prescriptions and requirements, a socio-technical approach is essential to define a relevant set of criticality safety rules favouring efficient and safe human activities. Indeed, a thorough knowledge of staff operating practices contributes significantly to the definition of appropriate technical and organisational provisions.

We will review organisational lines of defense underlying operators' inappropriate actions, especially those related to rules compliance, documentary reference framework and risk perception.

We will pay specific attention to showing that risk analyses and criticality safety frameworks need to be considered in the light of diversity of working situations and complexity of their organisational interfaces. It is also important to highlight that those documentations should take into account technical and organisational modifications throughout FCFs' lifecycle. Moreover, we will show that introducing and maintaining efficient and safe practices in the long term relies on appropriate staff risk awareness. For achieving this, in addition to the deployment of a relevant training program, the role played by local management and the support of criticality safety experts to operational staff is essential in order to make operating practices safer.

The paper refers to an example of an event that occurred in a French FCFs.

I. Introduction

Criticality accidents constitute sudden release of radiations without any previous warning signs. This is the reason why the corner stone of criticality risk management in nuclear facilities is the prevention of criticality accidents. Following the defense in depth principle, prevention of criticality accidents relies on technical and organisational lines of defense defined in the criticality safety framework.

In the light of the contribution of the socio-technical approach, the IRSN considers that criticality risk management and more generally nuclear risk management should rely on lines of defense which take into account work activities and technical and management support to those activities.

In this paper, the IRSN intends to show that, beyond prescriptions and requirements, a socio-technical approach is essential to verify the relevancy of the criticality safety rules and procedures, and to favour efficient and safe human activities. Indeed, a thorough knowledge of work situations and staff operating practices contributes significantly to the definition of appropriate technical and organizational lines of defense in order to ensure that operating situations are always compliant with situations authorized by the criticality safety framework.

We illustrate our point with an example from an event that occurred in a French fuel fabrication facility, FBFC[1].

1. FBFC (AREVA group) is a fuel-fabrication facility located in southeast of France.

II. The event

In September 2012, the company FBFC reported to the French Nuclear Safety Authority (ASN) an event involving non-compliance with rules and procedures to prevent criticality risks relative to conditioning, storage and internal transfer of containers known as "drums" which hold manufacturing scrap containing fissile material.

FBFC produces fuel for pressurized water reactors (PWRs) (See Annex for a description of the process). The main step of the fabrication process is the production of uranium oxide pellets from uranium oxide powder. Manufacturing scraps produced in the pellet fabrication workshop are calcinated before being re-injected in the fabrication process in powder form. A part of these calcinated scraps needs to be chemically purified in the recycling workshop located in another building. Normally, only dry fissile material was transferred between the two buildings. However, since the shutdown of the calcination furnace in the pellet fabrication workshop at the end of 2011, wet fissile material produced during grinding wheel cleaning cannot be dried in this workshop anymore. Parts of this wet material are thus transferred in 10-liters drums between the two buildings.

Figure 2: Specific label for a 10-liters drum with wet fissile material

To prevent criticality risk, operating rules define the conditions for using the drums. Depending on the type of fissile material in a drum (powder or pellet, dry or wet product), these rules set the maximum uranium oxide mass per drum as well as the conditions for identifying (dry or wet products), transporting and storing the drums. Since the criticality risk increases in the presence of moderator, the rules are stricter for wet products (i.e. specific identification, manual handling, individual transfer).

On the day of the detection of the event, an operator discovered, when opening a drum in the recycling workshop after its transfer from the pellet fabrication workshop, that it was containing wet fissile material[2], while it was placed in a tubular carriage dedicated to carry only dry fissile material between the two above mentioned buildings (and thus unauthorized to receive drums with wet material).

As soon as the event was reported, all transfers of fissile material within the workshops were suspended to proceed to an exhaustive verification of all the drums placed in tubular carriages. This verification process ended up with a total of six drums containing wet material not compliant with the criticality safety framework regarding the rules of identification, storage and internal transfer.

For IRSN, this event revealed shortcomings in the criticality risk management in this French Fuel Cycle Facility (FCF), reflected in weaknesses of existing lines of defense defined in the criticality safety

2. Drums with wet fissile material contain mainly sludge (instead of powder) and sometimes supernatant comprising water.

framework of the facility, especially those concerning the treatment of wet material in compliance with this framework.

Figure 3: Tubular carriage for the transfer of drums

III. Main lessons learnt from the analysis of the event

The identification of drums with wet fissile material is done with a red “centralized cleaning” label whereas there is no specific label for drums with dry material. The “centralized cleaning” label is the fourth label stuck on the drum (after the labels for enrichment, type of product, origin of the product and before the weight ticket). Therefore, as all the drums look identical, the red label is the unique provision to distinguish drums containing wet fissile material from other drums in order to apply the rules defined for storing and transferring drums with wet material, which state that wet material drums should be individually and manually transferred in the facility (between the two above-mentioned buildings).

For three out of the six non-compliant drums, the rules defined for storing and transferring drums with wet material were not applied because operators omit to stick on the red “centralized cleaning label” on the drums with wet material. These drums were erroneously placed on a tubular carriage later on.

Whereas human error is often invoked by licensees as a major factor that caused or contributed to an incident/accident, some technical and organizational configurations are more likely than others to generate inappropriate operators’ actions and to prevent from their recovery. The review of the work situation design reveals that no provision was foreseen to prevent or recover from an error of labeling such as distinct labels (in forms and colors) for both wet and dry material drums, coded pins on the drums containing wet fissile material, physical lock against the introduction of wet material drums on the tubular carriage, separated circulation flow for dry and wet material drums, separated temporary storage areas before transfer for wet material drums from areas for dry material drums and traceability of wet material drums in the nuclear material database.

Moreover, it appears that no provision for controlling the activity of labeling wet material drums, such as crosschecks, was carried out in the facility documentary framework.

As a consequence, two major lines of defense, work situation design and documentary framework, did not play their full role: a poor work situation design combined with shortcomings in the documentary framework relative to activity control lead to a situation unauthorized by the criticality safety framework (drums with wet material not individually transferred) following one single inappropriate operator’s action (label not stuck on the wet material drum).

The three other non-compliant drums containing wet material were misplaced in tubular carriage while they were correctly labelled. To understand the reason why operators did not apply the rules for storing and transferring those three drums, it is essential to tackle the issue of the consequences of the modification of material flow on their own activity.

Following shutdown of one of the old calcination furnace in the pellet fabrication workshop and its replacement by the furnace in the recycling workshop, the flows of fissile material within the facility were modified. Thus, operators had to perform the additional task to individually transfer drums containing wet material from the pellet fabrication workshop to the recycling workshop. The criticality safety framework of the workshop was then updated to take into account this rule.

The individual transfer of drums with wet fissile material is binding for the operators. Each transfer requires about nine minutes of delivery time per drum. Up to four drums may be transferred per shift work, representing a total period of forty minutes. Thus, in order to reduce the constraint of transferring the drums individually, the operators bundled several wet drums and transfer them in a grouped manner. This operating practice, yet non-compliant with the criticality safety framework, presents clear advantages to reduce time and number of transfers to the detriment of criticality safety rules. It turned out that operators deliberately circumvent the rules in order to optimize time allocated to transfer of fissile material, without being necessarily aware of the consequences of their actions.

The rule stating that wet material drums should be transferred individually in the facility was not supported by a criticality safety risk analysis addressing the compatibility of that way of transfer with operators' practices and activity constraints. Moreover, the extension of the rule for the individual and manual transfer of wet material drums to the recycling workshop should have been shared with operators in charge to ensure that the rule is well understood and that the link with criticality risk management makes sense to them.

As a consequence, failures of at least three lines of defense, risk analysis, documentary framework sharing and appropriation, and activity preparation, led to the same situation unauthorized by the criticality safety framework as for the three drums incorrectly labelled (drums with wet material not transferred individually), but this situation was caused by inappropriate operators' actions made intentionally.

This event also allows to reviewing the deeper organizational lines of defense which emerge under the inappropriate actions of operators, in particular those associated to the role of first-line managers and criticality safety support entities as well as perception of risks in complex work situations:

- **Criticality risk awareness**

The analysis made by the licensee showed that the non-compliance with the rule concerning the drums transfer was not only unauthorized by the criticality safety framework but also not covered by a specific criticality study. Consequently, operators are not aware to have breached the criticality margins and to be in a uncontrolled situation. The analysis emphasizes a work situation in which the switchover from an authorized situation to an uncontrolled one in which the remaining margins are unknown happens because of one single inappropriate operator's action without any technical and organizational provisions to recover from it. This situation is particularly problematic in the case of criticality risk management for which no forewarning is detectable before the accident is triggered[3].

This leads to the question of how to make understand this very particular risk to staff exposed to the risk of criticality. One response concerns the training program of staff; it must be designed in order to promote a more proactive role for the operators ahead of the operation in particular through the appropriation of the risk analysis. The involvement of criticality experts at this stage is crucial, to explain

3. The decision 2014-DC-0462 from the Nuclear Safety Authority of the 7th October 2014 relative to the criticality risk control in nuclear facilities specifies that a criticality accident must not in any case arisen from a single fault.

the importance of the rules compliance and its link to criticality risk management, to promote the criticality risk analysis and its implementation in connection with the working practices of the operators. Local appropriation by the operators is the key to success for a correct understanding and thus application of the prescriptions and rules, as long as they are adapted to the operational practices.

- **Role of first-line managers and criticality safety support entities**

This event shows that it is necessary to strengthen the link between operators, line managers (in particular shift supervisor) and criticality safety support entities (operational safety engineer, criticality engineer), in order to reinforce the support provided to the first line operators. This is also an opportunity to give sense to the working tasks by improving the criticality risk awareness individually and collectively, and a better understanding of criticality risk prevention. On the other hand, it could give the managers and safety experts a better view of the complexity of work situations that the operators have to deal with. In the present case, managers and experts lack of detailed knowledge of the actual activity on workstations and therefore have an erroneous perception of the tasks performed by operators. First example, the binary identification dry/wet of the drums as prescribed did not cover the actual work of operators. Indeed, operators had actually to manage 9 types of different products in removal from grinding machine for which the criticality safety framework did not list those authorized in the tubular carriages. The procedure did not indicate the wet or dry nature of those products. Second example, loading of the carriage and regulation set up is not performed by one operator. It’s the result of a collective work between 3 operators: the operator in charge of the oxidation furnace, the one performing the rectification of the pellets and the one sorting out the pellets. A detailed knowledge of the work activity performed allows a better understanding of the constraints of each other. It gives a better chance to mutual appropriate information sharing and to avoid or detect inappropriate human action. It also provides the opportunity to anticipate co-ordinations or safety matters.

IV. Safety management in socio-technical systems

Often inappropriate human actions are the consequence of the characteristics of the situation, which have not allowed operators (individual or team) to use their expertise in a relevant way, for reasons usually linked to failed lines of defense such as poor design of the systems, bad human-machine interface, lacking or inadequate prescriptions, ineffective organization, inappropriate training. Thus, the design of socio-technical systems plays a central role in effective performance of operators as it either facilitates or hinders their decisions and actions. The characteristics of the situations in which a human being is placed make certain types of behavior more likely. These characteristics can be local (design of a workstation, tools, procedures) or much broader in scope (company policy, management system, training programs).

An effective safety management should include two complementary components; one component, called “Rule-based safety”, is based on as complete as possible identification of possible failures in order to define provisions to prevent these failures and limit their consequences; a second component, called “managed safety”, aims to manage unforeseen situations in a safe way.

- “Rule-based safety” seeks to avoid all foreseeable failures through formal procedures, rules, automated safety mechanisms, the use of protective measures and equipment, training in “safe behavior” with management ensuring that rules are respected. This component makes it possible to predefine appropriate provisions (technical, human and organizational) to foreseeable situations. However, the approach to deal with safety in complex systems still tend to focus on the behavior of operators, on human error and on compliance with procedures derived from exhaustive risk analyses. Indeed, licensees still too often regard risk analyses and operators’ compliance with rules as a guarantee of safe facility operation. Operators are often considered as the weak link in the system. Their positive contribution to safety is usually neglected. Event analyses from nuclear licensees particularly reflect very often this erroneous view of safety, as the analyses are limited to the search for apparent causes, leaving aside less apparent essential causes.

- "Managed safety", the second component of safety management, develops the socio-technical system capacity to anticipate, to recognize and to formulate appropriate responses to unexpected scenarios that were not foreseen by the organisation because it is not possible to identify all the scenarios even for simple activities. It relies on provisions which foster competences and real-time presence of human expertise, the quality of initiatives, the way groups and organisations operate, and on management that is attentive to the situations and encourages coordination between the different type of knowledge that are useful for managing safety.

In other words, procedures and rules prepare the system for configurations that have been anticipated and play a major role in the ability to manage these situations. But situations also arise that are unforeseen or not (yet) analysed (hazards, evolutions following process modifications, degraded situations …). The way the system responds to these will depend on organizational lines of defense which allows the local resources of the teams and the management to be available in real time.

Formalizing the rules necessary to manage foreseeable work situations is essential especially when criticality risk is involved. Nevertheless, formalizing the response to foreseeable situations does not guarantee the relevance of the response to unforeseen situations. Worse still, organizations that base their entire safety policy on prescriptive formal procedures can find their robustness brought into question when a new or unforeseen situation arises.

To sum up, an organization contributes efficiently to safety when it facilitates an interaction between the formal rules, which provide general expertise, and the knowledge of specific operating situations and practices, which is held by the operators and managers on the field.

To reach this objective, an organization should be able to:

- Regularly reassess the assumptions and processes on which safety is based, in particular in case of evolution of safety hypotheses, of processes, of organization, etc.
- Collect operational experience feedback, analyse the data collected, and capitalize the lessons learned and share them among the different entities of the organization.
- Set up a collective functioning relying on effective activity co-ordinations and close coordination of entities involved.
- Carry out operations by detecting and locally managing variability linked to specific operating conditions.
- Involve operators in the design and the improvement of rules and procedures to take into account the characteristics of work situations, but also to encourage their adoption by operators. When a participative approach is encouraged/promoted and implemented, it contributes to reinforcing rigorous rule application. The same applies to the presence of management in the field, seen as participative leadership practices, which take the form of both support to the working activities and control.
- Establish a positive safety dialogue while encouraging certain improvements when applicable. In this way, operators, safety experts and managers participate in the coordination of "regulated safety" (top-down definition of the rules) and "managed safety" (integration of local characteristics).

V. Conclusion

To meet production and safety objectives, operators' work activity is not limited to the simple execution of procedures. Operators seek to achieve goals in specific working conditions. Safe production occurs only because each person manages many sources of variation while executing their tasks, with expertise acquired through experience. Hence, work activity is a response to a number of determining factors which present some variability: production and safety objectives, tasks to be performed, equipment available, working conditions, time constraints, abilities and knowledge of the operators, expertise acquired through experience, available collective resources, etc. As a consequence, global performance of a system in terms of production quality and safety is dependent upon interaction between social and technical components in workplaces.

The system into which operators evolve is complex. The actions of one and other interact very often, but not always explicitly. That's why being able to deal with the criticality risk often requires to finely analyze the activities performed by the operators individually and collectively, in order to define means of performance that make sense to their work and are compliant with the authorized safety framework. It is then crucial that operators clearly understand the relationship between criticality risk in the facility and the criticality safety prescriptions and requirements governing their daily activities in order to make them willing to abide by these rules every day. For the same reason, working practices and activity constraints in a given work environment should be taken into account when defining or modifying existing prescriptions or requirements.

Lessons learnt through the analysis of the FBFC event point towards three levers of action. Operators should be asked to participate to the definition of new criticality procedures and instructions and to any evolution of existing ones as experts of their own activity. They should also be encouraged to express any concerns about prescriptions they have to apply and possible limits their working environment. It goes without saying that any modification brought to the whole criticality safety framework should lead to check the overall consistency and relevance and the acceptability of the amount of the applicable documents so that those documents would be easily shared and used when required and needed.

The presence in the field of local management and criticality experts seen as a participative leadership practice is crucial to allow managers and criticality experts to fulfil their support function for activity besides their control function. Their role is twofold. They should ensure that any modification brought to the criticality safety rules is well understood by operators in the light of the criticality risk to manage. They also should be able to detect any unwanted evolution in the criticality safety practices and understand the reasons of its emergence, which could cover any evolution of production, quality or safety objectives but also any evolution of the work environment of the concerned operators.

Designing and operating social-technical systems, such as nuclear facilities, so that the ultimate goal of safe production is achieved, requires the deployment of specific skills on human and organizational factors. These skills are essential to take into account all the components of the variability of work situations (overall objectives, available competence and resources, procedures, technical devices, working conditions, etc.) and design appropriate and efficient technical and organizational lines of defense and to ensure they remain robust throughout facilities lifecycle.

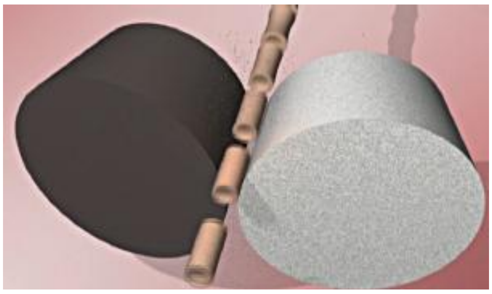
ANNEX

Description of the process in the PWR fuel fabrication facility in Romans-sur-Isere

At the Romans-sur-Isère site, the AREVA subsidiary FBFC operates two basic nuclear facilities: a fuel element fabrication plant for research reactors and a nuclear fuel assembly fabrication plant for pressurized water reactors (PWRs). The fuel assembly fabrication plant for PWRs consists of several buildings in which the different manufacturing operations are carried out:

- Building C1: conversion of enriched uranium hexafluoride (UF6) into uranium oxide (UO_2) powder;
- Building AP2: pellet fabrication from UO_2 powder, then fuel rod fabrication from these pellets and production of assemblies from these rods;
- Building R1: recycling of manufacturing scrap from the different steps of the process;
- HF station: treatment of gaseous effluents from the conversion process to recover hydrofluoric acid (HF).

In building AP2, one of the fuel fabrication steps involves grinding the sintered UO_2 pellets underwater. This operation adjusts the diameter of the pellets by passing them between two grinding wheels to obtain the required diameter before introduction into the fuel rod cladding.



The fuel fabrication operations produce:

- Manufacturing scrap in the form of entire pellets or pellet pieces, considered as dry products, i.e. containing a very low level of moderating material. This scrap is packaged in 10-liter drums;
- Sludge resulting from centrifugation of grinding process water, considered a wet product. This sludge is packaged in nacelles;
- Grinding wheel cleaning scrap (pellets, chips, dust, sludge, etc.), considered as a wet product. This scrap is packaged in 10-liter drums.

The manufacturing scrap is transferred between buildings AP2 and R1 in 10-liter drums, which are transported either individually (wet products) or in a "tubular" carriage (up to 18 drums per carriage). These operations are performed by personnel.

CRITICALITY SAFETY AND ORGANIZATIONAL PRINCIPLES AT THE CEA

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1. INTRODUCTION

The Commissariat à l'Énergie Atomique et aux Énergies Alternatives – usually referred to as the CEA has ten research centers in various regions of France, each one specializing in specific areas of research. These research centers/laboratories are located throughout France in the regions of Paris/Île de France, Rhône-Alpes, Languedoc Roussillon (Rhône Valley), Provence-Alpes-Côte d'Azur, Aquitaine, Central France and Burgundy.

Among them all, research on nuclear energy is conducted mainly in 3 civil research centers1: at Saclay in the region Paris/Île de France, Marcoule in the Rhône Valley and Cadarache in Provence.

In mid-1999, just a few months before the Tokai – Mura criticality accident, the CEA decided to review the organization regarding criticality safety in all of its nuclear facilities in order to improve it.

It's important to point out that until that time, the CEA's criticality safety organization had a very simplemodel. One high level criticality safety specialist appointed as the "ICC" (Ingénieur Criticien de Centre / Center's Criticality Engineer) was in charge of an entire nuclear center regardless of the number of nuclear facilities within. This criticality "officer" function was formally established in France in 1984 by the nuclear regulator (today's Autorité de Sûreté Nucléaire – more commonly known as the ASN).

2. CEA ORGANIZATION

As presented at the ICNC in 2003, the CEA's organization[2] is based on an operational line, backed up by support resources, and a control line. The organization is specified in a CEA internal instruction from the Nuclear Safety and Protection Division (DPSN) with specific instructions for each CEA center.

Regarding criticality safety and in particular the operational line and support resources, a three "level" organization was established as follows: (from "top" to "bottom"):

- A central (CEA) Criticality Safety Expertise Group (CSEG) in which criticality specialists perform among their other activities, criticality calculations for each of the nuclear facilities. The members of the CSEG are also in charge of criticality accident issues (until 2014, a Criticality Skill Team dedicated to criticality accidents was located in Valduc near experimental facilities. However, today it is closed.)

1. Nuclear research at the Grenoble research center ended in early 2000. Today, all of the nuclear facilities there have been decommissioned and their dismantling is still underway.
2. ICNC 2003, "The Organization of Criticality Hazard Prevention at the CEA" JAERI-Conference 2003-019.

- At each CEA center, there is a high level expert – ICC (He has a renewable term of 4 years)
- At every nuclear facility (Installation Nucléaire de Base - INB3 / Basic Nuclear Facility) where fissile materials are present, there is a local organization which is managed by a local criticality specialist named IQC (Ingénieur Qualifié en Criticité / Qualified Criticality Engineer with a renewable term of 4 years as well).

Regarding the control line and in particular the control function the aim of which is to enforce the observance, the adequacy and efficiency of the measures taken by the operational line managers in order to meet the nuclear safety goals set down by the director of each CEA Center, a team usually called the Safety Cell ("Cellule Sûreté") exists as well. The Safety Cell is totally independent of the aforementioned operational line, reports directly to the director of each CEA Center, and it performs a thorough check of the safety documents and all operational practices. This safety team includes one person with sufficient criticality expertise to be appointed the Criticality Specialist (CS).

These lines are implemented at every CEA center with licensed nuclear facilities (Installation Nucléaire de Base - INB) where fissile materials are present and where criticality safety issues can be met.

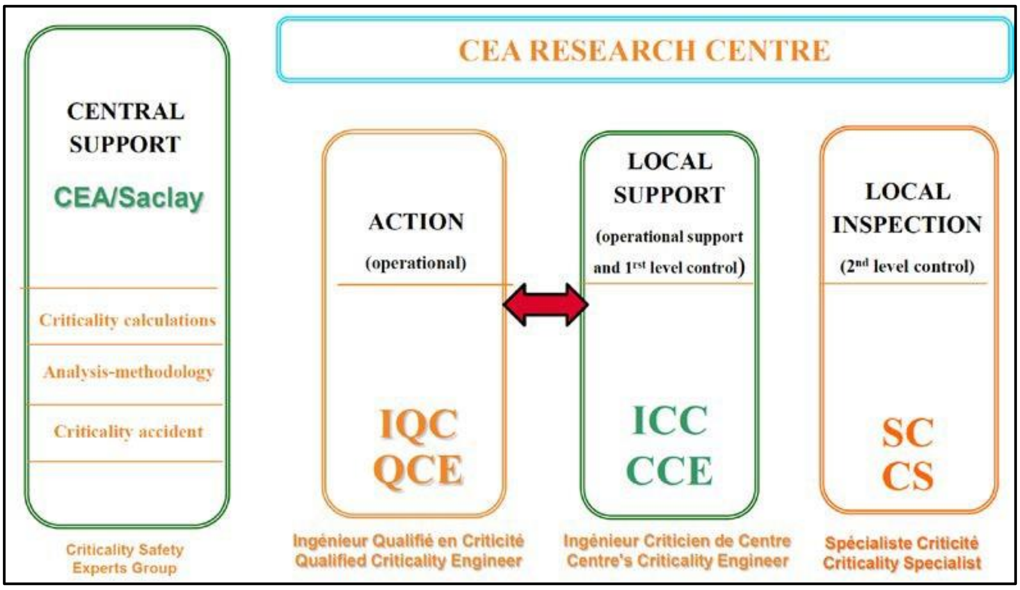

Figure 1: Organization chart

3. The list is available at www.asn.fr.

3. TRAINING FOR THE IQCS', ICCS', SCS'

3.1 Training of the IQCs

In order to reinforce this structure, an appropriate training in accordance with the responsibilities of the position is given to the appointees in charge of the operations or the control. Some of these positions require a "qualification agreement".

Training for the IQCs is a two times one week process. These "*initial training*" sessions are organized by the INSTN (National Institute of Nuclear Sciences and Techniques4) at Cadarache. The first one week session includes general theoretical aspects on criticality safety (an overview of fission phenomenology, cross sections, Monte Carlo codes, accidents, etc…..). During the second 3½ day session, the IQC's receive "*practical training*" in which they are trained using criticality reports/calculations, procedures, operating practices, etc. They are also given a quick overview of the regulatory framework regarding criticality, dealing with matters such as nuclear laws, nuclear regulatory authority decisions and internal CEA recommendations as examples.

For each training session, the trainees receive a certificate of attendance awarded by the INSTN. The nuclear facility manager in which an IQC is working must put in an application to the ICC for his/her "qualification". This initial qualification at the beginning of a criticality safety "career" is almost always accompanied by obligations and sometimes restrictions. The usual obligation is to stay in close touch with the ICC during the first months after his/her nomination as the IQC of the facility.

3.2 Training of the ICCs

On the whole, the ICCs have an overall good professional background in general nuclear safety and in criticality safety if possible. Recruitment is done among the ranks of reactor/neutron physicists, nuclear safety engineers or nuclear chemistry engineers and former IQCs.

Prior to their appointment, they receive an approximate nine-week overall training session which involves theoretical and practical aspects. An ICC applicant must pass a qualifying examination and spend one week as a trainee in a French nuclear facility during which he is required to write an internship report followed by a presentation / oral examination of the report to a committee made up of ICCs and senior French criticality experts.

Furthermore, a diploma is issued by the INSTN (training sessions take place every two years in France). It is the duty of the director of a CEA research center to request the "qualification" of the ICC for his own specific center, and a "qualification agreement" is issued by the Head of the Criticality Skills Team, following the appointment of the candidate by the director of the CEA center as an official ICC.

Criticality Specialists must have as a minimum requisite for training the same as that of the IQCs, but they often have had previous ICC training and experience in addition to this. It is the responsibility of the head of the "safety cell" to request the "qualification" of the SC for the specific center, and a "qualification agreement" is issued by the head of the Criticality Skills Team, after the appointment of the candidate as a SC.

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4. More information at .http://www-instn.cea.fr/Criticality-Safety.html

3.3 Training of the SCs

4. RESPONSIBILITIES OF THE IQCS, ICCS, SCS

4.1. Responsibility of the IQCs

The IQC carries out his responsibilities only in the area of one specific nuclear facility (INB) under the control of the INB's manager. He has the delegated authority to conduct every day controls within the perimeter of his nuclear facility. Either alone or in collaboration, he must write all the procedures, operational documents, verifies them (if he has not written them himself), and then submits all the regulatory procedures referenced in the general operating rules to the ICC for validation/control.

He must solicit the ICC for detailed technical advice, especially when he's unexperienced in the field or a beginner at the nuclear facility. In both cases it's mandatory.

As soon as he's experienced enough, he conducts training sessions for his own facility workers (either alone or possibly with the ICC) and holds periodical updated re-training sessions).

He carries out the criticality safety assessments and defines the necessary criticality calculations when new facilities are opened or when existing facilities are modified or whenever periodic regulatory assessments are led, under the guidance and support of the ICC.

4.2. Responsibilities of the ICCs

The ICC is the technical authority regarding criticality safety in a nuclear center. He is the upmost technical advisor for the IQC's, nuclear facility managers and the Center manager and as such, he controls and validates all relevant criticality safety documents, particularly criticality safety assessments.

He keeps in touch with the Criticality Skills Team on a routine basis and he attends internal CEA meetings dealing with general criticality issues and actively participates in the definition of internal regulations.

He makes sure that nuclear facilities are operated according to the specific rules and procedures of each, ensuring permanent sub-criticality. He meets all the demands of the center's facilities in the area of criticality safety.

He leads experience exchanges between the IQC's of "*his*" CEA center and takes into account the feedback on criticality safety coming from other CEA centers.

4.2. Responsibilities of the SCs

The SC is the final formal authority regarding criticality safety in a nuclear center. As member of the safety cell he has, among his other responsibilities, the task of checking that the facilities are being operated in conformance with the regulatory authorizations. This is accomplished through facility visits, documents and recordings inspections and by carrying out technical and quality assessments.

He implements lessons learned from significant criticality safety events that have occurred in nuclear facilities (located inside or outside his Center) and advises the Director of the center in case of any irregular or unusual situations arising in a facility and concerning the prevention of criticality hazards.

He controls all the relevant criticality safety documents on a formal basis before these are sent to the nuclear safety authority (the ASN).

5. DOCUMENTARY ORGANIZATION

5.1. General Documentary Organization

Criticality safety analysis is included in the baseline report of a nuclear facility. The outcome of a criticality safety analysis involves requirements (limits, rules, poisons, etc …). These are to be found in the Règles Générales d'Exploitation / General Operating Rules (RGE/GOR). The RGE/GOR are in general structured into several parts but regarding criticality safety one must two special chapters:

1. the definition domain
2. the general criticality rules.

All the operational requirements (limits, etc ...) can be found in the "Definition Domain" chapter. In the chapter entitled, "General Criticality Rules", we find all the general operating procedures and/or instructions. Sometimes, especially in old baseline reports, the "Definition domain" is actually the first part of the "General Criticality Rules" chapter.

All the documents (procedures, control parameters, operating instructions,…) referenced into the "General Criticality Rules" chapter MUST BE / ARE approved by the ICC who is the guarantor of their technical "quality".

5.2. Document Organization Examples

In some nuclear facilities, an organizational note called "criticality safety organization" summarizes in detail the hierarchy of the various documents pertaining to criticality safety such as procedures, operating rules, etc. Here's the chart of this tree structure:

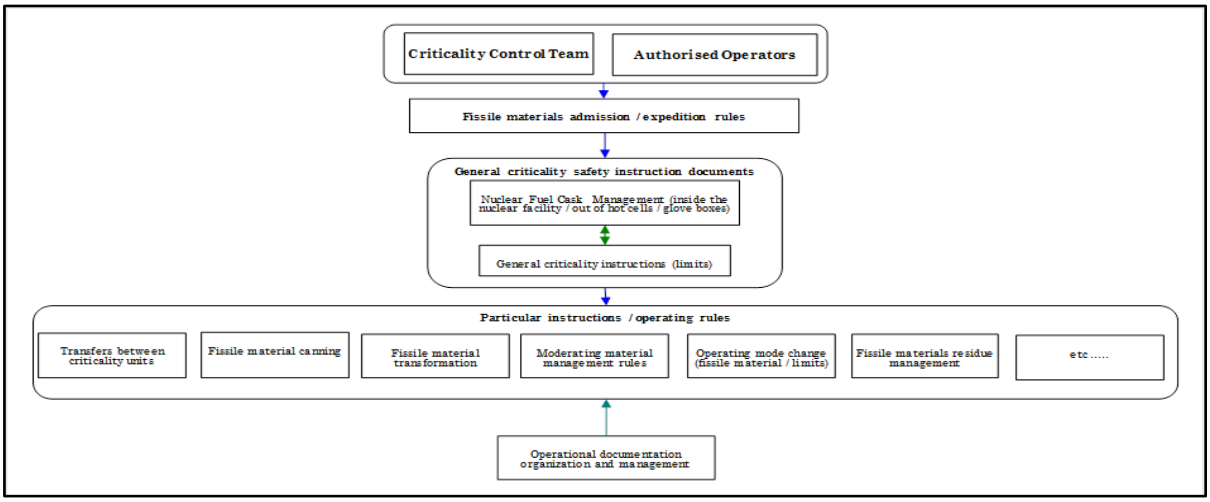

Figure 2: Documentary Organizations Chart Example

- **Criticality control team** = a team whose members trained by the IQC and ICC, have been appointed by the nuclear facility manager and who are distinctly identified; they never control an operation in which they participate.
- **Authorized operators** = workers appointed by the nuclear facility manager in order to perform operations on nuclear fuel/fissile material; they can "touch", transform/modify, transfer, containerize, etc the fissile material.

These first two documents are important regarding the double contingency principle.

- **Fissile materials admission / expedition rules** = a document describing all the steps to be taken before a cask enters or leaves the nuclear facility's area (perimeter). Example: control of the fissile material nature expected to be received versus the regulatory authorizations of the facility.

- **Nuclear Fuel Cask Management = a document describing all the procedures necessary for cask management (especially if the facility uses the nuclear transport Criticality Safety Index / CSI).**
- **General criticality instructions (limits) = document in which the regulatory general operation** rules (GOR) are transposed into more simple operational and understandable, everyday rules.
- **Particular instructions / operating rules = specific rules for specific operations such as:**
    - *Transfers between criticality units*: how to respect the addressee criticality unit's limits (example: mass limits).
    - *Fissile material canning*: how to make sure fissile material can be recognized after canning
    - *Fissile material transformation*: how to recognize the fissile material after transformation (example: what mass to assign to the various length parts of an initial fuel pin after cutting it)
    - *Moderating material management rules*: what the rules are and how to comply with the criticality unit's moderating material limits (quantity/amount, quality/grade)
    - *Operating mode change (fissile material / limits)*: what kind of operations must be undertaken in order to make sure that another kind of fissile material can be processed into the same criticality unit (new limits)
    - *Fissile material residue management*: how can we cope with fissile material residues in a hot cell / glove box (accumulation, rinsing, ..).

On an operational basis, daily monitoring regarding criticality issues is conducted through this documentary "structure". This "structure" is in accordance with the variety / complexity of the operations conducted in each facility, and of course nuclear workers are given the adequate training on how to use / fill in these documents that are necessary in conducting the various processes.

6. FINAL APPROVAL OF DOCUMENTS

Documents containing the word criticality must have to be approved by the ICC before continuing to the next level of internal control / validation / approval. This is not an optional way of proceeding, it's totally compulsory.

Documents such as safety reports, general operating rules, etc. must go through the SC's examination and obtain approval before the director of the CEA center or his delegated authority signs them in order to be sent to the Nuclear Safety Authority (ASN).

7. TRAINING PRACTICES – CRITICALITY EXERCISES

Among the training practices, classroom training is the most usual one especially when the safety report of a facility changes. A new process implies new RGE/GOR, new procedures, operational instructions, etc.

Nuclear facilities perform periodic alarm tests of the CAAS (Criticality Accident Alarm System), in accordance with the constructor's recommendations and also training, exercises and evacuation drills in order to check personnel awareness.

8. GENERAL GUIDES AND DOCUMENTS FOR CRITICALITY ENGINEERS

Criticality engineers (at all levels) have guides which are available everywhere (paper, local servers, intranets). One can especially find:

- CEA N – 2051 (French criticality standard also known as "the Maubert guide ")
- Guide Cards5 (fiches guides): a collection of files (card format) each dealing with one issue such as the minimal critical mass, etc. …
- A Guide for Criticality Accident Studies[6]

among others.

There are also national groups dedicated to information sharing such as the French Criticality Experts Group made up by a panel of experts from French operating companies and engineering entities such as: the CEA, AREVA companies (NP, NC, E&P…), EURODIF, ANDRA, and also workgroups on the French criticality safety codes package, CRISTAL (users, validation, calculation schemes, …).

9. CONCLUSIONS

The CEA has a strong criticality safety organization in general and in each center in particular (local organization).

Regarding the general organization, a homogeneous process of Qualification and Appointment for the different criticality safety functions provides that criticality engineers have the required level of knowledge and skills. The central CEA criticality safety expertise group who is in charge of methodological guides, criticality networks, lessons learned, provide high level information especially on the evolution of the criticality "*état de l'art / state of the art*".

In each CEA center, the local organization, with IQCs in each nuclear facility (laboratory, storage area, reactor, etc …) and the SC for the entire center ensure that the operational resources and control functions are distinctly separated. The ICC is the "man in the middle" between the support resources and the control functions.

The CEA has an overall criticality safety organization supported by a complete set of documents.

5. File collection (ICNC'07: A guide to summarizing the main notions and principles of criticality safety: the criticality guide files collection.
6. NCSD'05: Guide in Progress for Criticality Accident Studies.

From TA-18 (Pajarito Site) to NCERC: Lessons Learned and Management Perspective for the United States' Only General-Purpose Critical Experiments Facility

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ABSTRACT

Technical Area 18 (TA-18), also known as Pajarito Site, at Los Alamos National Laboratory operated the only remaining general-purpose critical experiments facility in the western hemisphere up until approximately 2005. The facility was referred to as the Los Alamos Critical Experiments Facility. In 1999, the Secretary of Energy made the decision to relocate the TA-18 mission to a new location. The initial time frame for shutting down TA-18, relocating the mission (which included moving on the order of one ton of special nuclear material, and designing and constructing a new Security Category I / Hazard Category 2 Nuclear Facility), and then successfully re-starting was estimated to be 2014. With the events of September 11, 2001, the security posture of all nuclear facilities within the United States (U.S.) increased dramatically and, as one would expect, there was a significant corresponding increase in the security costs associated with this change. The U.S. Department of Energy and the National Nuclear Security Administration then made the business decision to accelerate the relocation of the TA-18 mission to save money on anticipated security costs associated with operations in the out years. This paper discusses the very successful congressional line-item project to shut down TA-18 and ultimately relocate and reconstitute the mission in what is today the National Criticality Experiments Research Center located at the Nevada National Security Site. Lessons learned from operating experience, including those learned during the mission relocation and reconstitution project, are presented and discussed. A management perspective on recent successes is also provided.

BACKGROUND

The Los Alamos Critical Experiments Facility (LACEF) operated for 60 years at Los Alamos National Laboratory's (LANL) Technical Area 18 (TA-18), also called Pajarito Site, in Los Alamos, New Mexico (shown in Figure 1). Operations were initiated at TA-18 in 1946 following the first of two critical assembly accidents where the reactivity of the critical assembly was controlled by hand. Following the first accident involving Harry Daghlian at Omega Laboratory in 1946, critical assembly operations were relocated to TA-18. Unfortunately, critical assembly operations were still being manipulated by hand and, approximately one year later, Louis Slotin suffered the same fate as Harry Daghlian. As a result of this second fatal accident, all hands-on critical assembly operations were terminated and a remote controlled critical mass laboratory (now known as LACEF) was rushed to completion in 1947 [1].

Early experiments (pre-1955) were performed primarily in support of weapons development programs. Experiments performed after this early period evolved into basic nuclear physics and nuclear engineering research, research for the National Aeronautics and Space Administration (NASA), e.g., the Rover nuclear-propulsion program, and research in support of criticality safety, nuclear nonproliferation, and emergency response programs. TA-18 was a collection of general-purpose laboratories capable of subcritical, delayed, and super-prompt critical operations using large quantities of special nuclear material (SNM).

Figure 1. Technical Area 18 (Pajarito Site) at Los Alamos National Laboratory

In 1999, the Secretary of Energy made the decision to relocate the mission of TA-18 to a new location. The initial time frame for shutting down, relocating the mission (which included moving on the order of one ton of special nuclear material, and designing and constructing a new Security Category I / Hazard Category 2 Nuclear Facility), and then successfully re-starting was estimated to be 2014. With the events of September 11, 2001, the security posture of all nuclear facilities within the United States (U.S.) increased dramatically and, as one would expect, there was a significant corresponding increase in the security costs associated with this change. The U.S. Department of Energy (DOE) and the National Nuclear Security Administration (NNSA) then made the business decision to accelerate the relocation of the TA-18 mission to save money on anticipated security costs associated with operations in the out years.[m] The U.S. National Environmental Policy Act (NEPA) process was followed and an official Record of Decision (ROD) was issued in December of 2012 indicating the new location of the TA-18 mission was to be within the Device Assembly Facility (DAF) at the Nevada National Security Site (NNSS). The DAF and the layout of the NNSS are shown in Figure 2.

The official start of the TA-18 mission relocation project was December 8, 2000. This is the date Critical Decision-0 (CD-0) was approved by the NNSA Deputy Administrator for Defense Programs authorizing initiation of pre-conceptual designs and other environmental, technical, and cost studies necessary to support a sound business decision on the relocation of TA-18 missions [2]. CD-0 is the first step of rigorous, systematic process for the acquisition of capital assets as managed by the U.S. DOE. This process, which is required for any project whose total project cost exceeds $50M, is described in detail in DOE O 413.3B [3]. DOE O 413 projects are also referred to as "line item" projects as their funding

m. The security costs for the TA-18 facility prior to 9/11/2001 was on the order of $20M per year. The anticipated costs post 9/11 jumped to $60M per year. Therefore, for every year the TA-18 mission relocation project could be accelerated, or perhaps more accurately for every year the downgrade of TA-18 from a Security Category I facility to a Security Category III facility could be accelerated, there would be a realized savings of $40M. Thus a five year acceleration of the project would yield a $200M security cost savings and the project would pay for itself. At least this was the thinking at the time and the TA-18 Mission Relocation Project became the TA-18 Early Move Project.

requires congressional line item approval in the federal budget. Additional details on this process are provided in a section below.

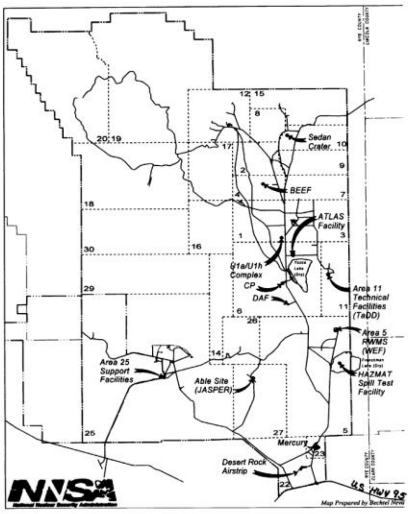

Figure 2. The Device Assembly Facility at the Nevada National Security Site

The TA-18 mission relocation project was ultimately a great success. Several other recent Hazard Category 2 / Security Category I nuclear facility projects within the DOE enterprise have not shared this same fate and have costs that have run into the billions of dollars, fallen years behind schedule and in some cases have been abandoned altogether. Our goal with this paper is to try to identify those aspects of the TA-18 mission relocation project which contributed the most to its success and to document lessons learned that can hopefully be applied to future large nuclear facility projects.

The *first key* to our success lies with decision to relocate the TA-18 mission to an existing nuclear facility that was designed for Hazard Category 2 / Security Category I operations. Moving into an existing facility as compared to new construction resulted in a significant savings in both cost and schedule. And just as important, if not more so, it likely reduced the impacts of implementing nuclear safety requirements into the design of a new facility as required by DOE O 1189, and the resulting iterative and (somewhat subjective) oversight processes that often result in severe cost escalation and construction delays.

UNIQUE CHALLENGES ASSOCIATED WITH TA-18 MISSION RELOCATION

Constructing and starting up any nuclear facility in today's regulatory environment represents a tremendous challenge. What was unique to the TA-18 mission relocation project however was we first needed to shut down operations at TA-18, followed by relocating nearly one ton of SNM to the new location 800 miles away, designing and constructing the new facility (once again 800 miles away), and finally starting up and operating a Security Hazard Category I / Hazard Category 2 facility at a remote location managed and operated by a different DOE contractor.

The NNSS in Nevada has a unique process for authorizing nuclear operations. This process is called the Real Estate / Operations Permit (REOP) process and it was absolutely key to our success in relocating the TA-18 mission and remains essential to our continued successful operations. What makes the REOP process unique from all others is that it allows National Laboratories other than the resident Management and Operating (M&O) contractor to own and manage their workspace as denoted in the REOP as if the space were at the home laboratory, in effect, 'sovereign" ground of LANL in this case. That is, organizations from outside of Nevada such as LANL can come to the NNSS and perform their own hands-on nuclear work and be responsible and accountable for their own activities. This solves the fundamental issue of defining and documenting roles, responsibilities, authorities and accountability. Within the DOE enterprise the M&O contractor is typically responsible and accountable for all activities within their site. The REOP process opens up the NNSS to all prospective users which was fundamental to the basic concept of relocating the TA-18 mission and having LANL continue to execute this vital work.

The REOP process not only allows outside organizations to come to the NNSS and manage their own work, it also allows these same outside organizations to bring their home laboratory processes, procedures, certifications and qualifications with them and ensures that these NNSA-approved home lab credentials are recognized and authorized. This is an unprecedented level of authority granted to an outside organization at a DOE/NNSA site. And, as one might expect, the price to be paid for such authority is accountability. The organization responsible and accountable for nuclear activities at the NNSS is whoever owns the REOP. If an outside organization's activity under their approved REOP results in an accident or an injury, then they are responsible and accountable for the consequences. All reporting requirements and ultimately any contract performance successes or deficiencies are attributed to the owner of the REOP.

It is the above attributes of the REOP process that make it the ultimate user model within the DOE enterprise and is the *second key* to the success of the TA-18 mission relocation project.

A PHASED APPROACH TO NUCLEAR STARTUP

The new facility to house the TA-18 mission was initially called the Criticality Experiments Facility (CEF) but was later renamed the National Criticality Experiments Research Center (NCERC) to emphasize its unique and vital role in the DOE national security enterprise. NCERC occupies approximately 40% of the floor space of the DAF at the NNSS and the layout of NCERC within the DAF is shown in Figure 3.

The tremendous challenge of closing one nuclear facility, relocating the mission to a different site, and then reconstituting that mission lends itself quite well to a phased approach. In particular we approached the startup of operations at the NNSS as a series of baby steps. Some, but certainly not all, of these small, *achievable* incremental steps were as follows:

1. Stood up Security Category I operations at the DAF
2. Established REOP and initial authorization basis, and performed federal readiness assessment [operational readiness review (ORR)] for receipt of SNM in containers

3. Received first shipment of SNM and demonstrated successful nuclear material handling, security operations, radiation protection and contamination control, criticality safety, and material control and accountability (MC&A)
4. Opened first containers and extended nuclear material and handling operations to include handling bare SNM and completing MC&A verification measurements
5. Constructed first radiation test object (RTO) to further extend criticality safety applications to high-multiplication systems and facilitated first scientific sub-critical experiments in NCERC
6. Conducted first training classes for criticality safety practitioners and emergency responders using sub-critical RTOs
7. Stood up portable radiography operations thus furthering emergency response research and development (R&D) and training capability
8. Conducted first international emergency responder exercise further expanding RTO activities to include portable radiography and expanded security operations to include cleared foreign nationals
9. Stood up critical assembly operations including full federal ORR; performed first critical

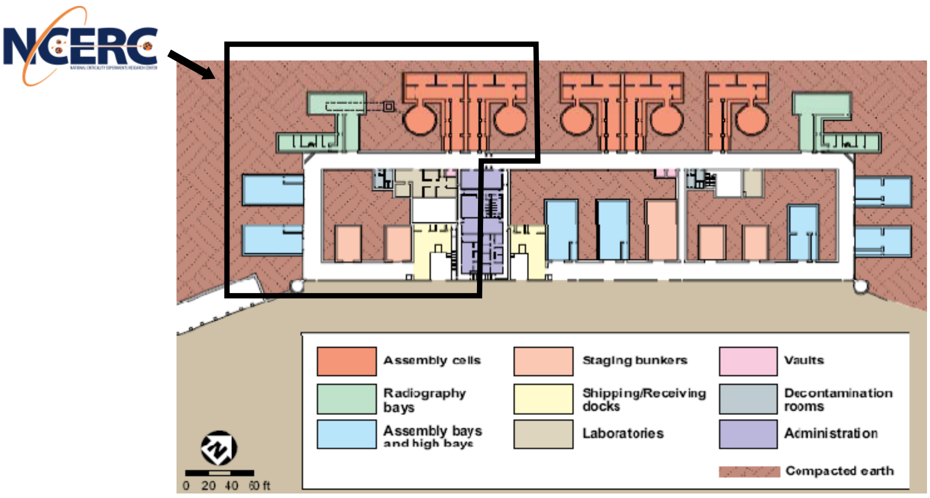

Figure 3. Layout of the DAF at the NNSS showing the location of NCERC within the facility

This phased approach to the startup of nuclear operations allowed us to demonstrate a series of incremental successes to our federal oversight and regulators thus building confidence with each step along the way. Another valuable lesson learned from past nuclear project failures is to not reach too far with your goals and expectations. The baby steps, phased approach to building confidence with you regulators and demonstrating incremental success is the *third key* to the success of the TA-18 mission relocation project.

The initial phases of NCERC operations (opening containers, handling nuclear material, performing subcritical measurements and constructing RTOs) started in June of 2007. This was followed by approval of portable radiography operations in August of 2009 and finally Critical Decision (CD) 4 approval of full NCERC critical operations in May of 2011. First critical on the Planet critical assembly occurred on June 15, 2011. Following CD-4 approval, NCERC embarked on a two-year-long, deliberate, and systematic startup plan to achieve first critical and then sustained critical operations on the four critical assemblies located at NCERC. The final objective of the startup plan culminated in the first Godiva super-prompt-critical burst operation which occurred on September 10, 2013.

MAINTAINING SKILLS AND PROFICIENCY DURING TRANSITION

One of the requirements of the TA-18 mission relocation project was to ensure that the skills and proficiency of the operators and nuclear material handlers was maintained during the transition. Our approach to achieving this was to utilize other operational nuclear facilities to the extent possible while re-establishing our nuclear material handling capability at the NNSS in the shortest possible time.

The phased approach to nuclear startup described in the previous section helped us out tremendously in this regard. By using this approach, the “down time” where we were forced to use other nuclear facilities was limited to only two years between 2005 when operations at TA-18 were terminated and 2007 when RTO operations at NCERC were initiated. Regardless, two years is a long time and we therefore made every attempt to maintain proficiency by using other facilities and collaborating with other laboratories and institutions.

The DOE Nuclear Criticality Safety Program (NCSP) entered into partnership with the French Le Commiserriat a l’Energie Atomique (CEA) to train U.S. scientists on the French critical assemblies at their Valduc Center. CEA had a fast burst metal reactor (Caliban) similar to Godiva and also a solution burst assembly (Silene) similar to Sheba. We jointly developed a formal training syllabus and qualification standard to provide auditable documentation of the training. Over the course of two years several of our LANL scientists were actually qualified to operate the French critical assemblies and did so. This was an important piece of maintaining the competence and proficiency of our LANL scientists as required by the DOE restart order DOE O 425. At the time of the readiness reviews all our scientists had auditable documentation of their current qualifications to operate critical assemblies and there were no findings of any kind related to this area of readiness to restart. Our colleagues at CEA Valduc contributed significantly to the success of NCERC startup.

In parallel with these activities, measurement and experiment campaigns were also conducted in the LANL Plutonium Facility (PF-4) and with our colleagues at Sandia National Laboratories.

THE DOE O 413 CONGRESSIONAL LINE ITEM PROCESS

Another requirement for the success of a large nuclear project is a clear roadmap to successful project completion. Without a rigorous, systematic and defensible process to define the steps, processes and deliverables, project success would likely not be achievable. The 413 line item project order provides this roadmap. We will not go into much detail here describing this process – reference 3 provides additional detail should it be desired. However, the importance of having such a process is the *fourth key* to the success of the TA-18 mission relocation project.

The 413 process can be characterized by the four critical decision points. These are:

CD-0 Approval of mission need. Specifically, that there is a need that cannot be met through other than material means. CD-0 is also characterized as the start of the conceptual design process.

CD-1 Approval of alternative selection and cost range. The selected alternative and approach is the optimum solution. CD-1 approval marks the completion of the project definition phase and the conceptual design.

CD-2 Approval of the performance baseline. Definitive scope, schedule and cost baselines have been developed. CD-2 is also characterized as the approval of the preliminary design.

CD-3 Approve start of construction/execution. The project is ready for implementation. During the period between CD-2 and CD-3 approval, the final design will have reached the level of maturity necessary to have confidence in the decision to initiate construction.

CD-4 Approve start of operations or project completion. The project is ready for turnover or transition to operations, as applicable.

These same four critical decision points are also illustrated in Figure 4 below. One of the aspects to having such a rigorous and systematic process is the defense of the project throughout its lifetime. Large nuclear projects typically take years to execute; spanning management and even political administration changes. And without question challenges to decisions made during project execution will continue to come up throughout its lifetime. The documentation requirements of the 413 process become not only the technical basis for the defense of these decisions, but also the legal basis.

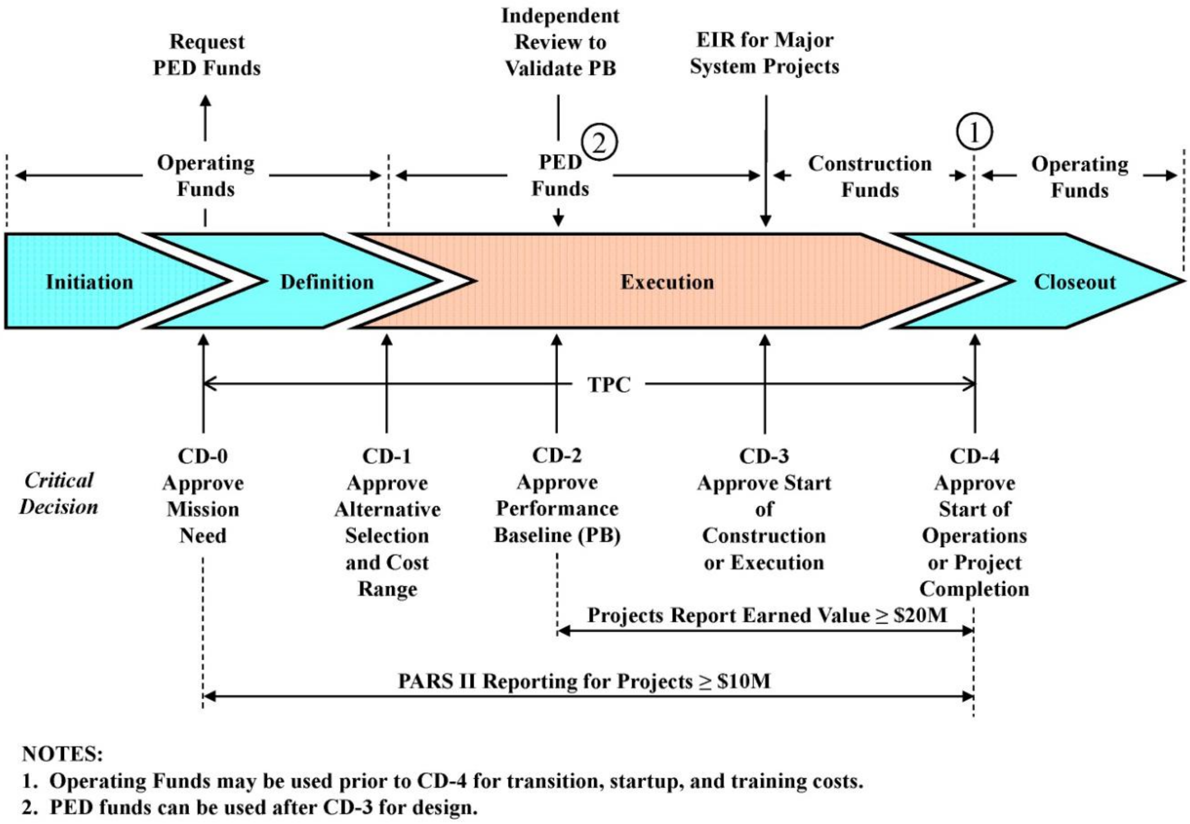

Figure 4. DOE O 413 Process for Line Item Capital Asset Projects

CRITICALITY SAFETY IMPROVEMENT MEASURES

During the TA-18 mission relocation project and the construction of NCERC, opportunities presented themselves for process improvements – specifically in the area of criticality safety. A few of these improvements are touched on here.

Seismic Protection

The DAF at the NNSS is a very robust, reinforced concrete facility that was originally designed for the assembly and disassembly of nuclear explosive devices. As such, the facility includes blast protection for working with high explosives and was designed from the beginning to meet seismic performance category 3. The facilities at TA-18 ranged anywhere from forty to sixty years old and were not designed or constructed with any seismic protection that could be credited by today's standards. Therefore, the relocation from TA-18 to the DAF inherently introduced a major increase in protection from seismic hazards.

Another aspect of seismic protection that was introduced as part of the relocation project was anchoring of the critical assemblies. Again, at TA-18 the assemblies were not anchored in any fashion. At NCERC in the DAF however, each of the four critical assemblies is anchored to the floor such that they meet seismic performance category 2.

Pre-action Fire Suppression

The DAF has modern fire detection and suppression systems including both smoke and heat detectors for detection and wet pipe sprinklers, and a deluge system in one case, for suppression. We know however that inadvertent sprinkler actuation in the presence of an operating critical experiment would likely lead to a reactivity excursion accident. Therefore, during the NCERC construction project, the wet pipe sprinkler system in the locations where RTO construction and critical assembly operations take place was modified to become a dry pipe pre-action fire suppression system. A pre-action system requires that fire first be detected before the system is charged with water, then the heat from the fire must still melt the fusible links in the sprinkler heads to initiate water flow. The introduction of this system reduced the probability of inadvertent sprinkler actuation by over two orders or magnitude.

Vault Design

The vaults at NCERC were designed as stainless steel racks with several hundred 2x2x2 ft. storage locations. The vault racks, unlike those at TA-18, were once again designed with seismic hazards in mind and were designed and built to seismic performance category 2. The vaults were initially intended to include a physical four inch spacing in-between each storage location thereby allowing higher storage limits. However, the engineering, implementation and ultimately cost of this design made it unattainable within the scope and funding of the project. So, contiguous 2x2x2 ft. locations were constructed with the intent of, at some point later in time, introducing two-inch hollow aluminum spacers on the interior of each storage location thus achieving the same effect.

Procedure Based Limits

One of the more significant changes at NCERC as compared to operations at TA-18 is the introduction of procedure based criticality safety limits. The criticality safety limits at TA-18 were generic to a storage location of a given size and were posted throughout the facility. This had the advantage of ease-of-use and the training to such limits was always consistent. However, today's nuclear safety expectations, specifically in the area of conduct of operations, requires that operations be procedurally based. Therefore all nuclear material handling operations at NCERC are conducted using in-hand procedures.

MANAGEMENT PERSPECTIVE OF THE TA-18 MISSION RELOCATION PROJECT

The relocation of the TA-18 mission from Los Alamos to NCERC at the NNSS in Nevada did cause some dismay among the long-time LANL staffers, but at the same time it also created an absolutely unique opportunity to transform and rebuild a sixty-year old facility. Some (but certainly not all) of what was accomplished during the transition included: (1) cleaning up the SNM inventory such that only national asset material was retained, (2) completely redesigning, rebuilding and updating the critical assembly control systems and instrumentation, (3) modernizing/updating the authorization basis documentation and all of the operating procedures, and (4) redefining the operational paradigm to leverage collaborative opportunities presented by the REOP process that is used to authorize nuclear operations at the NNSS.

NCERC is working towards having full plutonium critical experiment capability in the next year or two.[n] Together with the recently added ^{233}U fuel and vast assortment of diluent materials available, the utility of NCERC to provide integral data measurements is almost unlimited. It is the vision of our federal programmatic sponsors in the NNSA that NCERC grow to expand collaborations with the United Kingdom, France, and Japan on U.S. experiments of interest to our international colleagues.

The TA-18 mission relocation project had very strong support from both the federal government (DOE/NNSA) and the affected local governments (e.g., the state of New Mexico and the state of Nevada). This support along with the support of LANL senior management and the senior managers of the M&O contractor at the NNSS created the environment necessary for a large nuclear facility construction project to succeed. This is the *fifth key* to the success of the TA-18 mission relocation project.

Operations at NCERC are supported by the DOE NCSP, funded and managed by the NNSA for the DOE. A long term goal is to establish NCERC as a world class facility to be used for performing collaborative work with non-DOE domestic organizations and to collaborate with international partners. The scope of work for these collaborations can involve training, equipment testing, and data gathering for various applications that require subcritical and critical configurations of SNM. Non-DOE domestic organizations include NASA, the Department of Homeland Security (DHS), the Defense Threat Reduction Agency (DTRA), other government agencies, various universities, and some commercial partners. International collaborative partners include the Atomic Weapons Establishment (AWE) (England), the institute for radiological protection and nuclear safety (IRSN) and CEA (France), and the Japan Atomic Energy Agency (JAEA) (Japan). Unclassified benchmark experiments are published in the Organisation for Economic Co-operation and Development (OECD) International Criticality Safety Benchmark Evaluation Project (ICSBEP) handbook and we cooperate and contribute internationally to make these results useful to the international criticality safety community.

The first shipment of nuclear material from TA-18 to the NNSS occurred in September of 2004 and the final shipment which allowed downgrading TA-18 to a Security Category III facility occurred in October of 2005 – a full nine years ahead of the original schedule, thus saving the government $360M in security costs. Comparing this cost savings to the total cost associated with TA-18 mission relocation, which was on the order of $160M, would lead one to conclude that the business case for relocating the mission, and the subsequent decision to accelerate the move, were indeed sound.

CONCLUSIONS

The relocation and successful restart of the TA-18 mission in NCERC represents a major accomplishment for NNSA and DOE. NCERC is our nation's only general-purpose critical experiments facility and is only one of a few that remain operational throughout the world. NCERC's primary mission is to provide the integral experimental infrastructure of the DOE through the DOE NCSP providing nuclear data and technology needed for criticality safety, training, emergency response, and a host of other DOE and worldwide nuclear interests. NCERC and the people who operate it provide access to, and expertise in handling large (Security Category I) quantities of SNM for integral critical and subcritical experiments, R&D, and testing and evaluation. This capability is a vital component of a fully-functioning NCSP, and represents a national asset for our nation's nuclear R&D needs.

The successful startup of NCERC is particularly significant given the challenges associated with constructing and starting up Hazard Category 2 nuclear facilities. With the exception of NCERC, DOE/NNSA's recent efforts to design, construct and startup Hazard Category 2 nuclear facilities has reached into the billions of dollars and spanned decades, thus resulting in serious challenges to the their

n. Integral sub-critical experiments with large quantities of plutonium are already authorized.

future viability. NCERC stands as a successful example of how a congressional line-item project for a Security Category I / Hazard Category 2 nuclear facility can and should be executed.

Throughout this paper we have attempted to describe the challenges associated with relocating a major nuclear facility and highlight the keys to the success of the project. While there were thousands of individual activities and elements that had to come together for this project to succeed, the five below stand out as the most important.

1. Relocating to an existing nuclear facility rather than attempting new construction
2. The REOP process at the NNSS; the ultimate user model
3. A phased approach to nuclear startup; small, incremental steps to building confidence in your regulator
4. DOE O 413; a clear, systematic and defensible roadmap to success
5. Broad mission support across the enterprise and local governments

REFERENCES

1. H.C. Paxton, “Thirty-Five Years at Pajarito Canyon Site,” Los Alamos National Laboratory report LA-7121-H, Rev., May 1981.
2. U.S. DOE Memorandum from Madelyn Creedon to Roger Dintaman, Request for Critical Decision-0 (Phase I) for Los Alamos National Laboratory Technical Area-18 Mission Relocation, December 8, 2000.
3. Program and Project Management for the Acquisition of Capital Assets, DOE O 413.3B, November 29, 2010.