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Nuclear Installation Safety – Assessment

Scientific and Technical Support to the Regulatory Body Within the Framework of MDEP VVER WG Activity

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Abstract:

The Working Group (WG) on new VVER designs within the Multinational Design Evaluation Programme (MDEP) comprises the members from the nuclear regulatory authorities of China, Finland, Hungary, India, Russia and Turkey.

The VVER WG as well as the other four MDEP design specific working groups (EPR, AP1000, APR1400, ABWR) is successful in sharing information and experience on the regulatory safety reviews of new reactor designs with the purposes of enhancing the safety and enabling regulators to make timely licensing decisions.

The VVER WG is chaired and managed by SEC NRS (Russia) in arrangement of the group activity providing support to the regulators and reporting to the MDEP Steering Technical Committee and Policy Group on the benefits.

The VVER WG includes three technical expert subgroups: Severe Accidents Management, Fukushima Lessons Learned covered by design solutions, Reactor Pressure Vessel & Primary Circuit Components. The group activities include exchange of information on national legal framework related to new designs, regulatory safety requirements, safety assessment approaches, safety review results and experience.

SEC NRS supports the interactions between the VVERWG members and the Russian design and operating organizations to get additional information about new VVERs design solutions related to safety during technical visits to the new units under construction and commissioning in Russia and abroad.

1INTRODUCTION

The Multinational Design Evaluation Programme (MDEP) was established in 2006 following the proposal of the regulatory authorities of the USA, Finland, France and Russia.

MDEP is the multinational initiative to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities that are currently or will be tasked with the review of new nuclear power reactor designs [1].

The OECD Nuclear Energy Agency (NEA) facilitates MDEP activities by providing Technical Secretariat services for the programme.

The IAEA also participates in key aspects of MDEP's activities.

In accordance with the Terms of Reference the main objectives of MDEP are:

- Enhance multilateral co-operation within existing regulatory frameworks;
- Encourage multinational convergence of codes, standards and safety goals;
- Implement the MDEP products in order to facilitate the licensing of new reactors, including those being developed by the Generation IV International Forum.

A key concept throughout the work of the MDEP is that national regulators retain sovereign authority for all licensing and regulatory decisions.

The nuclear regulatory authorities of 15 countries participate in MDEP including:

- Canada (CNSC – Canadian Nuclear Safety Commission),
- People's Republic of China (NNSA – National Nuclear Safety Administration),
- Finland (STUK – Radiation and Nuclear Safety Authority),
- France (ASN – Nuclear Safety Authority),
- Hungary (OAH – Hungarian Atomic Energy Authority),
- India (AERB – Atomic Energy Regulatory Board),
- Japan (NRA – Nuclear Regulatory Authority),
- Korea (NSSC – Nuclear Safety and Security Commission),
- Russian Federation (Rostechnadzor),
- South Africa (NNR – National Nuclear Regulator),
- Sweden (SSM – Swedish Radiation Safety Authority),
- Turkey (TAEK – Turkish Atomic Energy Authority),
- United Arab Emirates (FANR – Federal Authority for Nuclear Regulation),
- United Kingdom (ONR – Office for Nuclear Regulation),
- United States (NRC – Nuclear Regulatory Commission).

The MDEP Policy Group (PG) and the Steering Technical Committee (STC) oversee the programme.

Working groups are implementing the activities in accordance with programme plans with specific activities and goals, and have established the necessary interfaces both within and outside of MDEP.

1.1 MDEP Structure

In accordance with the Terms of Reference, the MDEP carries out its work through design-specific and issue-specific working groups [2].

1.1.1 Design-specific Activities

Working groups for each new reactor design share information and co-operate on specific reactor design evaluations, construction, commissioning, and early phase operation. Participants in these working groups are MDEP member regulatory authorities (or their TSOs) who are interested in a specific reactor design and are willing and capable of contributing positively to the group’s activities.

There are five design-specific working groups on EPR, AP1000, APR1400,VVER and ABWR.

Under the design-specific working groups, expert subgroups have been formed to address specific technical issues.

1.1.2 Issue-specific Activities

Working groups have been established for selected technical and regulatory process areas within the MDEP's Programme of work. These currently include:

- Vendor Inspection Co-operation Working Group;
- Codes and Standards Working Group;
- Digital Instrumentation and Controls Working Group.

1.1.3 MDEP Library

MDEP information is communicated among the members through the MDEP library which serves as a central repository for all documents associated with the programme.

2 WORKING GROUP ON NEW VVER DESIGNS

2.1 Background

The Working Group on New VVER designs (VVER WG) was established in September 2013 to focus on the safety reviews of new VVER reactors. The VVER WG reports the status of its activities three times a year at the MDEP Steering Technical Committee (STC) meetings. The VVER WG interacts with key stakeholders involved in new VVERs construction including designers.

Rostechnadzor authorized SEC NRS to implement its function to lead and manage VVERWG and its subgroups.

Key stakeholders with whom the VVERWG interacts routinely include Russian design and operating organizations and utilities pursuing licensing and construction of new VVERs.

2.2 VVER WG Objectives

VVER WG objectives are:

- Leverage resources of the regulators and ensure that the VVER design reviews remain safety-focused;
- Exchange information on regulatory requirements and safety reviews taking into account design differences and differences in licensing processes in the following main areas:
  - Fukushima lessons learned enhancements;
  - RPV&PC;
  - Severe accidents.

2.3 New Nuclear Power Plants of VVER Design in Russia and Abroad

Currently nuclear power plants (NPPs) with new VVER designs (VVER-1000, VVER-1200, VVER-TOI) are being sited or constructed in China, India, Russia, Finland, Turkey, Hungary, Belarus, Egypt, Bangladesh and Vietnam.

2.4 VVER WG Members

The VVER WG includes regulatory bodies and TSOs of six MDEP members, namely China, India, Russia, Finland, Turkey, and Hungary.

2.5 VVER WG Structure

The VVER WG is chaired by Russia (SEC NRS).

The VVER WG consists of three technical expert sub-groups addressing specific technical issues (TESGs):

- Severe accidents;
- Fukushima lessons learned;
- Reactor pressure vessel and primary circuit.

2.6 VVER WG Activity

The VVER WG activities cover the following:

- Regular (twice per year) meetings with technical visits to new VVER NPPs under construction/commiissioning in member countries;
- Exchange of information and experience on regulatory activities, approaches and legal framework related to new NPP designs and design differences important to safety;
- Development of the comparison table of differences in the new VVER designs;
- Interactions (supported by SEC NRS) with Russian design organizations to provide necessary information, data on safety significant design solutions and to attend VVER WG and its TESGs meetings.

2.7 Technical Expert Sub-Group on Severe Accidents (TESG SA)

2.7.1 Introduction

Technical Expert Sub-Group on Severe Accidents (TESG SA) is chaired by Russia (SEC NRS, NPP Safety Division).

Its objectives are:

- Understand the differences in regulatory approaches and oversight practices used in VVER WG member countries related to severe accidents assessment and management as well as to identify commendable practices in this area;
- Prepare a common Technical Report covering following topics:
  - Methodology for severe accidents analysis;
  - Technical provisions for safety systems;
  - Severe accident management operating strategies (SAMG);
  - Radiological impact assessment.

2.7.2 Results of TESG SA Activity

SEC NRS developed and agreed with TESG SA members the Questionnaire related to SA. The members' answers to the Questionnaire underlined the following:

- Common approach of member countries´ representatives assumes that the issues relating to severe accidents have to be under control of national regulators and conform to international agreements, domestic laws, requirements and guidelines;
- The requirements of IAEA and other international organizations (WENRA, etc.) should be also taken into consideration;
- The volunteer initiative of the licensee is possible and encouraged, but the measures taken by the licensee on this basis have to be reviewed and agreed by the regulator.

The questionnaire and answers are basis for the Technical Report.

The Technical Report on regulatory approaches and criteria used in severe accident analyses and severe accident management covers the following aspects of SA and SAM:

- Procedures and Guidelines;
- Equipment for the severe accident management;
- Methods and approaches to SA analyses.

2.7.2.1 Procedures and Guidelines

The following procedures and guidelines have to be mentioned:

- The development of SAMG covering prevention and mitigation stages of severe accident are mandatory for licensee in all member states.
- SAMGs should be symptom-based and corresponding to administrative and technical requirements. Entry and exit criteria have to be clearly defined on the basis of measured parameters. Exit criteria are the set of conditions which define the stable and safe NPP state.
- In all member countries it is required that a decision making person has to be defined unambiguously. Usually, this responsibility is relying on the operator and assigned to the emergency director. The technical support center has a supporting role (to elaborate the recommendations on management strategy).
- In all member countries the review of SAMG and the corresponding technical basis are a part of licensing; verification and validation of SAMG are required. Plant simulator and results of SA analyses could be a basis of SAMG validation. The periodic emergency training and drills should be used to verify SAMG.
- The SAMG compliance with current NPP state is required and to be confirmed.

2.7.2.2 Equipment for Severe Accident Management

The following equipment for severe accident management is considered:

- Type of equipment;
- Mission time of equipment dedicated for SAM;
- Independency and single failure criteria;
- Safety classification;
- Requirements on I&C;
- Leak tightness and integrity of the containment;
- Heat removal from the damaged fuel;
- Devices for primary pressure decrease;
- Core catcher.

2.7.2.3 Methods and Approaches to SA Analyses

The below mentioned methods and approaches are implemented:

- List of SA;
- SA acceptance criteria;
- Large radioactive release;
- Severe accident analysis codes.

The methodology for SA analysis in general should be based on an as far as possible realistic approach, however in case of lack of knowledge conservative assumptions are credited (the conservativeness of the assumptions should be proved). An as far as possible realistic approach on SA analysis means to be based on best estimate codes plus sensitivity studies as far as applicable.

2.8 TESG SA Outputs

2.8.1 Conclusions

The following conclusions have been drawn:

- The results of SA analyses should confirm the successful recovery of the main safety functions (subcriticality of the damaged fuel, cooling, localization).
- The integrity and leak tightness of the containment under severe accident conditions must be proven based on SA analyses. SA analysis should consider all the phenomena that can aggravate the impacts on the containment and on the systems within containment). Elimination of hydrogen detonation must be confirmed.
- In all member countries the criterion applied with respect to cumulative CDF (core damage frequency) is equal to 10^{-5} 1/ reactor year. Large early releases should be practically eliminated.

2.9 TESG on Fukushima Lessons Learned (TESG Fuku)

2.9.1 Introduction

The Technical Expert Sub-Group on Fukushima Lessons Learned (TESG Fuku) is chaired by Turkey (TAEK) and co-chaired by Russia (SEC NRS, NPP Safety Division).

Its objectives are:

- Understand the differences in regulatory approaches and oversight practices used in VVER WG member countries related to how Fukushima lessons learned are considered in new VVER designs;
- Develop a common position addressing Fukushima related issues covering the following topics:
  - Accounting of external events in new VVER designs;
  - Design solutions to cover specific Beyodn-design basis accidents (BDBAs) such as station black-out (SBO) and loss of heat removal to ultimate heat sink (UHS);
  - Emergency preparedness and response and reliability of safety functions implementation.

2.9.2 TESG Fuku Outputs

The TESG Fuku common position covers the following issues important to safety:

- Accounting of external events in new VVER designs:
  - Site-specific characteristics (seismic hazards, external flooding, aircraft crash, hurricanes and tornados, external explosions);
  - Adequate protection against extreme external hazards and their credible combinations;
  - Multi-unit consideration;
  - Hazard assessments;
  - Periodic re-evaluation of external hazards;.
- Reliability of safety functions implementation
  - With the aim to guarantee solid DiD, the technical means designed for maintenance of three fundamental safety functions should conform the certain principles.
  - The main concept of new VVERs for providing fundamental safety functions are:

  - Passive means to deal with "design extension conditions" and "beyond design basis accidents" and provide back up for active safety systems;
  - Multiple train redundancy;
  - Diversity;
  - Physical separation of all four trains of safety systems and their control systems.

− Design solutions to cover specific BDBAs (SBO and loss of UHS):
  - Application of passive heat removal systems should be considered as engineering means for organization of reactor fuel heat removal.
  - Application of specially designated batteries with large discharging period should provide the additional possibility for monitoring the status of the fundamental safety functions along with the implementation of some accident management actions (etc., power restoration).
  - Measures facilitating the restoration of offsite power (hydro-electric power stations, gas-turbine power stations, etc.) should be considered in the NPP design.
  - Introduction of redundancy for the ultimate heat sink should be considered in the NPP design.
  - Application of mobile engineering means for accident management should be considered as a measure to ensure NPP safety in course of SBO or loss of UHS scenarios.

− Emergency preparedness and response and reliability of safety functions implementation:
  - The emergency plans should be comprehensively prepared and periodically demonstrated via full-scope exercises;
  - Training facilities should be extended to cover severe accident scenarios in order to support the preparedness of the personnel and improve the realistic character of emergency exercises;
  - Roles and responsibilities of all organizations involved in emergency management and response should be clearly identified and periodically checked during drills and exercises;
  - Accessibility and habitability of the control room, the emergency response center, and the local control points need to be adequately protected against internal and external hazards.
  - In the emergency plans and procedures, more emphasis should be provided on the protection of emergency workers in terms of provision of protective equipment and emergency dosimeters in appropriate number and of relevant strategies and procedures to avoid any unjustified risks during the response.
  - Instrumentation and controls qualified for accident conditions should be designed and installed to support the accident management measures by controlling the reactor and the spent fuel pools status.
  - Reliability and functionality of the on-site and off-site communication systems, equipment measuring radioactive releases, radiation levels and meteorological conditions need to be ensured, taking into account conditions related to extreme internal and external hazards.
  - On-site emergency plans, procedures and guidelines should cover long-term actions and possible influence of the facilities at the site;
  - Severe environmental conditions and possible degradation of the regional infrastructure may impact the emergency preparedness and should be considered in the emergency planning.

- For site facilities, the plant should be considered as a whole in safety assessments and emergency management and interactions between different units need to be analysed. External events that may simultaneously affect several site facilites should be explicitly considered in the emergency preparedness.

2.10 TESG on Reactor Pressure Vessel and Primary Circuit (TESG RPV&PC)

2.10.1 Introduction

Technical Expert Sub-Group on Reactor Pressure Vessel and Primary Circuit (TESG RPV&PC) is chaired by Finland (STUK), co-chaired by Russia (SEC NRS, Systems Structure & Components Integrity Division).

Its objectives are:

- Discuss the differences in regulatory approaches and oversight practices used in VVER WG member countries related to RPV&PC;
- Develop a Technical Report on regulatory requirements related to RPV&PC integrity.

2.10.2 TESG Technical Report on Regulatory Requirements Related to RPV&PC Integrity

Safety topics agreed on regulatory requirements related to:

- Application of LBB (Leak Before Break) concept;
- Manufacturing of primary circuit components;
- RPV radiation embrittlement regarding use of new base metal;
- Pre- and in-service inspection of primary circuit components;
- Design basis loadings and their combinations for primary circuit components;
- Cladding of primary circuit;
- Protection against overpressure of primary circuit.

2.11 VVER WG Milestones

The next VVER WG milestones are planned to be the following:

- Finalize the development of a common position addressing Fukushima related issues and submit to MDEP Steering Technical Committee by the end of 2016;
- Finalize the development of a Technical Report on regulatory approaches and criteria used in severe accident analyses and severe accident management for further submission to MDEP Steering Technical Committee for approval in 2017;
- Finalize the development of aTechnical Report on regulatory requirements related to RPV&PC integrity for further submission to MDEP Steering Technical Committee for approval in 2017.

2.12 Further VVER WG Activities

Further VVER WG activities will consider:

- Exchange of information on regulatory requirements and safety reviews taking into account design differences and differences in licensing processes in the following areas:
  - passive systems;
  - construction oversight;
  - radiation protection;

  - spent fuel pool and fire protection.
- Decision on practicality to establish TESG(s) to cover one or more of the above mentioned areas;
- Participation in preparation and conduct of the 4th MDEP International Conference on New Reactor Design Activities (September 2017).

2.13 Scientific and Technical Support to Regulatory Body

Sceintific and technical support provided by SEC NRS to regulatory body (Rostechnadzor) includes the following:

- Overall coordination of VVER WG and its TESGs activities;
- Transfer of experience and information on Russian regulatory requirements and safety reviews to VVER WG members;
- Coordination with Russian new VVER designers and operating organization to provide necessary information, data on safety significant design solutions;
- Invitation of representatives of Rosatom, Rosenergoatom and design organizations to attend VVER WG and its TESGs meetings to discuss and clarify new VVER design related issues;
- Arrangements (in collaboration with the industry) to organize technical visits to Russian reference NPPs (Leningrad NPP-2, Novovoronezh NPP-2);
- Carrying out the analysis of the regulatory requirements to new NPPs in member countries with further development of common positions and technical reports;
- Technical and informational assistance to member regulators which may be used for drafting the regulations, safety reviews and their decision making when licensing new VVERs.

REFERENCES

[1] http://www.oecd-nea.org/mdep/annual-reports/mdep-annual-report-2015.pdf

[2] http://www.oecd-nea.org/mdep/

Development and Application of Modern Safety Requirements as Part of GRS Technical Support for ANVS

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Abstract:

The Dutch Authority for Nuclear Safety and Radiation Protection (ANVS) contracted GRS as a TSO to support the improvement of the regulatory framework as well as review and assessment activities. One main task was the development of a modern set of Dutch Safety Requirements (DSR) for Nuclear Power Plants (NPPs) and research reactors where the most recent state of the art in science and technology was considered. As an example, the defence-in-depth concept proposed by WENRA is an integral part of the DSR. In 2011, development of the DSR started. The DSR were published in October 2015 following a positive review by the IAEA. In addition, ANVS and GRS developed a review plan to ensure an effective, comprehensive and transparent review of the Safety Analysis Report (SAR) in which the DSR are applied appropriately. Both the new requirements and the review plan are currently being applied during the review of the research reactor modification project at Delft University of Technology.

1 INTRODUCTION

ANVS as the regulatory body in The Netherlands being in charge of both licensing and oversight for nuclear facilities contracted GRS to provide enhanced technical support in the field of nuclear safety. This includes the development of the safety requirements for NPPs and research reactors as well. The DSR provides goal-oriented requirements where different technical solutions or approaches may provide an acceptable level of safety. However, where ANVS' technical safety expectations are precise, prescriptive and detailed, criteria are defined assuming a standard NPP (LWR) design. The Dutch regulator decided not to develop a dedicated regulatory framework for research reactors, but to apply NPP requirements by a graded approach. The new Dutch Safety Requirements as well as current Dutch legislation and further safety guides (in Dutch: Nuclear Veiligheids Richtlijn NVR) serve as a reference to review and assess a Safety Analysis Report (SAR).

To ensure an effective and thorough review and assessment a review plan, an organisational (Organisational Review Plan) and a technical part (Technical Review Plan) were generated. By developing such a review and assessment plan, the Dutch regulator follows an IAEA recommendation expressed in IAEA Safety Standard No. GS-G-1.2 [1]. The review plan also establishes a link between the content of the SAR and the regulatory framework in the Netherlands. The review plan primarily represents a guidance document for the reviewer, but also serves as guidance for applicants dealing with the preparation of a SAR that is in line with expectations of ANVS.

2 THE DUTCH SAFETY REQUIREMENTS FOR NUCLEAR REACTORS

In order to develop a modern set of Dutch Safety Requirements (DSR) for NPPs and research reactors, ANVS and GRS considered the most recent state of the art in science and technology described in the following documents. For new NPPs more stringent expectations towards nuclear safety have been formulated internationally and within Europe. The IAEA Safety Standards, primarily the IAEA Safety Standard No. SSR 2/1, forms the international basis for the development of the Dutch requirements; however, further IAEA safety requirements and IAEA safety guides were taken into account. On the European level the basis is formed by the directive of the European Council 2009/71/EURATOM and the WENRA safety objectives on new reactor designs [2]. In the WENRA report on “Safety of new NPP designs” [3] these expectations of the seven safety objectives have been elaborated in more detail. In addition, the WENRA updated the Safety Reference Levels for existing reactors to include the lessons learned from the accident at the Fukushima Dai-ichi plant. Another important European document is the “Technical Guidelines for the Design and Construction of the Next Generation of Nuclear Power Plants with Pressurized Water Reactors” [4], a report prepared by IRSN and GRS and finally adopted and published by the French GPR and German experts. Experiences from other countries currently updating their nuclear regulations were considered. Particularly the new German Safety Requirements for Nuclear Power Plants [5] with their supplementary Interpretations [6] served as a main reference. In addition, the new Finnish Guideline YVL B.1 [7] was consulted. The main difference compared to design requirements for existing reactors is that also accident conditions more severe than traditional design basis accidents have to be considered in the design. In the IAEA Safety Standard No. SSR 2/1 [8] those accident conditions have been introduced as design extension conditions. A similar approach was introduced by the European Utility Requirements EUR [9], an initiative by the European nuclear utilities. For the DSR the WENRA approach [3] with its more unambiguous terminology was adopted. Within an IAEA review mission the draft DSR were assessed. The main conclusion was that the DSR incorporated the most recent technical safety concepts of the recently published safety requirements, drawing upon developments made to enhance defence-in-depth (as required by WENRA), including lessons learned from the accident at the Fukushima Dai-ichi NPP and taking into account the more demanding requirements to meet the Design Extension Conditions (DEC) of the IAEA Requirements, setting high standards for safety in The Netherlands [10].

2.1 Implementation of an advanced defence-in-depth concept

The modern defence-in-depth concept as proposed by WENRA [3] is the basis of the new Dutch Safety Requirements [11]. It requires that during the design phase also accident conditions more severe than the traditional design basis accidents have to be taken into account. One advantage of the WENRA approach is the more clear and unambiguous terminology of the different plant states and the measures to be implemented to cope with such accidents. Table 1 provides an overview of the defence-in-depth concept implemented in the new DSR for nuclear reactors[1]. The new term “*postulated single initiating event*” replaces the traditional term “*design basis accident”*. A new group of events are the “*postulated multiple failure events”*, which includes more severe and complex accident scenarios than the postulated single initiating events, but still excluding accidents with severe fuel damage. Accidents with severe fuel degradation are called “*postulated core melt*

[1] The term “*nuclear reactor”* is used because the Dutch Safety Requirements shall be applied to both nuclear power plants and research reactors. Thus this term includes the whole facility not restrited to the reactor.

accidents" In this context, it has to be mentioned that accidents with severe fuel degradation in the spent fuel pool have to be practically eliminated. Following the IAEA approach the level 3 of defence-in-depth was split into level 3a to control postulated single initiating events and level 3b to control postulated multiple failure events. The reason for splitting level 3 was that for both plant states the same radiological objectives have to be met and aims for the prevention of severe accidents. A further argument is that in contrary to postulated core melt accidents no new physical phenomena are expected in case of postulated multiple failure events.

The implementation of the enhanced defence-in-depth concepts triggered the discussion on the understanding of the term "*design basis*". As the design encompasses all plant states including core melt accidents and the protection against hazards, the understanding of design basis needs to be changed. It is expected that each safety feature (e.g., safety systems, additional safety features, complementary safety features or SSCs used to protect the plant against hazards) will be designed in such a way that they can withstand the conditions of the plant state in case of demand as well as the impacts of the design basis events. A recently published IAEA TECDOC on "Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants" [12] discusses a similar approach to extend the design basis.

Table 1 Defence-in-depth concept implemented in the Dutch Safety Requirements [11] based on the concept proposed by WENRA [3].

Levels of defence-in-depth	Associated plant condition categories	Objective	Essential means	Radiological consequences
Level 1	Normal operation	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation, control of main plant parameters inside defined limits	Regulatory operating limits for discharge
Level 2	Anticipated operational occurrences	Control of abnormal operation and failures	Control and limiting systems and other surveillance features	
Level 3	Level 3.a Postulated single initiating events	Control of accident to limit radiological releases and prevent escalation to core melt conditions	Reactor protection system, safety systems, accident procedures	No off-site radiological impact or only minor radiological impact
	Level 3.b Postulated multiple failure events		Additional safety features, accident procedures	
Level 4	Postulated core melt accidents (short and long term)	Control of accidents with core melt to limit off-site releases	Complementary safety features to mitigate core melt, Management of accidents with core melt (severe accidents)	Limited protective measures in area and time
Level 5	-	Mitigation of radiological consequences of significant releases of radioactive material	Off-site emergency response Intervention levels	Off-site radiological impact necessitating protective measures

2.2 Implementation of a protection concept against internal and external hazards

It can be clearly distinguished between postulated initiating events (PIEs, e.g. pipe break, loss of off-site power, power excursion, etc.) and hazards (e.g. flooding, earthquake, fire, explosion, load drop, etc.). The consequences within the plant depend strongly on the kind of hazard and its severity. Thus, not every hazard might cause a PIE. The DSR require a protection concept against internal and external hazards. For natural hazards a design basis event with a frequency of 10^{-4} 1/a shall be used. For earthquakes, the design has to withstand at least a peak ground acceleration of 0.1 g. The protection concept has to ensure, that for hazards not exceeding the severity of the design basis events no redundant trains of safety systems will fail due to the impact of external hazards and in case of internal hazards only the affected train is allowed to fail.

In addition, combinations of hazards with other hazards or PIEs have to be considered if the combined events or hazards are causally related or if their simultaneous occurrence has to be assumed due to their probability and extent of damage.

To determine the loads due to hazards and to define the design basis accordingly, a hazard curve is required in the DSR to show the severity as function of the frequency.

The new Issue T of the WENRA Safety Reference Levels (RLs) published in 2014 [13] requires a systematic analysis of natural hazards exceeding the design basis events. In [13], particular RLs T6.1, T6.2 and T6.3 deal with hazards exceeding the design basis events. These three reference levels where implemented in such a way that an assessments of the effects of natural hazards exceeding the design basis events of the plant shall be performed. Therefore, analysis shall as far as practicable include:

- Determining the severity of the event at which fundamental safety functions cease to be available;
- Demonstration of sufficient margins to “cliff-edge effects”;
- Identification and assessment of the most resilient means for ensuring the fundamental safety functions;
- Consideration of events that could simultaneously challenge redundant or multiple SSCs, several units at multi-unit sites, site and regional infrastructure, external supplies and other countermeasures;
- On-site verification (typically by walk-down methods).

2.3 Including lessons learned from Fukushima

The development of the DSR started shortly after the accidents at the Fukushima Dai-ichi NPP site. Consequently, lessons learned from these accidents have been taken into account and were implemented. The implementation resulted in requirements for:

- Diverse ultimate heat sink;
- Accessibility and habitability of control room, supplementary control room and emergency control centre during or to conditions due to external events;
- Enhanced requirements for emergency power supply;
- Spent fuel storage pool within a containment.

2.4 Formulating safety requirements for research reactors

Intentionally, the safety requirements were formulated in a technological neutral manner but having light water moderated nuclear power plants in mind. However, these new requirements shall be applicable to research reactors as well. In addition, an Annex "Requirements for research reactors" was developed. This Annex includes specific requirements only applicable for research reactors and a structured method to grade requirements for nuclear power plants according to the specific hazard potential of research reactors. Central element is a structured method for grading safety requirements for NPPs. This method uses the fundamental safety functions to categorize the hazard potential of a research reactor.

No categorization is defined for the fundamental safety function "control of reactivity" because the nuclear chain reaction has to be controlled anytime. Three cooling categories and three risk categories are defined. The cooling categories consider the necessary cooling conditions to ensure heat removal from the fuel. The risk categories take into account the severity of radiological consequences due to a failure of the confinement function. To reflect the original hazard potential of the plant, the idea of an unprotected plant [14], i.e. a research reactor without additional safety features, is applied. For such a plant, a credible accident scenario needs to be defined to analyse the unmitigated consequences and allow for an appropriate categorization.

To provide further guidance for an applicant, a possible grading of the Dutch Safety Requirements is proposed for generic categories. Nevertheless, the responsibility to justify any grading rests with the applicant and requires a confirmation of the regulator. Experiences from pre-licensing discussions with applicants in the Netherlands showed that grading is a challenging process for the parties involved.

3 REVIEW PROCESS

Beside the DSR, a review process was developed to provide guidance for the experts dealing with the review and assessment of a SAR and the associated preparatory arrangements. The recommendations provided in the review plan shall contribute to an effective and efficient review process that is highly transparent and in which different applicants are treated equally.

Since the review process is available for (potential) applicants, the applicants have detailed insights on the regulatory expectations concerning content and scope of the SAR. The review plan also fosters a common understanding of the review process among the involved organisations and facilitates an early identification of challenges.

The first part of the review process covers organisational aspects that shall help to set adequate conditions in advance of the review phase (Organisation of the Review Process) whereas the second part focuses on the content of the technical review (Technical Review Plan).

3.1 Organisation of the review process

The report called' "Organisation of the Review Process" (ORP) describes a number of areas in which ANVS has identified a need for preparatory arrangements to ensure an efficient review phase. In the following examples for arrangements discussed in the ORP are given:

- Organisation-related aspects

In this area, the ORP highlights the importance and objective of pre-review meetings and gives recommendations on topics to be discussed such as success factors, role of the DSR and TRP, Request for Additional Information (RAI) process, establishment of time lines, responsibilities of the applicant, and implications due to the involvement of contractors. Another organizational arrangement presents the establishment of single point contacts in the involved organizations ("one-channel" communication) to ensure that responsible managers (i.e. within the applicant's and regulator's organisations) remain in control of the exchanged information.

- Document-related aspects

In this part, the ORP clarifies the documents contributing to the review process and their objectives/characteristics. One key aspect is the regulatory understanding of PSAR and FSAR in the respective licensing phase. Other documents such acceptance reviews statement, review and assessment report, request for additional information letter, technical response letter and safety evaluation report are also addressed in the ORP.

- Procedure-related aspects

In this area, the ORP specifies e.g. the management of the review and assessment including the development of work/project plans, necessary qualification and number of review experts, means for quality assurance, role and function of work package leaders, handling of crosscutting review issues, and traceable documentation of expert discussions. Also highlighted are implications of pre-licensing activities by the applicant and especially the adequate treatment of long-lead items without unnecessary risk of non-compliance.

- Communication-related aspects

In order to deal with insufficient performance of the applicant, the ORP suggests the implementation of an escalation scheme with three distinct escalation levels. Another topic concerns the handling of proprietary information with help of an adequate IT infrastructure. In this area, the ORP also describes the exchange of documents via an online documentation system.

3.2 Technical Review Plan

The Technical Review Plan (TRP) provides guidance for the experts dealing with the technical review and assessment of a SAR. Although the TRP was developed primarily for the review of a SAR of a nuclear power plant (standard LWR design), specific aspects of research reactors are also addressed. The overall objective of the TRP is to ensure that the review and assessment is performed in a thorough manner where all topics significant to safety are considered appropriately. In the following, the main three parts of the TRP are introduced briefly:

- Basic review recommendations

This part of the TRP provides recommendations on issues that need to be addressed before the start of the detailed technical review process. For instance, information on the performance of the acceptance review is provided. It further includes information on the application of the graded approach for research reactors as defined in the DSR. Another topic is the assessment of alternative acceptance criteria developed by the applicant in cases where technical criteria in the DSR (focus: standard light-water technology) are not applicable for the proposed (research) reactor design.

- Common review steps

In its core part, the TRP introduces a stepwise and systematic review approach to contribute to a technical review that is performed in a predictable and harmonized manner. The top-down scheme includes a set of common review steps in which the reviewer firstly assesses whether the applicant has addressed high-level safety aspects appropriately before moving gradually to the assessment of the technical implementation and evidences. Thus, each technical area is assessed strictly according to its contribution to the fulfilment of (fundamental) safety functions. The common review steps can be particularly applied to design-related issues. However, the underlying safety philosophy is also applicable for the review and assessment of non-design related issues. The common review steps are briefly introduced in the following:

a) **Scope:** In this first step, the reviewer assesses whether the applicant has defined the scope of the technical area and interfaces to related topics appropriately.
b) **Fundamental safety functions:** The reviewer then determines if the applicant has identified the fundamental safety functions that are affected by the technical area.
c) **Functional criteria:** Based on b) the reviewer assesses whether the applicant has described qualitatively the required performance (of the technical area) in order to contribute to the fulfilment of the affected fundamental safety functions.
d) **Safety requirements:** Subsequently, the reviewer verifies whether the applicant has identified all applicable (regulatory) criteria and requirements that are necessary to develop the safety demonstration for a particular technical area.
e) **Design basis:** The reviewer assesses if the functional criteria identified in c) have been translated appropriately into specific and measurable (design basis) requirements.
f) **Design description:** Here, the reviewer focusses on whether all relevant (design) information is given so that e) is fully implemented.
g) **Design evaluation:** In this part the reviewer assesses the evidences given by the applicant in order to support that the (design) measures described in f) are indeed capable of performing under design basis conditions and meeting their intended safety function.
h) **Instrumentation and signals:** In this step the reviewer focusses on whether for this technical area the provided information on instrumentation and signals is complete.
i) **Testing and inspection / monitoring:** In the last step, the planned measures for test and inspection as well as monitoring means are assessed.

- Specific review recommendations

While the common review steps inform the reviewers on the general review approach that can be applied to different technical issues, the TRP also provides review recommendation specifically for 22 review areas. These review areas correspond to 22 generic chapters expected in a complete SAR. For each review area, the TRP:

- defines the technical areas that are to be assessed,
- highlights interfaces to related technical areas,
- provides information on relevant regulatory requirements and IAEA standards or guides,
- specifies the information that is usually subject to review, and
- identifies issues specifically for research reactors.

Overall, the specific review recommendations together with the common review steps provide extensive guidance for the reviewer. However, the TRP is not to be confused with a review manual that eliminates expert’s judgement nor does it present a compilation of mandatory regulatory requirements. Instead, it promotes an acceptable review approach (common review steps) and a benchmark for the scope and depth of the expert’s review (specific review recommendations).

Although primarily developed for the reviewers, the TRP also helps (potential) applicants to derive scope and depth of information that is required in the SAR and to develop a robust and coherent safety argumentation.

4 SUMMARY

ANVS with support of GRS developed the new Dutch Safety Requirements (DSR) in which the most recent state of the art in science and technology with respect to nuclear safety is considered.

After a public consultation the DSR have been published as part of the "Guidelines on the Safe Design and Operation of Nuclear reactors" [11] in October 2015. The DSR serve as a guideline for the design of new reactors and are used as a reference for the regulatory assessment of nuclear safety (of existing reactors and new builds). Beside the DSR, a review plan has been developed to provide guidance for the experts dealing with the review and assessment of a SAR and the associated preparatory arrangements. In its core part, the review plan describes a stepwise review approach which, in a top-down scheme, helps to assess whether the information and safety argumentation provided in the SAR is sufficient. Although primarily developed for review experts, the review plan shall also help the applicant to understand the regulators expectations and developing a comprehensive SAR. Currently, first experiences of the application of the DSR and the review plan are gathered during the review of the SAR of the HOR[2] research reactor operated by the Delft University of Technology.

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Improving Control over Subcontracted Operations at Nuclear Power Plants in France

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Abstract:

The French licensee subcontracts approximately 80 % of the maintenance operations at its nuclear power plants. If not performed correctly, these activities may cause the failure of equipment important to safety, to the extent that more than 30 % of significant safety events reported by the licensee in recent years involve maintenance errors. Given the significant issues concerning the safety of these operations, the Nuclear Safety Authority ASN asked their technical safety organisation IRSN to examine measures taken by the licensee to control risks associated with subcontracted maintenance operations on reactors. The examination, which analyses the entire subcontracting management process, highlights the factors that ensure the licensee's control as well as certain organisational weaknesses that require correction. In this regard, the licensee must consider the overall quality of a service as the joint contribution of the project owner and the subcontractor.

1 BACKGROUND FOR IRSN'S ASSESSMENT PROCESS

Each year, the licensee turns to more than 22,000 subcontractor employees to perform maintenance on the reactors alongside their own 10,000 employees assigned to these tasks. The volume of subcontracted operations will increase in coming years due to the licensee's efforts to improve reactor safety to take into account lessons learned from the Fukushima Dai-ichi accidents in March 2011 and to extend the 40-years operating lifetime of reactors. If not performed correctly, these operations may cause the equipment failure with a possible impact on the nuclear safety. Indeed, operational experience indicates that more than 30 % of significant safety events reported by the licensee in recent years involve maintenance errors.

The licensee's decision to subcontract a significant share of the maintenance activities must be examined from the viewpoint of risk control. Operating experience feedback coming from industrial accidents shows that the relationship between project owner and subcontractor may have potentially destabilising effects for the respective organisations and with sometimes significant consequences for risk control. In addition, subcontracting in the nuclear industry is a frequent topic of discussion and controversy with regard to the safety of workers and reactors.

In this context, ASN requested IRSN to evaluate the measures implemented by the licensee to ensure control of risks associated with subcontracted maintenance activities at nuclear power plants. IRSN analysed in particular the following aspects:

- Ability of the project owner (the licensee) to make the subcontractor aware of safety issues and assess the subcontractor's response;
- How the subcontractor takes responsibility for safety issues and responds with appropriate technical actions;
- How the subcontractor communicates their operating experience to the licensee.

2 MEASURES IMPLEMENTED BY THE LICENSEE

Most maintenance operations take place during plant unit outages. These outages are required to replace spent fuel and to perform control and maintenance activities on parts of the facility that are not accessible during operation. Given the production requirements, the licensee seeks to make these outages as short as possible, however by guaranteeing optimum safety. This objective requires rigorous planning and preparation of operations and the involvement of qualified companies which have been selected on the basis of their expertise and the verified quality of their work. To meet all of these requirements, the licensee implemented general organisational measures at the national level which are applied at each nuclear power plant. These measures cover qualification of subcontractors and contractualization of the services to be carried out as well as evaluating these services and using insights and lessons learned from operating experience (OPEX) (Figure 1).

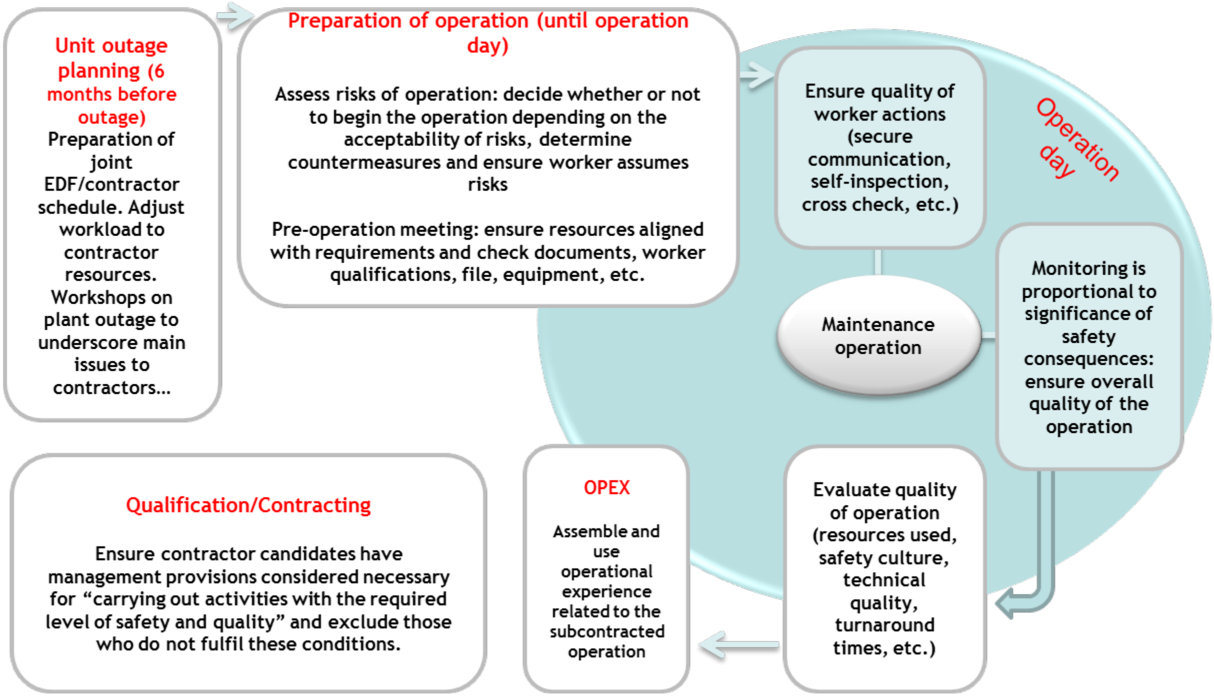

Figure 1: Organisational measures implemented by the licensee

3 IRSN'S ASSESSMENT APPROACH

IRSN's approach for assessing the organisational measures implemented by the licensee involved analysing the relationship between the licensee and their subcontractors and, in particular, how this relationship affects compliance with plant operational safety requirements.

In analysing the organisational measures used by the licensee to determine the conditions for subcontracting, IRSN has also evaluated the licensee's progress using a given number of measures that it has previously analysed (for example, risk assessments and monitoring of services).

3.1 Taking into consideration the requests of society at large

At the start of the assessment process, IRSN met members of local information commissions and environmental protection organisations. These meetings gave the opportunity to collect and specify the topics likely to have an impact on the safety issues of concern to stakeholders and which recur in public discussions, including the quality of monitoring the subcontractors by the project owner and the significant time constraints on workers.

3.2 Analysis of the licensee's organisational measures

First, IRSN examined the licensee's measures for carrying out maintenance operations by focusing on their actual effects on the activity of persons who had to apply them “in the field”. In particular, IRSN examined the adjustments and solutions adopted by workers when they encountered difficulties in applying these measures. To this end, IRSN visited three nuclear power plants during outages, conducted more than 160 interviews and observed in real time how some 40 maintenance operations were carried out. For each measure analysed, IRSN interviewed both employees of the licensee (project managers, monitoring managers, purchasers, etc.) and subcontractors. This mirror approach of the assessment process was useful for addressing both contributions of the licensee and their subcontractors in the overall control of risks related to subcontracted operations.

The objective of IRSN's analysis was to:

- Identify difficulties in the field (e.g. IRSN observed a subcontracted worker who encountered problems during an operation and who did not know the licensee’s employee to contact for help in order to tackle the problem);
- Demonstrating, by researching the deep organisational causes, a possible causal connection with the project owner/subcontractor relationship (in the example here above, one of the root causes identified was that a recent amendment has been made to the contract with the subcontractor involving the unplanned hiring of a worker who usually worked at another site).

Moreover, IRSN analysed the process for assembling and using operational experience (OPEX) from the subcontracted operation. The objective was to identify to what extent the licensee was able to reconsider its own organisation to improve the conditions for performing subcontracted operations, conditions which are indispensable for ensuring the safety of operations.

4 IRSN'S POINT OF VIEW

The strategy described above was used to examine the various stages in the subcontracting process. IRSN's assessment process showed that the licensee implemented a set of technical and organisational measures that make a solid contribution to the safety of subcontracted operations. Nevertheless, IRSN has found several areas for improvement, listed below, considered essential for ensuring the safety of these operations.

4.1 Ability of subcontractors to perform operations that impact safety

IRSN considers that the qualification and contracting stages are the occasion for the licensee to ensure in advance that subcontractors will be able to provide the management required for performing operations with the highest level of safety. In the field, IRSN observed that this approach was relevant but insufficient for the licensee to ensure the actual ability of subcontractors to implement appropriate management and to have sufficiently competent resources to carry out the maintenance operations assigned to them. To take into account IRSN's observations, the licensee undertook to study implementation of a “conditioned qualification”, which would be granted to a company on the single condition that it had already demonstrated its ability with work it performed during monitoring by the licensee.

4.2 Balance between workload and available resources

The ability of subcontractors to handle the workload is a basic condition for ensuring the quality of maintenance operations. Compliance with this condition requires the availability of the appropriate resources, both in quantity and competence. In view of this, the licensee recently implemented a number of measures so that subcontractors are more involved in planning operations for unit outages. On this point, the licensee has committed to assessing the actual effects of these measures in the field.

The licensee has also implemented compensatory measures (waiting times, last minute requests for service, etc.) to deal with contingencies (involving equipment or scheduling). IRSN has however observed that these measures may weaken subcontractor organisation and considers that the licensee must identify the potentially harmful effects of these measures on subcontractor working conditions in order to better anticipate these effects.

4.3 Risk assessment approach

A major measure in the control of risks that the licensee has implemented is the risk assessment performed prior to each operation. Before the operation, workers are expected to learn well this analysis and the methods provided for controlling risk. IRSN's assessment process confirmed the difficulties that the licensee has encountered for a number of years in producing analyses that take into account the risks actually encountered during maintenance activities. Aware of these difficulties, the licensee has committed to stepping up the current measures for applying the risk assessment approach to improve the quality of risk assessments and accurately measure their effectiveness in the field.

4.4 Licensee's monitoring of subcontracted operations

The monitoring by the licensee personnel of operations performed by subcontractors must contribute to avoiding certain deviations that may affect proper operation of safety-related equipment. Facing recurring difficulties related to monitoring of services provided (administrative rather than technical monitoring, problem with the legitimacy of those responsible for monitoring, etc.), the licensee implemented a new monitoring management policy in 2014. Reacting to IRSN's observations, the licensee is committed to clarifying the conditions for monitoring services, particularly for complying with regulations which require that the monitoring of a subcontracted safety-related operation cannot be subcontracted.

4.5 Collect and use operational experience from subcontracted work

The licensee has multiple channels for collecting and reporting information about subcontractors and services provided and using the operating experience (OPEX) of subcontracted operations. Nevertheless, during the assessment process, IRSN found weaknesses, both in the collection and handling of this information; this particularly concerns the difficulty for subcontractors to become involved in preparing OPEX and the lack of overall analysis by the licensee of all available data. In this regard, the licensee is committed to improving the process for assembling and using OPEX.

5 CONCLUSIONS

IRSN notes that certain weaknesses in the licensee's measures have been known for several years (in particular risk assessments and the monitoring of services) and believes that they continue in part due to an insufficiently thorough analysis of the causes. More generally, IRSN considers that the licensee's analysis of problems related to subcontracting is still too often limited to the direct causes and does not adequately consider the deeper causes, particularly those related to the licensee's own organisation. According to IRSN, the situation is linked to the licensee's tendency, when assessing the quality of a service, to overestimate the contribution of the subcontractor and underestimate its own contribution. From IRSN’s viewpoint, the licensee needs to adopt a vision that takes into account the overall quality of a service as the joint contribution of the project owner and the subcontractor. For IRSN, this change is a necessary condition if the licensee wants to have better control over subcontracted operations.

Challenges of Severe Accident Management at Ukrainian NPPs

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Abstract:

In June 2011, after the accidents at the Fukushima Dai-ichi nuclear power plants (NPPs), Ukraine joined the European Initiative on performing stress tests for NPPs in the EU member states and in neighbouring countries. In Ukraine, the stress tests were carried out following the methodology approved by the EC and ENSREG (Declaration of ENSREG, Annex 1 "EU Stress-Test Specifications", dated 13 May 2011 [1]).

Based on stress-test results, a package of measures has been developed aiming at severe accident management (e.g., SAMG, hydrogen removal system, containment filtered venting, etc.). Implementation of these measures showed a number of "open" questions (problem with validation of models, lack of experimental data, necessarily of in-depth investigation of several severe accident phenomena, etc.). For resolving these problems, a special investigation program (Program of Severe Accident Phenomena Investigation) has been developed and is being conducted.

This paper describes the main measures developed for severe accident management, main problems encountered in their development and implementation and proposed way for solving these problems.

1 GENERAL INFORMATION

There are currently 15 power units operated at four NPP sites in Ukraine with a total installed electrical power of 13,835 MW, which constitutes approx. 50 % of the total electrical power of all power plants in Ukraine.

There are three types of VVER units operating in Ukraine: VVER-440/213 (Rivne NPP Units 1 and 2), VVER-1000/320 (Zaporizhzhya NPP Units 1-6, South-Ukraine NPP Unit 3, Khmelnitsky NPP Units 1, 2 and Rivne NPP Units 3, 4) and VVER-1000/302 (South-Ukraine NPP Units 1, 2).

After the accident at Fukushima-1 NPP (Japan), the State Nuclear Inspectorate of Ukraine (SNRIU) Board approved an Action Plan for a targeted safety reassessment and further safety improvements of Ukrainian NPPs in the light of the Fukushima-1 accident and an Action Plan for a targeted safety reassessment. One of the actions defined in the Action Plans was a targeted safety reassessment of operating nuclear facilities at NPP sites (stress tests).

In the framework of the stress tests, the operators analysed in detail:

- Extreme external natural hazards (earthquakes, flooding, external fires, tornadoes, extremely high/low temperatures, extreme precipitations, strong winds, combinations of events);
- Loss of electrical power and/or loss of ultimate heat sink;
- Severe accident management.

At the operating nuclear power plants, the stress tests focused on nuclear fuel in the reactor cores, spent fuel pools, fresh fuel rooms and the dry spent nuclear fuel storage facility (Zaporizhzhya NPP).

2 MEASURES FOR SEVERE ACCIDENT MANAGEMENT FOR UKRAINIAN NPPs

Based on the completed stress test analyses, a number of recommendations for decreasing vulnerability of the Ukrainian NPPs were developed. These recommendations were considered and reflected in the Comprehensive Safety Improvement Program.

The following main measures related to severe accident management were evaluated and included in the Comprehensive Safety Improvement Program:

- Performance of severe accident analysis. SAMG development;
- Prevention of early containment bypass due to ejection of molten corium into containment;
- Implementation of containment hydrogen removal system;
- Implementation of containment venting;
- Analysis of implementation of the In-vessel Melt Retention Strategy.

3 STATUS OF IMPLEMENTATION OF MEASURES FOR SEVERE ACCIDENT MANAGEMENT

3.1 SAMG development

3.1.1 SAMG description

The development of SAMGs for pilot power units of Ukraine (VVER-1000/302, VVER-1000/302 and VVER-440/213) has been completed. These SAMGs passed state review and were agreed on with the SNRIU.

SAMG documentation package contains:

- SAMGs;
- Analytical justifications;
- Technical justifications;
- Validation and Verification Report.

The symptom-oriented approach was used for SAMG development. Analytical justifications analyses were conducted using MELCOR and RELAP5/SCDAP computer codes [2]-[4]. Mainly, best-estimate approach was used in analytical justification of severe accident management strategies. These analyses covered all main initiating events (IEs):

- RCS (reactor coolant systems) breaks;
- Primary to secondary breaks;
- Loss of power;
- Loss of feedwater;

Analyses covered all plant operation states. Under severe accident strategies development and justification, the special attention was paid to estimation of positives/negatives effects of SAM strategies implementation. The status of NPPs was assumed in the analyses (i.e. “post-Fukushima” measures were not accounted).

3.1.2 Review findings

The following main aspects were reflected in the regulatory review of SAMG for Ukrainian NPPs [5]-[7]:

- Cross-verification of MELCOR and RELAP/SCDAP models performed for station black-out (SBO) scenarios showed large difference in results (e.g., time of core damage, core relocation and reactor vessel failure).
- Implementation of water injection into reactor strategy cannot prevent VVER-440 vessel failure if water injection started after initiation of core damage. For VVER-1000, vessel failure can be avoided if water injection started any time before melt hit core lower plate.
- Reactor vessel failure and corium injection into containment may lead in most cases to containment failure due to the specifics of VVER-1000 and VVER-440/213 containments designs.
- High hydrogen concentration in containment is possible even if a hydrogen removal system (passive autocatalytic recombines) is implemented. This is result of reduction of oxygen concentration in containment due to hydrogen recombination.

3.1.3 Follow-up activities for SAMG development

Based on results of the regulatory review, it was mentioned that SAMG were developed with number of assumptions and simplifications. Further activities should be conducted for their improvement. The following special investigations were proposed by SSTC NRS [5]-[7]:

- Comparative validation of severe accident models for used codes;
- Corium formation and relocation in the vessel lower head;
- Corium stratification and vessel thermal loading (focusing effect issue);
- Fuel-coolant Interaction: leading to core melt fragmentation upon contact with water, steam production, dynamic loading of structures in case of steam explosion;
- Re-criticality in the melted core;
- Corium spreading and cooling in ex-vessel phase;
- Extra measures for post-accident management hydrogen.

3.2 Containment filtered venting

3.2.1 FCVS general information

At present, the technical concepts for containment filtering venting have been developed for each type of Ukrainian NPPs. Many justification activities have been performed for development of the technical solution. The main goals of these activities were to:

- estimate the possibility and conditions for pressure reaching the maximum design limit,
- assess the possibility of containment failure prevention using containment venting system,
- select the FCVS (filtered containment venting system) type possible for installation at NPPs, and to
- preliminary evaluate the FCVS efficiency.

The application of different filter types was analysed for Ukrainian NPPs. In these analyses the sand filters, Venturi scrubber and dry filters were discussed. Based on these analyses, the application of a dry filter and Venturi scrubber was selected as possible for Ukrainian NPPs.

According to stress test results and SAMG justifications, an in-depth analysis of the need for containment venting and justification of its implementation have been performed. All main types of Ukrainian reactors were covered. There are VVER-1000/320 (reference plant was ZNPP-1), VVER-1000/302 (SUNPP-1) and VVER-440/213 (RNPP-1). Analytical justifications of FCVS were performed using the MELCOR computer code and specific models for reference units. Those models were based on severe accident models created for SAMG development and justifications.

The following accidents were considered under FCVS for VVER-1000:

- large break loss-of-coolant accident with loss of power supply, and
- station black-out.

Results of severe accident calculations (see Fig. 1 and Fig. 2) showed the possibility of containment failure due to reaching the containment pressure limit (5 kgf/cm^2 (abs.)).

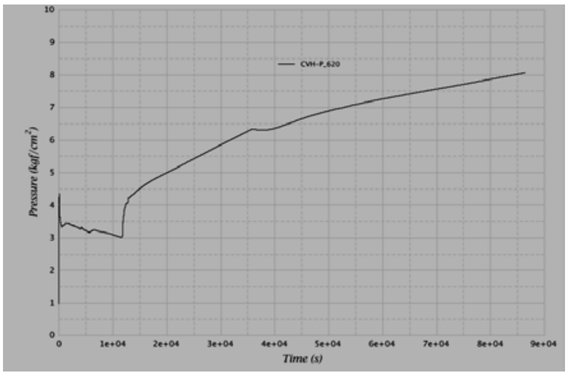

Fig. 1. Pressure in VVER-1000/320 containment under LB LOCA with loss of power supply

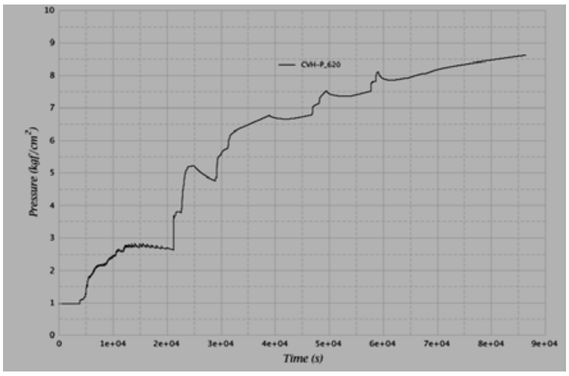

Fig. 2 Pressure in VVER-1000/320 containment under station blackout

Results from evaluation of the minimal diameter for containment venting system showed that implementation of FCVS with dump pipelines of less than 100 mm did not prevent increase in containment pressure. Implementation of FCVS with dump pipelines with a diameter of more than 100 mm prevents increase in containment pressure and prevents containment failure due to overpressure (see Fig. 3 and Fig. 4).

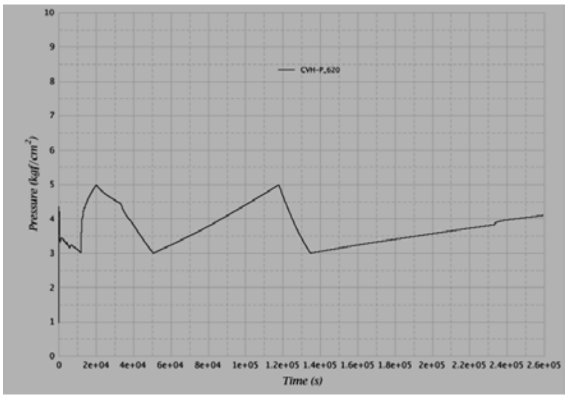

Fig. 3. FCVS application for VVER-1000/320 containment under LB LOCA with loss of power supply

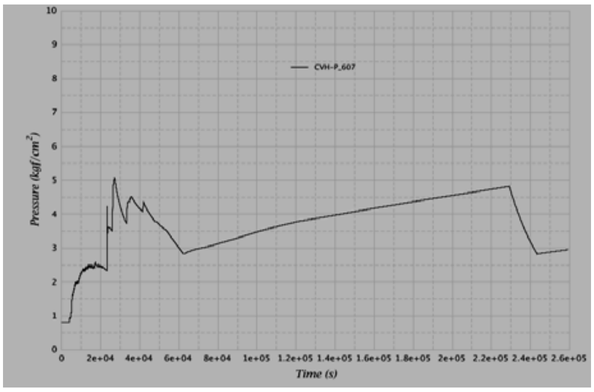

Fig. 4 Pressure in VVER-1000/320 containment under station blackout

According to results of analytical justification of FCVS for VVER-1000, preliminary requirements have been developed.

Results of the best-estimate analyses of severe accident progression for VVER-440 (see Fig. 5) did not confirm the possibility of containment pressure increase above the maximum design limit (2.5 kgf/cm^2 (abs.)).

Only a number of conservative assumptions can lead to exceeding the design limit for VVER-440 containment (see Fig. 6), such as:

- conservative assumption about cavity concrete content,
- failure to take into account containment leakage (more than 16% of initial mass per day), or
- consideration of possible water presence in cavity before reactor vessel failure.

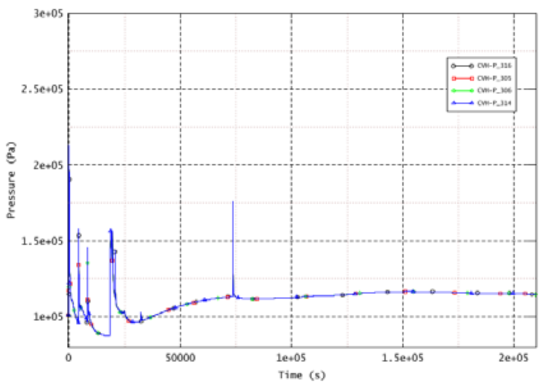

Fig. 5. Pressure in VVER-440 containment under LB LOCA with loss of power supply

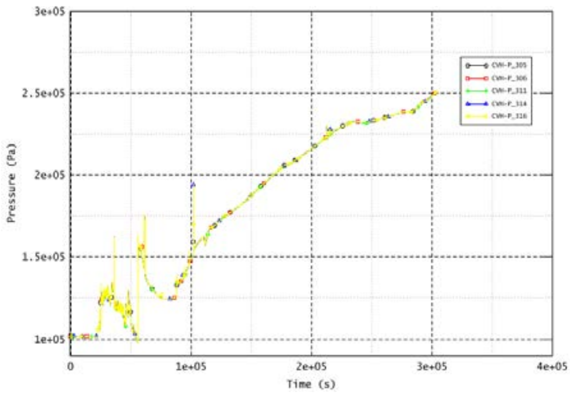

Fig. 6. Pressure in VVER-440 containment under LB LOCA with loss of power supply (conservative analysis

Based on results of analytical justification, the conceptual technical solution has been developed for VVER-440 containment filtered venting. This solution foresees implementation of the filtered containment venting system based on existing exhaust ventilation system.

3.2.2 Findings of the regulatory review

Results of analytical justifications of the FCVS for three reference units have been verified in the framework of the regulatory review. The main results of the review were related to analysis of calculations and technical justifications for correctness and review of technical solutions for validity (including an operation algorithm).

Under review of calculations and technical justifications for correctness, additional calculations (benchmarks) were conducted by SSTC NRS experts. The aims of these benchmarks were to check the correctness of justification results and confirm the validity of SSTC NRS expert comments and suggestions for the reviewed documents.

Analysis of the technical solutions is needed for confirmation than the implemented measure will solve the safety problem and will be the best solution for the selected NPP.

The set point of FCVS venting stop for VVER-1000 must be updated or additionally justified. The existing value (3 kgf/cm^2) can lead to deep containment vacuum. Besides, it is necessary to take into account that higher set point will lead to decrease of FCVS operation and, as a result, to lower radioactive release to the environment. SSTC NRS experts proposed to foresee additional investigation and justifications at FCVS implementation stage.

The review of the results for VVER-1000 showed overestimation of the aerosol mass and energy deposited on venting system filters. Benchmark calculations confirmed that these values were overestimated by more than 30 %. The utility checked the decay heat model, recalculated all analyses and conducted a number of sensitivity analysis. According to new results, maximum aerosol heat power and maximum aerosol mass deposited on filters were updated.

The review of VVER-1000 results showed that FCVS justifications were performed without taking into account passive autocatalytic recombiners (PARs). PAR operation leads to generation of additional heat (due to hydrogen recombination), resulting in decrease of FCVS efficiency. In addition, PAR accounting leads to earlier FCVS set point reaching, increase of containment temperature, decrease (below the PAR operation limit) of oxygen concentration in the containment atmosphere. SSTC NRS experts conducted benchmark calculations with special attention on above-mentioned assumptions and confirmed them.

For VVER-440/213 SSTC NRS experts mentioned the developed conceptual technical solution is based on international experience (first of all, on the solution for Kozloduy NPP), but it did not take into account VVER-440/213 features, containment design and high containment leakage (more than 16 % of initial mass per day). The majority of expert comments were related to this aspect. SSTC NRS experts proposed to investigate the possibility of FCVS use to reduce radioactive release. The benchmark analysis was conducted to investigate the efficiency of FCVS activation at early phases of severe accidents (e.g. immediately after core damage). Results of this benchmark analysis showed reduction in radioactive release in comparison to the case without FCVS activation. It was recommended to carry out in-depth analysis for this aspect in analytical justification. SSTC NRS conducted benchmark with FCVS modelling at outlet of air traps. In this case, additional radionuclide scrubbing through water on bubble-condenser trays is possible. The results of benchmark confirmed the expert assumption. The total radioactive release was decreased due to additional radionuclides scrubbing on bubble-condenser trays and due to dumping through FCVS quite “clear” atmosphere from air traps.

It is necessary to note that three main “post-Fukushima” modifications are planned for Ukrainian VVER-440. They include in-vessel retention modification, PAR installation and FCVS implementation. According to SSTC NRS opinion, coupled analyses should be performed for these modifications due to their interference.

3.2.3 Follow up activities for FCVS implementation

The results of regulatory review were implemented by the Utility in FCVS justification documents for VVER-1000. Additional investigation considering PARs was performed. The results of these calculations were taken into account under FCVS implementation at units with VVER-1000. Venting size increased to 125 mm (previous 100 mm). The first stage of FCVS is implemented for several VVER-1000 units.

At present, the FCVS conceptual technical solution for VVER-440 and analytical justification is being updated by the Utility.

3.3 Measures to reduce hydrogen concentration in containment during beyond design basis accidents

3.3.1 Measure description

The original designs of operating Ukrainian NPPs with VVER (VVER-1000/302, 338, 320, VVER-440/213) did not incorporate safety systems intended to reduce the concentration of hydrogen generated during accidents. Therefore, the development and implementation of measures to reduce hydrogen concentration in the containment during beyond design basis accidents (BDBA) were envisaged within the Comprehensive (Integrated) Safety Improvement Program for Ukrainian NPPs:

Implementation of these measures includes:

- Development of a conceptual technical decision to define the principal approach to ensuring hydrogen explosion safety during BDBA;
- Identification of technical characteristics of equipment (development of technical specifications) to be used to implement the accepted concept;

- Performance of required justifying calculations to confirm the adequacy of decisions (within development of the technical decision for mounting);
- Installation and operation of equipment.

The conceptual technical decisions for installation of passive autocatalytic recombiners (PARs) have been developed and agreed with the SNRIU for all NPP units.

During implementation of measures at Ukrainian NPPs, calculations were performed to justify the selection of representative accident scenarios accompanied by generation of a considerable amount of hydrogen. This also included preliminary assessment of the number and places of PAR installation at VVER-1000 of small series (302, 338) and VVER-1000/320. The preliminary safety analysis reports for implementation of measures have been developed.

At present PARs for BDBA have been installed on SUNPP-1,2 (RVK PAR) and ZNPP-1,2 (Westinghouse PAR).

It should be noted that PARS for hydrogen generated during design-basis accidents, were earlier implemented at KhNPP-2, RNPP-4 (ALSTOM PAR) and RNPP-1,2 (FRAMATOM).

3.3.2 Findings of the regulatory review

In framework of regulatory review on justifications of hydrogen removal system, several aspects that influenced PAR implementation were found. These are:

- Limited scope of investigations of spray system operation affects the hydrogen concentration in the containment. Start of spray operation at late phases of severe accident may lead to decrease steam concentration and, as results, to increase H_2 concentration in containment.
- Accounting of both sources (reactor and SFP) for VVER-1000;
- Accounting of specific concrete content for each NPP;
- Limited capabilities of MELCOR 1.8.5 in SFP modelling;
- Limitation of MELCOR 1.8.5 in modelling of corium spreading.

In addition, it is necessary to note that extra measures for post-accident management hydrogen are needed due to possibility of high hydrogen concentrations in containment at late phases of severe accidents even with PAR installation (see Figs. 7 and 8).

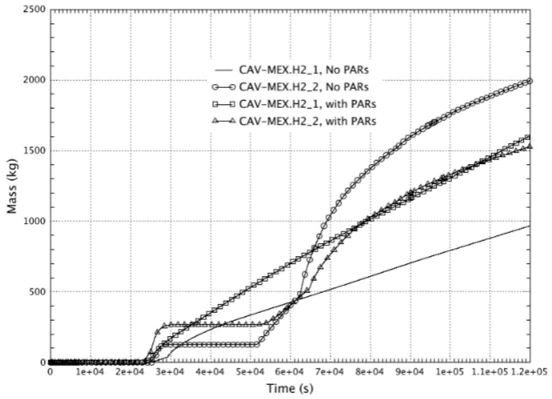

Fig. 7. Hydrogen mass generated in ex-vessel phase

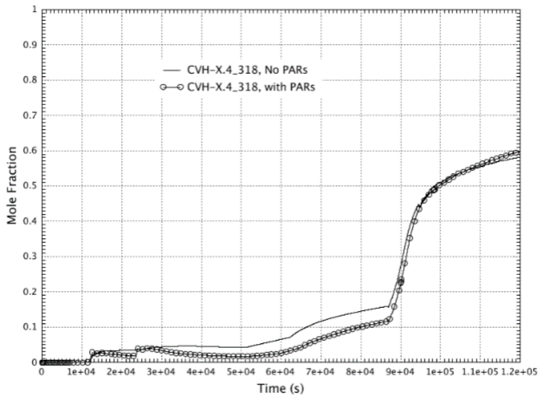

Fig. 8. Hydrogen concentration in containment under station blackout

3.3.3 Follow up activities for PAR implementation

The following issues should be investigated in detail:

- Hydrogen removal solution for late phase of severe accidents;
- Investigation of influence of specific concrete content on hydrogen generation in ex-vessel phase.

4 CONCLUSIONS

As mentioned above, the implementation of "post-Fukushima" measures (e.g., SAMG development, PAR and FCVS implementation) and their approval with the SNRIU, a number of assumptions and simplifications were made since Ukraine capabilities are to be enhanced to decrease uncertainties in modelling of severe accident phenomena. To resolve this issue, the utility developed the "Program of Activities on Analysis of Severe Accident Phenomena". This program is being implemented. The following activities will be conducted according to the program:

- Update of the existing version of computer codes or buy new codes;
- In-depth study of the selected phenomena (e.g., in-vessel and ex-vessel debris cooling, hydrogen generation and distribution during severe accident progression, in-vessel molten corium retention, possibility of criticality in core and/or in the spent fuel pool, etc.);
- Preparation of recommendations for modelling of selected phenomena;
- Improvement of the existing computer models for severe accident analysis (including their validation);
- Tutorial of NPP staff for new codes and new approaches for severe accident investigations.

5 REFERENCES

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[3] South-Ukrainian NPP. Unit 1. Analytical justifications for severe accident management strategies. 2012.

[4] Rivne NPP. Unit 1. Analytical justifications for severe accident management strategies. 2012

[5] State review of report "Zaporizhye NPP. Unit 1. Analytical justifications for severe accident management strategies". 2012.

[6] State review of report "South-Ukrainian NPP. Unit 1. Analytical justifications for severe accident management strategies". 2012.

[7] State review of report "Rivne NPP. Unit 1. Analytical justifications for severe accident management strategies". 2012

A New Method Taking into Account Physical Phenomena Related to Fuel Behaviour during LOCA

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Abstract:

The loss of coolant accident (LOCA) safety case was reviewed in France during a rulemaking process which started in 2008 and was finalized in April 2014.

In this context, EDF developed a new method to study LOCA (so-called CathSBI) taking into account physical phenomena related to fuel behaviour occurring during the transient: the fuel behaviour modelling is more accurate and improved.

The aim of this method is to calculate the peak cladding temperature (PCT) and the maximal cladding oxidation in order to compare them to the acceptance safety criteria, using CATHARE 2 system code and a statistical approach. The use of a statistical approach is a major change, in comparison to the current deterministic method. The first application is scheduled for the fourth decennial outage review of EDF's 900 MW_e nuclear reactors. The safety assessment of this new method has been made by IRSN in 2015.

The first results obtained by this new method show a significant sensitivity of the cladding temperature to the input parameters, appearing from 750 to 800 °C: this sensitivity is linked to the fuel rod phenomena (ballooning, burst, blockage, contact between rods and fuel relocation) activation, when the difference between cladding internal and external pressures is high enough and for cladding temperature levels above 750 to 800 °C : their cumulative effects lead in studies with CathSBI method to a significant increase of the cladding temperature (350 to 500 °C). Given this high sensitivity to input parameters, ensuring the robustness of LOCA safety studies based on this in progress method is still a challenge for EDF.

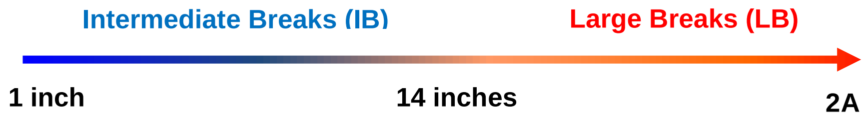
1 BACKGROUND

In the design of nuclear power plants, it is required that accident conditions (so-called design basis accidents, DBA) are considered and that the consequences of such occurrences are analyzed so that suitable mitigating systems can be designed. One of the DBA for pressurized water reactors (PWR) is the LOCA caused by a postulated coolant pipe break. More specifically, the initiating event is the double-ended guillotine break of one of the large coolant pipes between the reactor vessel and the main circulation pump.

A primary break would lead to a sudden depressurization and a loss of water inventory of the primary circuit. The fast pressure drop leads to large horizontal and vertical hydraulic loads on the internal structures of the reactor vessel and on the fuel assemblies. In particular, depending on the break location with regard to the vessel, the fuel assemblies can be subjected to horizontal movements, which may cause some fuel assemblies grids to impact the baffle and lead to their crushing. Also, due to the effect of vertical forces, the axial hold-on of the fuel assemblies can be jeopardized. Nevertheless, these components must retain their geometries to ensure the shutdown of the reactor by control rod cluster drop and to maintain core coolability. The loss of water at the break can cause a partial or total uncovering of the core which can lead to damage fuel rods up to the burst of a number of them. The discharge of steam into the reactor containment causes a large increase of its pressure and temperature.

In order to mitigate the consequences of this DBA, it is necessary to design the emergency core cooling system (ECCS) in order to ensure an efficient fuel cooling during all phases of the DBA. This safety principle naturally led to the requirement that the core must remain amenable to cooling through the whole LOCA sequence up to the long term. Thus, LOCA is taken into account for the mechanical design of the internal structures of the reactor vessel and the fuel assemblies. The LOCA transient effects are also taken into account for the containment design and for ECCS. Finally, the LOCA studies lead also to define the maximal lineic power of the core during operation.

The progression and consequences of a LOCA transient in terms of hydraulics and fuel behaviour are directly related to the location and the size of the postulated break on the reactor coolant system. As shown in Diagram, the spectrum of the potential breaks sizes extends from Intermediate Breaks (IB) to the “double-ended guillotine break” of one of the large coolant pipes between the reactor vessel and the main primary components (circulation pump, steam generator). The partition between IB and LB transients is conventionally made for 14 inches. The “double-ended guillotine break” is also called 2A break, with A being the flow area of the pipe. IB LOCA transients are slower than LB LOCA transients and lead to a primary pressure reduction that depends on the break size. Although the total duration of a LB LOCA transient does not exceed five minutes and the primary pressure drops very quickly to a fraction of MPa, the transient can last for about twenty minutes for smaller breaks and the core can get uncovered while the primary pressure remains at several MPa. During an IB LOCA transient, a more or less deep uncovering of the core intervenes in two phase liquid/steam flow, while for a LB LOCA transient, the uncovering of the core is complete and occurs in single phase steam flow. With regard to core coolability, and for a given break size, cold leg breaks are the most penalizing due to the assumptions that all ECCS water injected into the cold leg of the broken loop is lost at the break and does not therefore contribute to core cooling.



In France, since the start of the PWR program, breaks up to the 2A break (Fig 1) have been postulated to analyze some of the consequences of a LOCA transient such as core coolability, resistance of the reactor containment and radiological consequences. However for other consequences, such as the mechanical resistance of the reactor vessel internal structures and the fuel assemblies, only limited break sizes considering pipe whip restraints have been considered which are located at specific points along the primary loop.

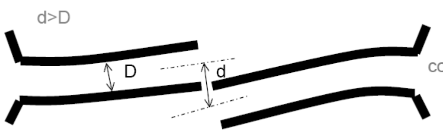


Fig 1: 2A Break

The U.S. NRC regulation [1] had been adopted in France at the start of the French PWR nuclear program and notably the clad oxidation rate (which must remain lower than 17 %) and the clad temperature (which must remain lower than 1204 °C) criteria. Compared to the situation forty years ago the discharge burn-up of the fuel rods has increased considerably. This has led to increased oxide thickness and higher hydrogen uptake in the fuel cladding, which influences its behaviour under LOCA conditions. Since then, the behaviour of fuel in LOCA conditions has been the subject of research and development (R&D) programs. The vast majority of research and development activities on fuel behaviour in LOCA conditions were focused on situations representative of LB LOCA scenarios. Thus, recent research findings have identified new phenomena under LOCA conditions with increasing burn-up, such as in particular the embrittlement mechanisms of fuel rods due to oxygen and hydrogen pickup [2] and the Fuel Fragmentation, Relocation and Dispersal (FFRD) [3], [4], [5]. Results of numerous experiments are also used to develop and improve the predictive models of software simulations of LOCA transients. These new physical phenomena related to fuel highlighted by R&D studies conducted since a few ten years have to be taken into account in the frame of safety demonstration.

Currently, U.S. NRC is proposing to revise the LOCA requirements and criteria for the ECCS design [7]. The proposed ECCS acceptance criteria are performance-based, and reflect recent research findings. The recently proposed rule replaces the current prescriptive ECCS acceptance criteria with a performance-based requirement to demonstrate adequate post-quench cladding ductility and adequate core coolant flow area to ensure that the core remains amenable to cooling.

Moreover, the operating conditions of the French plants have evolved (notably by stretch operation conditions) and new cladding materials have been introduced.

Because of these evolutions, the French Nuclear Safety Authority (ASN) has decided to review the LOCA safety demonstration concerning core coolability encompassing the following three main subjects:

(1) Definition of the LOCA reference transients;
(2) Physical phenomena to be taken into account and LOCA safety requirements associated with safety limits to be verified;
(3) LOCA analysis methods.

The LOCA safety case was reviewed during a rulemaking which started in 2008 and was finalized in 2014 when the acceptability of French utility EDF proposals were assessed by IRSN and reviewed by the Advisory Committee for Reactors for the Nuclear Safety Authority during two meetings in 2010 [8] and 2014 [9].

In 2015 and 2016, the new LOCA analysis CathSBI method proposed by EDF was reviewed by IRSN. This new method will be first applied for the fourth 10-yearly safety review of EDF’s 900 MW_e nuclear reactors starting in 2017, then for the next 10-yearly safety review of EDF’s 1300 and 1450 MW_e nuclear reactors. The objective of this paper is to discuss the new LOCA analysis method.

2 DEVELOPMENT OF LOCA REFERENCE TRANSIENTS

The development of LOCA reference transients is well described in the paper [18].

In the current French LOCA safety demonstration, EDF takes into account the same break sizes limited by pipe whip restraints for both thermal-hydraulic and mechanical analysis (maximum break size is below 28 inches and depends on plants design).

Regarding the thermal-hydraulic analysis, it is important to emphasize that such an evolution of LOCA reference transients leads to focus on a better modelling of the physical phenomena for IB LOCA conditions rather than focus only on the 2A break. This will motivate development of more appropriate methods to calculate the dominant physical phenomena during this type of transients.

Regarding the mechanical analysis, IRSN considered that EDF identified the key mechanisms associated with the irradiation effects on the behaviour of the vessel internal structures and fuel assemblies. Thus, EDF will have to take the irradiation effects into account in the future studies which will be performed starting with for the fourth 10-yearly safety review of EDF’s 900 MW_e nuclear reactors.

3 DEVELOPMENT OF LOCA ANALYSIS METHOD

For the upcoming 10-yearly reviews of EDF’s reactors, starting with the fourth reviews of 900 MWe reactors, EDF’s new CathSBI method for IB LOCA studies is still based on the use of CATHARE software associated with 1D fuel behaviour modelling but with multidimensional thermal-hydraulics modelling of the vessel. Regarding the new modelling of the vessel, the use of CATHARE 3D module is needed in the core and the downcomer to simulate thermal hydraulics 3D phenomena. In particular, the model of the cross-flows in the gas phase during high-pressure core uncovery takes into account the “chimney effect”. which has a direct positive influence on hot rod cooling. Moreover, the cold water injected by ECCS is able to go down in the boiling downcomer water and to reach the core. In contrary, with 1D module, cold ECCS water was “floating” above high void fraction mixture and was lost at the break. IRSN analysis was focused on CATHARE modelling qualification. This new modelling of the vessel leads to a significant beneficial effect on peak cladding temperatures. However, a lack of validation of the CATHARE 3D module was identified by IRSN. Consequently, justifications of modelling choices are still expected and experimental programs are ongoing to validate the CATHARE 3D module.

Another major evolution consists in a statistical approach as opposed to the current realistic method taking into account uncertainties with a deterministic approach. This method is based on taking into account the elementary uncertainties affecting the key parameters in the calculation of interest parameters and focuses on the impact of the relevant phenomena related to a particular scenario. Then the analysis of physical models and the equations of the code lead to select a list of potentially important parameters divided into three groups:

- ✓ Reactor initial and boundary conditions characterized by a quantified uncertainty;
- ✓ Code models and correlations characterized by measured and calculated uncertainties;
- ✓ Scenario parameters (such as break size, assembly burn-up), the range of variation of which are known but the penalizing values are not known a priori.

The new statistical approach takes into account coupled effects between key parameters (due to uncertainties propagation). After the advisory committee meeting in 2014 [9], based on IRSN analysis, ASN asked EDF to ensure the conservatism of the safety IB LOCA studies according to the statistical method in dealing with the most influential uncertainties (for example residual power and rod internal pressure) in a deterministic way or by an approach to define a penalizing range of variation (by range reducing). IRSN analysis was focused on the validity of this statistical approach proposed by EDF. After this review, IRSN considered that the statistical approach still needed more robustness and that EDF had not entirely fulfilled ASN’s requirements about the treatment of the most influential uncertainties. Moreover, IRSN pointed that some elementary uncertainties still needed to be justified.

The last, but not the least important, enhancement consists in a fuel behaviour modelling taking into account clad ballooning and burst, blockage, contacts between neighboring rods and fuel relocation. These points are developed in the following parts.

4 DEVELOPMENT OF FUEL MODELLING BEHAVIOUR

During a LOCA transient, the pressure decrease and the temperature increase in the primary circuit can lead to large inelastic deformation and eventually burst of the cladding due to the stress induced by the difference between internal and external pressures and temperature levels around 750 °C – 800 °C [10], [11]. Moreover, when cladding temperature reaches about 800 – 900 °C, cladding oxidation reaction speeds up and the transient oxide growth becomes significant [12], [11]. For the ballooned and burst fuel rods, a significant amount of hydrogen produced during transient oxidation is absorbed at the inner side of the fuel rods cladding [11]. This phenomenon is called transient secondary hydriding [2], [13].

4.1 Physical Phenomena during LOCA

Although the physical phenomena in IB LOCA and LB LOCA conditions are similar in nature, the fuel rods behaviour under IB LOCA transients is specific:

- ✓ Heating rates are lower, less than 10 °C/s, compared with LB LOCA transients around 30 °C/s. Also, the cooling kinetics are slower between 1 to 10°C/s compared with LB LOCA transients kinetics of 10 to 100 °C/s;
- ✓ Transient oxidation operates under high pressure between 20 and 80 bars, whereas for LB LOCA transients, the transient oxidation occurs at few bars.

During a LOCA, the fuel can be damaged according to two modes. During the core uncovering, the ductile mode deals with the phenomena of fuel rods ballooning and burst associated with fuel relocation which can lead to a partial blockage of the fuel thermal-hydraulic channels. Under a number of conditions (extended axial blockage, fluid velocity, …), this blocked geometry can jeopardize core cooling capability by reducing the heat exchange surface between the fuel rods and the coolant and by redistributing coolant flows [14].

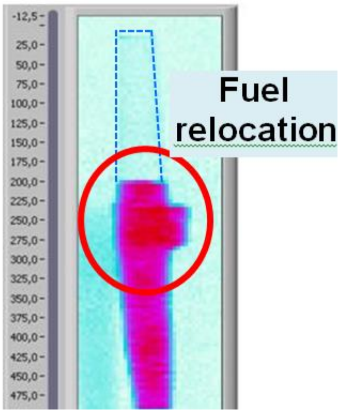


Fig.2: HALDEN test IFA-650.4 - 2006 OECD Halden Project

Some recent R&D programs conducted by ANL (Argonne National Laboratory), Studsvik and HALDEN (Fig 2) showed an accumulation of fuel fragments in the area of the ballooned and burst fuel rods. The fuel relocation can significantly modify the local heat generated in the fuel rods and tends to increase locally their temperature. Moreover, for the highly irradiated fuel, a dissemination of fuel particles outside the fuel rod was observed after its burst. This phenomenon is called fuel dispersal. The conditions required for the fuel dispersal occurrence have been studied experimentally and have previously been discussed elsewhere [3], [4], [5] but some of them are summarized here [15].

During the reflooding, the brittle mode deals with the high temperature cladding oxidation in steam environment including transient hydrogen pick-up which can lead to a loss of the fuel rods strength. Under the effect of the quench, the application of stresses on weakened fuel rods may lead to their rupture, jeopardizing core cooling capability. The thermal stresses due to the quench may be added with additional mechanical loads. The origins of such loads are discussed in detail in [16]. In addition, the possible effect of additional mechanical loads occurring after the LOCA transient, such as seismic forces, must be assessed.

For illustrative purposes, the following Fig 3 depicts the specificities related to IB and LB LOCA transients.

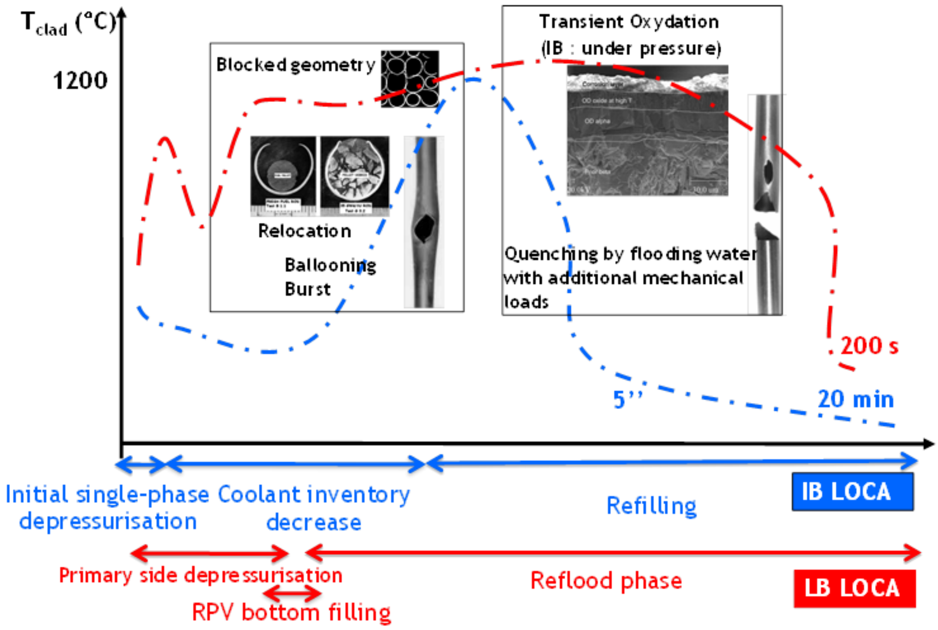


Fig 3: An illustrative example of fuel rods behaviour under IB LOCA (5 inches break size) and LB LOCA

4.2 The New French Regulation

The Safety Analysis Report of French PWRs specifies, for each design transient, the safety principles to be observed and their transcriptions into safety requirements in order to avoid unwanted physical phenomena to happen. Compliance with these requirements is followed by verification of acceptance criteria, which are computable parameters representative at best of the relevant physical phenomena.

At the start of the French PWR nuclear program, the fuel acceptance criteria established by the AEC (now the U.S. NRC) in the 1970s were adopted in France, based on the state of knowledge on fresh fuel on the basis of post-quench ductility tests, especially ring compression tests (RCT). The five well-known fuel acceptance criteria are currently specified in the 10CFR50.46 [1]. The two acceptance criteria connected with the brittle mode are recalled below:

(1) <u>Peak cladding temperature [PCT]</u>. The calculated maximum fuel element cladding temperature shall not exceed 2200°F [1204 °C]

(2) <u>Maximum cladding oxidation [ECR]</u>. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation […]

It is important to underline that for the ductile mode, the requirement of maintaining a coolable geometry has not been formally expressed in a decoupling criterion. In the current 10CFR50.46, the fourth criterion is worded thus:

(4) <u>Coolable geometry</u>. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

In 1994, EDF chose to calculate ECR taking into account the in service corrosion and the transient corrosion. In the USA, such a practice took place as from 2001 in compliance with the "NRC Information Notice 98-29" while the criteria of 10 CFR 50.46 have been formally amended. In 1999, the original requirement of a residual cladding ductility at the end of transient, based on ring compression test (RCT), changed into a requirement of fuel rod quenching resistance without additional load, based on leakage tests carried out on cladding sections oxidized at high temperature and having undergone a thermal shock quench (DEZIROX facility [11]). This change did not call the original criteria in PCT and ECR into question.

The Advisory Committee for Nuclear Reactors in 2010 about the French LOCA rulemaking led to examine the validity of the current safety requirements relating to the two modes of fuel degradation in the light of the state-of-the-art gained since the last thirty years. Then, the second Advisory Committee held in 2014 focused on the redefining of an acceptable safety limit concerning the brittle mode. It was also the opportunity of concluding on the physical phenomena to be taken into account in the safety demonstration. The main conclusions of the French nuclear safety authority ASN are summarized below.

Regarding the ductile mode, the question of an acceptable demonstration of the maintenance of a coolable geometry was widely discussed during technical meetings between the French utility EDF and IRSN. IRSN summarizes its technical position as the result of detailed reviews of the existing technical basis [10], [14] as follows:

- ✓ Concerning the formation of a flow blockage, even if many sources of axial and azimuthal temperature heterogeneities do exist that are expected to limit the local cladding deformations, the contacts between neighbouring rods would tend to homogenize the temperature in a plane section favouring the extension of the deformation in the axial direction. In addition, axial power profiles on neighbour rods will likely induce peak deformations at approximately similar axial locations, which could favour a significant coplanar blockage.
- ✓ Concerning the coolability of a partially blocked zone, the detailed review of the experiments performed in the 1980s had allowed to identify, in separate tests series, the main parameters that influence the cooling process in the blocked zone; but the results of these experiments do not allow to quantify a coolability limit.

With respect to the approach based on a 1D calculation [17], the core cooling is verified by a calculation of the cladding temperature during the transient, which has to stay below a value limited by default by the 1204 °C. Nevertheless IRSN underlined that the current EDF approach did not allow taking into account contacts between neighboring rods. The calculated temperature excursions in such blocked regions are thus underestimated and may exceed the PCT limit. Finally, the Advisory Committee for Nuclear Reactors followed the IRSN position in considering that this approach was acceptable within the current state of knowledge. In 2014 [9], the new safety demonstration proposed by EDF takes into account the negative effects of ballooning, burst and contacts between rods.

Up to now FFRD was not taken into account. Also, during the rulemaking [9], ASN asked EDF to model the impact of fuel relocation in calculations performed by the CATHARE software to verify core coolability. Moreover, despite fuel dispersal is not a safety concern in France with current core loadings and assembly burn-up limit (52 GWd/MTU), additional experiments shall be carried out to obtain data on MOX fragmentation behaviour during LOCAs.

Regarding the brittle mode, ASN asked EDF to review the way of defining the original limits derived from the U.S. NRC 10 CFR50.46 [1] and its Appendix K (17 % ECR and 1204 °C PCT). The main motivation was to integrate into the LOCA limits definition several physical phenomena that were not represented in the historical RCT approach such as transient secondary hydriding, axial loading during quench, wall thinning and hydrogen taken during normal operation. During the Advisory Committee in 2010, the safety principle of a strength-based approach (based on the LOCA semi-integral tests developed by JAEA) including an additional axial loading to be applied to the rod during the quench phase was accepted.

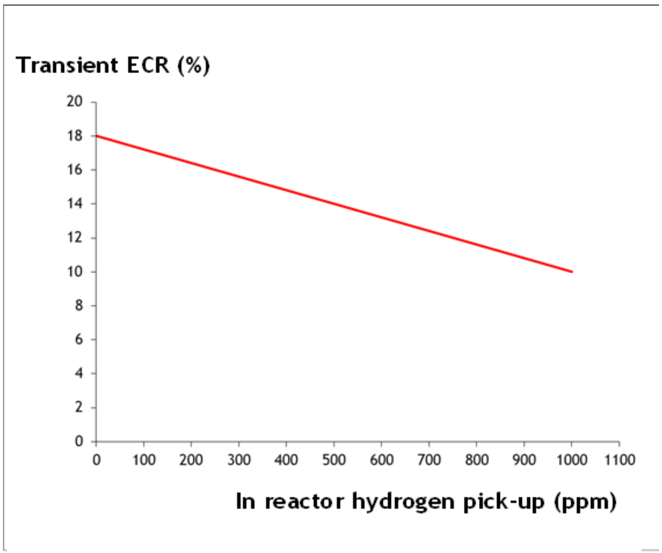


Fig 4: New French ECR design limit

A new French ECR criterion, expressed as a function of in-reactor hydrogen pick-up and combined with the historical 1204 °C peak cladding temperature criterion, was proposed by EDF (Fig 4). The revised LOCA limits were accepted by ASN in 2014 and will be implemented together in the new LOCA methodology. Details of the proposed approach and the new LOCA limit can be found in [16].

In addition, ASN stated that EDF shall demonstrate that an earthquake occurring during the phase of long-term cooling after LOCA does not prevent the core cooling. Indeed, the occurrence of this external hazard cannot be excluded in this phase insofar as it is probable that the fuel rods remain for a long time in the core after a LOCA [9].

4.3 Modelling of Physical Phenomena Related to Fuel Behaviour

In comparison with the current deterministic method, the new fuel behaviour modelling is improved and more accurate by taking into account more physical phenomena that were up to now either not properly modeled or not taken into account in the models.

Indeed, the new EDF CathSBI method takes into account the following physical phenomena:

- Clad ballooning and burst: Modelling is underway by EDF to improve the rupture criterion and to cover the IB heating rates.

- Coplanar clad strain: This parameter is subjected to a statistical approach based on a range of variation.
- Blockage of fuel channels hydraulics: The important enhancement of the new method consists in taking into account the balloons length. This parameter is subjected to a statistical approach based on a range of variation.
- Thermal exchanges between fuel rods and primary coolant and the reduction of the exchange surface due to the contacts between the rods: The new EDF approach is a notable improvement.
- Transient clad oxidation: The pressure effect due to IB LOCA transient on the clad high temperature oxidation kinetic is taken into account.
- Possible accumulation of fuel fragments in the ballooned section of the fuel rods (fuel relocation phenomenon): Various models are developed to take into account relocation consequences on clad temperature. These models are related to thermal conductivity of pellet fragments, the gap between the pellet fragments and the gap and linear power of the relocated fuel. Some parameters of these models are subjected to a statistical approach based on a range of variation. This phenomenon modelling is considered as a major evolution.

IRSN analysis was focused on the validity of this statistical approach proposed by EDF and on the fuel modelling qualification. The IRSN safety review identified a lack of justification for some uncertainties.

5 PENDING QUESTIONS AND CONCLUSION

PCT and ECR calculated without taking into account all the fuel physical phenomena may be underestimated: the first results obtained by this new method show a significant sensitivity of the cladding temperature to the input parameters, appearing from 750 to 800 °C. This sensitivity is linked to the fuel rod phenomena (ballooning, burst, blockage, contact between rods and fuel relocation) activation, when the difference between cladding internal and external pressures is high enough and for cladding temperature levels above 800 °C: their cumulative effects lead in studies with CathSBI method to a significant increase of the cladding temperature (350 to 500 °C).

This negative effect is partially compensated by the use of a statistical method instead of a deterministic one and by the use of new thermal hydraulics models for the vessel. However, the validation of this statistical method and of these new thermal hydraulics models still raise open issues.

Given the high sensitivity to input parameters in this temperature range, it is difficult to conclude with a high level of confidence on the respect of the safety criterion (1204 °C) for LOCA studies reaching cladding temperature levels above 800 °C with high enough difference between cladding internal and external pressures. In view of this high sensitivity area on the PCT, ensuring the robustness of LOCA safety studies based on this in progress method is still a challenge for EDF.

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Implications of Open Phase Conditions in the Electrical Grid Connections for the Safety System of NPPs

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Abstract:

Open Phase Conditions (OPC) and their consequences on the safety system of nuclear power plants (NPPs) were recognized for the first time in Byron NPP, Unit 2 on 2012-01-30. Such failures result from the interruption of one or two of the three phases in the active electrical grid connection of a NPP. Depending on the position of the phase-interruption, the load and the remaining phase(s) induce an asymmetric voltage inside the plants electrical system. This may affect both the operational and the safety buses. The induced asymmetric voltage may cause electrical drives (induction motors) to fail, either because overcurrent-protection relays are triggered or even due to thermal destruction of the drive. Since there is no separation between the redundant safety busbars during normal operation (they are all interconnected via the generator busbar) there is a high risk of simultaneous failures in multiple systems and redundancies. Therefore, it is necessary to ensure that OPCs are detected and handled reliably, so that all safety-relevant electrical consumers are capable to fulfil their safety function. This paper will give an overview about the operation experience with OPC, discuss the possible effects on the safety of NPPs and will present concepts to detect OPCs.

1 INTRODUCTION – NPP BYRON, JANUARY 30TH, 2012

On January 30th, 2012, Unit 2 of the Byron NPP was in full power operation, when a porcelain insulator on the high voltage side of the auxiliary transformers in the switchyard of the plant collapsed and caused an interruption of the associated electrical phase – a so called “Open Phase Condition” or OPC [1]. Even though one phase on the high voltage side was completely lost, due to the electromagnetic coupling of the three phases inside the auxiliary transformer, two of the three line-to-line voltages on the low voltage side did not drop to zero but to values of about 60 % of the nominal voltage amplitude while the third one remained at about 100 %. Such voltage conditions are called “asymmetric”. Due to different voting logics inside the onsite power voltage monitoring system the RPS sensed some problem and initiated SCRAM, but did not disconnect the plants safety buses from the offsite power supply to initiate EDG-operation.

Consequently, all electrical consumers in the plant remained connected with the fault position and were exposed to an asymmetric voltage. Two seconds after the collapse of the insulator the running essential service water (ESW) pump tripped due to overcurrent. In the following seconds a large number of other components which rely on inductions motors also tripped or failed to start because overcurrent protection devices triggered. Among them were a motor-driven auxiliary feedwater pump, a condensate pump, all reactor coolant pumps and also several fans. Attempts of the shift crew to start pumps manually failed due to repeated overcurrent trips of the pumps. For some minutes the reason for the behaviour of the components remained unclear, the shift crew opened connections to unit 1 to restore ESW supply. After about 8 minutes a report was received from the field that smoke arose from the auxiliary transformers. Based upon this report, the safety busses were manually disconnected from the auxiliary transformers and the emergency diesel generators (EDGs) started automatically as designed to supply the safety busses. By doing so, the fault was disconnected from the safety busbars. After the safety busses were supplied by the diesel generators the necessary components were started and plant conditions were normalized.

In summary, it can be stated that a single component failure outside the plant's safety system caused simultaneous failures of multiple components from different systems and redundant trains. The plant's automatic surveillance systems were not able to detect the faulty state in an appropriate way, so manual actions were necessary to identify and cope with the event.

2 THEORETICAL BACKGROUND

In the past, the availability of the offsite power sources of NPPs has been treated in a "binary way" – the source is either powered and can be used to supply the plant or it is not powered and alternate power sources like the EDGs have to be used. However, an electrical power source can only be regarded as available if certain requirements regarding its quality are met. The most important indicators for the quality of a three-phase alternating current are voltage amplitude, frequency and symmetry. The currently installed monitoring devices of the plant's power system focus on voltage amplitude and frequency, while the symmetry of the system is outside of its scope. A three-phase system is defined as "symmetric" when all three line-to-line voltage amplitudes are equal and the distance between the three phases is about 120°. In Fig. 1 an example of a symmetric 50 Hz three-phase system with the three line-to-line (Lx-Ly) and the three line-to-neutral (Lx-N) voltages is given.

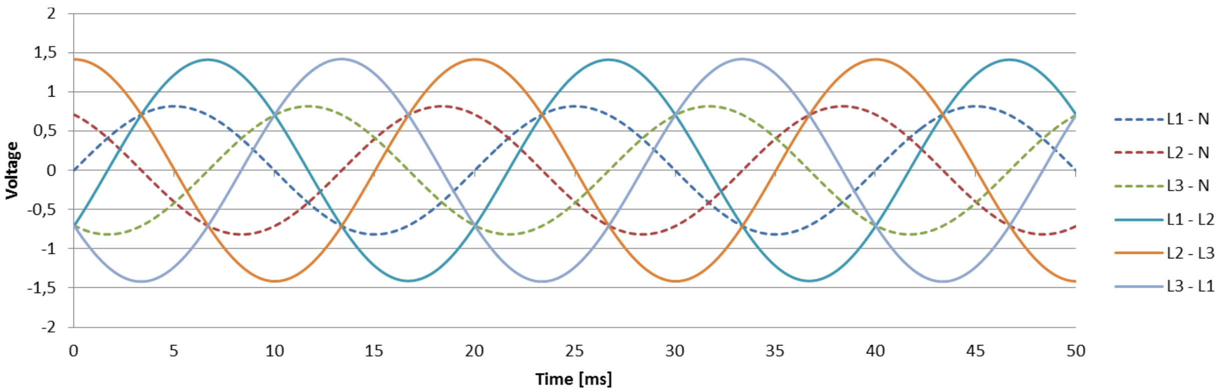

Fig. 1: Symmetrical three-phase system

In Fig. 2 an example of an asymmetrical voltage is given with an (OPC) in line 1 (L1). Such an OPC may be caused by failures of single breaker poles or interruptions of individual power lines. The line-to-neutral voltage in the affected phase drops to 0; the two related line-to-line voltages are affected as well. Transformers between the failure position and the electrical consumers may smooth the effect, so the voltage in the affected phase takes a value somewhere between 0 and the original value. Due to this "smearing" by the transformers the phenomenon is difficult to recognize. Even when all three line-to-neutral voltages are well above the threshold of the safety system, asymmetries may exist which are capable to impede the operation of almost all induction motors.

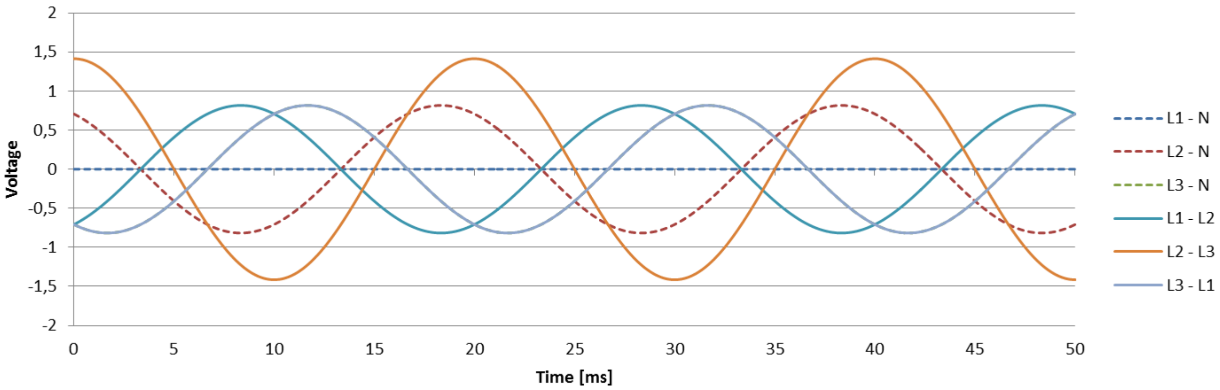

Fig. 2: Asymmetrical three-phase system

When an induction motor is operated with an asymmetric power supply, its integral current intake will rise while the delivered torque will drop. The increased current intake may trigger overprotection devices or may even destroy the motor due to overheating. Both effects have been observed in the recent world-wide operating experience. Even if only overcurrent-protection devices are triggered, the affected components have to be considered as unavailable for a certain amount of time since resetting the overcurrent protection devices in general requires manual action in the switchyard building.

3 OPERATING EXPERIENCE WITH OPEN PHASE CONDITIONS

Even though asymmetric conditions in three-phase alternating current power system are well studied and understood in electrical engineering departments, the phenomenon was completely out of the scope of NPP safety research and analysis up to the event in the NPP Byron. The importance of such events was confirmed on May 30th, 2013 where an OPC event due to a failure of the high-voltage grid breaker occurred in the NPP Forsmark [2] which led to comparable effects like in the Byron event.

3.1 Identified events with OPCs in NPPs

After the events in Byron and Forsmark GRS performed an in-depth analysis of the international operating experience to identify other events, where asymmetries in the plant's power supply system due to OPC had effects on the operation and/or the availability of the safety system. In total, 10 events have been identified; the earliest dating back to the year 1994 where OPCs in the active grid connection of a NPP lead to cross-system and cross-redundancy component failures in the affected NPP. These 10 events are listed below.

Date	Plant	Failure Cause
13.05.1994	Kalinin	Collapse of a transformer duct, OPC in one phase
25.02.1997	Balakovo	Unintended closure of a single breaker pole
31.03.2001	South Texas	One breaker pole in the switchyard failed to close
11.11.2005	Koeberg	One breaker pole in the switchyard failed to close
31.07.2006	Vandellos	Mechanical failure of a disconnector
14.05.2007	Dungeness-B	One pole of a HV-transformer breaker failed to close
30.01.2012	Byron	Collapsed Insulator caused a line interruption
01.12.2012	Bruce	Mechanical line failure during severe weather (storm)
30.05.2013	Forsmark	Failure to open on command of a single breaker pole
27.04.2014	Dungeness-B	Open breaker pole in the switchyard

Only events with actual component failures in more than one electrical redundancy as a direct consequence of an asymmetry in the plant's onsite power system are listed. Events where only standby grid-connections or single redundancies were affected are excluded.

3.2 Occurrence frequency of OPC events

The integral international operating experience with NPPs since the first identified event adds up to about 9500 reactor years. With the ten OPC events listed in the table above, the frequency for an OPC in the active grid connection of a NPP can be estimated to

≥ 1 · 10^{-3}/ry. It has to be noted that there is no systematic reporting of OPCs, so this list of events given above is most likely incomplete and therefore the estimation of the occurrence frequency has to be regarded as non-conservative.

4 IMPLICATIONS OF OPCS ON NPP SAFTEY

In Fig. 3 a schematic layout of a typical NPP's power system is given. It can be seen that there is no separation between the electrical redundancies of a NPP during normal power operation. As long as the plant is not in loss of offsite power (LOOP) condition (when the safety relevant busbars are supplied by the EDGs) all electrical busbars including the safety busbars are connected either via the generator busbar or the high voltage side of the standby transformer.

Any failure which affects the connected sections of the plant's power system may therefore influence all redundant electrical trains of the plants power supply simultaneously. In case of an OPC in the positions marked with an all electrical consumers of the plant – including the consumers within the NPP´s safety system – are affected by the asymmetry and may therefore fail to function as designed. Systems with pumps that rely on alternative drives like diesel-engines or steam turbines may also be affected, since motor driven valves which might be necessary for the operation of the systems may also fail due to the asymmetry.

Thus, electrical asymmetries due to OPCs have the potential to render well established concepts of reactor safety like redundancy and diversity useless.

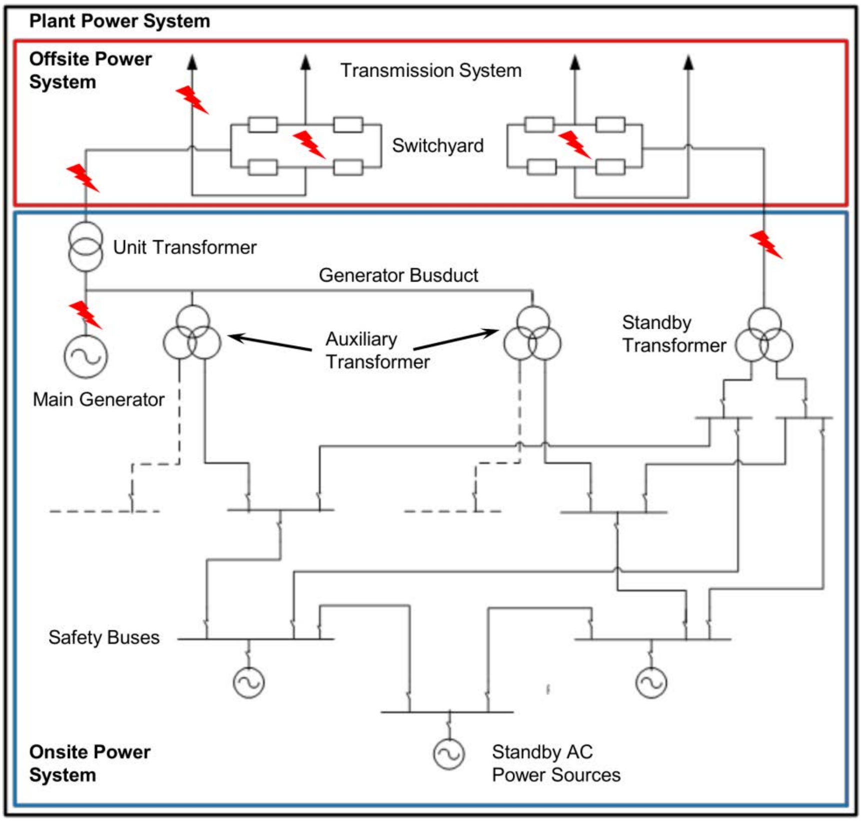

Fig. 3: Schematic layout of a NPP power system (Source: IAEA, modified)

The actually observed consequences of an asymmetry in a NPP differ from event to event. In some of the events, the asymmetry led to temporary component failures which caused the RPS to trip the reactors by SCRAM – in most cases because the RPS sensed a failure of the main coolant pumps. In 7 of the 10 events, the RPS/ESFAS was not able to detect the faulty state of the plant's onsite power system and therefore did not disconnect the power source affected by the OPC automatically so manual actions of the crew were necessary to identify the problem and to cope with the situation. Some OPCs even remained undetected up to several days. During this period, several electrical consumers tripped but could be restarted afterwards.

In the worst case scenario, asymmetric conditions in the plant's onsite power system due to OPC have the effect of an "undetected station blackout". The electrical consumers – including those of the safety system – which rely on three-phase-current as power source fail either because they are tripped by their overcurrent protection devices or destroyed due to overheating.

Operating experience showed as well that coping with a correctly identified electrical asymmetry is easy to achieve – once the grid connection affected by the OPC is disconnected from the onsite power system the safety system will work as designed and either switch to an alternative grid connection or start up the EDGs.

4.1 Risk quantification

In current Probabilistic Risk Assessments (PSA) for NPP OPCs are not treated as possible initiating event. Integrating them into the scope of the PSAs may therefore contribute to the core damage frequency (CDF) calculated by the PSA. This additional frequency ΔCDF_{OPC} that a NPP suffers core damage as result of an accident sequence caused by an OPC can be quantified as shown below:

$$\Delta CDF_{OPC} = f_{OPC} \times CCDP_{OPC}$$

In this equation, f_{OPC} describes the frequency how often a NPP is affected by an OPC while $CCDP_{OPC}$ describes the conditional core damage probability in case of a given OPC. As outlined in section 3.2 f_{OPC} can be estimated to $1 \cdot 10^{-3}$/ry. In case of an event like the ones in Byron or Forsmark in a plant with no additional automatic detection devices for asymmetric conditions, $CCDP_{OPC}$ is equivalent to the probability that the shift crew

1. is able to recognize that the observed component failures are caused by an asymmetry in the electrical onsite power system,
2. is able to detect the OPC causing the asymmetry correctly and
3. opens the correct breaker(s) to disconnect the onsite power system from the fault position.

Additionally it is vital that all required systems and components (breakers, EDGs, etc.) perform as designed.

A precursor analysis of the event at the Byron NPP done by U.S. NRC [3] showed that this sequence is dominated by the Human Error Probabilities (HEP) of the crew actions. Depending on the used HEP modelling assumptions the conditional core damage probability $CCDP_{OPC}$ ranges from $1 \cdot 10^{-4}$ (standard SPAR-H method with additional qualitative factors) to $3 \cdot 10^{-3}$ (SPAR-H method without the qualitative factors).

The above mentioned precursor analysis was made specifically for the Byron NPP and the event from January 30th, 2012. So it cannot be transferred directly to other plants and possible events. Nevertheless, since the sequence of crew actions – detect the asymmetry as such, locate the failure source and isolate the failure – would be the same for all plants, the order of magnitude of the results can be regarded as a generic result.

With the above mentioned values for f_{OPC} and $CCDP_{OPC}$, the additional core damage frequency ΔCFD_{OPC} due to OPC events can be quantified with $1 \cdot 10^{-7}$/ry to $3 \cdot 10^{-6}$/ry. This corresponds well with an analysis performed by the Nuclear Energy Institute (NEI) which resulted in an increase in core damage frequency of $3 \cdot 10^{-6}$/ry [4].

4.2 (Automatic) Detection of OPCs

As mentioned before, OPCs are difficult to detect but easy to cope with once they are detected. The currently implemented RPS-graded (class 1E, category A, etc.) automatic surveillance of the onsite power system, which focuses on voltage amplitude and frequency is – as seen in the operating experience – not able to detect asymmetries caused by OPC reliably.

Each automatic detection system for the detection of electrical asymmetries in the onsite power system of a NPP has to face two challenges: it has to be able to detect all those asymmetries that endanger the capability of the plant’s electrical consumers to fulfil their (safety) function fast and reliably but it may not be triggered due to “normal” asymmetries which exist as the result of normal grid operation or transients like lightning strikes. Furthermore, the detection system must be able to detect existing asymmetries due to OPC in the active grid connection as well as latent OPC existing in standby grid connections.

Several methods have been developed and presented in the last years. A detailed description of the methods and parameters would exceed the scope of this paper, so only a non-comprehensive list of parameters which could be used in order to detect asymmetric conditions is provided:

- Symmetric components of voltage;
- Symmetric components of current;
- Line-to-line voltages;
- Magnetization current;
- Zero sequence current;
- Start point currents / voltages (depending on transformer grounding).

The appropriate detection method has to be determined specifically for each plant.

5 CONCLUSIONS

Asymmetric conditions in the onsite power system of NPPs due to OPC as they were observed in several NPPs worldwide during the last years have the potential to cause multiple, simultaneous component failures throughout different systems and redundant trains. This type of failure is not included in current PSA for NPPs but will most probably have an non-negligible effect on such PSAs so reliably precautionary measure have to be implemented.

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Bending Observed on the Carbon Steel Liner of the Pre-stressed Concrete Primary Containment of a Nuclear Reactor: Safety Assessment

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Abstract:

On September 7th, 2014 during the periodical visual inspection of the carbon steel liner of the pre-stressed concrete primary containment of a Belgian nuclear reactor, the authorized inspecting organization (AIO) observed a bending on the carbon steel liner.

Rapidly after the observation, several actions were taken by the Licensee in order to evaluate the importance of the observed bending. The bending was confirmed being 2.7 m wide, 1.9 m high and with a maximal curving of 14 cm. Several tests were also carried out (a magnetic control of the carbon steel liner thickness, an ultrasonic control of the welds and a drilling).

On January 30th, 2015 the regulatory body was informed about the situation and started then a safety assessment in order to evaluate the acceptability of the situation regarding the safety functions to be ensured by the carbon steel liner and by the pre-stressed concrete primary containment itself.

This paper aims to share the safety assessment carried out by the Belgian TSO in the framework of the bending observed on the carbon steel liner of the pre-stressed concrete primary containment of a Belgian nuclear reactor. It starts with historic and some useful feedback from experience. It includes the Belgian TSO analysis of the subject and the actions carried out by the Licensee and ends up with the Belgian TSO conclusions and further defined actions.

1 INTRODUCTION

On September 7th, 2014 during the periodical visual inspection of the carbon steel liner of the pre-stressed concrete primary containment of a Belgian nuclear reactor, the authorized inspecting organization (AIO) observed a bending on the carbon steel liner. Ten percent of the surface of the carbon steel liner has to be inspected at each programmed unit stop. The observed bending was located above the opening for the material air lock.

Rapidly after the observation, the Licensee carried out several actions in order to:

- Determine the exact geometry of the bending;
- Verify the welds of the liner;
- Verify the presence of concrete right behind the buckled carbon steel liner;
- Determine the origin of the bending;
- Verify the (non-)existence of a similar phenomenon at other Belgian nuclear reactors;
- Evaluate the impact of the bending on the reactor operation.

A 3D laser scan allowed the Licensee to determine the exact geometry of the bending confirming bending dimensions of 2.7 m width, 1.9 m height and a maximal curving of 14 cm.

A magnetic control of the carbon steel liner thickness and an ultrasonic control of the welds were carried out by the Licensee in order confirm the integrity of the buckled liner by verifying its thickness and the condition of its welds.

A 17 mm long and 10 mm diameter drilling was also carried out in order to verify the presence of concrete right behind the buckled carbon steel liner. A first endoscopic inspection was carried out through this drilling.

In its evaluation to determine the origin of the bending, the Licensee indicated it was already present in 2004.

Taking the performed investigations into consideration, the Licensee concluded that the observed bending had no impact on the safety of the nuclear reactor operation.

On January 30th, 2015 the regulatory body was informed about the situation and started then a safety assessment in order to evaluate the acceptability of the situation regarding the safety functions to be ensured by the carbon steel liner and by the pre-stressed concrete primary containment itself.

Bel V, the Belgian Technical Support Organisation (TSO), started the safety assessment collecting some useful feedback from experience. This showed that there was no bending reported in national and international literature on the carbon steel liner of the pre-stressed concrete primary containment of nuclear reactors similar to the one observed by the Belgian Licensee. It continued with the analysis of the subject and ended up with Bel V statement regarding the acceptability of the situation and the definition of the further actions to be carried out by the Licensee.

This paper aims to share the safety assessment carried out by Bel V in the framework of the bending observed on the carbon steel liner of the pre-stressed concrete primary containment of a Belgian nuclear reactor. It starts with historic and some useful returns of experience. It includes Bel V analysis of the subject and the actions carried out by the Licensee and ends up with Bel V conclusions and further defined actions.

2 HISTORIC INFORMATION

In the nuclear reactor concerned, the pre-stressed (better called '*post-tensioned*') concrete primary containment has a diameter of 42 m, a height of 62 m and a thickness of 0.70 m. All its inner side is covered by a 6 mm thick carbon steel liner (with a yield strength of 255 MPa).

The carbon steel liner covers the cylindrical part, the hemi-spherical dome part and the raft of the inner primary containment. It is anchored in the pre-stressed reinforced concrete primary containment by bended 10 mm diameter steel profiles. These anchors are placed every 150 mm (in both directions: vertical and horizontal) and are welded to the liner on all rectilinear portions in contact therewith. The same anchoring system is used in the dome. On the raft, the carbon steel liner is welded to I-beam profiles anchored in the concrete.

The loads considered on the primary containment according to the design are the following:

- Normal conditions:
    - self-weight,
    - equipment loads,
    - depressurize inside the primary containment in normal operation,
    - depressurize outside the primary containment in normal operation,
    - thermal loads due to normal operation.
- Accidental conditions:
    - self-weight,
    - equipment loads in accidental conditions (line break),

- o thermal loads in accidental conditions,
- o pressure loading accidental conditions,
- o safe shutdown earthquake (SSE) load.

The carbon steel liner ensures the tightness of the inner primary containment under the entire, above mentioned normal and accidental conditions. The carbon steel liner forms a tightness element and not a resistance element. It is considered as having no self-stiffness, however it transfers the loads (equipment loads, pressure or depression loads) to the pre-stressed reinforced concrete primary containment.

Therefore, the carbon steel liner safety function is to ensure the tightness of the inner primary containment (tightness safety function), while the pre-stressed reinforced concrete primary containment safety function is to resist to the applied normal and accidental loads (structural safety function).

The carbon steel liner and the pre-stressed reinforced concrete primary containment are key elements to nuclear safety.

During the construction of the pre-stressed concrete primary containment discussed in this paper, the carbon steel liner was used as a lost formwork. The related concrete load had been taken into consideration in the design calculation of the carbon steel thickness. The concrete deformations (drying shrinkage and creep effects) had also been taken into consideration during the carbon steel liner design.

Per design, the carbon steel liner material had been calculated in order to avoid any plastic deformation under normal operating conditions. Also the anchors spacing had been determined in order to avoid any bending in normal operation conditions.

3 EXPERIENCE FEEDBACK

The Licensee and Bel V both checked the occurrence of similar events worldwide and in Belgium.

The internally available operating experience revealed the existence of bending on carbon steel liners of other pre-stressed concrete primary containment nuclear reactors. However, the bending reported had a maximal surface of around 1 m² (and had generally a surface of 0.5 m²). The studies carried out with such buckled carbon steel liners showed that this phenomenon had no impact on the safety functions ensured by the carbon steel liner and so no impact on the safety of the nuclear reactor operation. Nevertheless, these dimensions are not in the same order of magnitude as those of the bending observed in the Belgian nuclear reactor discussed in this paper. So, for Bel V the same conclusions are not directly applicable without further actions.

Looking for similar events in the national operating experience led Bel V to a Licensee report [20] indicating that in 1992 a local deformation (bending) of the carbon steel liner was observed in its lower part (at - 2 floor level of the reactor building, in front of the perimeter sumps). The carbon steel liner bending was located at around 1.80 m above the raft and was approximately 1 m long and 50 cm to 60 cm high. At that time, impact testing showed up that there was a void behind the carbon steel liner at the location of the bending. This void was then completely filled with a grout mixture made from cement, sand and binder. The Licensee demonstrated that this bending existed since the construction of the unit. The Licensee explained this bending was caused by the formwork difficulties encountered at this location of the pre-stressed reinforced concrete primary containment where the primary containment has a conic section.

The Licensee controlled and confirmed the integrity of the carbon steel liner by a global type A pressure test realized in 2004. Moreover, during the reactor stop in 2014, the Licensee controlled the progression regarding the dimensions and the location of this repaired bending. No progression has been observed.

As a conclusion, national and international literature have shown no reported bending on the carbon steel liner of the pre-stressed concrete primary containment of nuclear reactors similar to the one observed by the Belgian Licensee regarding dimensions and location.

4 ANALYSIS OF THE OBSERVED PHENOMENON

Seen the conclusions of analysing the operating experience, Bel V carried out a safety assessment in order to evaluate the possible safety consequences considering:

- Upholding of the tightness safety function of the liner (and so the upholding of the bending) in accidental conditions:
  - The purpose of the carbon steel liner is to ensure the tightness of the primary containment in normal and accidental conditions and to transfer the loads to the pre-stressed reinforced concrete primary containment.
- Upholding of the structural safety function of the pre-stressed reinforced concrete primary containment in accidental conditions taking into consideration:
  - that the observed carbon steel liner bending could have damaged the pre-stressed reinforced concrete primary containment (due to tension forces, …);
  - that the observed carbon steel liner bending could be caused by a defect in the pre-stressed reinforced concrete primary containment (delamination, void, loss of pre-stressing …).

In its safety assessment Bel V insisted to verify whether the bending was evolving or not. This was done in order to make the Licensee take the necessary actions in the future and in order to define an adapted follow-up plan. Bel V also kept in mind to determine the causes of the bending in order to define rather this phenomenon is possible in other units or in other places of the same unit.

4.1 Historic

In its evaluation to determine the cause of the bending observed on September 7th, 2014, the Licensee found out that the observed bending was already present in 2004 [21]. At that time the bending was classified as "acceptable with remarks" by the inspecting organism. No trace of further actions in order to characterize the observed bending was found out by the Licensee. On Bel V demande, the Licensee increased the inspection organism awareness of communicating all carbon steel liner defects reported during the carbon steel liner periodic inspection.

During his safety assessment, Bel V pointed out the difference in the bending curving values measured in 2004 (30 cm, [21]) and in 2014 (14 cm, [15]). The Licensee explained that no detailed measures were carried out in 2004 as no 3D scan was carried out. The observations carried out in 2004 therefore cannot be taken into consideration in order to determine if the bending has progressed during the last 10 years.

On Bel V demande, the Licensee carried out a 3D scan of the bending in 2015 and confirmed no progression of the bending as its location and dimensions were exactly the same than the ones measured in 2014. For more certitude Bel V asked the Licensee to carry out again a 3D scan of the bending in 2016.

4.2 Causes of the Bending

The Licensee explained the observed bending by the failing of the scaffolding at the time of the pre-stressed reinforced concrete primary containment construction. For the Licensee, the thrust of the fresh concrete on the non-supported carbon steel liner used as formwork caused the today observed bending. Indeed, the formwork, the concreting works and the scaffolding at the location of the observed bending are complicated to execute due to the presence of the material air lock. Moreover, based on the formwork drawings, the Licensee could identify the area where the bending is located as being filled by a concrete of second phase. Finally, the Licensee explained the shape of the bending (half-moon) by the presence of the material air lock and confirmed by this argument that this was the only place where the formwork could be non-supported due to the difficulties to realize the scaffolding at this location. By this, the Licensee concluded that the phenomenon was isolated and that there was no reason to observe a similar bending on any other unit.

The Licensee could find out an evidence demonstrating that the phenomenon of small bending on the carbon steel liner was known at the construction time. Though, the bending observed today with its specific dimensions was not mentioned in any of the construction documents.

From Bel V point of view other possible causes for bending on the carbon steel liner of the pre-stressed reinforced concrete primary containment are the thermal loads and more specifically the fatigue phenomenon caused by the thermal loads. The Licensee justified that a curving of 14 cm cannot be caused by the fatigue phenomenon due to the thermal loads. Thermal loads are considered in the design of the primary containment and can only cause smaller bending between the anchors (around 0.5 m^2 and a few millimetres of curving).

A bending of the carbon steel liner could also be caused by a defect in the pre-stressed reinforced concrete primary containment (delamination, void, gravel nest, loss of pre-stressing, …). According to the Licensee point of view tensile loads are too low to cause a delamination of the pre-stressed reinforced concrete primary containment.

Bel V asks to the Licensee to justify the cause of the bending by evidence in order to evaluate the exact safety impacts on the unit affected as well as on other units. Therefore, Bel V asked the Licensee to carry out necessary inspections and investigations in order to determine with exactitude and by evidence the causes of the bending.

4.3 Safety Issues

According to Bel V, the bending on the carbon steel liner of the pre-stressed concrete primary containment could lead to two potential safety issues:

- The tightness of the liner: The purpose of the carbon steel liner is to ensure the tightness of the primary containment under normal and accidental conditions and to transfer the loads to the pre-stressed reinforced concrete primary containment. It is therefore necessary to ensure the strength of the buckled liner in accidental conditions.
- The structural integrity of the primary containment: A loss of the structural integrity of the concrete primary containment could be the cause of the observed bending. According to the Belgian TSO, the bending could be hiding a structural defect (gravel nest, presence of foreign body, loss of pre-stressing, …) which could jeopardize the structural integrity of the concrete primary containment.

4.3.1 Tightness safety function

Several actions were carried out by the Licensee in order to demonstrate that the tightness safety function of the carbon steel liner was still properly ensured.

Inspections (magnetic controls) of the liner were carried out in order to confirm that the thickness of the liner remained above 6 mm.

Inspections (ultrasonic tests) of the welds between the liner and the anchors were carried out in order to confirm the presence of all welds and their thicknesses. However, this test could not confirm the correct anchorage of the liner anchors in the pre-stressed reinforced concrete primary containment. In order to confirm this last point, the Licensee carried out a calculation considering the observed bending and demonstrating that there is no risk of breakage neither of anchors nor of the welds between the anchors and the carbon steel liner.

The Licensee also carried out a pressure test (global type A pressure test) in 2005 confirming the integrity and tightness of the buckled liner in accidental conditions (the bending was already observed in 2004, see paragraph 4.1). According to the Technical Specifications [2], the purpose of the global type A pressure test is to determine the overall leakage rate of the primary containment and is carried out at a pressure of 1.6 bar (value slightly higher than half of the relative accident pressure P_a = 3.1 bar). The type A test pressure value (1.6 bar) was considered as sufficient by a workgroup held in 1988 [1]. Other tests of conformity are carried out as a compensatory measure. According to the Technical Specifications, the Licensee is requested to carry out this test every 10 years (with a margin of 18 months).

The Licensee also carried out a finite element analysis [4] in order to demonstrate the strength of the buckled liner in the following conditions:

- Normal conditions: self-weight, concrete deformation, operation temperature;
- Disturbed conditions: self-weight, concrete deformation, operation temperature, Design Basis Earthquake (DBE);
- Accidental conditions: self-weight, concrete deformation, accidental pressure (3.1 bar), SSE;
- Pneumatic test conditions: self-weight, concrete deformation, test pressure (1.15 × accidental pressure);
- Type A test conditions: self-weight, concrete deformation, type A test pressure (0.5 × accidental pressure).

In the finite element model, a spherical and smooth bending with a diameter of 2.6 m and a curving of 14 cm is considered. In the model, the material air lock located below the buckling is not considered. The model considers a void bending (which means that no anchorages in the area of the bending are considered to be working; only the anchorages located out of the bending allow the anchoring of the liner in the pre-stressed reinforced concrete). Calculations showed that the following results:

- For the anchors: Both calculated loads and displacement are below the maximum admissible load and below the maximum admissible displacement.
- For the liner: According to ASME III div 1, deformations imposed on the concrete and temperature are “secondary” loads. These “secondary” loads will create secondary strains in the liner that will limit themselves and will never reach the maximum admissible strain value of the carbon steel liner. Regarding displacements, the liner will undergo a plasticity that will remain far below the maximum admissible value of around 27 %. The liner will therefore remain tight.
- For the welds: These are justified for all load cases.

The conclusion of this finite element model is that the structural integrity of the liner anchors and welds being ensured and that the carbon steel liner will remain tight regardless of the applied load.

The model is conservative in some aspects (void considered between the buckled carbon steel liner and the pre-stressed reinforced concrete) and non-conservative in other aspects (spherical bending). According to BEL V, considering a spherical bending can lead to underestimate the strains in the liner and in the anchors. However, the material air lock located below the bending (and not considered in the finite elements model) helps stiffening the liner and it will therefore reduce the tensile load in the anchors located in the lower part of the bending. Moreover, the margins between the calculated loads and displacements and the maximum admissible loads and displacements are comfortable.

A second 3D scan was carried out by the Licensee in June 2015 [16] with the same equipment as in 2014 and showed no progression in the bending. It was located at the same place and had the same dimensions (2.7 m wide, 1.9 m high and a maximal curving of 14 cm) as in 2014.

The last global type A pressure test was carried out in 2016, giving a possibility to confirm the tightness of the carbon steel liner. Bel V asked the Licensee to carry out a 3D scan before and after this global type A pressure test in order to confirm that the bending of the liner is a non-evolving phenomenon. The 3D scan results realized in June 2016 [17] and in August 2016 [18] showed no progression of the phenomenon (same dimensions, same location).

According to the above mentioned elements, Bel V concluded that the tightness safety function of the carbon steel liner was fulfilled.

4.3.2 Structural safety function

The Licensee carried out a 17 mm long and 10 mm diameter drilling in order to verify the presence of concrete right behind the buckled carbon steel liner. The Licensee also carried out an endoscopic inspection through this drilling. This inspection showed the presence of a void and of concrete behind the carbon steel liner at the location of the drilling.

The Licensee also carried out a visual inspection of the outer part of the pre-stressed reinforced concrete primary containment from the annular space. No damages were observed.

According to BEL V, the structural integrity of the pre-stressed reinforced concrete primary containment behind the buckled carbon steel liner needs to be demonstrated. The following elements need therefore to be checked and their correct working needs to be verified and confirmed:

- The reinforcement and the pre-stressing (better called '*post-tension forces*') of the concrete primary containment:

  According to the Licensee the reinforcement plays a limited role in the structural strength of the primary containment in case of an accident. Indeed, the pre-stressing of the primary containment is calculated in order to avoid any tensile load in the concrete in case of accidental overpressure or thermal loads.

  Post-tension is a fundamental feature of the primary containment design as it allows fulfilling the strength requirements for the accidental loading situations, particularly for what concerns the pressurization and thermal effects. As post-tensioning forces tend to decrease with time as a result of tendon steel relaxation and concrete long-term deformations such as drying shrinkage and creep effects, a correct evaluation of these post-tension losses is therefore a key factor of a successful design. At the time of the design of the primary containment[1], the post-tensioning losses were taken into consideration in accordance with the available knowledge and models. Allowable lower bounds of the post-tension forces were therefore defined for the wall vertical and horizontal tendons as well as for the dome tendons. These post-tension lower bonds are directly related to design limits expressed in terms of concrete long-term deformations allowable upper bounds. The concrete long-term deformations of the wall (and the dome) have since then been monitored and the measured concrete deformations are periodically (once a year since 1994 and twice a year since 2006) assessed with respect to the above design limits.

  In the demonstration of the structural integrity of the pre-stressed reinforced concrete primary containment behind the buckled carbon steel liner, the verification of the post-tension losses (monitoring) is a key factor.

  At the construction of the unit (1974), the post-tension losses have been monitored with embedded vibrating wire extensometers during the initial pressure testing. The further evolution of the post-tension has been monitored from October 1994 on, by

[1] The unit under investigation was built in 1974.

re-using the existing embedded vibrating wire extensometers. These analyses show that the average deformation of the wall in horizontal and vertical direction is within the design limits.

According to BEL V, these analyses tend to demonstrate the structural integrity of the pre-stressed reinforced concrete primary containment. However, as far as these analyses consider the average deformation of the wall in horizontal and vertical direction, the structural integrity of the pre-stressed reinforced concrete primary containment behind the buckled carbon steel liner is not strictly demonstrated. Therefore, Bel V asked the Licensee to carry out necessary inspections and investigations in order to exclude that the observed bending was linked to a loss of post-tensioning.

At the time being, Bel V is assessing the post-tension losses of the primary containment for all Belgian power plant units [6], [7], [8].

- The characterization of the properties and of the actual performance of the primary containment concrete. According to Bel V the absence of any structural (possibly evolving) unacceptable defect (gravel nest, presence of foreign bodies, voids, delamination …) located in the area where the bending is observed has to be checked and confirmed. The first endoscopic inspection limited to a length of 17 mm may not be representative of the situation overall behind the bending. According to the Licensee seen the dimensions of the primary enclosure in pre-stressed reinforced concrete local defects would not impact on the overall structural behaviour of the primary enclosure because they have no impact on the overall stiffness of the structure. According to BEL V, the presence of evolving defects (delamination, …) could jeopardize the structural integrity of the pre-stressed reinforced concrete primary containment.

  The TSO asked the Licensee to carry out additional inspections in the pre-stressed reinforced concrete primary containment in order to check the absence of any structural (possibly evolving) unacceptable defect and in order to check the presence of concrete everywhere behind the bending (and to determine the cause of the bending).

  In a first stage and in order to avoid destructive tests, the Licensee carried out a study to check the possibility to perform non-destructive tests (radar, ultrasonic testing, infrared thermography) on the pre-stressed reinforced concrete primary containment from the annular space (in order to avoid interferences from the carbon steel liner). This study includes the realization of non-destructive tests (ultrasonic testing) on well-known pre-stressed reinforced elements (beams) and other reinforced elements (columns and concrete of secondary containment) and concludes that the system does not work for elements having a thickness above 90 cm, the interpretation of the resulting measures of non-destructive testing is too complex and can therefore not confirm with certitude the absence of any structural (possibly evolving) unacceptable defect nor the presence of concrete everywhere behind the bending.

4.4 Further investigations on the primary containment

In order to answer to Bel V questions related to the cause of the bending and related to the structural safety function of the pre-stressed reinforced concrete primary containment, the Licensee decided in June 2015 to carry out two drillings inside of the bending through the carbon steel liner [9, 10].

The drillings were located based upon the 3D scan realized in June 2015 and far enough from the drill carried out in 2014. Before drilling through the carbon steel liner, the Licensee carried out ultrasonic tests and penetration tests in order to check that the thickness of the liner at the location of the drillings was conform to the design and to the previously measured thickness (6 mm).

A team of qualified welders was in charge of the realization of a first diameter 16 mm boring through the liner. Then, the Licensee realized one sclerometer test in order to measure the local concrete compression resistance. The team of qualified welders welded the system aimed to restore the tightness of the carbon steel liner. The repair of the carbon steel liner was realized in accordance with ASME XI [11]. The weld was tested (dye penetrant testing) in order to control its tightness. The 40 mm diameter boring was then carried out by the operator in charge of the boring through the carbon steel liner and the 30 mm diameter concrete cores extracted. The drilling machine adhered by suction to the carbon steel liner. The boring was air-cooled rather than water-cooled in order to avoid any water ingress inside of the pre-stressed reinforced concrete primary containment. In order to avoid any damage on the reinforcement bars or pre-stressing cables of the concrete primary containment, the diameter of the boring was limited to 30 mm and the length of the boring to 14 cm (which corresponds to the measured curve of the bending). The AIO was present during all the operations.

A small scale mock-up (150 cm high and 150 cm wide) of the primary containment with the carbon steel liner was realized in order to train the operators and to test the system before the realization of the operation through the carbon steel liner of the primary containment. The small scale made-up has been realized in order to reproduce the real situation (geometry, materials, anchors used are identical to the ones of the primary containment). The construction method for the small scale made-up was the same than the construction method for the primary containment (mainly the carbon steel liner was used as lost formwork). During the construction of the small scale made-up, the Licensee also observed a bending on the carbon steel liner.

Two concrete cores have been extracted from the drillings carried out through the bending and two endoscopic inspections have been realized through these drillings.

The drillings have been sealed with a chemical grout mixture. Specific class 2 nozzles were ordered in order to repair the carbon steel liner at the location of the drillings. The Licensee justified by calculation the resistance of the repaired liner in accidental conditions. These nozzles were welded to the carbon steel liner. The welds have been tested (dye penetrant testing) and the tightness of the complete system has been successfully verified (dye penetrant inspection).

The two cores have been visually examined. This inspection revealed two different material compositions. The first material was a compact cementitious compound similar to mortar type. It was located just behind the metal liner and had a thickness around 8 cm. The second material located directly behind the first is a conventional concrete including relatively small limestone aggregates. The two cores did not show any gravel nest. The test carried out with the sclerometer showed up a concrete compressive strength of 70 MPa. Endoscopic tests did not reveal any delamination or void.

The small size of the cores did not allow for carrying out compression tests. Chemical analyses were carried out and demonstrated that the bended area was filled with grout which cement content per percent of concrete mass was significantly different from the cement content of the concrete composing the pre-stressed concrete primary containment [19]. Based on this analysis and other previously considerations and justifications the Licensee concluded the observed bending was present since the construction of the pre-stressed concrete primary containment and that, at that time, it was filled with a grout (or by a concrete of second phase). The phenomenon is therefore not systematic in other units or in other places of the same unit.

5 CONCLUSIONS

On September 7th, 2014 during the periodical visual inspection of the carbon steel liner of the pre-stressed concrete primary containment of a Belgian nuclear reactor, the inspecting organism observed a bending on the inspected carbon steel liner. A 3D scan allowed the

Licensee to determine the exact geometry of the bending confirming bending dimensions of 2.7 m wide, 1.9 m high and a maximal curving of 14 cm.

A review of the national and international literature did not show reported bending on the carbon steel liner of the pre-stressed concrete primary containment of nuclear reactors similar to the one observed by Belgian Licensee regarding dimensions and location.

Therefore, Bel V carried out a safety assessment in order to evaluate the possible safety consequences considering:

- Upholding of the tightness safety function of the liner (and so the upholding of the bending) in accidental conditions:

  The purpose of the carbon steel liner is to ensure the tightness of the primary containment in normal and accidental conditions and to transfer the loads to the pre-stressed reinforced concrete primary containment.
  The two last years, the Licensee carried out several 3D scans and magnetic controls of the liner, ultrasonic tests on the welds and several global type A pressure tests. These results showed no progression of the phenomenon (same dimensions, same location). The Licensee also realized a finite elements calculation of the liner considering all operating conditions.
  As the results of all these tests were satisfactory, Bel V concluded that the tightness safety function of the carbon steel liner was fulfilled.
- Upholding of the structural safety function of the pre-stressed reinforced concrete primary containment in accidental conditions taking into consideration:
  - that the observed carbon steel liner bending could have damaged the pre-stressed reinforced concrete primary containment (as for instance due to tension forces) and
  - that the observed carbon steel liner bending could be caused by a defect in the pre-stressed reinforced concrete primary containment (as for instance due to delamination, voids, loss of pre-stressing).

In order to answer to the Bel V questions related to the cause of the bending and related to the structural safety function of the pre-stressed reinforced concrete primary containment, the Licensee decided in June 2015 to carry out two drillings inside of the bending through the carbon steel liner. No void, neither delamination, neither foreign body were found out. No loss of post-tensioning forces was observed. Chemical analyses demonstrated the bending was present since the construction of the pre-stressed concrete primary containment.

Bel V concluded that the structural safety function of the pre-stressed reinforced concrete primary containment in accidental conditions is ensured.

6 ACKNOWLEDGMENTS

The authors want to acknowledge the Licensee for the factual checking and the Bel V colleagues who contributed to this safety assessment.

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Seminar 2

Nuclear Installation Safety – Research

Study of Safety and International Development of SMR

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Abstract:

In the last years, several well-developed SMR designs from different international vendors were announced. Such reactors with low power are not only attracting sparsely populated areas but also heavy populated cities to provide electricity, potable water and heat. So it is not excluded that SMRs will be deployed in Europe, too. For that reason, GRS performed a study of safety and international development of SMR two years ago [1], which was presented as well in [2], [3] and [4]. The goal was to create an overview about current SMR designs in order to identify essential issues for reactor safety research, which are needed to specify special needs of adaptation of system codes used at GRS for reactor safety research. This is a precondition for performing safety assessments of these designs in the future.

While SMR stands for Small Modular Reactor in general, the IAEA uses this abbreviation for Small and Medium Sized Reactor neither excluding the modular character nor forcing it. The GRS study used both definitions. It consists of sound overviews of 69 SMR concepts divided into 32 LWR, 22 LMR, 2 HWR, 9 GCR and 4 MSR designs. Information gathered from public accessible sources (e.g. [5], [6]) including data of e.g. cooling circuits, core and safety systems. Safety relevant issues were identified using the German safety requirements for nuclear power plants and common fundamental safety functions. Finally it was evaluated, whether the different parts of the reactor design can already be simulated by GRS simulation tools [7] and where further code development and validation is necessary. This paper summarizes the outcome of this study.

1 INTRODUCTION

In the past, the power of commercially operated nuclear reactors became larger and larger, owing to shrinking specific costs per kWh. However, looking at current construction sites like Flamanville or Olkiluoto shows that this argument may not be valid anymore. The financial risk for the NPP vendors is currently very high, so that new builds are possible with massive subsidies by the countries only, like in the UK. This financial risk could decrease, if building NPPs with lower power output. However, while the overall costs of such a reactor are lower than current designs, the specific costs per kWh may be higher. For further cost reduction, the industry is designing so called Small Modular Reactors (SMR). They expect that such reactor designs would have a very compact design, which could be prefabricated in a central facility and transported to the construction side as a whole or in a few parts only by truck, train or ship. This prefabrication concept would lead to short production times, high qualities owing to standardization and the possibility of mass production.

The vendors also see the potential to deploy SMR in sparsely populated areas, where large power plants would suit hardly. These reactors could produce electricity, potable water and heat. They also could easily be upgraded by deploying additional modules.

Whereas since the 1950s the USA and USSR used small nuclear reactors to empower icebreakers and submarines, the idea of small reactors is not a new one. Currently five SMR concepts are under construction (CAREM, CNP-300, KLT-40S, HTR-PM and PFBR-500) and even three are operating (CEFR, CNP-300 and PHWR-220). The following table shows additional information about the mentioned SMR.

Table 1 SMR concepts currently operating and under construction

Name	Type	Manu-facturer	Coun-try	P [MW_e]	Status	Site
Currently operating						
CEFR	LMR	CIAE/ CNEIC	CN	20	Operating, Prototype for CDFR-1000	Tuoli (China)
CNP-300	LWR	CNNC	CN	325	Operating, additional planned	Qinshan 1 (China), Chashma (Pakistan)
PHWR-220	HWR	BARC	IN	236	16 operating, additional planned	Rajasthan, Madras, Narora, Kakrapar, Kaiga (India)
Currently under construction						
CAREM	LWR	CNEA	AR	27	Start of construction: February 2014	Atucha (Argentina)
CNP-300	LWR	CNNC	CN	325	2 blocks under construction	Chashma (Pakistan)
KLT-40S	LWR	OKBM Afrikantov	RU	35	2 reactors in Akademik Lomonosov, deployment: 2016	Akademik Lomonosov (Barge)
HTR-PM	GCR	INET	CN	105	Demonstration plant under construction since 2012 (2 modules)	Shidaowan (China)
PFBR-500	LMR	IGCAR	IN	500	Under construction, first criticality planned in mid of 2015	Kalpakkam (India)

The USA as well as the UK announced funding for development of SMR concepts. The funding in the USA (done by the US Department of Energy) has a volume of about $452 m and is intended for factory built SMR. The first funding round was won by B&W (mPower) in 2012, the second by NuScale in 2013 [5]. While it seems, that for NuScale a first construction site was found near the INL and the design certification application for US NRC is already under preparation [8], the funding for other designs was reduced. B&W for example announced in 2014 a financing decrease of the mPower to $15 m/a [9]. In the UK in 2016 a so-called small modular reactor competition was announced by the Department of Energy and Climate Change (DECC). The budget is about £250 m [10].

2 OVERVIEW OF SMR CONCEPTS AND SPECIAL FEATURES

Beside light water SMR (LWR), also heavy water reactor (HWR), gas cooled reactor (GCR), liquid metal cooled reactor (LMR) and molten salt reactor designs were covered by the GRS study. In the following Table 2, concepts with planned deployment and further concepts without deployment in the near future are shown.

Table 2 SMR concepts with planned deployment and without deployment in the near future

Name	Type	Manufacturer	Country	P [MW_e]	Status	Site
Concepts with planned deployment						
ACP-100	LWR	CNNC	CN	100	Planned construction (Start 2015)	Zhangzhou, later: Jiangxi, Hunan, Jilin
ALFRED	LMR	Int	Int	125	Planned construction (Start 2017)	Mioveni, RO
BREST-OD-300	LMR	NIKIET	RU	300	Planned construction	Beloyarsk, RU
CNP-300	LWR	CNNC	CN	325	Operating, additional construction planned	PK
G4M	LMR	Gen4 Energy	US	25	Planned construction	Savannah River, US
GT-MHR	GCR	Int	Int	285	Planned construction	Seversk, RU
MHYRRA	ADS-LMR	SCK CEN	BE	Heat only	Planned construction (Start 2015)	Mol, BE
PHWR-220	HWR	BARC	IN	236	16 operating, further planned	IN
RITM-200	LWR	OKBM Afrikantov	RU	175 MW_{th}	Completion expected: 2018, 2 more in 2019 and 2020	Icebreaker LK-60
SVBR-100	LMR	AKME	RU	101.5	Planned construction	RIAR in Dimitrovgrad
VK-300	LWR	RDIPE	RU	250	Planned construction (Current status unknown)	Kola peninsula, Archangelsk, Primorskaya
Further Concepts						
4S	LMR	Toshiba/CRIEPI	JP	10-50	Well-developed, possible construction site: Galena (Alaska)	
ABV-6M	LWR	OKBM Afrikantov	RU	6	Well-developed	

Adams Engine	GCR	Adams Atomic Engines Inc.	FR	10	2010 folded
AHWR300-LEU	HWR	BARC	IN	304	Well-developed, site selection started
ANGSTREM	LMR	OKBM Gidropress	RU	6	n/s
ANTARES/SC-HTR	GCR	AREVA	US	250	Developing phase
ARC-100	LMR	ARC LLC	US	100	Developing phase
ASTRID	LMR	CEA	FR	600	Conceptual design phase till 2015
CAP200	LWR	SNERDI/SNPTC	CN	200	Conceptual design finished
ELENA	LWR	Kurchatov Institut	RU	0.1	-
Em²	GCR	GA	US	240	Early state
ENHS	LMR	University of Calif.	US	50-75	Well-developed, demonstration plant till 2025
FBNR	LWR	Federal University of Rio Grande do Sul	BR	70	Early state
Flexblue	LWR	DCNS	FR	160	Developing phase
Fuji	MSR	TTS	Int	200	Market maturity planned till 2018-2025
GTHTR	GCR	JAEA	JP	274	Development after Fukushima doubtful
IMR	LWR	MHI	JP	350	Licensing earliest 2020
IRIS	LWR	Int	Int	335	Just before licensing of US NRC, needs investors
LSPR	LMR	Titech	JP	53	Developing phase
mPower	LWR	B&W	US	180	Well-developed, DOE funding, financing reduced since 2014
MRX	LWR	JAERI/ JAEA	JP	30	no up to date information available
NHR-200	LWR	INET	CN	Heat only	n/s
NIKA-70	LWR	NIKIET	RU	15	Apparently folded in favour of KLT-40S und VBER
NP 300	LWR	AREVA	FR	300	no current information available
NuScale	LWR	NuScale Power Inc.	US	45	Well-developed, funded by DOE
PB-AHTR	MSR	UCB/ORNL	US	410	Early state
PBMR	GCR	ESCOM	ZA	165	International commercialization
PEACER	LMR	NUTRECK	KR	300-550	Development phase, planned demonstration plant (PATER)
PRISM	LMR	GE-Hitachi	US	311	Well-developed, US NRC licensing pending
RADIX	LWR	Radix Power Systems	US	10-50	n/s
RAPID	LMR	CRIEPI	JP	1	Development phase
RAPID-L	LMR	CRIEPI	JP	0.2	Development phase
RUTA-70	LWR	NIKIET	RU	Heat only	Development phase, lacking funding
SC-GFR	GCR	SNL	US	100/200	Conceptual phase
SCOR600	LWR	CEA	FR	630	Development phase
SHELF	LWR	NIKIET	RU	6	Early design phase
SmAHTR	MSR	ORNL	US	50	Early design phase
SMART	LWR	KAERI	KR	100	Licensing completed
SMR-160	LWR	HOLTEC	US	160	Well-developed, US NRC licensing shall start in 2016
SSTAR	LMR	ANL/LLNL	US	20	Well-developed
STAR-LM	LMR	ANL	US	175	Development phase
STAR-H2	LMR	ANL	US	Heat only	Development phase, construction till 2030 planned
SVBR-10	LMR	AKME	RU	12	Development phase
TRIGA	LWR	GA	US	11,8	Focus of GA lies on GT-MHR and EM2
TSMR	MSR	SINAP	CN	45	Development phase
TWR	LMR	Terrapower	US	500	Construction of a demonstration plant between 2018 and 2022 planned
U-Battery	GCR	Int	Int	5-10	Development phase
UNITHERM	LWR	RDIPE/ NIKIET	RU	2.5-6.0	n/s
VBER-300	LWR	OKBM Afrikantov	RU	295-325	Well-developed
Westinghouse SMR	LWR	Westinghouse	US	225	Well-developed, decreased financing since 2014
WWER-300	LWR	OKBM Gidropress	RU	300	n/s

2.1 Characteristics of SMR

One goal of the GRS study was to assess the considered SMR designs, whether they can be simulated with the computer codes used at GRS. Therefore at first general features of all considered SMR were gathered, followed by characteristics of the used safety systems used in SMR to fulfil the fundamental safety functions [11]:

- Control of reactivity,
- Fuel cooling and
- Containment of radioactive substances.

The safety systems were put into a defence in depth scheme used by the German safety requirements for nuclear power plants, which contains four levels:

1. Normal operation
2. Anticipated operational occurrences
3. Accidents
4. Very rare events involving multiple failures and severe fuel assembly damages

While at first in this chapter general features of SMR are shown, in part 2.1.2 selected safety systems for decay heat removal, emergency core cooling, etc., mainly used in tier 3, for the different kinds of reactors are shown.

2.1.1 General features of SMR

2.1.1.1 Light water SMR

Some main characteristics of light water SMR can be found also in larger LWR. For example, the reactivity coefficients for void and temperature of fuel and coolant are negative. For reactivity control some concepts are boron free, which implies in general an even lower temperature coefficient of the coolant and space savings owing to a left out of a boron system. In order to achieve high burnups and cycle lengths, excess reactivity is compensated by burnable absorbers in the core (e.g. Gd_2O_3, IFBA, B_4C or Er) or by control rods, which are also used for short-term control of the reactor. Materials used for control rods are Ag In-Cd, B_4C and $Dy_2Ti_2O_7$. The following Table 3 shows light water SMR (all PWRs except the VK-300, which is a BWR) using boron acid or burnable absorber for compensation of excess reactivity. Furthermore, the mean power densities are shown, which are quite lower than common values of German PWRs of about 100 kW/l. Lower powers and lower power densities lead to slower transients in general.

Many of the SMR designs are proposed as an integral reactor. Integral means in general, that pressurizer and the steam generators are located within the reactor pressure vessel. SMR designers say that such a construction would exclude a large break loss of coolant accident (LBLOCA) by design, since no large connection lines are needed. In some cases, also the control rod drives are integrated into the reactor pressure vessel, excluding an accidental control rod ejection. Owing to the compactness, maximizing of heat transfer areas is done by choosing special heat exchanger geometries like helical tubes or plate heat exchanger.

Beside the integral design, also loop designs with very short coaxial connection nozzles can be found (e.g. KLT-40S, CAP-200, etc.). Here the hot leg is located in the inner pipe while the cold leg is in the outer part of the coaxial pipe in order to minimize temperature losses.

Many of the light water concepts are operating under natural circulation without the use of main coolant pumps (e.g. NuScale, CAREM, ABV-6M, etc.). Consequently, in these concepts, no pump trips have to be considered, but especially during start-up phase this may lead to flow instabilities like geysering or density wave oscillations, the designers have to deal with. Descriptions of such phenomena for the IMR design can be found in [12].

Table 3 Light water SMR using boron acid or burnable absorber for compensation of excess reactivity

Name	Thermal Power [MW_{th}]	Boron acid	Burnable Absorber	Planned FE cycle length	Planned mean burnup [MWd/kg_U]	Mean power density [kW/l]
ABV-6M	38	n/s	n/s	10-12 a	n/s	n/s
ACP-100	310	x	-	24 M	n/s	n/s
CAREM	100	-	x	48 M	n/s	n/s
CAP200	660	x	?	24 M	37	66.9
CNP-300	1.000	x	x	18 M	n/s	n/s
ELENA	3,3	-	-	21,7 a	27.39	7.1
FBNR	218	x	-	25 M	15.3	45
Flexblue	530	-	x	40 M	n/s	n/s
IMR	1.000	x	x	26 M	46	40
IRIS	1.000	x	x	30-48 M	40-65	51.26
KLT-40S	150	-*	x	28 M	45.4	119.3
mPower	530	-	x	48 M	35	n/s
MRX	100	-	x	42 M	22.6	n/s
NHR-200	200	-	x	60 M	30	n/s
NIKA-70	70	-	x	n/s	n/s	n/s
NP 300	1.000	n/s	n/s	18-28 M	n/s	n/s
NuScale	160	x	x	24 M	> 30	n/s
RADIX	40 - 200	n/s	n/s	10 a	n/s	n/s
RITM-200	175	n/s	n/s	7 a	n/s	72.0
RUTA-70	70	-	x	800 d	28.7	n/s
SCOR600	2.000	-	x	n/s	n/s	75.3
SHELF	28	n/s	n/s	56 M	n/s	n/s
SMART	330	x	x	36 M	31	62.62
SMR-160	525	-	-	n/s	n/s	n/s
TRIGA	64	-	x	n/s	35	n/s
UNITHERM	30	-	x	25 a	> 30	27.3
VBER-300	917	x	x	1-2 a	47.9	63.4
VK-300	750	-	x	18 M	41.4	n/s
Westinghouse SMR	800	x	x	24 M	n/s	n/s
WWER-300	850	x	x	24 M	65	n/s

* No H_3BO_3, but $Cd(NO_3)_2$ for emergency shutdown

2.1.1.2 Heavy water SMR

Both the AHWR-300 LEU and the PHWR-220, which are considered in the GRS study, are pressure tube reactors. In difference to the PHWR-220, where the tubes are arranged horizontally, the tubes in the AHWR-300 LEU case are arranged vertically. Owing to this arrangement, the AHWR-300 LEU is operating under natural convection even in normal conditions. In both cases, the pressure tubes are located within a large moderator tank named calandria filled with heavy water.

2.1.1.3 Gas cooled reactors

The core of gas cooled reactors may have either a pebble bet structure or is designed of a hexagonal graphite structure with drilled holes for fuel, control rods and coolant. The neutron spectrum can be fast or thermal. Used coolants are helium and nitrogen.

Core cooling is done in all treated concepts by forced convection. Mainly two types of reactivity control are used: Either by inserting of control and absorber rods or by using so called control cylinders. These cylinders are integrated into the lateral located reflectors. They are vertically divided into two parts: one made of reflector material and one made of absorber material. The cylinders are mounted on a mandrel and can be rotated in order to turn the respective part towards the core.

Residual heat can be removed by natural circulation, heat conduction and radiation when forced air-cooling is unavailable. The high temperatures of the fuel of about 1,600 °C needed for the high temperature gradient between fuel and environment can easily be covered by the design temperature of the used fuel.

2.1.1.4 Liquid metal cooled SMR

Normally liquid metal type SMR have a so-called pool type design, which means, that the most parts of the reactor (core, steam generator, intermediate heat exchangers, pumps, etc.) are located in a large pool filled with liquid metal. Used coolants are sodium, lead or LBE. Owing to the high saturation temperatures of the used coolants (Na: 883 °C, Pb: 1,749 °C, LBE 1,670 °C) the pressure inside the reactor vessel can be maintained at atmospheric pressure or just slightly above (e.g. 4S: 3 bar, CEFR: 6 bar). This low pressure differences decrease the possibility for a LOCA. In the event of a LOCA, however, the pressure inside the containment is not going to rise, since there will be no evaporation effects like in LWRs.

2.1.1.5 Molten salt SMR

There are mainly two types of molten salt SMR: the ones with liquid and with solid fuel. In the solid case, the fuel is located within the core in holes of hexagonal shaped graphite blocks with additional channels for the coolant or inside fuel pebbles. This kind of core is quite similar to those of gas-cooled reactors. The case of liquid fuel the fuel is located within the coolant but not critical outside the core. It becomes critical inside the core owing to reflector and moderator materials. Reactivity control is done by control rods and burnable absorbers. Liquid fuel reactors can be shut off by draining the coolant into storage tanks.

2.1.2 Characteristics of safety systems

2.1.2.1 Light water SMR

For decay heat removal, both active and passive systems are used. The passive possibilities are shown in Figure 1. Mainly five different ways were realized.

In option A and B, a heat exchanger is connected to the steam generator. Steam is flowing into the heat exchanger. It condenses there and the condensate is flowing back to the steam generator and is evaporating again. The heat exchanger can be cooled either by a water pool (A) or by airflow (B).

In option C, a closed circuit is installed connected by heat exchangers to the primary system and to a heat sink (e.g. a water pool). No activated fluid leaves the primary systems. In difference to that in options D and E the cooling circuit is coupled to the primary system. Here single phase and two-phase flow is possible.

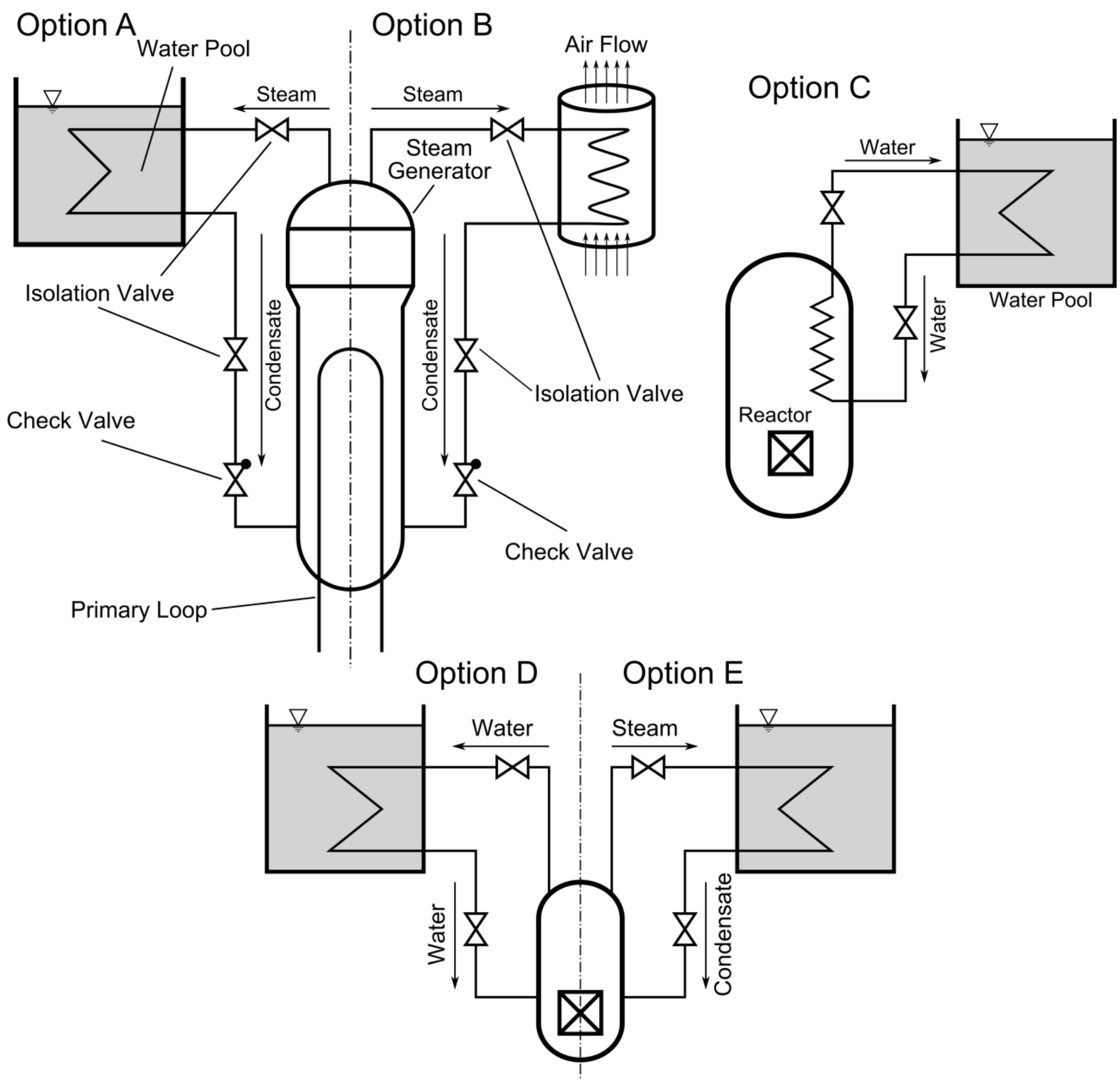

Figure 1 Possibilities in light water SMR to remove decay heat [13]: steam generator connected with extra loop to heat exchanger with water (A) or air (B) cooling, closed loop with heat exchanger inside RPV (C), primary system connected on heat exchanger (single phase (D), two phase (E))

During a LOCA, it has to be ensured, that the core is still cooled by replacing the lost coolant by means of emergency core cooling systems (ECCS). In Figure 2, several ECCS used in SMR concepts are shown. The option (1) in Figure 2 is the injection of coolant with accumulators. They inject water into the primary system, when the primary pressure drops under a certain level and the corresponding check valves open or rupture disks burst. Option (2) in Figure 2 is a core make-up tank, connected to the primary system on an elevated level. When opening the corresponding valves, the water inside the make-up tank flows into the primary system. It is possible to inject the water of the make-up tank also at high pressures. In contrast to that, connected external pools (see (5)), also elevated, can inject water at low pressures only, since the driving force is just the geodetic pressure gradient.

A large amount of steam is entering the containment during a LOCA, condensing on the containment internals and the containment inner wall. The condensate is accumulating in the sump. Some designs are providing an active emergency injection with a pump and a heat exchanger in order to reinject the lost water inventory back to the primary circuit (see (3) in Figure 2). In other concepts, the sump is a quite narrow gap, where the liquid level is rising fast because of the inflowing condensate. In that case, the condensate can drain back to the pressure vessel by a direct vessel injection (DVI) or recirculation valves built in the reactor pressure vessel wall passively driven by the geodetic pressure drop (see (4) in Fig. 2).

Steam in the containment atmosphere can also drain into a pressure suppression pool, like in current BWR concepts with dry well and wet well. However, in difference to that a pump may feed the water back from the pool to the primary circuit.

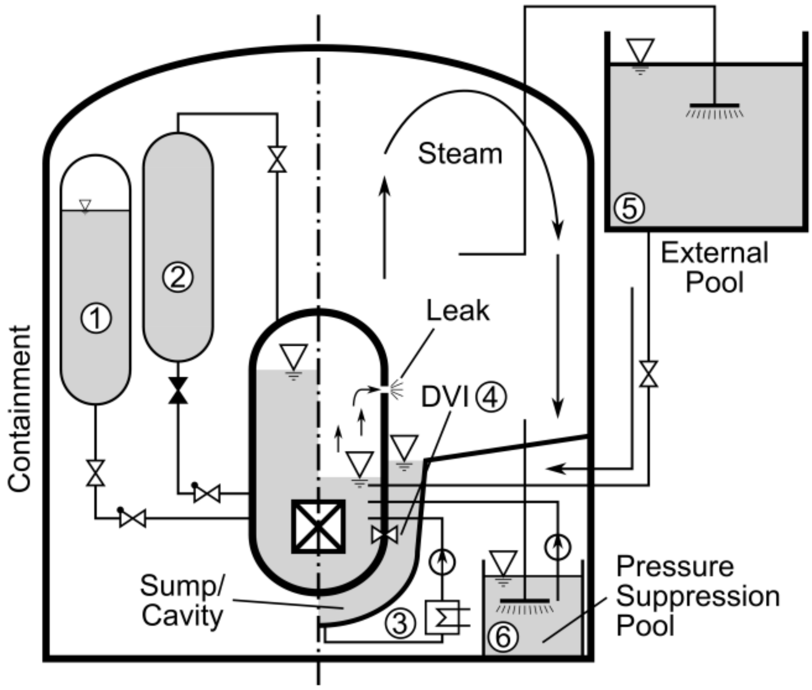

Figure 2 Systems for emergency core cooling;
(1)… Accumulator
(2)… Core Make-Up Tank
(3)… Active emergency injection (from sump)
(4)… Direct Vessel Injection (DVI) / Recirculation Valve
(5)… High leveled external water pool
(6)… Active emergency injection (from PSP)

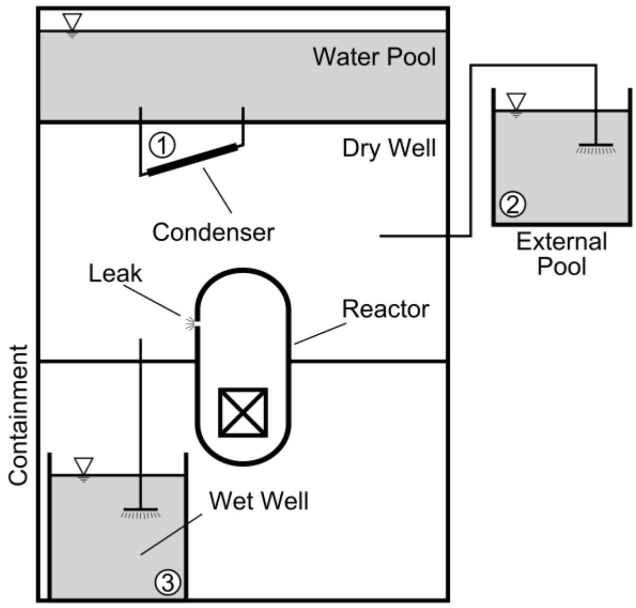

Figure 3 Pressure suppression systems of the containment;
(1)… Containment Condenser
(2)… Condensation in external pool
(3)… Condensation in wet well

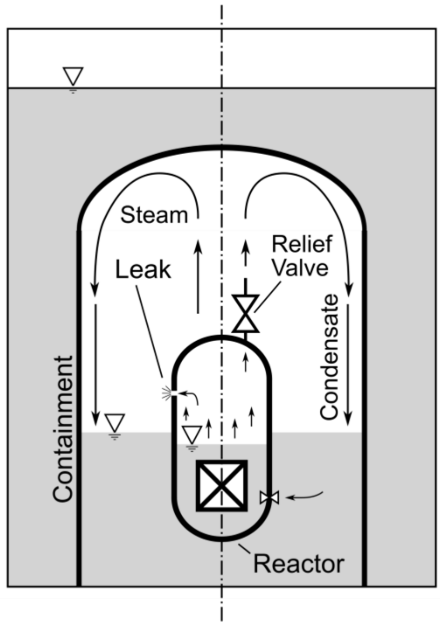

Figure 4 Pressure suppression by convective cooling of the containment

Table 4 Selected safety systems of light water SMR

<table>
<tr><th colspan="2">Principle</th><th>Reactor</th></tr>
<tr><th colspan="3">Decay Heat Removal</th></tr>
<tr><td rowspan="2">Steam generator cooling</td><td>Passive with water pool</td><td>IMR, IRIS, KLT-40S, NuScale, SMART, VBER-300</td></tr>
<tr><td>Passive with air flow</td><td>ELENA, mPower (air flow may achieved by fan) , IMR, NuScale</td></tr>
<tr><td rowspan="4">Primary Side</td><td>Passive in water pool (one phase natural convection)</td><td>ACP-100</td></tr>
<tr><td>Passive in water pool (two phase natural convection)</td><td>ACP-100, CAREM, mPower</td></tr>
<tr><td>Passive in water pool</td><td>CAP200, Flexblue, MRX</td></tr>
<tr><td>Passive by extra loop</td><td>SCOR600, TRIGA, SMR-160, Westinghouse SMR</td></tr>
<tr><td colspan="2">Active auxiliary systems</td><td>KLT-40S, SMART, VBER-300</td></tr>
<tr><th colspan="3">Emergency Core Cooling</th></tr>
<tr><td colspan="2">Accumulator</td><td>ACP-100, CAREM, CNP-300, IMR, KLT-40S, RITM-200, VBER-300, WWER-300</td></tr>
<tr><td colspan="2">Active low and/or high pressure injection</td><td>CAP200, KLT-40S, SCOR600, SMART, UNITHERM, VBER-300, WWER-300</td></tr>
<tr><td colspan="2">Make-Up-Tank</td><td>ACP-100, CAP200, CAREM, CNP-300, IRIS, SMR-160 (poss.), Westinghouse SMR</td></tr>
<tr><td rowspan="2">Higher water pool</td><td>Inside containment</td><td>ACP-100, CAP200, mPower</td></tr>
<tr><td>Outside containment</td><td>VK-300</td></tr>
<tr><td rowspan="4">Long-time cooling</td><td>Passive with sump/cavity or from top of the RPV</td><td>ACP-100, CAP200, Flexblue, IRIS, NuScale, SMR-160, Westinghouse SMR</td></tr>
<tr><td>Active with sump/cavity</td><td>KLT-40S</td></tr>
<tr><td>Active with pressure suppression pool</td><td>SCOR600</td></tr>
<tr><td>Passive with external pool</td><td>VK-300</td></tr>
<tr><th colspan="3">Primary Depressurization</th></tr>
<tr><td colspan="2">Relief in water pools/tanks</td><td>ACP-100, CAREM, CAP200, CNP-300, Flexblue, IRIS, mPower, SMART, TRIGA, VK-300, WWER-300</td></tr>
<tr><td colspan="2">Relief in containment</td><td>ACP-100, CAP200, NuScale, UNITHERM (poss.), VBER-300 (poss.), Westinghouse SMR</td></tr>
<tr><th colspan="3">Pressure Suppression in the Containment</th></tr>
<tr><td colspan="2">Wet well/Pool</td><td>CAREM, Flexblue, IRIS, KLT-40S, SCOR600, VK-300</td></tr>
<tr><td colspan="2">Containment condenser</td><td>ACP-100, KLT-40S, VBER-300</td></tr>
<tr><td colspan="2">Spray in containment</td><td>CNP-300, SMART</td></tr>
<tr><td colspan="2">Containment surrounded by water</td><td>CAP200, NuScale, SMR-160, Westinghouse SMR, FLEXBLUE, MIT offshore, ACP-100</td></tr>
<tr><th colspan="3">Additional Components</th></tr>
<tr><td colspan="2">Flow limiter</td><td>KLT-40S, VBER-300</td></tr>
<tr><td colspan="2">Venturi nozzles</td><td>SCOR600, TRIGA</td></tr>
<tr><td colspan="2">Heat Pipes</td><td>CAP200</td></tr>
</table>

Depressurization measures of the primary system are steam relief from the pressurizer into a water tank (like a pressurizer relief tank in current PWR), into a pressure suppression pool or into the containment atmosphere. When the relief tank is located above the reactor pressure vessel, some designs have the possibility of a back flow of its inventory into the vessel.

When steam enters the containment atmosphere, either by a LOCA or due to steam relief from the pressurizer, the pressure inside the containment is rising. In order to supress the pressure inside the containment, several systems are provided by the SMR designers, which are shown in Figure 3. Some designs have a containment cooling condenser (see (1) in Figure 3). This heat exchanger is connected with a large water pool, generally on the top of the containment. Steam in the containment atmosphere is condensed on the outer tube walls of the condenser. Water inside the condenser is heated up and possibly evaporating. The heated fluid is flowing upwards into the water tank and is condensing there. Another possibility is to guide the steam inside the containment into water pools distributed by a sparger. This water pools can be located inside the containment like a wet well (see (3) in Figure 3) or outside (see (2) Figure 3).

In Figure 4 a design is shown, were the pressure inside the containment is decreased by cooling the containment from the outside by a large water tank, in which the containment is immerged. Steam entering the containment by a leak or a relief valve is condensed on the inner containment wall and the condensate flows to the bottom and may even be reinjected into the primary system by a recirculation valve. Another possibility is to spray water into the containment atmosphere in order to condense the incoming steam.

In Table 4, the safety systems of the SMR based on light water technology are compiled. At the bottom, also so-called additional components are mentioned. Flow limiters are located mainly in connecting pipes and are used for providing different pressure losses for different flow directions. When using it in a cold leg, it would limit the flow in a case of a cold leg LOCA in the direction from the reactor pressure vessel to the leak. In normal direction from the steam generators to the reactor pressure vessel, the limiter has only little influence. Another component is the so-called venturi nozzle used for blocking and clearing flow paths, depending on the flow velocity. Finally one light water concept is using heat pipes for transferring heat over a certain distance.

2.1.2.2 Heavy water SMR

Table 5 Selected safety systems of heavy water SMR

Principle	Reactor
Decay Heat Removal	
Passive with secondary steam relief or condenser	PHWR-200
Passive with isolation condenser in large water pool	AHWR-300 LEU
Active calandria cooling	PHWR-200, AHWR-300 LEU
Emergency Core Cooling	
Accumulator	PHWR-200, AHWR-300 LEU
Active with sump	PHWR-200
Higher water pool	AHWR-300 LEU
Active injection by fire fighter pumps	PHWR-200
Pressure Suppression Containment	
Wet well/Pool	PHWR-200, AHWR-300 LEU
Passive containment condenser	AHWR-300 LEU
Active fan cooler	PHWR-200
Passive concrete structure cooler	AHWR-300 LEU

Safety systems of SMR based on heavy water technology are quite similar to those of the light water SMR mentioned above. Decay heat removal can be done apart from normal operation systems by using the steam generators and guiding the steam to the condensers or relieving it into the containment atmosphere. In addition, a passive cooling by an isolation condenser or active calandria cooling is used.

For emergency core cooling accumulator tanks (tiered for different pressure levels), active injection of sump water, a high-levelled water pool for passive injection of water and the possibility of using fire fighter pumps are provided by the different heavy water SMR designs.

Pressure suppression of the containment is done by active fan cooler, containment condenser, pressure suppression pools and cooler on the concrete surfaces of the containment, which are cooled passively.

The above-mentioned systems are summarized in Table 5.

2.1.2.3 Gas cooled SMR

Since in gas cooled reactor designs decay heat can be transported safely to the environment by natural convection, thermal conduction and radiation, only few systems are provided by the designers to support the heat removal. This is possible owing to low power densities and high temperature reliabilities of the reactor structures. At first, decay heat removal can be done by so-called direct auxiliary cooling systems (DHRS). In the case of the EM^2, this system is working passively. Here the decay heat is transported by natural convection to an auxiliary cooling circuit, which is working with water as coolant. This circuit transports the heat to a water pool. In a case, when there is no water inside the auxiliary circuit, a second system is installed, cooling the primary system with airflow with natural circulation. The heat is transferred in this case to the environment. Other possibilities to support the decay heat removal is to place cooling bodies inside the reactor cavity, which are heated by thermal radiation and convection and which are finally cooled by water or air circuits.

In order to prevent the containment of an over pressure, most of the treated designs provide a filtered or unfiltered venting system. The concepts Adams Engine and EM^2 are provided by a full pressure containment. This, in turn, improves the heat transfer from the core to the environment, due to higher densities and specific heat capacities.

The above-mentioned systems are summarized in Table 6.

Table 6 Selected safety systems of gas cooled SMR

Principle		Reactor
Decay Heat Removal		
Direct auxiliary cooling system	Passive with water circuit	EM^2
	Passive with air circuit	EM^2
	Active with condenser	All
Cooling of reactor cavity	Passive with closed water circuit	ANTARES, HTR-PM, PBMR
	Passive with open air circuit	GT-MHR, GTHTR
	Direct heat conduction to environment	Adams Engines, GT-MHR, PBMR, U-Battery
Pressure Suppression Containment		
No system designated		Adams Engine, EM^2
Venting	Filtered	GTHTR, PBMR
	Unfiltered	HTR-PM, PBMR
	No data	ANTARES, GT-MHR, U-Battery

2.1.2.4 Liquid Metal cooled SMR

Within liquid metal cooled SMR, in general decay heat removal is done with three different kinds of systems: Systems, which provide additional cooling circuits, systems cooling an intermediate loop and systems, which cool surfaces directly. DRACS, DHRS and SGAHRS belong to the first kind. The so-called DRACS (Direct Reactor Auxiliary Cooling System) and DHRS (Decay Heat Removal System) have immerged heat exchanger inside the hot pool. This heat exchanger is connected with a separate circuit, cooled passively at a higher level by airflow ((1) in Figure 5). While the DRACS or DHRS is cooling the primary side, the SGAHRS (Steam Generator Auxiliary Heat Removal System) is cooling the water/steam circuit of the LSPR (3). In the steam drum, a heat exchanger is located, removing the heat to an air cooler.

In the case of the IRACS (Intermediate Reactor Auxiliary Cooling System) sodium/air heat exchanger is located in the intermediate sodium loop of the 4S to remove the decay heat.

Finally, some designs provide systems cooling surfaces. For example the RVACS (Reactor Vessel Auxiliary Cooling System, or in the TWR case: Reactor Vessel Air Cooling System): This is a convective external air-cooling of the reactor vessel or the containment vessel (shown in (2) of Figure 5). The system consists of multiple air inlets. The air is guided to the bottom and is then distributed over the vessel surface and cooling it. The heated air is released to the environment by chimneys. Another possibility is to cool the steam generator outer surfaces by airflow, shown in Figure 5 (4). This system is called ACS (Auxiliary Cooling System). In the BREST-300 case, air channels are introduced into the reinforced concrete structure surrounding the reactor vessel, cooling it by an air flow.

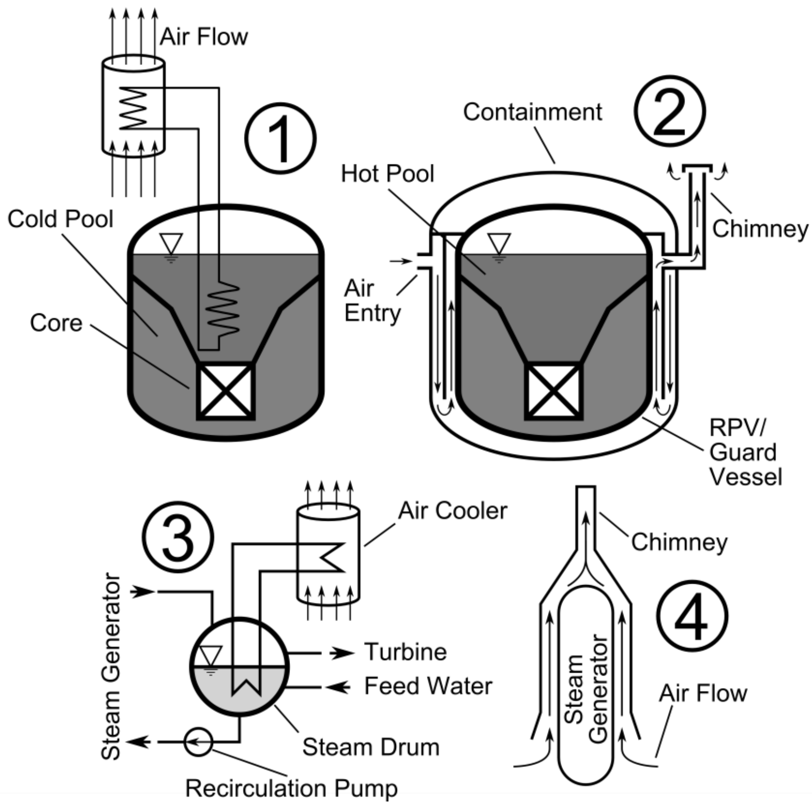

Figure 5 Decay heat removal systems for liquid metal cooled SMR; (1) DRACS/DHRS [14], (2) RVACS [15], (3) SGAHRS [15], (4) ACS [16]

The treated liquid metal cooled SMR do not provide any special systems for emergency injection. The pressure of the primary system is mainly limited to atmospheric pressure since the saturation temperatures are very high. Most of the SMR provide a so-called guard vessel made of steel or reinforced concrete. This vessel surrounds the reactor vessel by a thin gap only. It is designed in such a way, that the core keeps covered by the coolant even in a case of a LOCA.

From special interest are steam generator ruptures. Using sodium as primary coolant, it could react with the steam to hydrogen and sodium hydroxide. This reaction is very exothermic. In order to lower the risk, that contaminated sodium react with steam, all sodium reactors have an intermediate sodium circuit. In some concepts (e.g. 4S and PRISM), sodium storage tanks are located below the steam generators. Rupture disks burst, when the pressure increases owing to a sodium steam reaction and unblock the tanks. The sodium is then finally drained into the tanks and is finally being isolated. In the 4S case, double walled steam generators are used to limit the risk of a sodium steam reaction.

While LBE and lead reactors usually do not need an intermediate circuit, some of them provide one. In the BREST-300, LSPR and SVBR-10/100 case, the steam generators are located within the hot pool of the reactor. In an inert gas room above the liquid level of the coolant, steam is collected in the case of a steam generator tube rupture. The BREST-300 and LSPR design provide a pressure suppression pool to limit the pressure inside the reactor vessel. In the SVBR case, the inert gas chamber is connected to a cooled condenser as well as to a large water pool surrounding the reactor vessel separated by a rupture disk. A small release of steam by a small tube rupture only leads to a low pressure increase covered by the condenser. When the steam release is higher, due to a larger break, the pressure increase is much higher and the disk is bursting. Thus lets the steam flow into the water pool and condense there.

Table 7 Selected safety systems of liquid metal cooled SMR

Principle		Reactor
Decay Heat Removal		
Separate cooling circuits	DRACS/DHRS*	ARC-100, CEFR*, PFBR-500 SSTAR, STAR-H2, STAR-LM, TWR
	SGAHRS	LSPR
Cooling of intermediate circuit	IRACS	4S
Direct cooling of surfaces	RVACS	4S, ARC-100, ENHS, LSPR, PEACER, PRISM, SSTAR, STAR-H2, STAR-LM, TWR
	ACS	PRISM
	Air channels inside reinforced concrete structure	BREST-OD-300

2.1.2.5 Molten salt SMR

As already mentioned, in general, there are two types of molten salt SMR: Reactors with solid fuel and reactors with fuel within the coolant. On the left side of Figure 6, decay heat removal systems of molten salt SMR with solid fuel are shown. In normal operation, the fluid flows from the core bottom through the core and is then pumped down through the intermediate heat exchanger back to the core bottom. The heat is finally removed to the main heat sink or the residual heat removal system. If the pumps are unavailable, the flow path in the primary system will change. Without the pumps, a natural circulation is established upwards through the core and then downwards through the auxiliary heat exchanger and the so-called fluid diode back to the core bottom. Heat is removed from the system to the environment. The fluid diode is constructed in such a way that in one direction the pressure drop is much higher than in the other direction. Without the diode, the fluid would flow through the auxiliary heat exchanger as well as through the core. However, in normal operation, with the diode in reverse direction, this flow path is blocked. If both the residual heat removal system and the auxiliary heat exchanger were unavailable, the heat would be removed by radiation and convection to the housing structures and from there to the environment.

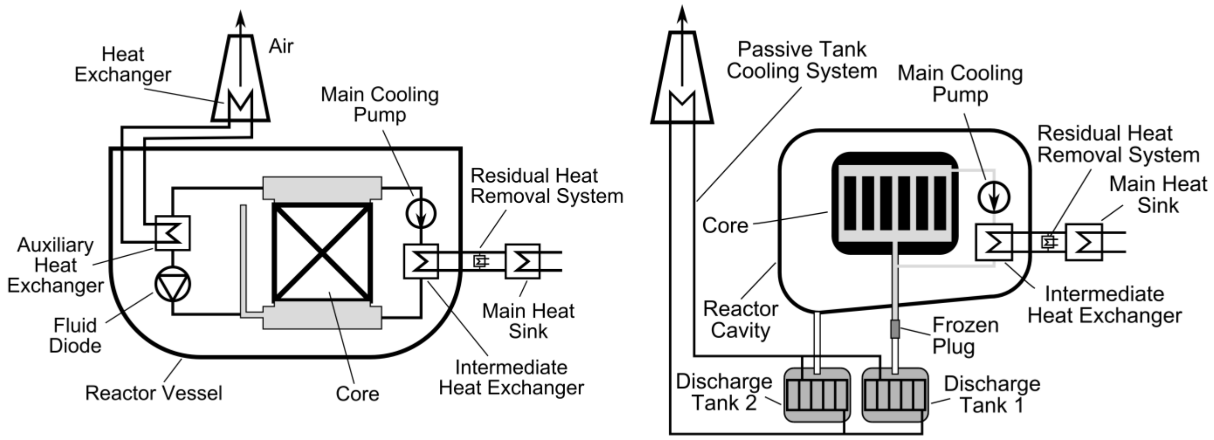

Figure 6 Decay heat removal systems in MSR with solid fuel (left) and liquid fuel (right)

The right picture of Figure 6 shows the heat removal concept of SMR with liquid fuel. In normal operation, the heat is removed by the intermediate heat exchanger to the main heat sink or the residual heat removal system. This is also possible by natural circulation in the primary system, when the pump is unavailable. If the heat removal is faulted, the temperature of the coolant rises. Owing to the high temperatures, the frozen plug is melting giving free a flow path to discharge tank 1, where the coolant is stored and cooled passively. During a LOCA, the fluid would flow into the reactor cavity, which is the lowest point, and from there into the discharge tank 2, also passively cooled. Inside the discharge tanks the fluid remains in a subcritical state.

All molten salt SMR have an integral designed primary system. Thus, a LBLOCA is excluded. The reactor vessel is surrounded by a guard vessel filled with a buffer salt, which reduces the discharge of contaminated salt into the gap between reactor vessel and guard vessel. Finally, it is possible to inject additional salt into the reactor cavern in the case of a LOCA in the reactor vessel and the guard vessel, to ensure a covered core.

For reactors with liquid fuel no additional measures have to be considered for a LOCA event, since the coolant with the fuel is cooled within the discharge tanks.

Table 8 Selected safety systems of molten salt SMR with solid and with liquid fuel

Principle		Reactor
Decay Heat Removal		
Natural convection in primary loop	Residual heat removal system	FUJI
	Main heat sink	SmAHTR, TMSR
Primary loop	Direct auxiliary heat removal system	PB-AHTR, SmAHTR, TMSR
Heat conduction through structural components		PB-AHTR
Passive cooling of discharge tanks		FUJI
Emergency Core Cooling		
Double wall (integral system)		PB-AHTR, TMSR
Flooding of reactor cavity with stored salt		PB-AHTR, TMSR

3 SELECTED NECESSARY IMPROVEMENTS OF THE GRS CODES FOR SAFETY ASSESSMENTS

The overview about the different SMR designs in [1], which was presented in the chapter before, was the basis to identify special improvement needs for computer codes used at GRS for assessment of nuclear plants. While the code ATHLET (Analysis of Thermal-Hydraulics of Leaks and Transients) is used for calculation of the behaviour of the cooling circuit, the tool COCOSYS (Containment Code System) simulates the fluid behaviour outside of the circuits in the containment. Core calculations are done by tools like QUABOX/QUBBOX. Coupling the tools with each other is also possible and provides more detailed simulation than stand-alone calculations.

Currently in ATHLET and COCOSYS, several working fluids can be simulated, summarized in Table 9. Within ATHLET, the most important working fluids light and heavy water, lead, LBE, sodium as well as helium and other gases can already be simulated. SMR using the fluids NaK, Lithium, CaBr and R-114 are ARC-100, TWR, RAPID-L, STAR-H2 and TRIGA. As mentioned in Table 2, the construction of a first reactor of one of these SMR concepts is in the far future, so the implementation of the missing components has a low priority only. Missing gases can easily be input by tables (regarding the specific gas constant, specific heat capacity and specific enthalpy) into the ATHLET data set.

While in COCOSYS all of the needed gaseous working fluids are already implemented and useable, the only available liquid component is light water. That means for example, a coupled calculation of a liquid metal cooled reactor is currently limited in such a way, that no liquid metal may enter the COCOSYS domain.

Implementation of new components means not only the input of the fluid properties but also the validation and, when needed, adaptation of the corresponding correlations for friction and heat transfer.

Table 9 Availability of components used in the treated SMR concepts in ATHLET and COCOSYS

Component	ATHLET	COCOSYS
Light Water	x	x
Heavy Water	x	-
Lead	x	-
LBE	x	-
Sodium	x	-
NaK-Eutekticum	-	-
Lithium	-	-
CaBr	-	-
R-114	-	-
Molten Salts	-	-
Helium	x*	x
Nitrogen	-*	x
CO_2	(available aus user defined gas)*	x
Air	-*	x

* Part of the multicomponent model of ATHLET

Beside of the fluids, also structure and fuel materials have to be considered. Within ATHLET it is possible to input missing material data by tables, regarding heat capacities, heat conductivities and densities. Already implemented are UO_2, MOX, circaloy, ferritic and austenitic steel, SiC and graphite. For core calculations with neutron kinetic programs it is more complicated. Tools like QUABOX/QUBBOX are using in general basic neutronic data from databases like ENDF-VII or JEFF 3.1. A special data processing sequence is needed to receive the macroscopic cross sections needed for calculations. This sequence is currently validated for thermal light water reactors only.

Because of economical and proliferation reasons, the reactor core should be replaced as a whole and very rarely. The resulting fuel cycle lengths are very long (4 – 5 years, up to 30 years, current GRS experience: 12 – 22 months). To achieve this, the burn ups are very high (70 – 75 MWd/kg, current GRS experience: 50 MWd/kg) and the enrichment rates can be higher than 5 %. In order to compensate the excess reactivity the concepts are using boron acid, burnable absorber and/or control rods. Some concepts are leaving out a boron system since it makes the temperature coefficient of the coolant more negative, it safes space and for some coolant it is not applicable (e.g. He). While burnable absorbers can only be used at the beginning of a cycle, because of their fast burnup, essentially only control rods and moveable reflectors remain for long-term compensation of excess reactivity, leading to large inhomogeneity in the power distribution over the core. Phenomena related to compensation of excess reactivity without a boron system are not validated yet for GRS codes.

Table 10 Core properties of the different SMR designs

Designs	LWR	HWR	MSR	GCR	GCR	LMR
Neuronic spectrum	thermal				fast	
Coolant	H_2O	H_2O/D_2O	Molten salt	He, N_2	He, CO_2	Pb, LBE, Na
Fuel	U	U, Th	U, Th	U, Th	U	U, Pu
Cycle length	From 12-22 months, max. 22-48 months			18-24 months	4-5, max. up to 30 years	
Planned burnup	15-90 MWd/kg	7-40 MWd/kg	-	ca. 70 MWd/kg	70-110 MWd/kg	60-100 MWd/kg
Planned enrichement	Up to max. 20 %				Upt to max. 20 % Some up to 50 %	
Moderator	H_2O	D_2O	Graphite		-	
Core grid	quadratic, hexagonal	hexagonal	Molten salt, spherical	Hexagonal-Block, sperical	Hexagonal-Block	hexagonal
Structural materials						
Cladding	Circaloy, Zr-Nb, E110	Circaloy	-	Steel, Graphite, SiC	Steel	Steel: HT-9, Fe-Cr-Al, D9, EP-823
Reflector	Wasser, Stahl	D_2O	Graphite	Graphite, BeO	Be_2C	Steel, stainless steel
Control rods	Ag-In-Cd, B_4C, B_4O, $Dy_2Ti_2O_7$	B_4C	B_4C	B_4C	B_4C, Hf	B_4C, Hf
Burnable absorber	Gd_2O_3, B_4C, IFBA, Er	-	-	-	-	-
Excess reactivity compensation / Power control	Born acid, burnable absorber, control rods	Boron acid, control rods	Control rods	Control rods	Control rods	Control rods, moveable reflectors

Safety systems in several SMR concepts use heat exchangers with geometries, which differ from the usual designs, known from current built plants. While vertical (U- or straight tube) heat exchangers or horizontal steam generators of the VVER-kind are well validated, other geometries need more investigations. More validation work is needed for slightly inclined horizontal heat exchangers (e.g. CAREM), helical steam generators (e.g. NuScale), bayonet heat exchanger (e.g. SCOR) or plate heat exchanger (e.g. TWR). The heat sink of the heat exchangers are often large water pools, where 3D flow phenomena occur like thermal stratification and 3D flows, which may have an effect on the heat transfer into the pools. Since ATHLET is a 1D system code, 3D related phenomena can be simulated roughly by special nodalization schemes or using the 3D model of ATHLET, which is currently under development. Another possibility is the coupling with a CFD code. Simulation of 3D related phenomena with COCOSYS is also limited. Possibilities are to couple COCOSYS with ATHLET and using the 3D model of ATHLET or ATHLET/CFD coupling or using the CoPool tool, developed by the Fraunhofer-Institut für Techno- und Wirtschaftsmathematik ITWM. When coupling with CoPool, COCOSYS can simulate one phase 3D flows [17].

Finally, another important point is related to the simulation of natural convection in SMR with very compact design. Natural circulation is used in various SMR designs not only in accident cases, but also under normal operation. ATHLET, however, was primarily developed for forced convection conditions within generation II plants. It has to be validated, whether ATHLET is able to simulate free convective flows induced by small driving forces (e.g. small pressure differences). Of course, that includes also the behaviour of used passive safety systems. Here, validation against to single component tests of the passive systems is needed in order to determine their behaviour. Additionally validation against integral tests is important, too. While the different systems (passive and active) are interacting with each other, uncertainties become more and more important and may influence the behaviour of the whole system massively [18].

4 SUMMARY

In this paper, an overview about the result of the GRS study for safety and international development of SMR done in 2014 is presented. The different concepts were reviewed carefully using public available information only. Using the gathered information the safety systems of the different SMR were divided into the German safety requirements for nuclear power plants and common fundamental safety functions. Finally it was assessed, what code improvements and validation work was necessary to perform safety assessments of the treated SMR designs. The most important points are:

- Implementation and validation of new working fluids corresponding correlations (friction, heat transfer, closing equations, etc.)
- Adaptation of heat transfer correlations for new heat exchanger geometries (plate heat exchanger, helical heat exchanger, etc.)
- Completion and validation of the 2D/3D model of ATHLET
- Implementation and validation of new components (e.g. venturi)
- Validation of the integral behaviour of passive safety systems
- Analysis of uncertainties of the nuclear basic data used for core calculations for new materials (e.g. fuel, structures)
- Validation of the complete nuclear calculation chain for the new reactor concepts

5 ACKNOWLEGDEMENTS

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OECD/NEA THAI program for containment safety research: main insights and perspectives

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Abstract:

The OECD/NEA THAI joint research program aims at investigating open questions on fission product and hydrogen behaviour in the containment of water cooled reactors in addition to the national THAI research program which started in the year 1998. First two phases of the OECD/NEA THAI program, namely THAI (2007 - 2009) and THAI-2 (2011 - 2014) have been successfully completed. The ongoing phase 3 of the OECD/NEA THAI project was launched in February 2016 for the duration of 3.5 years.

Experimental data produced in OECD/NEA THAI program have been continuously used for the validation and development of lumped parameter (LP) and computational fluid dynamic (CFD) based codes in the area of reactor safety. Major progress in measuring and analysing spatial hydrogen distributions, slow hydrogen deflagrations, performance of Passive Autocatalytic Recombiners (PAR) under accident-typical conditions, and fission product distribution and their interaction with PAR has been demonstrated with the THAI test facility representing conditions with regard to the containment. Code models have been improved based on THAI experimental data. Model validation and code applications using the complex experimental results confirmed the progress made, e.g. on hydrogen distribution (OECD/NEA THAI HM-2 code benchmark), PAR performance (OECD/NEA THAI-2 HR-35 code benchmark), and hydrogen combustion behaviour (ISP-49). Important progress has been demonstrated also in modelling and analysing aerosol and iodine behaviour in the containment and the coupling of such phenomena with containment thermal hydraulics in severe accident analysis codes.

The present paper provides main insights gained from the previous two phases of OECD/NEA THAI projects and remaining open issues are discussed; some of them are being investigated in the ongoing OECD/NEA THAI-3 project.

1 INTRODUCTION

Safety assessment and accident management in nuclear power plants (NPP) necessitate investigating complex phenomena and processes with adequate accuracy. In support of such activities, THAI national and OECD/NEA THAI joint research programs investigate open questions on fission product and hydrogen behaviour in NPP containments.

The experimental investigations carried out in the frame of THAI projects have contributed significantly to hydrogen and fission products related issues under severe accident conditions. Experiments are performed by use of representative aerosol and H_2 concentrations and thermal-hydraulic conditions. Spatio-temporal fission product behaviour can be studied by use of radiotracer I-123.

Details of OECD/NEA THAI projects [1, 2] and their respective experimental work programs are provided in Appendix 1; main insights gained from previous projects and remaining open issues are discussed in the paper. A wider overview of experimental investigations conducted mainly during first phase of OECD/NEA THAI projects as well as detailed overview of experiments conducted in national programs during time frame of 2000 – 2015, including their application for code verification and validation purposes, is provided in [3, 4, 5].

2 TEST FACILITY

The technical-scale THAI test facility (Figure 1) is operated by Becker Technologies in close co-operation with AREVA, Erlangen, and GRS, Cologne. THAI$^+$ is the extension of the containment test facility THAI, which construction was most recently completed. The original facility and its extension are designed to allow investigating thermal hydraulic processes in the atmosphere of NPP containments during postulated accidents. Furthermore, the behaviour of hydrogen and fission products (iodine and aerosols) is investigated. THAI$^+$ is an acronym for **T**hermal-hydraulics, **H**ydrogen, **A**erosols and **I**odine in multiple (**+**) compartments.

The main components of the facility are two connected (DN500 piping) cylindrical steel vessels, the THAI Test Vessel (TTV: 60 m^3, 9.2 m high, and 3.2 m in diameter) and the new vessel called PAD (Parallel Attachable Drum: 17.7 m^3, 9.73 m height, and 1.6 m diameter), with a sump compartment at the lower end of each vessel (Figure 1). Vessel and pipes are fully insulated by rock wool enveloped with aluminium cladding. Both vessels wall temperatures can be controlled through an external thermo-oil circuit.

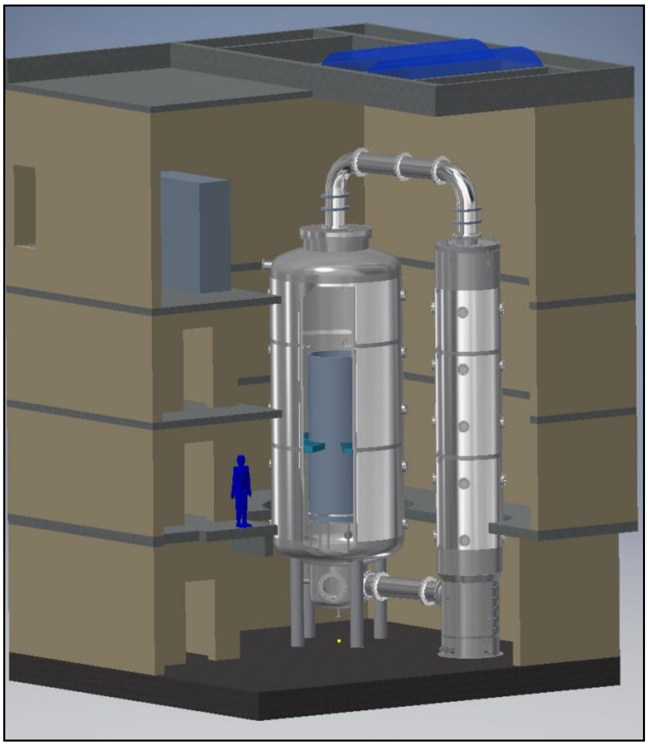


Figure 1: THAI$^+$ test facility

The THAI$^+$ test facility has same design boundary conditions as in the original THAI test facility (14 bar at 180 °C) and also retain its unique experimental features, e.g. use of hydrogen and iodine tracer I-123, differential wall heating / cooling. Moreover, by establishing independent desired flow and temperature conditions in each of the two vessels, it will be possible to broaden the THAI capability to obtain closer similarities to reactor scenarios.

As per the experimental requirement, it will be possible to perform the experiments in single-vessel configuration alone or in the extended THAI$^+$ two-room vessel geometry. Depending on a test requirement, it is possible to divide the test facility in more compartments than previously studied in the THAI vessel (i.e. 5-compartment geometry for iodine distribution experiments).

3 MAIN INSIGHTS FROM OECD/NEA THAI AND THAI-2 PROJECTS

3.1 Thermal-hydraulics / gas distribution

Atmospheric stratification and locally enhanced hydrogen concentration in the containment contribute to the risk of early containment failure in case of strong combustions. In the OECD/NEA International Standard Problem ISP-47, stratified atmospheric conditions were extensively studied by using the THAI experiment TH-13 [6]. Analysis of the test results indicated that the light gas cloud erosion by the buoyant plume from the lower steam injection in the experiment was over-predicted by nearly all CFD- and LP-codes resulting in fully mixed atmospheric conditions. Application of such codes with a tendency to predict lower uniform concentration instead of locally enhanced hydrogen concentration to a reactor case could lead to non-conservative underestimation of the risk from hydrogen combustion. Apart from providing data for code validation purpose, the main objective of the Hydrogen / Helium Material scaling (HM) tests performed in OECD/NEA THAI was to validate the transferability of experimental findings with helium to hydrogen problems. The HM test series was similar to TH-13 but with simplified test procedure to answer remaining open issues from ISP-47. It was shown that comparable atmospheric distributions, pressure and temperature levels can be obtained if the volumetric concentrations of hydrogen and helium are comparable. Test results and application of test data for code validation purpose and use of other related code benchmarks on THAI gas distribution tests are summarized in chapter 4 and are discussed in [3, 4, 6]. No other gas distribution experiments have been performed within OECD/NEA THAI projects; the main focus rests on hydrogen and aerosol/iodine related topics described here after.

3.2 Hydrogen issues

3.2.1 PAR Performance

Efficiency of hydrogen mitigation by Passive Autocatalytic Recombiners (PAR) under severe accident typical conditions was always an important issue in the OECD/NEA projects [7]. Three commercially available PARs from AREVA, AECL, and NIS were used for the testing in the THAI facility. PAR test results enlarge the knowledge base on start-up behaviour, per-formance and ignition potential under severe accident conditions and provided useful infor-mation for PAR model validation for all types of codes in addition to data gained from earlier experiments. The experimental database has been extended by investigating influence of oxygen lean atmosphere on PAR start-up behaviour and influence of modified gas composi-tion (reduced O_2 concentration in air) on PAR ignition.

The start-up behaviour of PARs was not affected by O_2 starvation conditions. The start-up was even faster when exposed to slowly increasing oxygen concentration under nearly inert THAI atmosphere containing hydrogen. However, lean oxygen concentration is shown to adversely affect PAR (designed) capacity of hydrogen depletion.

Hydrogen recombination rate of a PAR develops almost proportionally to hydrogen concen-tration at the inlet and also proportionally to pressure (for the same hydrogen concentration at the PAR inlet). Based on THAI tests, a threshold value for oxygen starvation could be de-fined in terms of O_2 surplus ratio defined as $\Phi = 2 * C_{O2} / C_{H2}$ where C_{O2} and C_{H2} are oxygen and hydrogen concentrations (by volume) at the PAR inlet. For an unimpaired PAR perfor-mance, critical minimum O_2 surplus ratios $\Phi = 2.2$ (AREVA), 2.3 (AECL) and 2.75 (NIS) were found to be necessary, which is indeed much higher than the stoichiometric ratio of one for O_2/H_2 mixture. At O_2 surplus ratio equal to one, H_2 recombination rate falls below 50 % of the design capacity.

Three H_2 recombination regimes are observed during the tests. For an oxygen surplus ratio value $\Phi \leq 1$, an oxygen lean gas mixture exists at the PAR inlet and the rate of oxygen diffu-sion through the catalyst boundary layer mainly governs the recombination rate. A transition in hydrogen recombination rate occurs for $1 < \Phi \leq 2$, and H_2 recombination is governed by both oxygen and hydrogen diffusion through the catalyst boundary layer. For oxygen surplus ratio value $\Phi > 2$, hydrogen lean gas mixture prevails at the PAR inlet and the recombination rate is mainly governed by the rate of hydrogen diffusion through the catalyst boundary layer.

Hydrogen recombination through PAR is incomplete and this can be quantified by hydrogen depletion efficiency γ (in %) calculated from the measured H_2 concentrations at the PAR inlet and outlet, $\gamma = (C_{H2in} - C_{H2out}) / C_{H2in} \cdot 100$. The hydrogen depletion efficiency is almost inde-pendent of the steam content and in oxygen lean atmosphere increases mainly by increasing oxygen surplus ratio. It remains constant once oxygen surplus ratio exceeds respective criti-cal value for specific PAR design. The hydrogen depletion efficiency was determined to be varying between 40 - 70 % in an atmosphere containing sufficient oxygen surplus for the tested AREVA, AECL and NIS PARs.

Possible hydrogen deflagration inside containment and especially the question if ignition can be caused by PAR operation is regarded as an important reactor safety concern and there-fore tests were performed to investigate conditions for ignition by PAR. THAI test results indicate that PAR exposed to a high hydrogen concentration acts as an ignition source for the combustible gas mixture present in the PAR environment (respectively in the test vessel vol-ume) and can initiate a hydrogen deflagration. For AREVA and AECL PARs, minimum hy-drogen concentration at which ignition occurs was measured between 6 and 9 vol.% depend-ing on the steam content. An example (AREVA PAR) of possible ignition area on ternary dia-gram by using minimum required H_2 / O_2 / steam concentrations is shown in Figure 2.

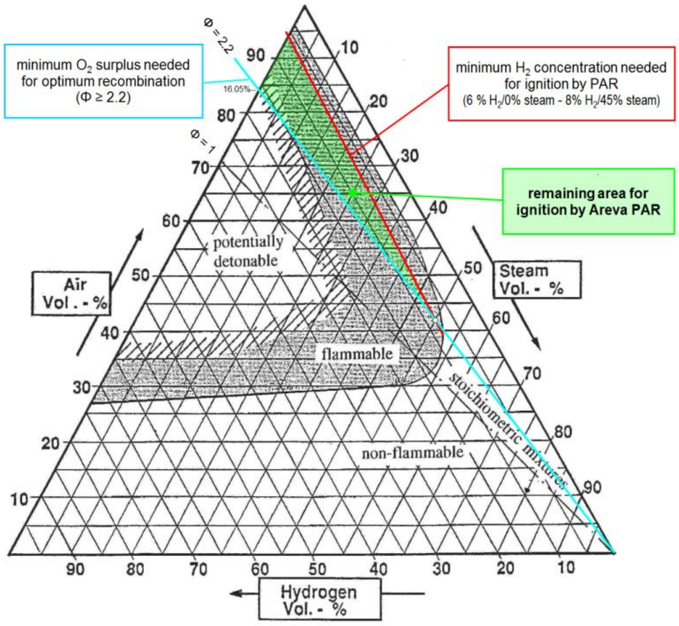


Figure 2: Ternary diagram: Area of possible ignition by PAR, resulting from HR test

Ignition potential for NIS PAR was also investigated in THAI tests. In the test conducted at 1.5 bar, 74 °C and steam content of 25 vol.%, only a weak combustion event with an ex-tremely low pressure rise (about 0.1 bar) was observed. One specific feature of the NIS-PAR under investigated test conditions with 25 vol.% steam was the expulsion of glowing particles ("glow worms") into the bulk in case of high load (>5.2 vol.% H_2 at the PAR inlet). This visible effect coincided with a marked additional H_2 recombination (in the range of 10 % of total measured recombination) in the bulk without adverse pressure effects. Tests conducted with high steam content (> 40 vol.%) did not show any glow worm. Either "glow worms" did not exist, or were not visible due to a too low surface temperature. In any case, the higher mini-mum required ignition energy for the steam-rich mixture could not have been provided.

In oxygen lean mixtures, the concentration of oxygen is the limiting factor for the hydrogen recombination rate and results in lower catalyst temperatures if O_2 starvation conditions are reached. PAR induced ignition potential in O_2 lean mixtures is low, since the catalyst temper-atures are directly correlated to PAR self-ignition.

The ternary diagram depicted in Figure 2 is based on air/steam/hydrogen gas mixture sys-tem, which needs to be adapted if gas mixture system differs, e.g. for the late phase of MCCI due to reduced O_2 content in air. In OECD/ NEA THAI-2 project, tests were performed with reduced O_2 content in air by establishing gas composition of O_2/N_2/steam at a defined dilu-tion ratio ($\delta = c_{O2} /(c_{O2} + c_{N2})$). The main objectives of these tests were to assess the code capabilities to predict PAR ignition potential under transient accident conditions. The data indicate that a change of dilution ratio also modify the steam intertisation limit. In case of a test conducted with dilution ratio of $\delta = 0.1$ at PAR inlet and H_2 concentration of 10 vol.%, inertisation limit of steam concentration was determined to be 30 vol.% instead of 55 vol.% in air/H_2 mixture. It should be noted that the ternary diagram depicted in Figure 2 is prepared for ambient pressure and about 100 °C temperature conditions and an increase in gas tempera-ture will further widen the indicated ignition limit on this diagram.

After ignition, PAR continues to recombine unburned mass of hydrogen available in the gas atmosphere. Tests with continued or restart of hydrogen release shortly after ignition show repetition of PAR induced ignition as soon as favourable conditions for ignition as discussed earlier are met. In one of the THAI "multi-ignition test" the first two ignitions occurred rather early with an H_2 concentration in the dome area at the lower ignition limit (4 vol.% H_2 for H_2-air mixtures at ambient temperature). Consequently, the deflagration was very smooth and the pressure rise very low. The third ignition occurred at or near the PAR inlet. The flame proceeded first into the vessel bottom zone, and from there upwards into the inner cylinder and annulus zones, and finally into the dome zone. Due to the combustion in the bottom zone with relatively high local H_2 concentration (6.6 vol.% H_2) and flame propagation in the entire vessel volume, the pressure increase following the third ignition was higher than after the previous burns.

Hydrogen concentration at the PAR inlet describes the load of the PAR in the moment of ignition but not necessarily the atmosphere above the PAR into which the hydrogen deflagration propagates. Hydrogen concentrations and flow conditions upstream and downstream of a PAR may differ significantly depending on an accident scenario and reactor geometry, which in-turn may have an influence on PAR induced ignition behaviour. In majority of the PAR tests, the atmosphere stratification developing above PAR during its operation resulted into low pressure peak after ignition. In addition to the multi-ignition test mentioned above, during other PAR tests conducted with no or weak thermal/gas stratification above PAR, relatively high pressure peaks were observed [8].

Performance behaviour of an operating PAR when exposed to aerosol and iodine containing atmosphere under severe accident scenario is of high significance as it may have an impact on in-containment fission product source term. Two tests on PAR interaction with fission product were conducted in OECD/NEA THAI project [5]. No poisoning was observed if PAR is exposed to fission product after onset of hydrogen recombination and thus rendering catalyst surfaces hot. Combining this finding with earlier discussed results of O_2 starvation effect, it is possible that during late phase of accident involving Molten Core-Concrete Interaction (MCCI), O_2 lean atmosphere may lower catalyst surface temperatures but other available heat sources including continuous steam release in containment will keep PAR sufficiently hot to prevent deposition of potential poisons on catalyst surfaces.

Another investigated issue was related to the possible thermal decomposition of metal iodides passing through an operating PAR. Based on test results, decomposition of CsI aerosol to gaseous iodine in the range of 1 - 3 % was observed. If sufficiently high amount of metal iodides is present in LWR containments, gaseous iodine produced due to thermal decomposition may increase fission product in-containment source term. Detailed analyses taking into account reactor conditions will be necessary to confirm the impact of measured conversion rate on potential source term to environment.

3.2.2 Hydrogen deflagration

Other hydrogen related tests performed in OECD THAI projects focused on hydrogen deflagration behaviour in free volume of THAI vessel. The main parameters varied were hydrogen and steam concentration, temperature, pressure, burn direction upward and downward, well-mixed and stratified atmosphere. Majority of the tests have been conducted with an initial pressure of 1.5 bar and an initial gas temperature between ambient and 140 °C. The initial hydrogen concentration has been varied between 6 and 12 vol.%. In the follow-up OECD/NEA THAI-2 project, influence of water spray operation on hydrogen deflagration was investigated. Hydrogen deflagration tests from previous project were taken as reference tests to quantify the influence of water spray on hydrogen combustion.

Based on results obtained for tests without spray, the limit concentration for downward flame propagation was determined to be 8.7 vol.% H_2 in air (at 1.5 bar and 20 ^{0}C gas temperature) and 12 vol.% H_2 in a steam-air mixture containing 47 vol.% saturated steam (at 1.5 bar and 90 °C). Higher initial temperatures (up to 140 °C), which are more typical for severe accidents conditions, lead to lower peak pressures (because of the lower density/energy inventory in the gas mixture) and also give rise to more steady flame front propagation and a more complete combustion. High initial temperature conditions cause slower upward and faster downward flame propagation because of changes in buoyancy forces. Upward flame propagation is supported by buoyancy and proceeds at comparatively low hydrogen concentration with higher velocity and shows convex flame surfaces. In contrast, downward flame propagation shows lower velocities and more flat flame surfaces. A high steam content (48 vol.%, at saturation state) in the combustible gas mixture leads to an irregular ("erratic") combustion both for upward and downward burn direction with lower flame velocities and lower peak pressures as compared to "dry" mixtures. Figure 3 shows the effect of steam content on flame front propagation behaviour.

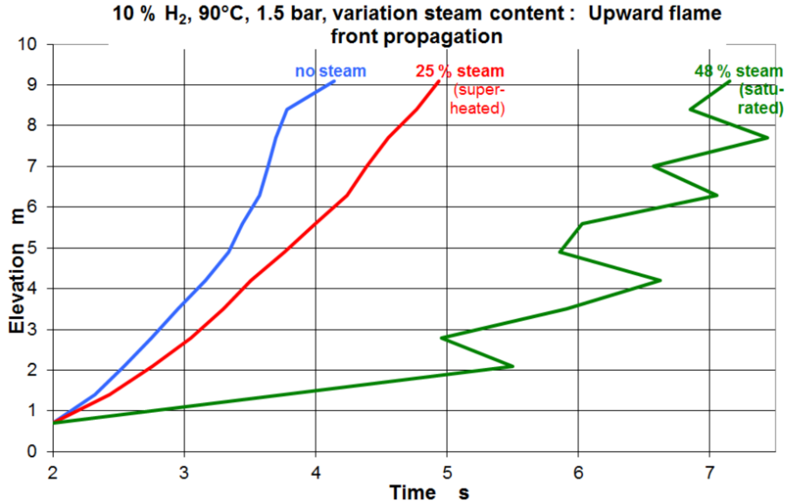


Figure 3: Upward flame propagation: effect of steam content

Potential severe accident scenarios leading to hydrogen deflagration in an atmosphere with density difference were also investigated with stratification tests. The comparison between tests with the same mean hydrogen concentration but with thermal stratification established in upper vessel plenum during one of the tests shows that the upward flame propagation is supported by the gas mixture with higher density in the lower half and lower density in the upper half. The steep and higher pressure rise and complete hydrogen combustion in the test with density gradient in comparison to the test with initial homogenous density supports this conclusion about faster flame speed. Large portions of unburnt gas mixture were displaced by flame induced convection during the deflagration, particularly in the tests with upward burn direction. Hydrogen deflagration induced mixing effects turn originally non-flammable mixture into flammable. In case of downward burn direction, the convection effect is weak and combustion stops when the flame front enters into a mixture which is not burnable for downward burn direction.

Hydrogen deflagration tests with spray conducted in OECD/NEA THAI-2 project further elaborated the effect of spray induced convection and turbulence on flame propagation behaviour. Test results show that convection generated by spray operation enables downward flame propagation for mixtures too lean (or below limit H_2 concentration) for downward combustion under quiescent conditions. Tests were conducted with single spray nozzle installed at the elevation of 7.4 m and with spray angle of 30 °C. Both spray water and initial gas temperatures were varied to investigate spray induced condensation/evaporation effects by operating cold water spray in hot gas atmosphere or hot water spray in hot has gas atmosphere, respectively. For studying the mixing effect, two different nozzles with spray droplets Sauter mean diameters of 670 µm and 970 µm were used.

All tests with H_2 concentration up to 10 vol.% and upward burn exhibited clearly a suppressing effect of spray with respect to peak pressures and peak temperatures. The results were independent from initial gas temperature as well as spray characteristics, e.g. spray water temperature, droplet size. For the test conducted with a H_2 concentration of 12 vol.% and 25 vol.% steam and downward burn direction, the peak pressure exceeded that of the quiescent (without spray) test by 10 % due to spray induced turbulent flow pattern and the related high flame speed as compared to the reference test without spray as shown in Figure 4.

A higher degree of combustion completeness occurred in the test with spray. For the flame quenching process, the total available droplet surface is a decisive parameter. Spray nozzle with large droplet size indicated less cooling effect, enhanced gas mixing and consequently slightly increased pressure. Spray water temperature indicated no observable effect on combustion suppression.

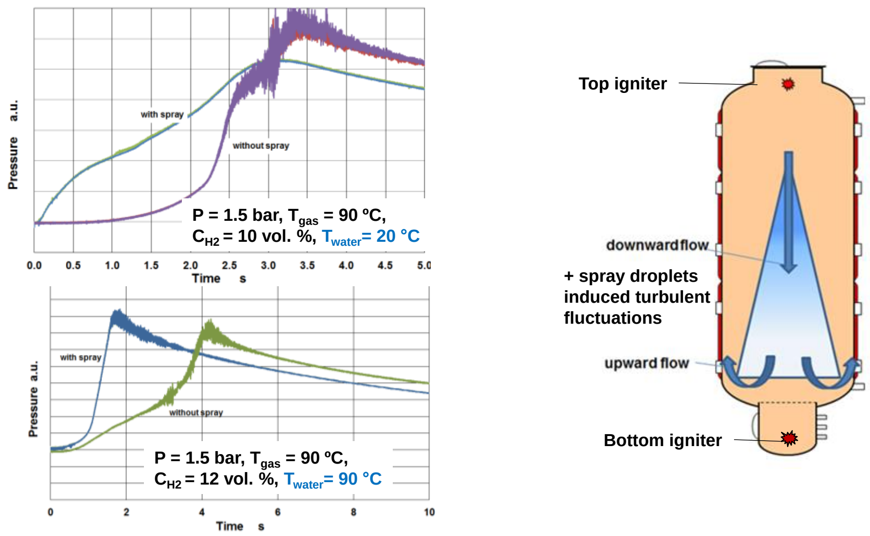


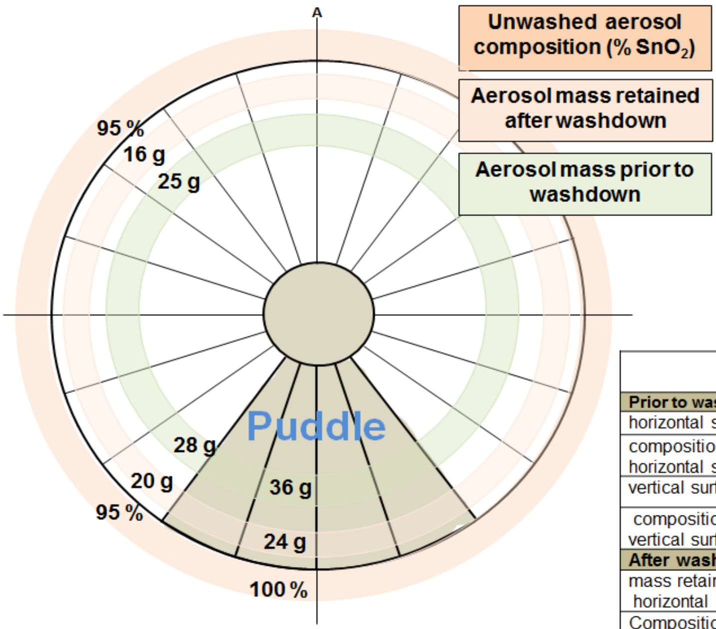
Figure 4: Upward (top) and downward (bottom) burn with spray: comparison with reference tests without spray

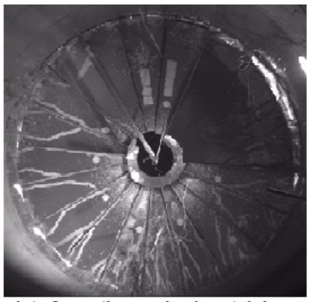
3.3 Aerosol and Iodine issues

The THAI aerosol wash-down (AW) test conducted in OECD/NEA THAI project addressed the main phenomenon related to the wash-down process of soluble CsI aerosol. Test procedure was defined to generate relevant database for model verification and validation purpose. On injection of CsI solution by two-fluid nozzle, evaporation of CsI droplets occurs in hot gas atmosphere and dry aerosol particles are deposited under superheated thermal hydraulic conditions on vertical walls, horizontal surfaces or in a small pool (initially dry during aerosol injection and deposition phase). Horizontal deposition surfaces consisted of 20 stainless steel sections and covered the complete THAI vessel cross section. Out of total 20 sections, 16 sections form a flat surface (plate section) and 4 sections form a water puddle (39 mm deep, volume: 26.9 litre). The stainless steel sections and the puddle were inclined to vertical centreline with a downward gradient of about 2°. Aerosol injection was done near the top of the vessel and the injected dry mass of CsI (particle size: 1.0 µm MMD, 1.5 GSD) was 1178 g. From total injected aerosol mass, 51% deposited over horizontal deposition surfaces and 49 % over vertical surfaces. The thermal-hydraulic conditions were established in such a way that wash-down of deposited aerosols occur only by steam condensing over the vessel mantles and no volume condensation takes place during steam injection. The condensate drained out from the plates and the puddle was analysed separately for determining the aerosol concentration. The test results indicate that horizontal surfaces are subject to aerosol wash-down in the time range of minutes to hours. The puddle water acts as an intermediate storage of aerosol material, which leads to a considerable delay in the wash-down transport. Laboratory experiments performed in parallel showed, that the main processes which were found to have an influence on the observed aerosol wash-down behaviour included condensation induced flow patterns, such as rivulets or closed water films. In case of rivulets, which was the prevailing case during the tests, their formation and dissolution mechanisms, partial aerosol wash-down as compared to closed water-film, and strong retention of even soluble aerosols in the water filled puddle was observed.

Additional tests to further investigate wash down process for non-soluble aerosols have been conducted with single component aerosols Ag as well with aerosol mixture (CsI/SnO_2) in THAI national projects [5]. In the presence of non-soluble aerosol, the washed out aerosol fraction was drastically reduced. In case of non-soluble aerosol Ag, very low wash-down efficiency (defined as the ratio of washed-down silver mass to the total initially deposited aerosol mass over respective surfaces) was observed. It was 16.5 % for the vertical wall section, 9.5 % for the combined wall/horizontal plate section, and 10.2 % for the combined wall/puddle. THAI tests on aerosol wash down were accompanied by two lab-test programs in which parameter variations like aerosol loading, condensate mass flow, particle sizes and surface coating were investigated. A major outcome of these tests was the observation of long-lasting stability of rivulets in the presence of non-soluble aerosols surface loadings. Furthermore, the width of the individual rivulets scales with surface inclination, but not with condensate mass flow.

Main results of the test conducted with mixture of CsI/SnO_2 are presented in Figure 5.





Rivulets formation on horizontal deposition surfaces in THAI

	16 plates	puddle	total
Prior to washdown			
horizontal surface	529 g	174 g	703 g
composition horizontal surface			65% SnO_2 + 35 % CsI
vertical surface			395 g
composition vertical surface			90% SnO_2 + 10 % CsI
After washdown			
mass retained horizontal surface	370 g	124 g	494 g
Composition horizontal surface			94.5 % SnO_2 + 5.1% CsI

Figure 5: Results of CsI/SnO_2 aerosol mixture wash down test AW-2 [5]

The two THAI tests on the gaseous iodine and aerosol reactions investigated the effect of I_2 removal from containment atmospheres by interaction with containment aerosol, the reactive silver (Ag) aerosol providing a high reactivity, and the inert tin oxide (SnO_2) a low reactivity under the given boundary conditions (i.e. dry atmosphere, 1.5 bar pressure), probably linked to different involved mechanisms, mostly physisorption with SnO_2 and chemisorption with Ag, which also impact desorption. The test results demonstrated that the removal of gaseous molecular iodine from the vessel atmosphere by interaction with the reactive silver aerosol was twenty-five times faster as compared to the test with the inert SnO_2 aerosol. The I_2 chemisorption reaction with the Ag particle surface was as high as for the higher range of I_2 / decontamination paint reactions, as evident from comparison of other large-scale THAI tests conducted in THAI national projects and associated laboratory-scale tests [9]. Real mixed aerosol in the containment can be expected to exhibit I_2 removal rates between the bounding cases "reactive Ag" and "inert SnO_2".

One integral iodine (Iod-29) test was conducted to deliver experimental data on release of gaseous iodine from a flashing jet under high pressure and high temperature thermal hydraulic conditions. The design of the experiment was oriented towards PWR design-basis accident “steam generator tube rupture during reactor shut-down”. A pressure vessel installed outside THAI vessel was used to simulate the pressure drop of 40 bar (at 250 °C saturation temperature) same as in a real accident scenario (110 bar on primary side and 70 bar on secondary side). The molecular iodine was injected in outside pressure vessel filled with cold water. The total iodine inventory available at the time flashing was about 14 g. The flashing evaporation rate was estimated to be approximately 23 %. Test results indicated no release of gaseous iodine by flashing. Although molecular iodine was injected before heat-up, only the iodide form was found, and iodate as a product of iodine hydrolysis was not detected above the detection limit of 5.4 % with respect to the sum of all iodine species. This indicates that the injected molecular iodine had quickly dissolved and hydrolysed, with intermediate hydrolysis species such as HOI and I_2OH^- reacting quantitatively with the steel wall of the primary vessel or metal ion impurities in the aqueous phase during heat-up to produce the non-volatile iodide form. Results from Iod-29 design calculation, mainly on chemistry part, indicated lack of validation of existing iodine chemical models in the aqueous phase for the Reactor Coolant system (RCS) conditions, e.g. effect of high temperature.

4 ANALYTICAL ACTIVITIES

The experimental investigations carried out in the framework of OECD/NEA THAI projects have been strongly supported by accompanying analytical activities performed by the partners of the Analytical Working Group (AWG) [10]. The analytical activities delivered effectively by voluntary contribution from participating organisations included code calculations for pre-test assessments, result evaluations considering blind/open code benchmarks and extrapolation of the experimental results to reactor situations.

The experimental data obtained from the different series of tests demonstrated the capabilities of the THAI facility in producing high-quality, high-resolution data on gas mixing/stratification issues, slow hydrogen deflagration, passive autocatalytic recombiners, aerosol wash-down, gaseous iodine and aerosol interaction with airborne aerosols, release of gaseous iodine from flashing jet in a technical scale test facility. The THAI projects contribute to the validation and further development of advanced LP codes (e.g. COCOSYS, ASTEC, MELCOR) and CFD codes (e.g. CFX, FLUENT, GOTHIC, GASFLOW, FLACS, OPENFOAM) used by the project partners by e. g. providing experimental data for code benchmark exercises. Based on THAI experimental results, important progress has been demonstrated in modelling of aerosol and iodine behaviour and their coupling with containment thermal hydraulics in severe accident analysis codes, such as COCOSYS-AIM, ASTEC-IODE, JAEA code ART. Summary of analytical activities performed within THAI and THAI-2 projects is provided in Table 1; some examples follow.

Table 1: Analytical work in OECD/NEA THAI projects

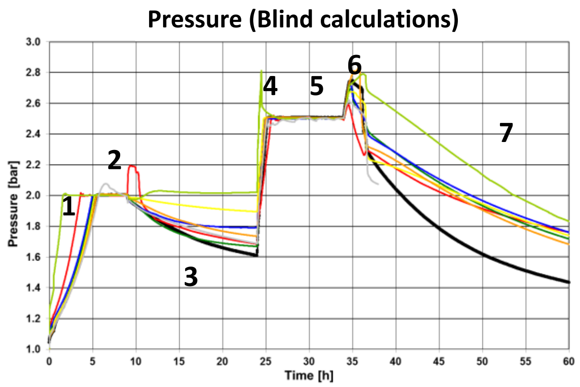
Project/frame	Issue	THAI experiment, activity
Thermal-Hydraulics/Gas distribution		
OECD/NEA THAI	Hydrogen distribution and containment thermal-hydraulics	HM-2, blind and open code benchmarks
Hydrogen mitigation – Passive Autocatalytic Recombiners		
OECD/NEA THAI	PAR performance under accident conditions	HR test series, validation and further development of PAR specific models / empirical correlations, PAR ignition models, code validation and application to reactor situations
OECD/NEA THAI-2	PAR performance in O_2 lean atmosphere	HR-35, blind and open code benchmarks
Hydrogen combustion		
OECD/NEA THAI	Hydrogen deflagration in premixed gas atmosphere (upward/downward burn)	→ HD test series, code validation → HD-2R (open) & HD-22 (blind), International Standard Problem (ISP) no. 49
OECD/NEA THAI-2	Hydrogen deflagration under spray	→ HD test series, code validation
In-containment fission product (Aerosol, Iodine) behaviour		
OECD/NEA THAI	Soluble aerosol wash-down from surfaces and puddle	AW, model verification and validation
OECD/NEA THAI-2	Gaseous iodine deposition on soluble and insoluble aerosol particles in dry gas atmosphere	Iod-25 & Iod-26, model verification and validation

4.1 Thermal-hydraulics / gas distribution

The analytical work performed for the HM test benchmark on containment thermal-hydraulics and gas distribution resulted in an improvement of the LP and CFD codes and provided valuable experience to code users. Recommendations for improved LP model nodalisation for hydrogen distribution analysis were successfully applied. Application of HM-2 test data for code validation purpose and use of other related code benchmarks on THAI gas distribution tests are widely discussed in [3, 4].

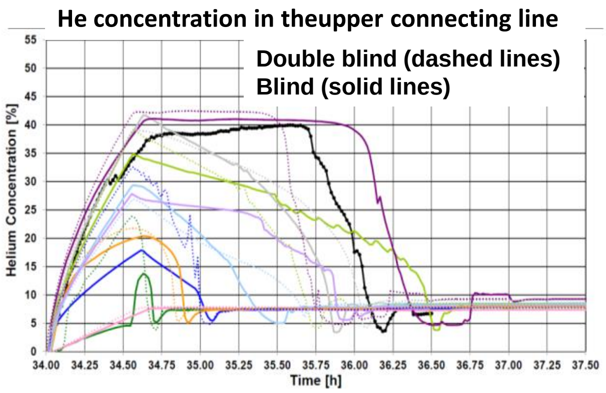
Recently, test TH-27 performed as commissioning test for the THAI+ test facility in the national THAI project was offered for an international code benchmark involving several LP and CFD codes. Some results are presented here exemplarily to demonstrate that knowledge gained through analyses of previous THAI experiments is useful for code applications. The TH-27 code benchmark provides new challenge to analyses due to significant extension of the THAI facility. The goals from this benchmark were set to quantify thermal losses of the THAI+ system, volumetric exchange between two vessels, and dissolution of stratified helium gas layer by natural convection.

Double blind simulations (that is, simulations before the test was actually performed) were performed by 12 organizations using 6 different codes (ASTEC, COCOSYS, CFX, FLUENT, GASFLOW and GOTHIC). In the follow-up activity, blind simulations (that is, simulations after the test was performed, but experimental results were not disclosed) were performed by 9 organizations using 4 different codes (ASTEC, COCOSYS, GASFLOW and MELCOR). Double blind simulations started during construction of the THAI+ facility and blind specifications were revised according to the actual measured test boundary conditions. The test procedure and the resulting pressure evolution and helium concentrations as measured in TH-27 test and as calculated by the codes in the blind analyses are shown in Figure 6.

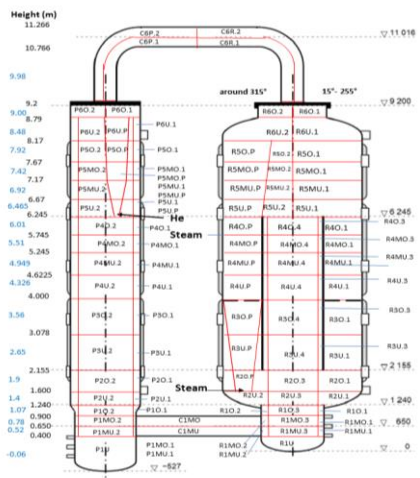


Test includes seven individual phase

1.Steam injection for THAI+ system heating
2.Pressure stabilization phase
3.Heating of THAI vessel walls only
4.2nd steam injection into THAI vessel and pressure stabilization
5.Vessel cooling by thermal oil mantles of PAD
6.Helium injection
   - Measurement of gas distribution
   - Effects of light gas stratification
7.Vessel cool-down after stopped steam injection



Black solid line: experimental results



Example nodalisation with COCOSYS

Figure 6: TH-27: test procedure, example nodalisation, and main results [11]

The benchmark exercise confirmed generally good prediction of pressure and temperature evolution but in spite of defined boundary conditions, still large discrepancies in predicting formation and erosion of helium stratification were observed. The main reasons behind differences between experiment and calculations were related to heat loss calculation of the facility and modelling differences like control of wall temperatures, condensate distribution on walls and possible re-evaporation from walls back to gas atmosphere as well as user effects in code application [11]. Importance of nodalisation for LP and modelling gaps in CFD codes related to wall condensation, turbulence, thermal radiation effect, injection modelling, free jet modelling are still in process of being further improved for predicting mixing of inhomogeneous gas atmosphere by buoyancy or momentum dominated flows.

4.2 Hydrogen issues

4.2.1 PAR Performance

In the case of the PAR investigations, a deep insight into their performance and ignition potential under accident typical conditions has been obtained. Main analytical activity was already performed in OECD/NEA THAI along with the huge amount of additional data about the behaviour of PARs under typical severe accident conditions. This allowed analysts to set-up or improve PAR models implemented in LP and CFD codes [12, 13] as well as specific codes with detailed physical and chemistry modelling of the catalytic reactions e.g. REKO-DIREKT, SPARK [14, 15], and consequently to design and assess hydrogen mitigation measures properly, to take maximum benefit for accident management.

Several organizations used these results for updating their code models. For new safety analyses, one example comprises a re-evaluation of the PAR concept for a German PWR by using the GRS code COCOSYS with an updated PAR model based on THAI results [16]. A detailed containment nodalisation considering all plant specific aspects together with well validated PAR models and a selection of representative severe accident scenarios (LOCA and transients) were applied at GRS for the re-analysis of the PAR concept for large dry PWR containments. The analyses of the implemented PAR concept demonstrated in general a high safety profit. Nevertheless, mainly in the inner containment (inside missile shield) combustible gas mixtures exceeding 10 vol.% H_2 could be developed locally for short times even with the PAR system (Figure 7). In some scenarios like transients with loss of steam generator (SG) feed water supply, steam inertisation prevents any combustion (Figure 7 right); in others, like large break LOCAs, maybe not (Figure 7 left). As PARs could act as igniters, local combustions are still possible, but the PARs would ignite the mixture at relatively low H_2 concentrations, well below the detonation limit.

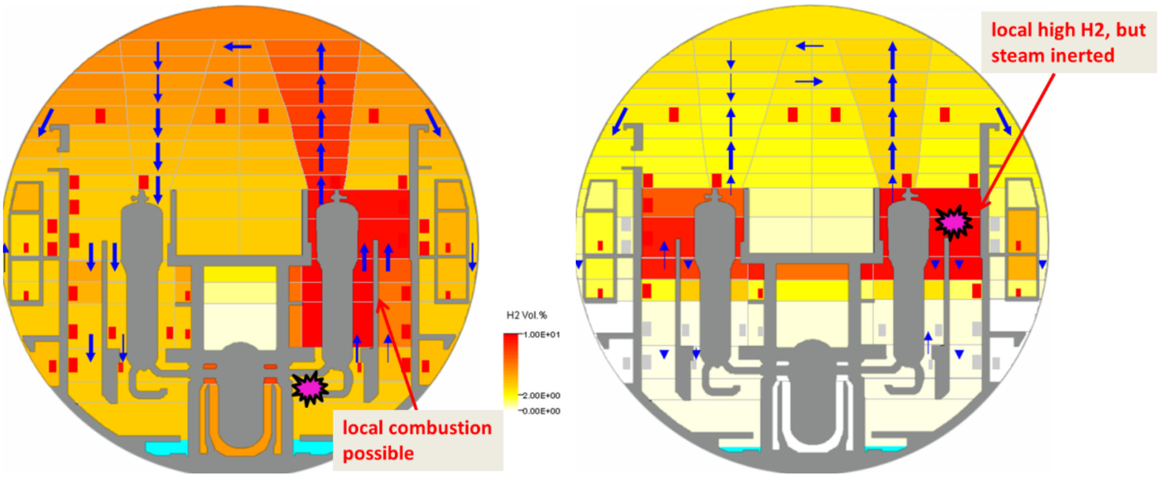


Figure 7: H_2 concentration in PWR containment and PAR operation (red box), core degradation phase: left) large break LOCA, right) transient with loss of SG feed water supply [16]

In OECD/NEA THAI-2 analytical activities based on PAR performance in oxygen lean atmosphere have been performed and resulted into recommendations for improving existing empirical correlations for the prediction of the recombination rate as well as other recommendations. In the frame of a benchmark exercise, blind pre-test and open post-test calculations for test HR-35 have been performed [17]. Five institutions participated in the blind simulation exercise and a total of six simulation results were submitted for evaluation. Thereby lumped parameter codes (MELCOR, COCOSYS and GOTHIC) have been used as well as CFD codes (GASFLOW and CFX). The internals of the recombiner have either been resolved within the nodalisations or modelled by specific code modules, or by black box type models. The physics of the PAR were modelled using REKO-DIREKT or a model based on it, the AREVA correlation, similar types of correlations or self-developed correlation functions. Thereby the simulation benchmark addresses a large variety of different approaches which are used today to analyse the containment conditions during accident scenarios where hydrogen has been released. The hydrogen recombination rate and the fraction recombined are shown in Figure 8 as an example of the analyses results.

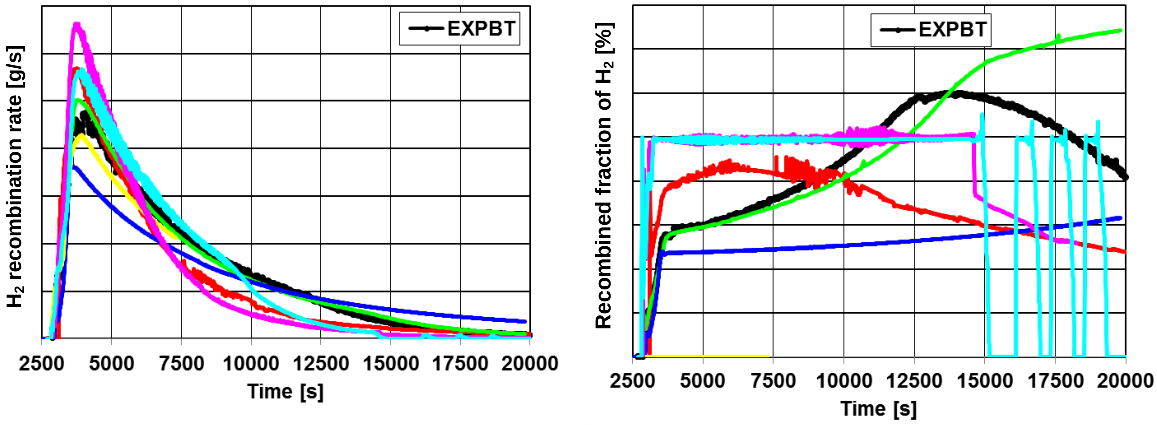


Figure 8: HR-35 blind exercise: Hydrogen recombination rate and fraction recombined [17]

It turned out that several criteria are essential for such a prediction: The internal behaviour of the PAR, i.e. its onset, its recombination rate and overall efficiency, a proper modelling of the released energy and the correct prediction of the natural convection through the PAR which transports the reactants to the catalyst plates. Comparison of test results with blind simulations indicate that codes using H_2 recombination rate correlation overestimate the recombination at the beginning of the experiment under oxygen starvation conditions. Physical bases of diffusion type models enforce good prediction of H_2 recombination rate. The coupling of the recombiner model with the containment codes is as well of importance since a thermal stratification might be induced by the recombiner which influences the atmospheric containment conditions [17].

4.2.2 Hydrogen deflagration

The application of LP and CFD codes for deflagration tests provided useful insights. Of special interest for model development are tests with downward burn and flame instability and extinction. The results of the performed simulations clearly demonstrated that currently existing nodalisation rules for LP codes require further improvement for general application purpose. Combustion models implemented in CFD codes are required to further improve prediction of flame dynamics especially by including turbulence effects (e.g. turbulence generated by flame), which are implemented in the existing codes.

The deflagration tests in the presence of operating spray system covered a broad range of accident typical conditions by parameter variations. The data proves clearly that interaction between airborne water droplets and flame deflagrations need to be considered in code models and for new safety analyses, since it not only affects the pressure and temperature load on the containment, but also the propagation of flames. Several organizations started already to incorporate these results into their code models in order to improve prediction of flame dynamics and pressure response. Both LP and CFD codes indicated difficulty in reproducing turbulence enhancing effect by spray to correctly simulate spray and combustion interaction. Indeed, the role of water sprays on premixed flame propagation is complex and depends strongly on several parameters such as the airborne liquid water fraction (a function of spray water mass flow and the mean droplet falling velocity), droplets distribution inside the containment atmosphere and droplet size. Work on development and further improvement of models is currently underway to calculate spray induced entrainment, heat and mass transfer between droplets and gas [18, 19, 20].

4.3 Aerosol and Iodine issues

Concerning fission product transport, calculation of the wash down tests with LP codes revealed the importance of further phenomena, which should be considered in the modelling: (a) partial aerosol wash-down due to rivulet formation at low wall condensation rates, and (b) strong retention of aerosols in the water puddle. Further development of wash down model e.g. AULA module in COCOSYS is currently underway by including aerosol wash down tests conducted in OECD and national THAI projects [21].

The newly developed model AULA describes the erosion of insoluble particles by down-flowing condensate under reactor conditions. It is based on a model for sediment erosion applied in geology. Particles on vertical walls and floors erode and are washed-down when certain flow conditions are given. Two types of flow patterns are considered: water films and rivulets. The THAI AW tests delivered a suitable data base for model validation. THAI tests AW-2 and AW-3 as well as first calculations (Figure 9) indicate that a complete wash-down of deposited silver aerosol in containment of a LWR is not likely. For instance, new reactor designs are incorporating more and more passive safety or inherent safety features. One of such examples is design of PCCS (Passive Containment Cooling System) in Westinghouse design AP1000. The free falling liquid water film spreading on the outer surface of containment helps to keep the inside temperature and pressure under safety limits. The liquid flowing down the vertical and particularly on the curved surfaces of containment building may prevent efficient fission product wash-off, as rivulet flow velocities may be too slow to initiate erosion and the rivulets leave parts of the surface dry [22]. Therefore an accurate wash-down simulation for insoluble aerosols like silver which are source term relevant has to be of major concern in accident analyses.

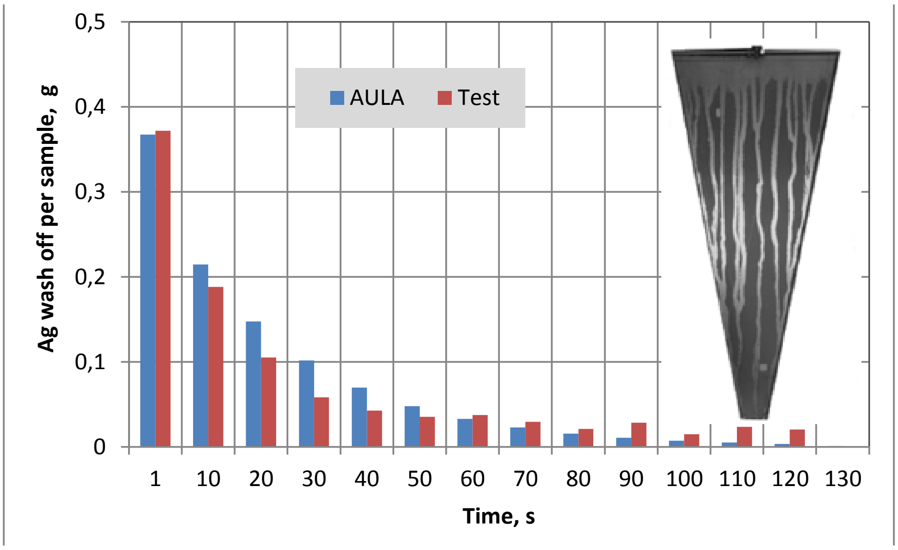


Figure 9: COCOSYS-AULA result: Measured and calculated erosion rate and rivulet flow pattern (right) in the THAI AW-3 LAB test 4 [21]

The tests on iodine deposition on chemically reactive and non-reactive aerosol particles provided extension to existing knowledge and contribute to assessing the safety relevance of this effect. Model developments to incorporate this effect into source term prediction tools are currently ongoing but first assessments indicate a necessity to take such transport mechanism into account, for the sound quantification of air-borne fission products [9].

Design calculations were performed to define test procedure for the test investigating SGTR DBA scenario to determine the release of gaseous iodine from a flashing jet under high pressure and high temperature thermal hydraulic conditions. Results from design calculation indicate lack of validation of existing iodine chemical models in the aqueous phase for the investigated test conditions (high temperature).

5 PERSPECTIVES ON OECD/NEA THAI-3 PROJECT

In spite of significant improvements demonstrated by the available computer codes, there are still deficiencies in modelling certain phenomena related to hydrogen and fission product behaviour in light water reactor containments. Code validation and development work based on previous THAI PAR tests under natural convection flow conditions demonstrated significant progress towards PAR performance analysis and their response to severe accident typical conditions. Nevertheless, studies of representative accident sequences indicate that performance of PARs might be affected by adverse flow conditions, e.g. counter-current flow. Therefore, experimental database related to PAR start-up behaviour and PAR performance (H_2 recombination rate, H_2 depletion efficiency) under counter-current flow conditions and by considering representative gas composition (e.g. carbon monoxide) is required to be investigated in order to enhance the predictive capabilities of safety analysis tools.

Regarding fission product related open issues, environmental measurements made during the Fukushima accident indicated that a significant fraction of radionuclides has been released into the environment. However, estimation of this released fraction proved uncertain not only due to limited knowledge about the accident scenario but also because of uncertainties in quantifying fission product retention and release (“pool scrubbing”) from water pools under accident conditions. The need for more comprehensive understanding of pool scrubbing is required under different hydrodynamic conditions, chemical boundary conditions, and characteristics of release gas into the water pool (e.g. mass flow, non-condensable gas fraction).

Some of the above-discussed open issues are addressed in OECD/NEA THAI-3 project launched in February 2016 for the duration of 3.5 years. By employing the THAI$^+$ test facility, it will be possible to generate data in THAI-3 project on hydrogen flame propagation in compartmentalized geometry for which uncertainties still show up in the modelling. Experimental data on this topic are sparse for severe accident scenarios under which large scale natural convection flow loops may develop within the containment driven by the mass and heat releases from the reactor circuit and the heat sink of the containment walls or even by safety systems, e.g. fan cooler, spray.

The foreseen source-term relevant experiments, namely fission product release from hot water pool involving “pool scrubbing” related phenomena and resuspension of pre-deposited fission products by hydrogen deflagration (“delayed source term”) are considered to be high priority at international level in the light of Fukushima accident, e.g. in CSNI report on Filtered Containment Venting System [23] released in 2014, and “pool scrubbing” e.g. in a report under preparation by OECD/NEA Senior Expert Group on Safety Research Opportunities Post-Fukushima (SAREF) [24].

6 SUMMARY

An overview of the OECD/NEA THAI projects and their role in validation and further development of lumped-parameter and computation fluid dynamics containment codes in the areas thermal hydraulics, hydrogen, aerosols and iodine is provided.

Experimental data produced in OECD/NEA THAI program have been continuously used for the validation and development of LP and CFD based computer simulation programs in the area of reactor safety. Major progress in measuring spatial hydrogen distributions, slow hydrogen deflagration behaviour, performance of PARs under accident-typical conditions, and fission product distribution inside containment and their interaction with operating safety devices has been demonstrated with the THAI test facility. The improved models based on

THAI experimental data have demonstrated reliable simulation of complex experiments, e.g. hydrogen distribution (OECD-THAI HM-2 code benchmark), PAR performance (OECD-THAI-2 HR-35 code benchmark), and hydrogen combustion behaviour (ISP-49), etc. Based on THAI test Iod-29 investigating I_2 release from flashing jet, lack of validation of existing iodine chemical models in the aqueous phase for reactor coolant circuit conditions relevant for design basis accident was also identified.

Remaining open issues related to hydrogen and fission product in containment of a water cooled nuclear reactor are being investigated in OECD/NEA THAI-3 project, which has been launched in February 2016 for the duration of 3.5 years. Hydrogen related investigations on PAR performance under counter-current flow conditions and hydrogen deflagration tests in two-compartment system are part of the work program. Additionally, in light of Fukushima Daiichi accident, experimental investigations are foreseen to study the re-entrainment of fission product (aerosols and gaseous I_2) from water pool at elevated temperature due to continuous heat-up of pool or depressurization (venting) induced boiling. Another planned experiment is related to "delayed" source-term in order to investigate resuspension of aerosols as well as iodine deposits from steel/painted surfaces due to hydrogen combustion. Analytical activities including blind and open code benchmarks are planned to assess prediction capabilities of LP- and CFD- based severe accident codes.

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APPENDIX 1: OECD/NEA THAI PROJECTS

Projects	OECD/NEA THAI	OECD/NEA THAI-2	OECD/NEA THAI-3
Period	01.2007 – 12. 2009	08.2011 – 07.2014	02.2016 – 07.2019
Project budget	2.8 M€	3.6 M€	4.75 M€
Experimental program			
Thermalhydraulics / gas distribution	H_2 and He stratification break-up by steam plume (**2 tests**)		
Hydrogen combustion **Hydrogen mitigation**	Hydrogen deflagration in premixed/ stratified gas atmospheres, single compartment tests (**29 tests**) PAR performance under severe accident conditions **(30 tests)** Tested PAR units: AREVA, NIS, AECL →onset, Ignition, O_2 starvation → Impact of PAR on I_2 In-containment source term (**1 test**) → PAR poisoning by fission products (**1 test**)	Hydrogen deflagration under spray operation, single compartments tests (**7 tests**) PAR performance In oxygen lean atmosphere **(10 tests)** → AREVA, NIS → onset, ignition	Hydrogen deflagration in premixed/stratified atmosphere, tests in two interconnected vessels (**6 tests)** PAR performance under counter-current flow conditions (**5 tests**) → AREVA, NIS
In-containment fission product behaviour (Iodine, Aerosol)	Aerosol (CsI) wash-down by wall condensate **(1 test)**	I_2 deposition on aerosol particles (**2 tests**) → Non-reactive (SnO_2) → Reactive (Ag) I_2 release from flashing jet under SGTR DBA scenario **(1 test)**	Fission product (aerosol, Iodine) re-entrainment from water pool at elevated temperature due to (**3 tests)**: → continuous heat-up → depressurisation induced boiling Aerosol and iodine re-suspension from deposits by hydrogen deflagration “delayed source term” (**1 test**)

Host country: Germany, **Project partners:**

OECD/NEA THAI [9 countries]: Canada, Czech Republic, Finland, France, Germany, Hungary, Republic of Korea, the Netherlands and Switzerland

OECD/NEA THAI-2 [11 countries]: Canada, Czech Republic, Finland, France, Germany, Hungary, Japan, Republic of Korea, Sweden, the Netherlands and UK

OECD/NEA THAI-3 [17 countries*]: Belgium, China, Czech Republic, Finland, France, Germany, Hungary, India, Japan, Korea, Luxembourg, Poland, Russian Federation, Slovak Republic, Sweden, Switzerland, and UK
**signing of THAI-3 Agreement is still in progress by some countries mentioned here; discussions are ongoing with additional countries*

Secondary side corrosion of SG tube alloys in typical secondary side chemistries

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Abstract:

SG tube alloys of all grades have been found susceptible to corrosion problems in laboratory tests simulating model secondary side crevice environments. However, the test results do not compare very well with the operating experience. Thus, IRSN performed an analysis of typical secondary side local environments based on data from operating steam generators. Relying on these data, corrosion testing has been performed to assess the potential susceptibility of SG tube alloys in such chemical conditions, and not solely model environments. The conditions of election of the local chemistries will be presented, as well as the results from the corrosion testing. Corrosion tests were performed on alloys 600TT and 690TT industrial tubes with typical chemistries derived from the operating experience of plants. This presentation will present some expertise of corrosion features and cracks encountered in the specimens.

1 SAFETY AND INDUSTRIAL FRAME

Secondary side chemistry-related issues have long plagued the operation of NPPs with a non-negligible impact on the safety level of NPPs. Among the many encountered problems one may mention denting, flow-accelerated corrosion, TSP clogging or stress corrosion cracking [1]. The latter, often named IGA/ODSCC can be described as the initiation and propagation of cracks in steam generator tubes. If these cracks propagate enough, they may lead to a primary to secondary leaks and, in the worst case, to a steam generator tube rupture (SGTR) which is a problem from both an operating and a safety points of view. In addition to impairing the functionnality of the steam generators, it also often results in primary to secondary leaks, i.e. a breach in one of the containment barriers. In case of SGTR, this could even result in release of radioactive species to the atmosphere. This is accordingly a topic of strong safety significance. SG tubes have a thickness of about 1 mm. Hence, when a corrosion phenomenon initiates, it is not necessarily long before it may perforate the tube. This causes heavy inspection campaigns of the SG tube bundles to detect these problems.

IGA/ODSCC occurs essentially in confined areas, such as at the tube-to-tubesheet gap, or at the tube-to-tube-support plate one, where chemistry deviates from the nominal one and some local stresses may be encountered. In a typical recirculating steam generator, there are about half a million of such crevices, which emphasizes the potential for damage and the complexity of the non-destructive examination (NDE) surveillance. In these crevices, the chemistry is everything but nominal. In the bulk secondary fluid, the level of impurities (halides, sulfur species, sodium, aluminosilicates …) is in the range of dozens of ppbs. However, in crevices, the amount of impurities can reach a few thousands ppm, i.e. 10^5 more concentrated than in the bulk fluid. This is due to a combination of factors:

- a crevice environment with restricted flow;
- some heat transfer through the crevice;
- the presence of corrosion products deposits (magnetite) which may act as a sponge for impurities.

Accordingly, in crevices, the chemistries will be heavily concentrated and the local pH may vary in the range between 2 and 11, prompting various possibilities for corrosion phenomena to occur. All of these phenomena have been extensively studied in laboratory until the mid 90s and the resulting operating experience recorded, see for instance [1 - 4].

IGA/ODSCC behaviour of steam generators has been greatly improved by the application of some mitigation measures:

- use of more corrosion-resistant SG tube alloys (600TT, then 690TT and 800);
- design including limited crevices;
- a more stringent control of the secondary fluid impurities (for instance sodium or raw-water ingresses).

2 CURRENT SITUATION AND TESTS PERFORMED BY IRSN

In spite of all these efforts, IGA/ODSCC was not totally suppressed, even though it affects only a limited amount of tubes per SG. Where in the eighties, a SG could contain thousands of ODSCC cracks, current steam generators only have a few dozen tubes affected. The situation has been clearly improved, albeit not totally from a safety perspective since a SGTR affecting only 2 or 3 tubes would prove to be a real safety issue.

The situation is however different than that of the 90s. Indeed, ODSCC is found to also affect SG tube alloys which are considered among the most corrosion-resistant ones, namely alloy 800 [3], sometimes even only after a decade or so of operation [5]. 600TT tubes also begin to be affected everywhere in the world [6]. This was at first surprising as the chemistry, even in crevices, had been largely improved, with local pH far less extreme than those from the past. Many expertises performed on pulled SG tubes affected by ODSCC have shown that the main chemical factor influencing the current cracking trend is the presence of residual sulfur or lead (Pb) in steam generators deposits [7], often in association with hard sludge deposits on the tubesheet.

In this new situation, one may not simply rely on past R&D tests to try assessing the cracking risk of SG tubes, hence the associated safety impact. Indeed, the pH now mentioned by utilities are rather in the range 4 to 9, in the range where very few laboratory tests used to be performed. In addition, the Canadian industry pointed out the significant effect that magnetite deposits could have on the electrochemical conditions in crevices. The current chemistries are rather the consequence of a “slow” build-up of residual impurities in the deposits on the secondary side surfaces, which is slightly different from past chemistries due to sudden ingresses of pollutants.

Thus, IRSN decided to launch a new round of testing to assess the ODSCC risk for alloys 600TT and 690TT, with the following parameters:

- pH in the range 4 to 9;
- use of specimens sampled from actual industrial SG tubes;
- use of pre-passivated specimens, as the pollutions are not present at the first start of a steam generator;
- an amount of detrimental pollutants slightly above the average amount in actual SGs;
- use of secondary side deposits in adequacy with the analysis of deposits removed from plants over the last 20 years;
- consideration of a triphasic environment (sludge, crevice-simulating liquid and wet steam) to represent all the boiling locations on the top of the tubesheet.

These parameters, adapted to the analysis of the maintenance operating experience over two decades provide, according to IRSN, with results more accurate to the present situation of steam generators, with “typical” local chemistries.

The presentation associated with this paper presents the main current results, considering that many analyses remain to be performed. However, some interesting first results are already available:

- 600TT is more susceptible to ODSCC than 690TT, whereas some past tests implied the opposite in presence of Pb;
- The ODSCC features observed of 600TT and 600MA reference specimens are in agreement with expertises results from the field;
- In mildly acidic conditions, 600TT and 690TT are susceptible to ODSCC, in the test conditions, even without Pb but with S alone.

These preliminary results are summarized in the table below. (GC: generalized corrosion)

		pH_T= 4			pH_T= 7,5		
		WF422 600MA	WF489 600TT	116201 690TT	WF422 600MA	WF489 600TT	116201 690TT
Pb (PbO)	Solid	C	C		C		
	Liquid	-		C	-	-	-
	Steam	-		C	-	-	-
Pb and S ($PbSO_4$/PbS)	Solid	C	C		C	C	
	Liquid	-	C	C	-	C	
	Steam	-	C	C	-	C	
S (Na_2SO_4/Na_2S)	Solid	C					
	Liquid	-	C	C	-		
	Steam	-	C		C	C	-

No analyzis	-
GC and/or IGA	
No indications	
C	Cracks observed from the surface

Some other tests in mildly alkaline conditions are still in progress and will end by end of december 2016 and the specimens will be analysed next year, in addition to the specimen tested in quite-neutral conditions.

3 CONCLUSIONS

Due to the evolution of the chemistry management of steam generators, ODSCC has become a much less prominent problem for NPPs. However, the operating experience indicates that despite this, even SG tubes alloys as resistant as alloy 800 may crack in a decade. The safety risk is still present.

Reviewing of the past tests and the current chemistries of steam generators lead the IRSN to conclude that some new series of tests had to be launched to assess the current risks of ODSCC in modern steam generator tube bundles. IRSN launched a program of tests integrating typical chemistries based on review of the operating experience and actual industrial materials.

The first results of these tests cope with the operating experience of alloy 600TT and 600MA which indicates the relevance and accuracy of the test conditions. The preliminary results also show some risks in mildly locally acidic conditions for 690TT, not easily derived from past tests, but that an expert opinion could expect.

Accordingly, IRSN considers that this R&D program, although not yet finished, already provides, and will go on providing, with results reinforcing the safety assessment capability for steam generators.

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Fuel coolant interaction modelling in sodium cooled fast reactors

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Abstract:

One of the important challenges in modelling the core melt progression during a severe accident in a sodium cooled fast reactor is related to the consequences of energetic fuel sodium interaction. Currently the applicability of the existing fuel coolant interaction codes for the fuel sodium interaction is under examination.

The aim of the paper is to highlight the research priorities for the melt-sodium interaction in comparison to the melt-water interaction. Namely, the modelling capabilities of the fuel coolant interaction codes to cover the most energetic events in the light-water reactors were already demonstrated in the frame of international programs.

Our analysis has shown that the main challenges for the fuel coolant interaction modelling in sodium are related to the continuous melt fragmentation, the heat transfer in highly sub-cooled conditions and to the approach to model pressurisation. In future, more experimental data with sodium are necessary for validation.

1 INTRODUCTION

The constructions of several demonstration-scale sodium cooled fast reactors (SFR) are planned in different countries, one of them being the ASTRID reactor (Advanced Sodium Technological Reactor for Industrial Demonstration) in France. In the frame of safety studies for the demonstration-scale reactors the risk for the environment in a severe accident must be estimated.

An unprotected transient over-power or a loss of coolant flow can result in the core melt of SFR. Relocation of melt to the region containing liquid sodium or relocation of liquid sodium to the region containing melt could result in a complex thermo-dynamical interaction between melt and sodium named fuel coolant interaction (FCI). During the melt relocation into sodium the continuous melt fragments into the melt droplets of an order of mm in diameter, the melt droplets quench in a period up to a few seconds and finally the melt droplets might form a debris bed. To stop the melt progression and to maintain the integrity of the surrounding systems, structures and components, the debris bed must be coolable. Additionally, a vapour explosion may be triggered. In this energetic FCI phenomenon a part of the melt energy is rapidly transferred to the coolant during the fine fragmentation process of the melt droplets in a very short time scale (i.e. few ms), leading to a fast vaporization (or important density changes) and thus to the generation of shock waves through a process similar to detonation. Potentially severe dynamic loadings of the vapour explosions on the surrounding systems, structures and components could be induced. Evidently, the issue of FCI phenomena is important for nuclear safety. Explosive events are thus considered to occur only through the development of those two steps: preliminary mixing, called premixing, and secondary explosive mixing.

It is noteworthy that FCI modelling, in the frame of reactor safety, started concurrently for both cases of interaction with sodium and with water. In the past, several experiments (e.g. BETULLA (JRC, Italy), FRAG (SNL, USA), MFTF (Winfrith, Great Britain),TERMOS (JRC, Italy), THINA (KfK, Germany)) were launched to help understanding and characterizing the FCI phenomena where corium is poured in sodium [1-3]. Some of the experiments (e.g. CORRECT (CEA, France), MFTF-B (Winfrith, Great Britain)) were also performed to help understanding the FCI phenomena during the re-entry of sodium onto melt [1]. However, in the nineties, the focus has been put on the case with the water, and the applicability of the current understanding and modelling of FCI to the case with sodium is not clear.

The capabilities of the FCI codes to cover the fuel water interaction in the reactor cases were demonstrated in the frame of the OECD SERENA (Steam Explosion REsolution for Nuclear Applications) and EU SARNET (Severe Accident Research NETwork of Excellence) programs [4, 5]. Because of large differences in the thermo-dynamical and physical properties of sodium compared to water, the applicability of the FCI codes for sodium must then be investigated. For example, the applicability of the premixing and explosion models in the MC3D code (IRSN, France) [6, 7] to simulate FCI with sodium is currently under examination.

The objective of the paper is to review the key FCI processes shown in Figure 1 for the premixing and explosion phases. The aims are to identify the current status of the processes understanding and modelling and to highlight research needs.

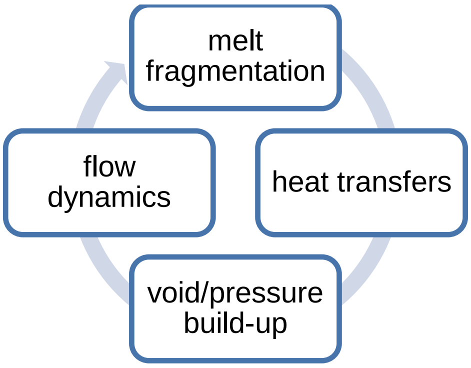

Figure 1: Schematic diagram of complex interaction during FCI phenomena.

2 IMPLICATIONS OF DIFFERENT PHYSICAL AND CHEMICAL PROPERTIES OF SODIUM COMPARED TO WATER

The differences of the physical properties may already induce a quite different behaviour. Thus, the main differences between the water and sodium properties given in Table 1 are considered in the discussion.

Table 1: Basic sodium and water properties [8, 9].

Property		H_20	Na
At 0.1 MPa	T_{sat} [K]	373	1153
	ρ_{liq} [kg/m^3]	959	743
	ρ_{vap} [kg/m^3]	0.59	0.28
	$c_{p,liq}$ [J/kg/K]	4215	1270
	$c_{p,vap}$ [J/kg/K]	2078	2565
	μ_{liq} [μPa·s]	283	159
	μ_{vap} [μPa·s]	12	18
	λ_{liq} [W/m/K]	0.68	49
	λ_{vap} [W/m/K]	0.02	0.05
	L [kJ/kg]	2258	3626
Critical point	T_{crt} [K]	647	2504
	p_{crt} [MPa]	22.06	25.64
	ρ_{crt} [kg/m^3]	322	219

It is firstly noticed that the difference in density is rather mild, compared to the corium density, and may not play a role. Similarly, the difference in viscosity may not lead to a noticeable behaviour.

On the contrary thermo-physical properties as the heat capacity and the thermal conductivity of the liquid phase are very different. The thermal conductivity of sodium is higher by nearly two orders of magnitude. The implications should be very strong. In particular, as long as the temperature is lower than saturation, the void production (boiling) should be strongly limited. Also, the temperature should be more homogenous in the liquid. Finally, the smaller heat capacity (nearly a factor 4) leads to consider that the energy storage capacity of sodium is rather limited. Considering these properties together, an improved transfer of energy into the liquid part may be expected, up to the moment when the liquid will reach saturation (quite homogeneously), with a sudden and intense bulk boiling.

Concerning the thermo-dynamical properties, it is highlighted that the saturation temperature is much higher for the sodium. Combined with the effect of thermal conduction, this should induce rather different boiling processes. In water, film boiling is the dominant boiling mechanism. In the case of sodium, this mechanism might play only a minor role, at least during the premixing phase.

Another important feature is anticipated from the vapour pressure curves in Figure 2. The boiling at high melt temperatures is possible at all pressure levels for sodium and water. At lower melt temperatures the boiling is still possible for water but it becomes less probable for sodium. As it is seen in Figure 2 for sodium, the saturation temperature is strongly evolving with pressure. The critical temperature is very high, about 2500 K, i.e. nearly the melt temperature. Then, in contrast with water, pressure far beyond the critical one may not be achievable with sodium during a vapour explosion.

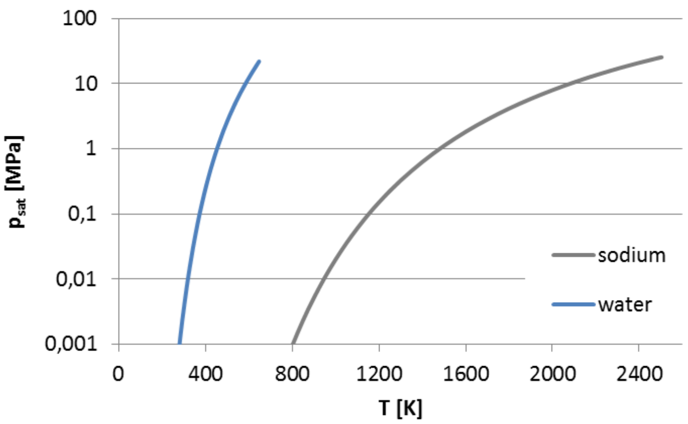

Figure 2: Comparison of sodium and water vapor pressures [8].

To summarize, it is clear that the different thermo-physical properties of sodium should affect strongly the heat and mass transfers in both premixing and explosion processes. Following the diagram of Figure 1, this may affect in turn the fragmentation. Also, since higher heat transfers may be expected (large conductivity, small void), a faster solidification of the melt (at comparable superheat) is awaited.

Another important difference, definitely positive, is the absence of chemical interactions as oxidation. In the interaction with water, oxidation may be very important and induces strong complications and uncertainties for modelling and evaluating the FCI.

3 MODELLING CHALLENGES IN PREMIXING PHASE

In this section the current understanding and approaches to model the melt fragmentation, the heat transfer and the void build up in the premixing phase are discussed. It is recalled that the premixing phase is important to determine the initial conditions of a possible vapour explosion. In the absence of vapour explosion, premixing will nevertheless drive the formation of the debris bed on a core catcher and thus the coolability of the corium.

3.1 Melt fragmentation

In the premixing phase the melt fragmentation process is often divided into two stages. The primary fragmentation is related to the continuous melt fragmentation and is the key process that defines the initial and boundary conditions for subsequent processes. The continuous melt could be either a jet either a stratified pool. The secondary fragmentation process is related to the further melt droplet break-up. In fact, this differentiation is often more a matter of convenience for the modelling than a reality in which the processes, at different scales, are often intermixed. Such two-step fragmentation model is available in the latest version of the MC3D code. Basically, the primary fragmentation can be used to evaluate the global fragmentation rate and thus the break-up length of the jet, whereas the secondary fragmentation is the process which determines the final drop sizes.

The jet fragments are due to various instabilities created at the melt-coolant contact. For water the major difficulty is related to the quite intense boiling which can occur at low sub-cooling and/or large scale. This boiling process leads to the formation of a more or less large vapour film around the jet, in the core of the mixture. However, in the frame of the OECD SERENA project a consensus on the dominating role of the Kelvin-Helmholtz (KH) mechanisms on the vertical jet fragmentation seems to be achieved for water [4]. As already discussed previously, the differences of physical properties are not sufficiently important to anticipate differences in the fragmentation rate. Indeed, as seen in Figure 3 the average jet fragmentation rate for sodium may be considered as comparable to the water cases.

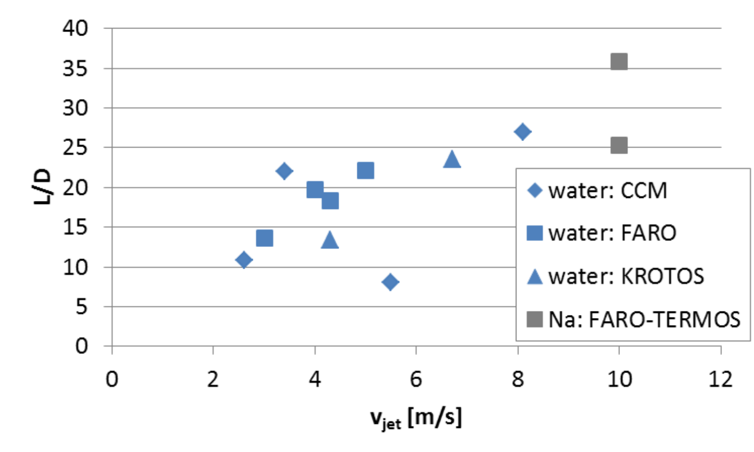

Figure 3: Ratio of jet break-up length (L – characterizing the position of complete jet fragmentation) and jet impact diameter (D) for experiments with water and sodium [2, 6]. Uncertainties for the given sodium case are importantly affected with the approach to assess the impact velocity from free fall, impact diameter from conversation of mass and length from destroyed meshes used in the FARO-TERMOS T2 experiment.

However, in the case of sodium, as long as the saturation is not reached, we do not await an important boiling. In fact, we might anticipate two different behaviours: a quasi liquid/liquid behaviour with small impact of boiling or a strong impact of the boiling process as it is known that transition boiling (and also nucleate) is a quite dynamic process. For the strong impact of the boiling process the thermal processes might then play a more important role compared to the water case. With the first hypothesis, which may happen when film boiling occurs (the film being very thin), the models based on the KH formulation for fragmentation show a weak impact of the velocity for the break-up length. The dimensionless break-up length is then quite constant, of the order of 20±10 for the considered conditions. The results shown in Figure 3 might lead us to such conclusion. FARO-TERMOS T1 and T2 tests are the most representative experiments at our disposal, involving 60 and 45 kg of UO_2 in interaction with sodium [2]. In both cases, the dimensionless fragmentation length (above which the jet if fully fragmented) was of the order of 30. However, the experiments with sodium all show a turbulent behaviour, attributed to transition boiling, accompanied by “pressure events”. In the

FARO-TERMOS experiments, these "pressure events" looks like local steam explosion, with very sharp pressure peaks. However, these events do not transform into generalized explosions. These thermal effects on the fragmentation rate should then be studied with more precision.

The detailed debris characteristics of FARO-TERMOS test are not at our disposal. The report indicates debris smaller than 1 mm on the mean [2]. More details are given in [3] regarding the BETULLA tests which were done for the preparation of FARO-TERMOS. A test with UO_2 involved about 2 kg of melt. The analysis of the debris highlights a different behaviour compared to cases with water. As seen in Figure 4, the median particle size is slightly above 1 mm quite similarly to KROTOS test with water and similar masses of melt. However, there is an important part of very small particles on the BETULLA tests, leading to a much smaller mean Sauter diameter. The analysis of debris of the BETULLA experiments shows however a predominance of final debris with a fractured shape. However, it is difficult to infer for a precise specific mechanism since such small diameter is representative of an explosion test in water, so it seems unlikely that this fine fragmentation occurred while the melt was hot. In contrast, according to the BETULLA report, the large fragments where with smooth rounded shape, and with a large hole in the centre showing that the melt fragmentation in liquid state occurs in mm sized particles and the solidification is likely to proceed as a growing crust. As a consequence, the fragmentation processes need to be clarified.

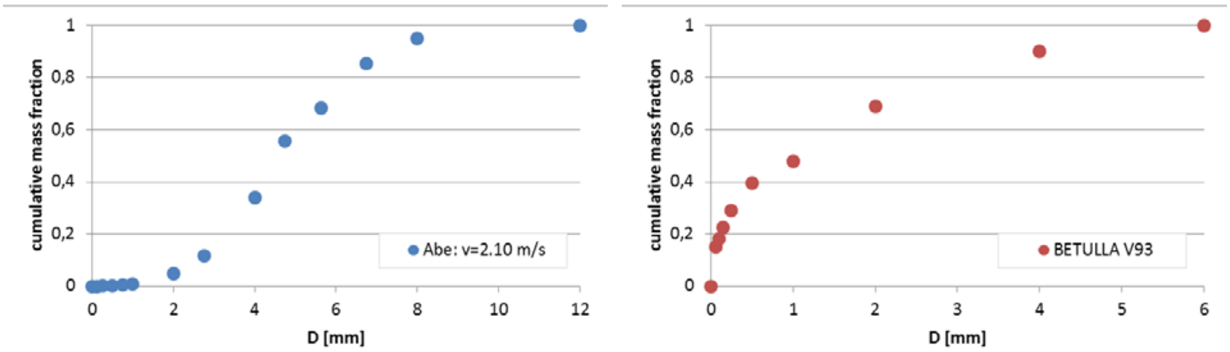

Figure 4: Melt mass distributions in a test with low boiling in water (left) and a case with UO2 in sodium (right) [3, 10] .

If the boiling has a low impact, one can compare with the experiments of liquid-liquid fragmentation with low (or no) boiling (Figure 4). It is clear that the distributions are not comparable, due to the very large fraction of very small particles in the UO_2/Na case. Thus, for further research, there is a clear need to characterize the impact of the thermal fragmentation and the impact and mechanism of the observed very fine fragmentation.

The solidification is recognized, in the corium-water case, to be a strong mitigating process. In the case with sodium, due to the particular properties of sodium discussed above, solidification of the melt is anticipated to be even more important. Indeed, the data tend to indicate a fast solidification:

- the large particles show a solidification progression as a growing crust, whereas the process of solidification is still a matter of question for water;
- the fine fragmentation, present in water cases but at far lower amounts, is likely to originate from thermal stresses, which indicates then also a strong and fast solidification.

Then apart from an improved understanding of the solidification processes and impact on fragmentation (limiting for water and may be promoting for sodium), it seems compulsory for the modelling to represent it adequately. It is hard to account all the subsequent processes if all the melt droplets are presented with one group. In the MC3D code, the multi size group (MUSIG) approach is considered both for sodium and water cases. It is considered at the moment that the very fine fragmentation originates from solidification and occurs quite lately. Thus, at first order, it might be not taken into account.

The data indicate that large droplets solidify through a growing crust. In such case, for partly solidified droplets, a modified Weber number was proposed to assess critical conditions for the hydrodynamic fragmentation in the presence of a crust [11]. In this case modelling of melt droplet solidification is required and proposed in most of the current FCI codes [12]. The applied melt droplet quenching models and the modified Weber numbers are physically based and could be therefore applicable not just for water but also for sodium. In fact, the applicability of such models for real cases in water is still debated, particularly if one considers the question of oxidation. They might be better suited for the sodium case.

Finally, in sodium the thermal secondary fragmentation might be more important than in water. It is suspected that this phenomenon is at the origin of spontaneous explosions with alumina melts in some KROTOS tests. Although there are still some debates, it is largely recognized that, at least with UO_2/ZrO_2 melts, this phenomenon should be of second order in water. In sodium, the thermal models might be considered for rapid bubble condensation in the sub-cooled sodium near the liquid melt droplet surface potentially being violent enough to break up the droplet. The models should also consider the existence of the thermal stresses in the crust [13].

Due to uncertainties in the primary and secondary fragmentation (e.g. successive repeated fragmentation) both parts could be combined together in the modelling approaches for water. Similar approach might be considered applicable also for sodium.

3.2 Heat transfer

During the premixing phase the heat transfer mainly occurs between the melt droplets and coolant. Typical melt surface temperatures are in the range of more than 3000 K and the coolant bulk temperature. In this range the radiation, convection and conduction must be considered. The boiling curve is used to characterize the various heat transfers. The boiling curve is commonly divided into the boiling regimes. In the typical temperature range the quenching melt should normally experience the film boiling regime, the transition boiling regime, the nucleate boiling regime and the convection regime.

As discussed in Section 2, in water the heat is mainly transferred in the film boiling regime whereas in sodium the heat is most probably importantly transferred not only in the film boiling regime but also in the transition boiling regime.

3.2.1 Radiative heat transfer

The Stefan-Boltzmann law defines the heat transfer by radiation between the melt and coolant. The effective emissivity is defined by the emissivity of the melt and surrounding coolant. The emissivity in water is approximately 0.9. For sodium and for temperatures below 900 K the total normal emissivity is below 0.05 [14]. The sodium and water vapour could be considered as a transparent medium.

3.2.2 Film boiling regime

The film boiling heat transfer in water is well characterized [15]. The theoretical background of the Epstein-Hauser (EH) approach makes it the preferred choice for the characterization of the film boiling heat transfer in the FCI codes [16]. In our EH based approach the saturated and the sub-cooled part of the EH correlation were reweighted and the EH coefficient was readjusted. The results of our approach are presented in Figure 5. It may be seen that the ratio of the experimental and calculated heat fluxes are in reasonable agreement (ratio around 1). On the theoretical level the EH approach could also be applicable for sodium. However applicability shall be demonstrated with experiments.

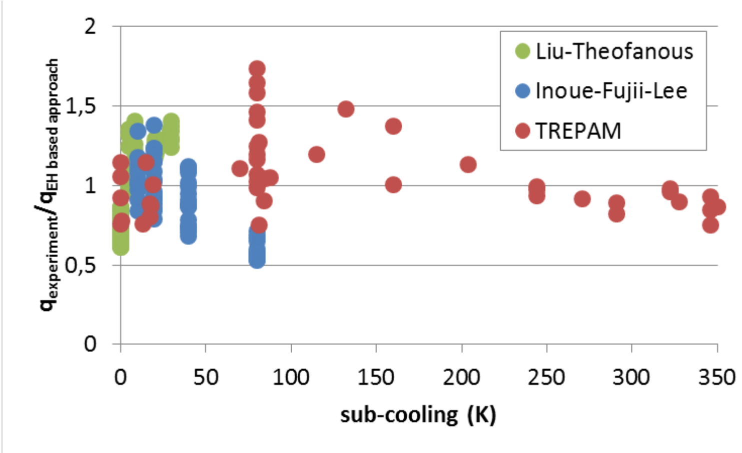

Figure 5: Comparison of experimental data [15, 17, 18] and the EH based approach.

As seen in Figure 6, in some experiments with the sub-cooled water and the surface temperature above the homogeneous nucleation temperature the heat transfer was higher than typically observed in the film boiling regime [19]. Thus, it seems that at such conditions hypothetically two modes of the film boiling regime could be considered and discussed using the vapour film destabilisation (VFD) temperature as a criterion for the transition between the modes. If the surface temperature is above the VFD temperature the heat transfer occurs in the stable mode of the film boiling regime. Below the VFD temperature the heat transfer increases due to the presence of instabilities on the strongly reduced film thickness. However, in water, this should occur for conditions when the melt is already strongly cooled and solidified. The impact is thus weak and it is not needed at first order. This is clearly not the case in sodium. The existence of such conditions during FCI in sodium shall be experimentally investigated because the expected sub-cooling in the sodium cooled fast reactor is in the range of few hundreds K.

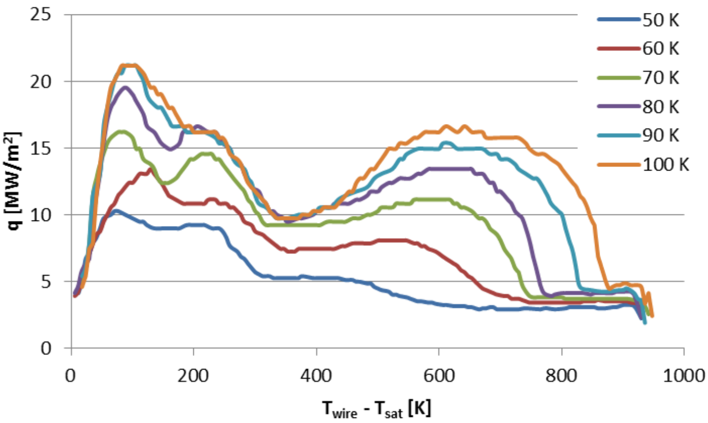

Figure 6: Experimental measurements of heat flux during quenching of 0.3 mm wire in 50-100 K sub-cooled water and relative velocity of 1 m/s. Data are extracted from [19].

The performed experiments with water and sodium, dedicated to the minimum heat flux (MHF) temperature could be used to propose the models for the minimum film boiling (MFB) and VFD temperatures [19-22]. But, experimental results are uncertain and those uncertainties could be related to the surface conditions, diameter, sub-cooling and flow velocity. Additionally, experimental investigations on the transition from the film boiling towards the transition boiling regime shall be performed.

3.2.3 Transition boiling regime

At the moment the interpolation approach to model the heat transfer between the maximal and minimal heat fluxes could be considered as sufficient for the FCI codes to model the FCI interaction with water and sodium.

3.3 Void build-up

The amount of void in the mixture importantly affects the vapour explosion strength. The modelling approaches strongly depend on the modelling of the heat dissipation and on the size of the bubbles and are discussed hereafter.

The main issue in the modelling of radiative heat transfer is related to the dissipation of heat in the surrounding of the droplet. The heat is dissipated either at the vapour interface in case of the film boiling or in the surrounding coolant. The long-range or the intra-cell radiation model could be considered for dissipation to the surrounding coolant. The importance of the issue increases with the presence of void in the mixing zone. Different radiative heat transfer models were developed and implemented into the FCI codes. For liquid sodium the approach to model dissipation to the bulk seems to be less relevant than for water. It seems that the same approach could be applicable for the sodium and water vapour.

Useful data related to the void build up in the film boiling regime are hard to be obtained with experiments. Thus analytical assessments of the energy distribution are being performed. As seen in Figure 7, the assessed fraction of the heat used for the vapour production during the TREPAM experiments significantly depends on the coolant sub-cooling [17]. Recently for sodium the scaling analysis was used to estimate the heat flux distribution for the liquid heating and for the vaporization [23]. The scaling analysis allows estimating the heat fluxes within a correct order of magnitude when compared to the pool-boiling experiments with sodium. The analysis has shown that the vaporization is very weak at sub-cooling around 30 K. Since the bulk temperature of sodium is typically sub-cooled by several hundred K the heating up of the bulk is expected prior to any vaporization.

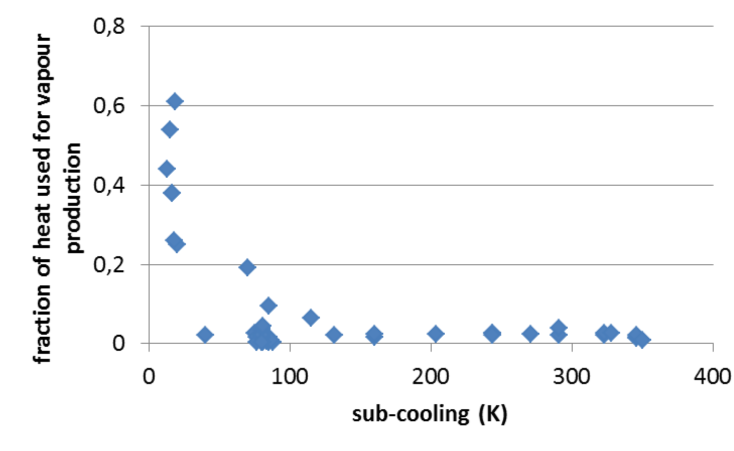

Figure 7: Fraction of heat used for vapour production in TREPAM experiments [17].

For the film boiling regime a simple parametric approach might be used in the FCI codes. In such an approach, all the heat is used for vaporization when the coolant is at the saturated conditions. If the coolant sub-cooling is above a limiting value all the heat is used for liquid heat-up if the film exists. Between the limits an interpolation might be used. The size of the generated bubbles is a user parameter and should be adopted for different sub-cooling and coolants. Another approach could be based on the continuous vapour generation and the bubble detachment. In the sub-cooled liquid the created bubbles condense. The EH approach could be used for assessment of the generated bubbles with diameter equal to the diameter of melt droplets. The generated bubbles condense in the sub-cooled bulk. Both approaches seem to be applicable for sodium and water.

Data from the FARO-TERMOS experiments [2] and the first evaluations with the MC3D code indicate a particular behaviour for the sodium case due to the particular physical properties. Due to the very strong conduction, the heat should be easily diffused in the liquid, thus limiting the void production for a long time. However, the heat capacity is small, compared to water. The conjunction of these two properties will thus lead to an important bulk boiling since large portions of the sodium might reach simultaneously the saturation temperature. This effect is balanced by the much more important saturation temperature and thus potential sub-cooling (so that the $\rho c_p \Delta T$ is quite comparable with water). Nevertheless, the global boiling picture should be different, with a much gradual boiling with water, with a same global heat sink capacity but a smaller conduction, i.e. heat diffusion.

4 MODELLING CHALLENGES IN EXPLOSION PHASE

The fine fragmentation is one of the key processes affecting the strength of the vapour explosion. The fine fragmentation process rapidly increases the melt surface area, vaporizing more coolant and increasing the local vapour pressure. Additionally, the strength of vapour explosion also depends on the presence of void and the ability of the coolant to evaporate. In this section the current understanding and approaches to model the melt fine fragmentation, the heat transfer and the pressurisation are discussed.

4.1 Fine fragmentation

The fine fragmentation could be due to the hydrodynamic forces or due to the thermal forces.

For water cases the hydrodynamic fragmentation is considered as dominant. In the modelling approaches for the hydrodynamic fine fragmentation the critical fragmentation conditions, the fragmentation rate and the fragments size are considered. The applicability of the Weber number (based on the surface tension) and the modified Weber number (based on the elasticity of crust) for defining the critical conditions for the fine fragmentation of liquid and partly solidified droplets in water was already demonstrated with the comparison to experimental data [11]. Both dimensionless numbers are physically based and therefore considered applicable for sodium and water. The fragmentation rate depends on the break up mode for which different mechanisms of fragmentation are possible (e.g. bag, shear, striping). The size of fragments could be evaluated, considering the local conditions and the melt properties using the Weber number, or could be a user parameter. It seems that similar approach to define the fine fragments size is applicable for sodium and water.

Current models for thermal fragmentation consider the fragmentation as a result of the vapour film destabilisation around the melt droplets. However, actual mechanisms are still under discussion.

For sodium the importance of the thermal fragmentation has to be examined.

4.2 Heat transfer

As indicated in Figure 5, it seems that the Epstein-Hauser (EH) approach could be sufficient also for the explosion phase. However as seen in Figure 8, for water the experiments mainly cover the premixing conditions. For water additional experimental data would be useful for the explosion phase where the expected relative velocities are higher. To the best knowledge of the authors no experiments with sodium were dedicated to the film boiling regime at relevant conditions. As indicated in Section 3.2.2, the EH approach could be considered appropriate also for sodium.

Experiments with sodium covering also significant velocities, high pressures and large sub-cooling shall be performed to assess the mode of heat transfer and the EH approach.

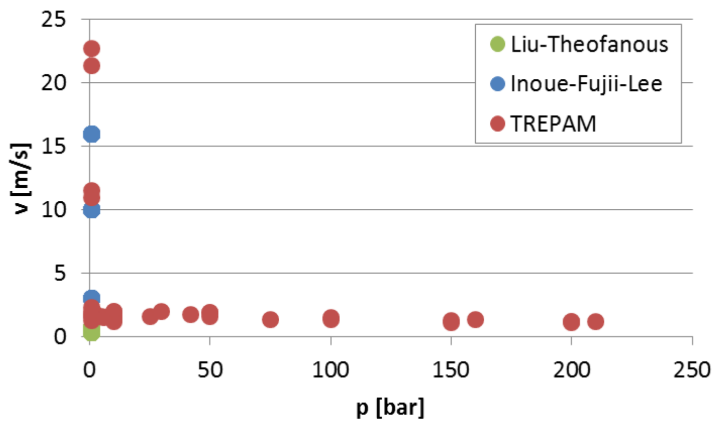

Figure 8: Parameters map for different heat transfer experiments performed at conditions relevant for FCI [15, 17, 18].

4.3 Pressurisation

One of the main challenges in modelling the explosion phase is related to the precise nature of the pressurization mechanisms. Pressurisation during the explosion phase is mainly governed by significant increase in the melt-coolant surface area during the fine fragmentation. The fine fragments limit the energy for fast heat transfer. Understanding the heat transfer processes between the fine fragments and the coolant seems to be crucial for the pressurization modelling.

As seen in Figure 9, two modelling approaches are currently used in the FCI codes for the evaluation of pressurization during the explosion phase in water [12]. In the direct boiling approach, applied also in the MC3D code, the pressurization is due to the direct boiling at the interface of liquid and vapour. The energy used for the coolant vaporization is equal to the difference between the energy leaving the melt fragment and the energy used for the bulk liquid heating. Currently the main challenge is to define the rate of heat used directly for vaporization if the bulk temperature is not close to the saturation temperature. Next, the range of the heat transfer to the coolant, i.e. only in the vicinity of fragments or to the bulk, must be defined. Additionally, the importance of the vapour condensation in the sub-cooled conditions on the heat transfer must be assessed. Other challenges are related to the mode of heat transfer at significant velocities and high-pressures. In contrast, the micro-interaction approach states that the heat is propagated in the coolant up to a given distance (more properly to a given volume). The rational is not so clear and the distance (volume) is selected on the basis of confrontation of results to experimental data. This is called the entrainment factor. In the approximation of this concept the pressurization is due to the heating and possible bulk boiling of part of the coolant that is in thermal equilibrium with the melt fragments. It is assumed that the fine fragments quenching time is shorter than the fine fragmentation time. One of the main challenges of the micro-interaction concept is to define the entrainment factor that connects the fragmentation rate and the entrainment rate of coolant into the mixture. The entrainment factor is selected on the basis of confrontation of results to experimental data. Such an approach neglects the mode of the heat transfer between the fragments and coolant.

An important future challenge is to assess the applicability of both approaches for vapour explosions in sodium. As discussed in Section 2, in average in sodium the boiling might be limited to moderate pressures (with a limit close to critical pressure). Moreover, the pressurization might be limited due to strong condensation in the sub-cooled bulk of sodium. Higher thermal conductivity (nearly a 2 orders of magnitude) and lower heat capacity for sodium than for water promotes bulk heating and faster heating of sodium. Thus for the micro-interaction approach the entrainment factor is probably to be readjusted. Experimental data are necessary for this. At least, an in-depth analysis is necessary.

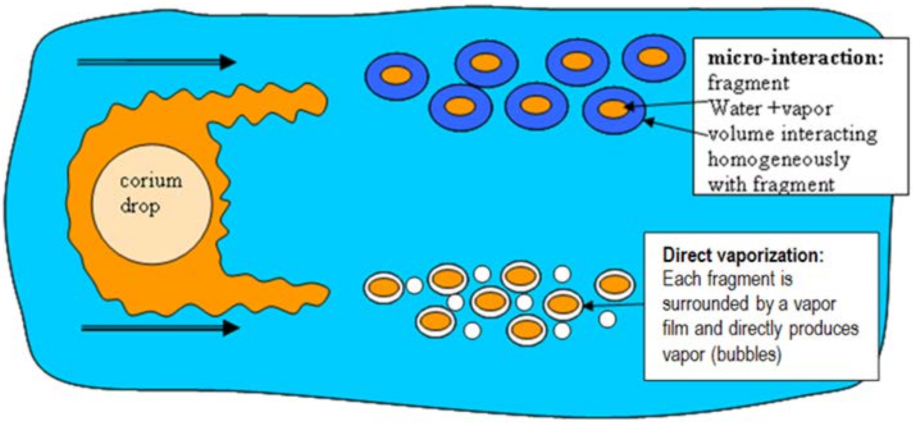

Figure 9: Major differences between modeling approches for evaluation of pressurisation during the explosion phase.

The MC3D code model is based on the direct boiling approach. However, preliminary results showed that strong vapour explosions, beyond the critical point, could be obtained. Indeed, in MC3D, the pressurization is also obtained by the change of density of the liquid with temperature. The difference with the micro-interaction concept is that it does not consider a limiting distance or volume for heat transfer (heat is transfer homogenously to the liquid in each cell).

5 CONCLUSIONS

To enable FCI simulations with sodium the status of understanding and modelling approaches were reviewed in the paper. Furthermore, the needs for future improvements were highlighted. The main conclusions related to sodium are stressed in Table 2 for the premixing phase and in Table 3 for the explosion phase.

Table 2: Modelling of premixing phase in sodium: overview of understanding and modelling status, and needs for further improvements.

	Status		Needs	
	Understanding	*Modelling*	*Experimental*	*Analytical*
Melt fragmentation	Experimental data and comparable governing sodium and water properties are indicating that similar jet fragmentation mechanisms are acting in water and sodium. An important fine fragmentation occurs in premixing in sodium, the origin should be understood. The secondary fragmentation might be important. The impact of thermal fragmentation must be assessed.	Combined jet and secondary fragmentation. Kelvin-Helmholtz approach. Multi-size group modelling.	Impact of jet diameter, velocity and sodium sub-cooling on break-up length and debris size spectrum.	Tuning of models to experiments. Impact of solidification on melt fragmentation. Assessment of impact of thermal fragmentation (fragmentation through thermal stresses process, thermal fragmentation).
Heat transfer	Film boiling and transition boiling regime are important. On theoretical level the Epstein-Hauser approach to model film boiling might be applicable. At strong sub-cooling the behaviour in film boiling regime might change from stable to unstable mode.	Epstein-Hauser based approach for stable mode of film boiling regime. Various correlations available. Interpolation in transition boiling regime.	Quenching experiments covering film boiling and transition boiling forced convection experiments. Impact of sub-cooling on film boiling heat transfer and on transition from film boiling regime towards transition boiling regime.	Tuning of models to experiments. Film boiling models for highly sub-cooled conditions. Criteria for temperature range of film boiling and transition boiling regimes.
Void build-up	Sub-cooling strongly effects vaporization and condensation. Higher surface temperatures are needed for important vaporization in sodium than in water (differences in vapour pressures). Bulk heat-up and vaporization might be more important in sodium than in water (differences in thermal conductivity).	Long-range or intra-cell dissipation of radiative heat transfer. Parametric dissipation in film boiling regime. Continuous vapour generation in film boiling regime.	Integral experiments with void history.	DNS like simulations for assessing fraction of heat used for vaporization. Long-range vs. intra-cell radiation. Dissipation of radiation at vapour-coolant interface. Vapour bubble condensation in sub-cooled conditions.

Table 3: Modelling of explosion phase in sodium: overview of understanding and modelling status, and needs for further improvements.

	Status		Needs	
	Understanding	*Modelling*	*Experimental*	*Analytical*
Fine fragmentation	Importance of thermal fine fragmentation is not clear.	Weber and modified Weber numbers for critical conditions of liquid and partly solidified melt droplets. Parametric determination of fragments size or use of Weber number for liquid droplets.	Hydrodynamic fine fragmentation to define fragments size, effect of solidification, break-up mode. The impact of thermal fragmentation on triggering, propagation and loads should be assessed.	Tuning of models to experiments. Impact of solidification on droplet fine fragmentation. If necessary, model for thermal fragmentation
Heat transfer	On theoretical level Epstein-Hauser approach to model film boiling might be applicable.	Epstein-Hauser based approach.	Impact of relative velocity, ambient pressure and sodium sub-cooling on heat transfer.	Tuning of models to experiments.
Pressurisation	Higher surface temperatures are needed for important vaporization in sodium than in water (differences in vapour pressures). Limitation to sub-critical pressure is not demonstrated yet.	Direct boiling approach. Micro-interaction approach.	Explosion data for assessment/tuning of models.	DNS like simulations to understand heat transfer processes during explosion, to assess pressurization and to tune model parameters.

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Seminar 3

Waste Management and Decommissioning & Dismantling

Establishing decommissioning plans and the decommissioning of the fuel facility FBFC in Belgium

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Abstract:

FBFC, subsidiary of AREVA, produced fuel assemblies for nuclear reactors of the PWR type. Most of the activities (production of uranium / gadolinium tablets, pins and elements, stoppers and springs for fuel elements) were stopped in 2012. Only the MOX fuel assembly facility was maintained until 2015.

A first dismantling license was granted by Royal Decree in 2010 for the dismantling of buildings 1 (laboratory), 2 (gadolinium production), 3 (waste water treatment) and 5M (MOX). After the notification by AREVA in 2012 to the FANC that AREVA had decided to stop all activities in the FBFC facility, a dismantling license was granted by Royal Decree in 2013 for the dismantling of building 5 (Uranium). These decommissioning licenses set the conditions to ensure the safe decommissioning of the buildings with specific attention to the use of subcontractors with necessary training and experience, the use of best available dismantling techniques, including decontamination of contaminated metals, the clearance of decommissioning waste and the final release of the buildings.

After ending the production activities, a number of technical risk-reducing measures were taken (removal of fissile materials, disconnection of electric cables, …).

FBFC started the decommissioning activities in 2012. The goal is to have all buildings dismantled, as well as the FBFC site released from nuclear control (after the treatment of the remaining contaminated soil) by 2018.

1 INTRODUCTION

FBFC International, affiliate of the AREVA group, operated a Low Enriched Uranium fuel manufacturing facility in Dessel from 1958. Since 1997, FBFC also operated a large scale MOX fuel assembly facility starting from sealed MOX pins delivered by subcontractors.

FBFC produced fuel assemblies for nuclear reactors of the PWR type. Most of the activities (production of uranium / gadolinium tablets, pins and elements, stoppers and springs for fuel elements) were stopped in 2012. Only the MOX fuel assembly facility was maintained until 2015.

The present paper will give an overview of the decommissioning and dismantling process, starting from the licensing procedure for decommissioning till the current situation end of 2016.

2 LICENSING PROCEDURE FOR DECOMMISSIONING AND DISMANTLING

In order to obtain this license, a decommissioning report was written, containing the description of the installation and its safety systems, a radiological and toxic inventory, the dismantling strategy (purpose, dismantling alternatives, safety principals and criteria, destination of the site, …), project management (information about personnel, documentation and financing), a description of all the dismantling activities, including planning, decontamination and dismantling techniques, radioactive waste clearance, re-use of materials, risk analysis, incident analysis, emergency planning and security.

All safety related modifications during the decommissioning had to be approved by FANC and Bel V.

A first decommissioning license was granted by Royal Decree in December 2010, based on the decision to centralize the nuclear activities from two old buildings into the newer uranium fuel manufacturing building (building n° 5) and to decommission the old buildings. This license also included the dismantling of the MOX building, in the case of a definitive stop of these activities in Belgium by AREVA.

In May 2012, AREVA officially notified the Federal Agency for Nuclear Control the decision to stop all its activities in the FBFC facility in the coming years.

The production of uranium fuel was stopped at that time and immediately after ending the production activities, a number of technical risk-reducing measures were taken (removal of remaining fissile materials, disconnection of electricity cables, …). Due to this decision, FBFC submitted in December 2012 a decommissioning license application for the uranium fuel manufacturing building. Following review of this license application by the FANC and by the Scientific Council, and following consultation of the local authorities, a decommissioning license was granted by Royal Decree in October 2013.

The decommissioning licenses set the conditions to ensure the safe decommissioning of the buildings with specific attention to the use of subcontractors with necessary training and experience, the use of best available dismantling techniques, including the decontamination process of contaminated metals, the clearance of decommissioning waste and the final release of the buildings.

3 DECOMMISSIONING PHASE

FBFC started the decommissioning activities in 2012.

During the decommissioning phase, Bel V performed monthly inspections in the installations and on the site.

The dismantling of the buildings is proceeding on schedule.

The dismantling of building 1 (lab) was completed in 2015.

In building 2 (GADO), the dismantling will be completed and the majority of the release measurements will be carried out in 2016.

Building 3 was demolished down to the foundations in 2015. Removing the foundations and measuring their radiological cleanliness began in December 2015.

In building 5, the dismantling of the installations began in December 2013.

The last MOX campaign in building 5M was completed in April 2015.

The last 25 spare MOX pins were removed from the site in September 2016. No more fissile material is present on the site of FBFC.

4 WASTE MANAGEMENT PROGRAM

All materials are selectively collected in waste categories, according to the criteria defined by NIRAS-ONDRAF, the Belgian Agency for Radioactive Waste and Enriched Fissile Materials.

Specific waste drums are used to collect the waste.

All drums are radiologically characterised.

The results of the characterisation determine the removal paths of the material.

All information of each drum is saved in a data management system.

The waste management program allows removal of all radioactive material, while reducing also the amount of radioactive waste, using up-to-date measurement techniques and methodologies.

5 CONCLUSION

The goal is to reach the complete dismantling of the buildings, as well as the release of the FBFC site from nuclear control (after the treatment of the remaining contaminated soil) by 2018.

Immediate Dismantling of a Large Fleet of LWR NPPs: Consequences for Spent Fuel and Waste Management

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Abstract:

International experience feedback shows that the dismantling of one Light Water Reactor (LWR) is now well under control. However some specific difficulties may arise in view of the dismantling of numerous LWRs simultaneously. Indeed, in some countries, many LWRs may be permanently shut down then dismantled "as soon as possible" over a period of few years in the next decades. Such a situation notably addresses the issue of the overall management of large quantities of Spent Nuclear Fuel (SNF) and decommissioning waste. One issue is to remove the SNF from all the LWRs, even if the removal is conducted simultaneously in many LWRs and the SNF is grouped in a few storage facilities. Similar issue has to be taken into account regarding the Radioactive Waste (RW) produced by the dismantling actions. One method followed by IRSN to assess these issues is the use of estimates of flows of SNF and RW, based notably on assumptions defined to dismantle one Nuclear Power Plant (NPP) and to phase out all the LWRs of the fleet. These estimates can be compared to the feedback of SNF and RW flows for LWRs under operation, in order to identify risks when facing decommissioning. The risks highlighting are driven by key-parameters (as duration of the main decommissioning phases) of the estimates which can be adapted to minimize their impact. On this basis, it is possible to identify key-factors to dismantle each LWR and phase out the fleet regarding SNF and RW management.

1 INTRODUCTION

About 270 Pressurized Water Reactors (PWR) and 100 Boiled Water Reactors (BWR) have been commissioned in the dozen countries[1] with the largest number of Nuclear Power Plants (NPP) worldwide (references [1] to [5]). Among these 370 LWR units, more than 70% of them have been commissioned in the 70's, 80's and 90's. So, a large part of these units are already in operation for 30 to 40 years. Extension of the lifespan of the units or their permanent shutdown is an issue that falls first to the operators, but also to the state authorities and the governments, which also interests people. In view of a permanent shutdown of these LWR units at a similar rate to that of their commissioning, some specific industrial difficulties may arise if dismantling of numerous LWR units has to be done simultaneously. Indeed, such a situation addresses the issue of the overall management of large quantities of SNF and decommissioning waste for the concerned units.

For example, in France, the legislative and regulatory framework for the nuclear facilities favors their dismantling "as soon as possible" after their permanent shutdown which implies as well to limit the duration of the transition period from operation to decommissioning. Furthermore 58 PWR units have been commissioned between 1977 and 1999 in France – on average more than 2 units per year. In this context, the operator (EDF) plans to remove quickly the SNF then to perform dismantling actions immediately after the permanent

[1] : Canada, China, France, Germany, India, Japan, Russia, South-Korea, Spain, Sweden, Ukraine, United-Kingdom and United-States of America.

shutdown of the PWR units. One issue is to remove the SNF from all the relevant units, even if this removal is simultaneous in many units (permanently shut down or still under operation). Similar issue has to be taken into account regarding the management of the RW produced by the dismantling and clean-up actions, notably the RW that cannot be disposed of in a near surface repository.

One method followed by the French technical support organization (IRSN) to analyze these issues is the use of estimates of flows of SNF and RW allowing comparisons. These estimates are based notably on a phasing-scenario and a planning template defined for the dismantling of the units of one NPP and coupled to an overall schedule for phase out all the units of the fleet. This method is described later in the paper and, to be more comprehensive, a dedicated illustration has been built. In this illustration (case study), a situation of a “dummy” country is considered, where a fleet of 32 LWR units are under operation and located over 10 sites named A to J (2 or 4 units per site). To simplify the estimates of SNF and RW flows, only one kind of reactors has been hold: twinned pairs of 900 eMW PWR units (3 loops Westinghouse’s / Framatome’s design, described in the documents [6] and [7]). For the same reason, it is supposed that all the 32 units have been commissioned within 10 years, between the late 70’s and late 80’s. Additional information about the fleet of PWR units is given in TABLE I.

TABLE I. Information concerning the fleet of twinned pairs of 900 eMW PWR units

Type[a]	Commissioning years	Number in operation	Sites of units	Units designation[b]
0	Late 70’s / Early 80’s	5	A, B & C	[A-1, A-2], [B-1, B-2], [B-3, B-4], [C-1, C-2] & [C-3, C-4]
1	Early / Mid 80’s	6	D, E, F & G	[D-1, D-2], [D-3, D-4], [E-1, E-2], [F-1, F-2], [F-3, F-4], & [G-1, G-2]
2	Mid / Late 80’s	5	H, I & J	[H-1, H-2] [I-1, I-2], [I-3, I-4], [J-1, J-2] & [J-3, J-4]

a: design evolution to improve operation and safety.
b: [X-i, X-i_{+1}], twinned pair of 900 eMW PWR units No. i and i_{+1}, located on site X.

2 EXPERIENCE FEEBACK FROM ENTIRE DISMANTLING OF A PWR UNIT

Currently, worldwide, 6 PWR units[2] with a power exceeding 100 eMW have been decommissioned (termination of their authorization) including the demolition or release of their buildings, all operated in United States of America (USA). The experience feedback from the SNF removal, primary circuit loops (PCL) rinsing, dismantling and clean-up actions and RW management of these PWR units is consigned, for example, in the EPRI’s reports [8] to [13]. The general lessons that may be learned or observed therefrom are the following:

- the dismantling actions started immediately or a few years later after the permanent shutdown of the units;
- among the first operations performed, there is often the PCL rinsing;

[2] : Connecticut Yankee (560 eMW), Maine Yankee (860 eMW), Rancho Seco (873 eMW), San Onofre 1 (436 eMW), Trojan (1 095 eMW) and Yankee Rowe (167 eMW).

- the dismantling actions of the PCL equipment, reactor vessel and its internals were implemented over a period less than or equal to 5 years;
- the reactor vessel internals mostly were cut under water;
- the reactor vessel and its closure head mostly were removed whole;
- all the dismantling and clean-up actions were implemented over a period less than 15 years, site remediation and buildings demolition included;
- the SNF and intermediate level-long lived (IL-LL) RW transfer from the storage pool of the PWR unit to the dry storage facility built and commissioned on the same site, may sometime last almost as long than the decommissioning stage;
- at the end of decommissioning, no further building of the PWR unit (or other superstructure) remains on the site.

These items are taken into account in the present paper to define a “generic” phasing-scenario and its planning template for the decommissioning of the LWR units of one NPP. These phasing-scenario and planning template are used to estimate the annual flows of SNF and RW during the transition, dismantling and clean-up actions of several LWR units of different NPPs.

3 GENERIC PHASING-SCENARIO AND PLANNING TEMPLATE FOR NPP DECOMMISSIONING

The generic phasing-scenario and planning template for the decommissioning of the LWR units of one NPP are based on assumptions which are defined to be consistent with the national context (LWR units operated, legislative and regulatory framework…). So, some assumptions of the present paper take into account the peculiarities of the French context, as the national strategy of a dismantling as soon as possible after the permanent shutdown of a facility and also some EDF’s considerations, but can be modified as needs. These assumptions are the following:

- the permanent shutdowns of the NPP’s units are shifted against each other and for each LWR unit, the unloading of the last SNF core from the reactor vessel to the storage pool is performed immediately upon the permanent shutdown of the concerned unit;
- during the transition period and for each LWR unit, the SNF removal from the storage pool of the fuel storage building (FSB) is performed in a few years, the removal of the operational IL-LL RW and the PCL rinsing too;
- the turbine hall (TH) of the first LWR unit permanently shut down on a site is refurbished to manage RW from decommissioning actions of all the NPP’s units and the other THs are decommissioned in the same time than the nuclear buildings;
- the dismantling actions of the main systems are performed successively in each LWR unit;
- in the reactor building (RB), the dismantling actions phasing distinguishes the reactor vessel with its head and its internals, the PCL equipment and the other equipment located in the RB;
- the dismantling actions of the reactor vessel and its internals are based on their cutting under water;
- the dismantling actions in the other nuclear buildings are performed in the same time than those in the RB;
- the clean-up actions are performed in the working areas of a nuclear building after completion of the dismantling actions (removal of all equipment) in this building;
- the superstructure of each building is demolish after completion of the dismantling and clean-up actions.

As indicated above, to illustrate the method, a case study has been built based on a fleet of 16 twinned pairs of 900 eMW PWR units in 10 NPPs, each NPP having 1 or 2 twinned pairs.

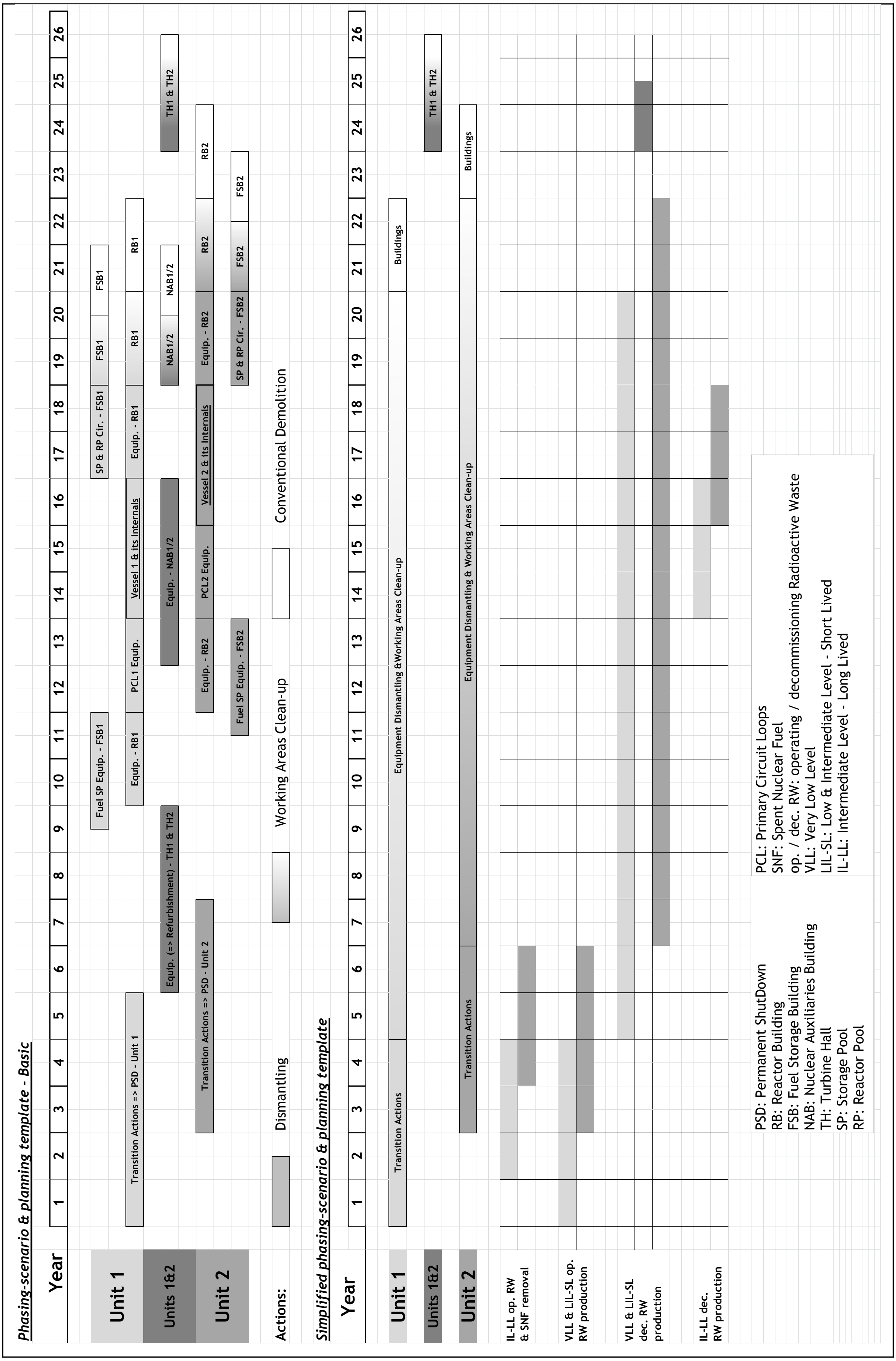

Figure 1. Decommissioning of a NPP with 1 twinned pair of 900 eMW PWR units (all types)

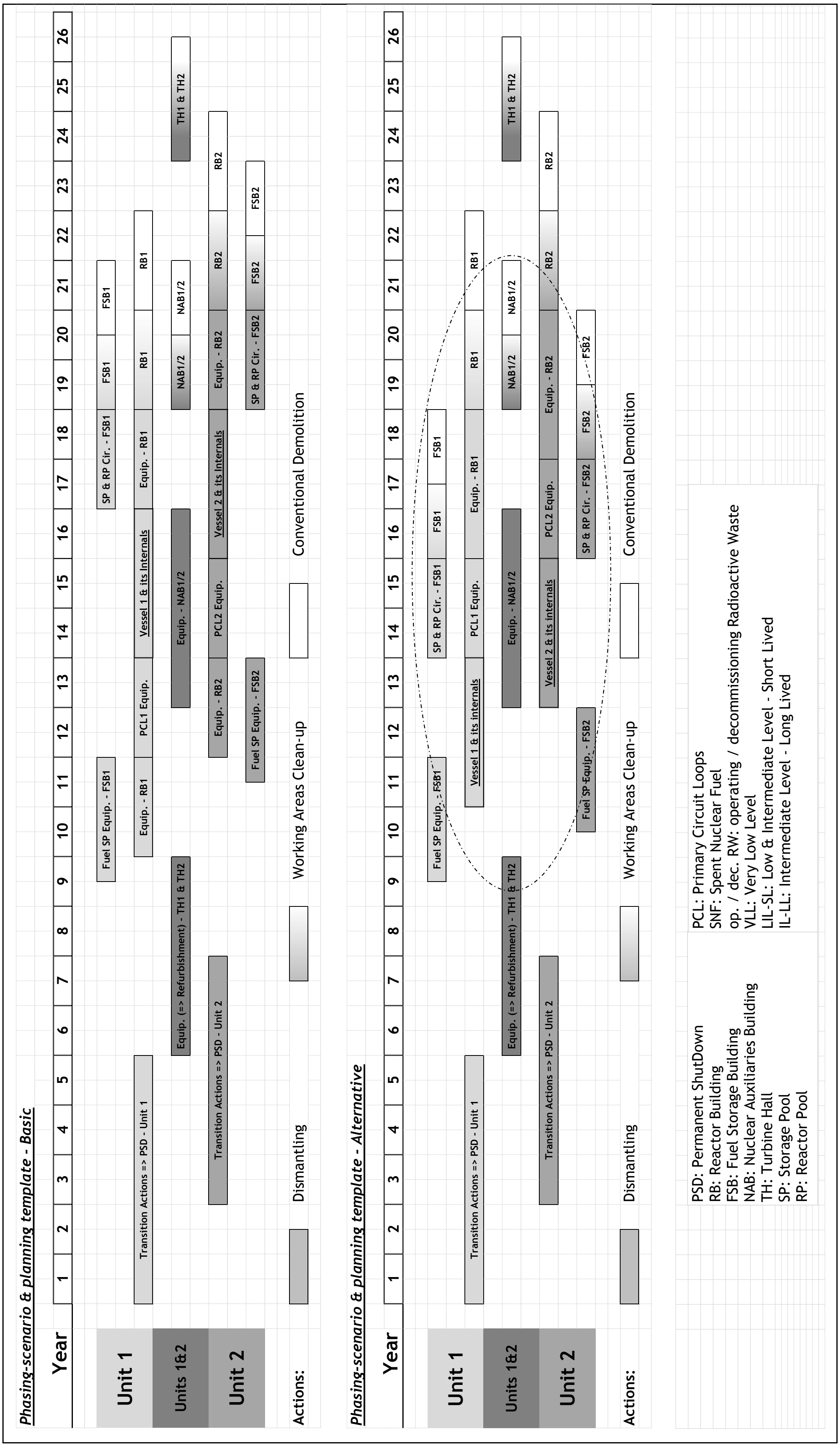

Figure 2. Basic / alternative phasing of dismantling actions in the RB and FSB (NPP with 2 PWR units)

Compared to the previous assumptions, the gap between the permanent shutdowns of the NPP's PWR units is fixed identical and equal to 2 years (smoothing of the SNF and operational IL-LL RW removal), the duration of the SNF and operational IL-LL RW removal, identical and equal to 3 years for each PWR unit. The durations of the dismantling actions are fixed identical for each PWR unit. They are equal to 3 years for the reactor vessel and its internals, and to 2 years for the PCL equipment. The durations of the other dismantling, clean-up and conventional demolition actions, as well as their phasing, are shown on the Figure 1 for a NPP with 2 PWR units (type 0, 1 or 2) and are similar for a NPP with 4 PWR units. Concerning the phasing between the dismantling actions of the vessel and its internal, those of the PCL equipment and those of the other equipment located in the RB, "basic phasing" and "alternative phasing" are considered, as defined on Figure 2. Compared to the previous items, it must be underlined the main assumption which consists for one NPP (with 2 or 4 PWR units), to limit to approximately 20 years the duration of the dismantling, the clean-up and the conventional demolition actions performed in all its PWR units.

However, to estimate the annual flows of SNF and RW at the scale of an entire fleet of LWR units, it may be difficult to use the phasing-scenario and planning template defined for a NPP. For a large fleet (dozens of LWR units), "simplified" phasing-scenario and planning template are considered, built on the basis of the entire phasing-scenario and planning template relative to one NPP and where each LWR unit is treated as a single entity and not as a set of several buildings. The simplified phasing-scenario and planning template may be used depending on the accuracy of the estimates of RW flows (for example, main categories of RW, without consideration on their nature and detailed pre-disposal management solution). For the case study, simplified phasing-scenario and planning template have been used to estimate the RW flows (excepted for the IL-LL RW); these are shown on the Figure 1 for a NPP with 2 PWR units. The average annual flows are calculated for one LWR unit, by dividing the total amount of SNF and those of different categories of RW by the duration of their phase of removal or production. Then, all units flows are added at the scale of the fleet and the contribution of the decommissioned LWR units to the total flow may be analyzed. To do that, it is necessary, if it does not exist, to define before an overall schedule concerning the phase out of all the LWR units of the fleet.

4 OVERALL SCHEDULE FOR PHASE OUT ALL THE LWR UNITS OF THE FLEET

The permanent shutdown of one LWR unit may be a decision taken by the operator, for technical and/or economic reasons, but also a decision imposed by the local or national political authorities. The phase out of all the NPPs units operated in a country is a decision that seems more political, although economic and technical factors are taken into account. So, an overall schedule for phase out all the LWR units of the fleet is something which in practice never exists. Nevertheless, it seems necessary to perform such analysis of the SNF and RW flows in the next decades.

In this context, alternative assumptions may be used to build a theoretical and realistic overall schedule to phase out all the LWR units of the fleet. Meanwhile, a continuity of the nuclear power generation, by the commissioning of new reactors, also has to be taken into account. For that, alternative assumptions may be used too. Finally, the use of a set of alternative assumptions allows assessing the influence of the alternatives on the total flows of SNF and RW (sensitivity study).

For the case study, the alternatives assumptions use to define the phase out of all the 32 PWR units of the fleet are the following:

- the lifespan of each PWR unit is similar and is approximately equal to 50 years;

OR

- the lifespan of each PWR unit depends on its type and is approximately equal to 40, 50 and 60 years respectively for the types 0, 1 and 2.

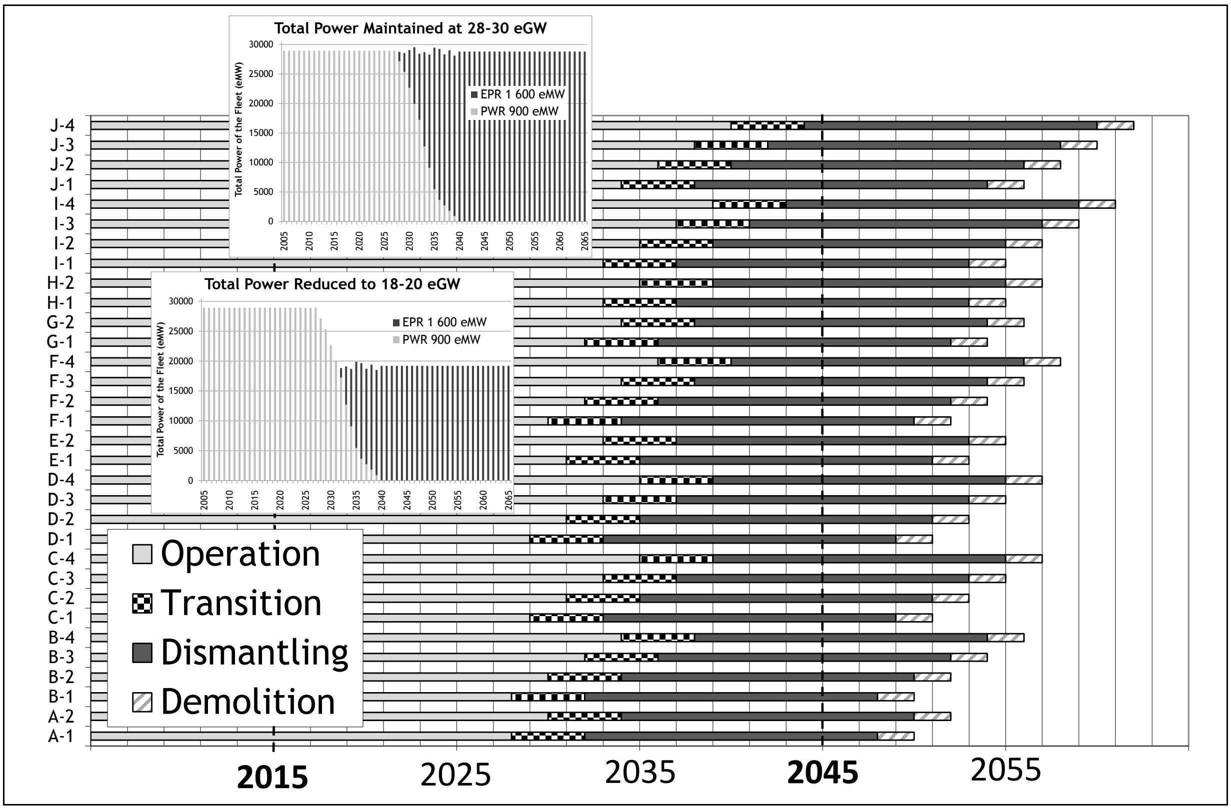

Figure 3. Homogeneous schedule to phase out the fleet of 900 eMW PWR units
(& corresponding commissioning of EPR units to maintain or reduce the total nuclear power)

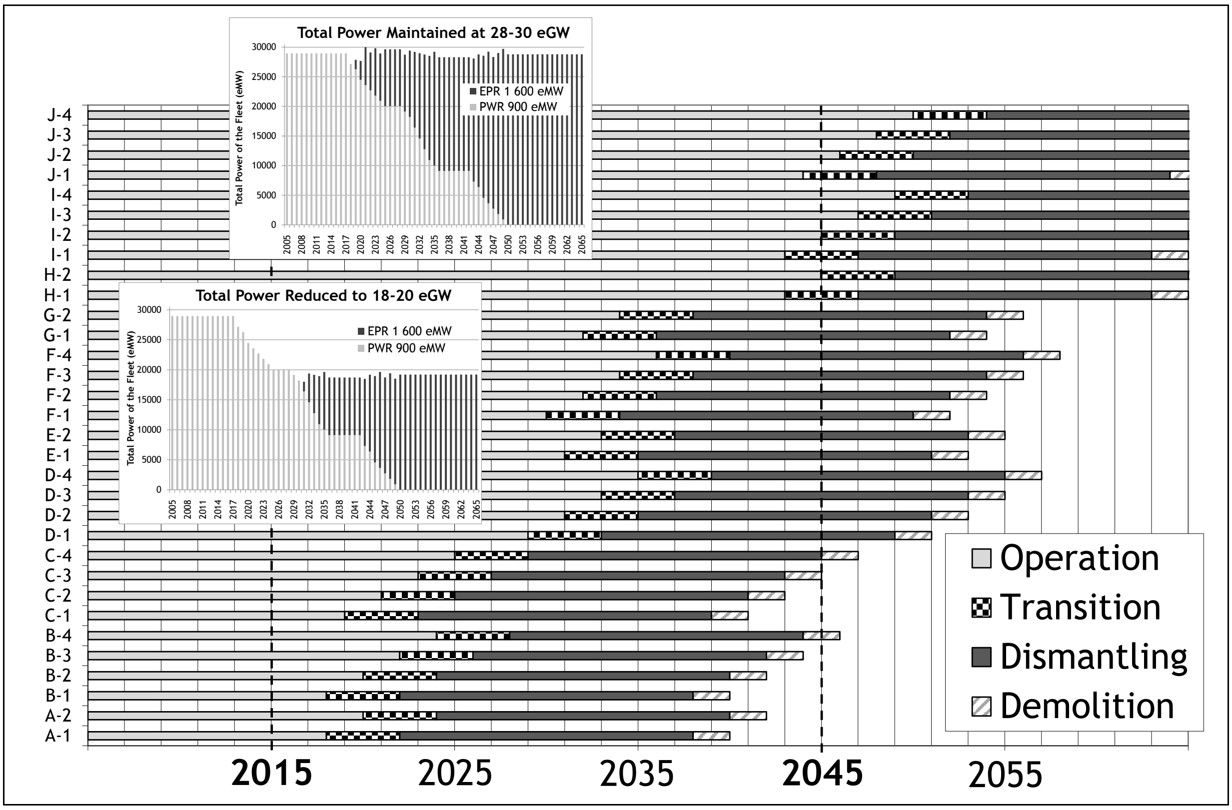

Figure 4. Heterogeneous schedule to phase out the fleet of 900 eMW PWR units
(& corresponding commissioning of EPR units to maintain or reduce the total nuclear power)

The previous values, assumed for the lifespan of each 900 eMW PWR unit, are considered realistic. In the second assumption, the link between the lifespan of a PWR unit and its type is supposed reflect a possible extension of this operating life upon technical and economic considerations, according to the upgrading of the initial design from one type to the following.

Then, the alternative assumptions are coupled with the simplified phasing-scenario and planning template for the decommissioning of the PWR units of one NPP. On this basis, the assumptions relative to the livespan of the PWR units govern the year of the permanent shutdown of the first unit of the NPP and for the over NPP's units, their years of permanent shutdown are given by the simplified phasing-scenario and planning template (gap of 2 years from one unit to the following). The assumption of a similar lifespan, approximately equal to 50 years for each PWR unit, leads to the overall schedule to phase out all units of the fleet shown on the Figure 3, called "homogeneous overall schedule". The alternative assumption, lifespan of the PWR unit approximately equal to 40, 50 or 60 years depending on its type, leads to another overall schedule to phase out all units of the fleet, shown on the Figure 4 and called "heterogeneous overall schedule".

In each alternative schedule to phase out the fleet, an equal period of 5 decades (2015 – 2065) is considered to perform the flow calculations. Nevertheless, to look at the results and to learn lessons, it may be more relevant to focus on a shorter period (first 2 or 3 decades).

As indicated above, to assess the total flows of SNF and RW, the continuity of the nuclear power generation has also to be considered (by the increase, the maintaining or the reduction of the total installed nuclear power). Such a continuity supposes that a sufficient number of new reactors have to be commissioned in parallel that reactors of the current fleet are permanently shut down. For the case study, two alternative assumptions are used to quantify the continuity of the nuclear power generation and all the new reactors commissioned are 1 600 eMW PWR units (EPR type, AREVA's design, described in the document [14]). So, the alternative assumptions are the following:

- the total installed nuclear power of the country is maintained at 28-30 eGW (by the gradual commissioning of 18 new EPR units, not before 2019);

OR

- the total installed nuclear power of the country is progressively reduced at 18-20 eGW then maintained at this level (by the gradual commissioning of 12 new EPR units, not before 2031).

Finally, the total flows of SNF and RW have three main contributors: the PWR units of the current fleet still in operation, the PWR units of the current fleet permanently shut down then dismantled and the new EPR units commissioned (Figure 3 and Figure 4). For the EPR units, their lifespan is supposed equal to 60 years. So, the permanent shutdown and the dismantling of the EPR units (later than 2065) are not taken into account.

5 ANNUAL FLOW OF REMOVED SNF

5.1 Method and additional assumptions

The total annual flow of SNF depends on the SNF quantities removed each year from each LWR unit in operation (current fleet and new LWR units) and from each LWR unit permanently shut down (during the transition period). To estimate this flow, additional assumptions and inputs are needed, relative to the operation of the LWR units, the irradiation of their nuclear fuel (core management) and the SNF quantity stored in the pool when the LWR units are permanently shut down. More accurately, for each LWR unit, the additional assumptions concern:

- the coefficient of productivity;
- the SNF quantity definitively unloaded each year from the core and those annually removed from the storage pool;

- the core management evolutions and the facility modifications during the operating period;
- the total quantity of SNF to remove during the transition period (sum of the SNF amounts of the last core and still in the storage pool at the permanent shutdown).

For the case study, the additional assumptions used to estimate the total annual flow of SNF are simplified, but considered realistic; they are the following:

- the coefficient of productivity is the same for each PWR unit and equal to 80%;
- the SNF quantity removed each year from the storage pool is equal to those definitively unloaded each year from the core;
- the SNF removal from the storage pools begins the 9th year of operation for the new EPR units;
- the nuclear fuel type and its irradiation cycles, defined for the PWR units of the current fleet and for the new EPR units, are those shown in the TABLE II;
- the core management and the facilities (equipment and buildings) are identical over the lifespan (commissioning to permanent shutdown);
- the total quantity of SNF to remove during the transition period of a PWR unit permanently shut down is those shown in the TABLE III.

TABLE II. Core management of the PWR units

Kind of reactor	**Nuclear fuel type and irradiation cycle characteristics**	**Quantity of SNF removed (t_{ihm}/y)**
900 eMW PWR unit (types 0, 1 and 2)	natural U oxide enriched in U-235 at 4,00% 0,460 t_{ihm} per fuel assembly 52 new fuel assemblies per irradiation cycle (1/3 core) 394 equivalent days of irradiation at full power per cycle average burn-up of 45 GWd/t_{ihm} at the definitive unloading	17,7
1 600 eMW PWR unit (EPR type)	natural U oxide enriched in U-235 at 4,50% 0,529 t_{ihm} per fuel assembly 61 new fuel assemblies per irradiation cycle (1/4 core) 392 equivalent days of irradiation at full power per cycle average burn-up of 55 GWd/t_{ihm} at the definitive unloading	24,0
t_{ihm}/y: tone of initial heavy metal per year *GWd/t_{ihm}: gigawatt day per t_{ihm}*		

TABLE III. SNF amount to remove after the permanent shutdown of a PWR unit

Type of 900 eMW PWR unit	**SNF of the last core (t_{ihm})**	**SNF stored in the FSB's pool at the permanent shutdown of the unit (t_{ihm})**	**Total SNF in the unit at its permanent shutdown (t_{ihm})**
Type 0	72,2	48,1	120,3
Type 1 and 2	72,2	72,2	144,4

5.2 Results of the estimate

After completion of the estimate of the total annual SNF flow over the next decades, the possible impact of the LWR units permanently shut down can be analyzed. Notably, the total annual SNF flow when all the LWR units of the fleet are operated can be compared to that when a part of the LWR units are permanently shut down (SNF removal during the transition period). In the situation where the total annual SNF flow increases, it may be necessary to anticipate. Various solutions are then possible. For example, the duration of the transition period may be extended or the removal of all the SNF stored in the FSB's pool may be performed before the permanent shutdown of the considered LWR units. Another way may be adapting the SNF management strategy to take into account the increase of the total annual flow of SNF. In consequence, it could be necessary to design, build and commission in timely manner needed independent SNF storage facilities and also, if required, casks to transport the SNF from NPPs to these storage facilities.

For the case study, the total annual SNF flows calculated are shown on Figure 5. Regardless of the overall schedule considered to phase out the fleet of 900 eMW PWR units and the total nuclear power considered for the country, the SNF flow increases for several consecutive years comparatively to that when no unit of the fleet is permanently shut down (566 t_{ihm}/y). For the heterogeneous overall schedule to phase out the fleet, the SNF flow increases up to +15% for a few years firstly around 2020 and secondly around 2030. For the homogeneous overall schedule, this increase reaches up to +23% for a few years around 2030 only. Whatever is the considered case, such variations of the SNF flows have to be analyzed to set out the SNF management strategy for the next decades so that the transition actions of PWR units permanently shut down are not unduly disrupted.

6 ANNUAL FLOWS OF PRODUCED RW

6.1 Method and additional assumptions

The total annual flows of RW depend on the RW quantities produced each year by each LWR unit in operation (current fleet and new LWR units) and from each LWR units permanently shut down (during the decommissioning period). To estimate these flows, additional assumptions and inputs are needed, relative to the RW produced by the LWR units in operation, the physical inventory of the facilities, the activation and contamination of the equipment and in the working areas. More accurately, for each LWR unit, the additional assumptions concern:

- the flows of operating RW, which may be defined on the basis of inputs taken from the operating experience feedback;
- the amounts of the activated dismantling RW, which may be estimated on the basis of neutron transport and materials activation calculations;
- the amount of the contaminated equipment and those of corresponding dismantling RW, which may be defined on the basis of the physical inventory and inputs taken from the operating and decommissioning experience feedback;
- the contaminated working areas and the amounts of corresponding dismantling and clean-up RW, which may be defined on the basis of the physical inventory and inputs taken from the operating and decommissioning experience feedback.

As indicated previously, for the decommissioning RW, the average annual flows are calculated for one LWR unit, by dividing the total RW amounts by the duration of their phases of production. For the case study, only the different RW categories are taken into account (using of simplified phasing-scenario and planning template relative to the decommissioning of one NPP). Nevertheless, some additional assumptions are required to estimate the total annual flow of RW; they are the following:

- the flows of operating RW are identic for each PWR unit and based on operating experience feedback notably detailed in reference [16];
- the operating IL-LL RW (control rods, absorbent bundles…) are stored in the FSB's pool over the lifespan (commissioning to permanent shutdown) of the PWR unit;
- the flows of operating LIL-SL and VLL RW during the transition period are supposed the half of those when the PWR unit is operated;

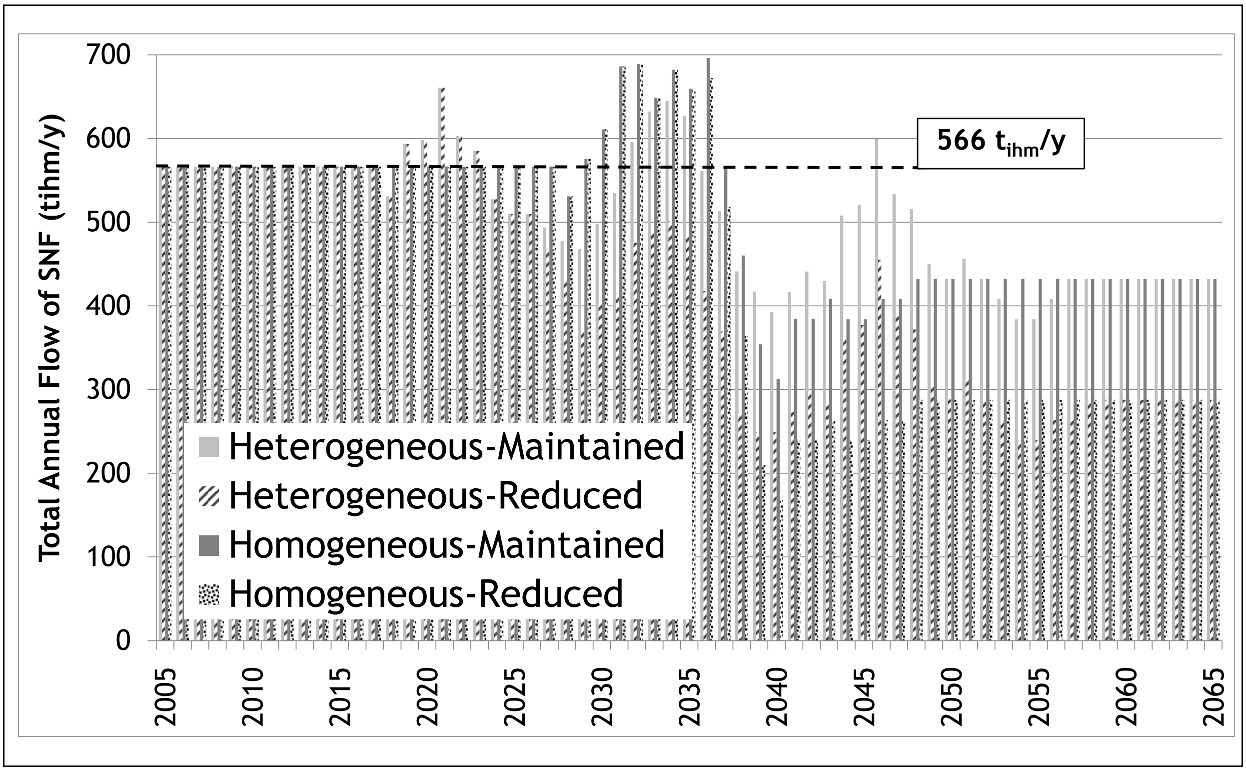

Figure 5. Total annual flows of SNF for the alternative assumptions

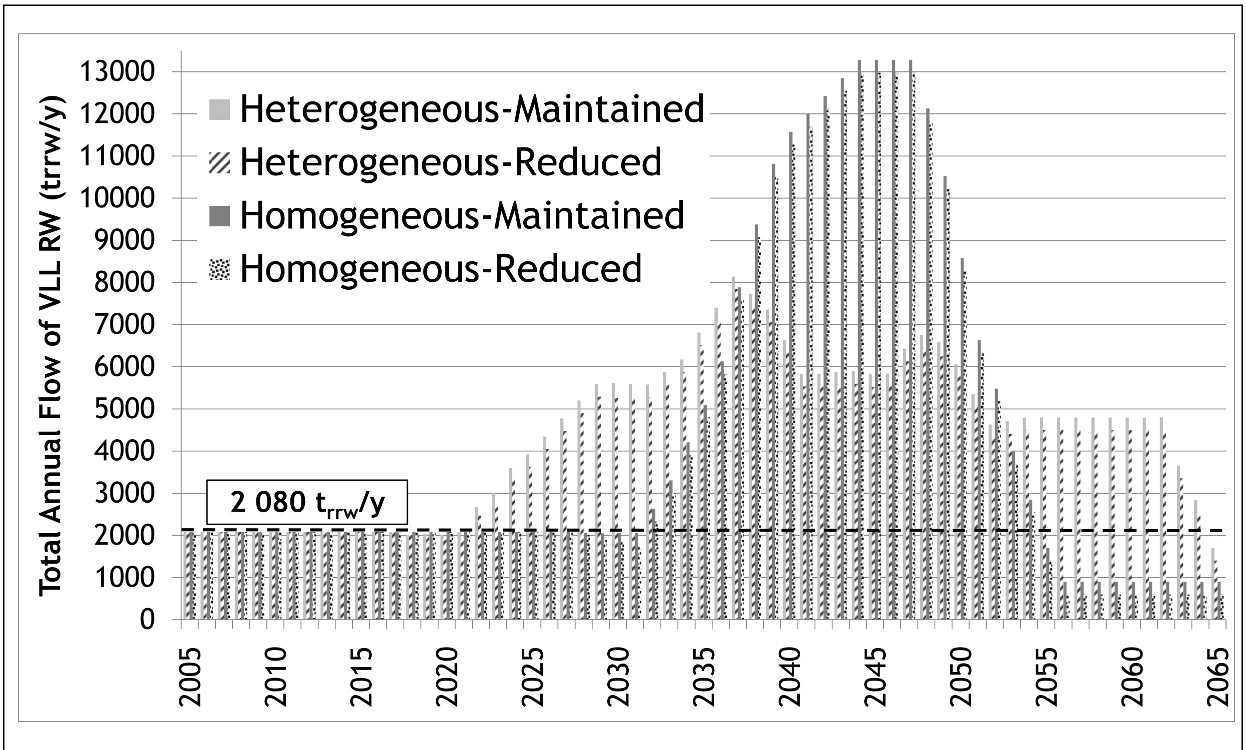

Figure 6. Total annual flows of VLL RW for the alternative assumptions

- the amounts of the activated RW are identic for each decommissioned PWR unit and estimated on the basis of neutron transport and materials activation calculations detailed in EPRI's report [15];
- the amounts of contaminated equipment are identic for each decommissioned PWR unit and match to the vessel, its head and its internals, also the PCL equipment, the auxiliary and emergency circuits and the ventilation equipment; their spread in the categories of dismantling RW are defined accordingly to inputs taken from the operating and decommissioning experience feedback notably detailed in reference [16];
- the working areas are supposed all contaminated in each decommissioned PWR unit; the corresponding amounts of dismantling and clean-up RW are defined and spread in each RW category accordingly to inputs taken from the operating and decommissioning experience feedback notably detailed in reference [16].

Finally, on the basis of the previous items coupled with the physical inventory of one 900 eMW PWR unit extract from documents [6], [7], [17] and [18], the amounts of RW generated during operating and decommissioning are those shown in TABLE IV, estimated for each category.

TABLE IV. Amounts of operating and decommissioning RW

RW category	**RW amount produced annually by a 900 eMW PWR unit in operation (t_{rrw}/y)**	**Total amount of RW produced by the dismantling of a 900 eMW PWR unit (t_{rrw})**	**RW amount produced annually by an EPR unit in operation (t_{rrw}/y)**
IL-LL	0,4 to 0,5 (stored in the FSB)	50	(not considered)
LIL-SL	130 (65 during TA)	2 500	100
VLL	65 (32,5 during TA)	5 500	50

IL-LL RW: 10^6 Bq/g < specific radioactivity < 10^9 Bq/g
LIL-SL RW: 10^2 Bq/g < specific radioactivity < 10^6 bq/g
VLL RW: specific activity < 10^2 Bq/g
t_{rrw}: tone of raw radioactive waste
TA: transition actions

6.2 Results of the estimate

After completion of the estimate of the total annual RW flows over the next decades, the possible impact of the LWR units permanently shut down can be analyzed. Notably, the total annual RW flows when all the LWR units of the fleet are operated can be compared to those when a part of the LWR units are permanently shut down (RW production during the decommissioning period). If the total annual flow increases for some kinds (metal, concrete…) or categories of RW, it could be necessary to anticipate this notably to avoid any accumulation of RW within the buildings of the decommissioned LWR units. Various solutions are then possible. For example, the duration of the dismantling and clean-up actions may be extended for some LWR or these actions may be further shifted from one LWR unit to another. Another way may be adapting the RW management strategy to take into account the increase of some total annual RW flows. As a matter of fact, it could be necessary to design, build and commission in a timely manner needed RW storage facilities on the NPPs sites and, if required, adapt the means to transport from NPPs, to process, to store elsewhere then to dispose of these RW.

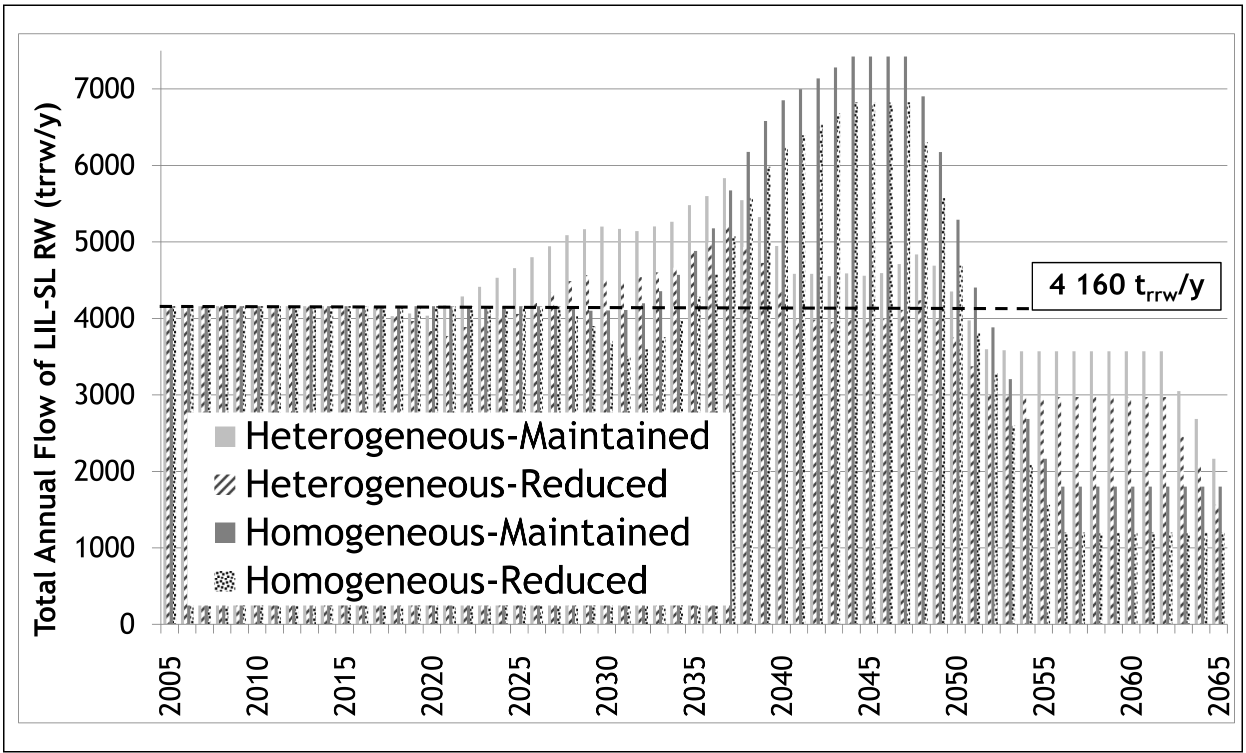

Figure 7. Total annual flows of ILL-SL RW for the alternative assumptions

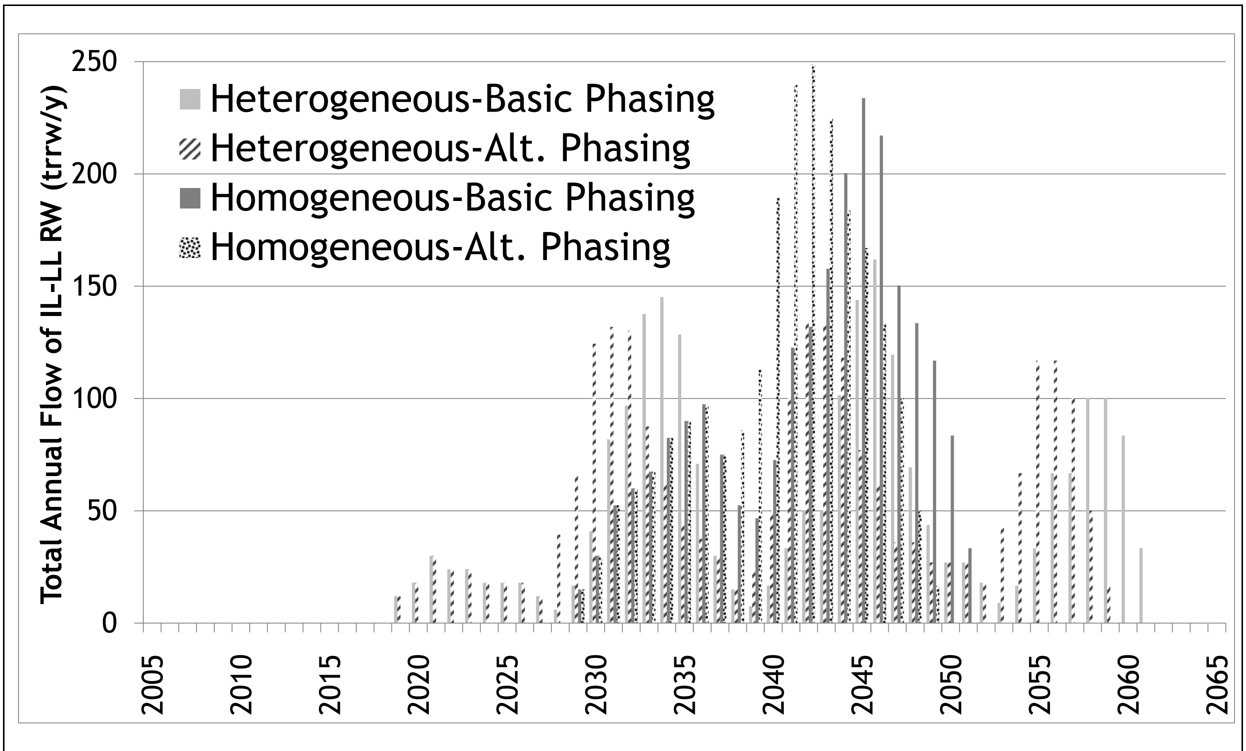

Figure 8. Total annual flows of IL-LL RW for the alternative assumptions

For the case study, the total annual flows calculated are shown on Figure 6 (VVL RW), Figure 7 (ILL-SL RW) and Figure 8 (IL-LL RW). For the VLL RW, their flow, in comparison with the situation where all 900 eMW PWR units were under operation (2 080 t_{rrw}/y), increases by a factor 2 to 6 over 2 to 3 decades after 2025 or 2035, essentially according to the overall schedule considered to phase out the fleet. For the ILL-SL RW, their flow, comparatively to that when no 900 eMW PWR unit is permanently shut down (4 160 t_{rrw}/y), increases between +10% and +60% over 2 to 3 decades after 2025 or 2035, according to, first, the overall schedule considered to phase out the fleet, second, the total nuclear power considered for the country. Concerning the IL-LL RW, their flow rises up to 150 t_{rrw}/y over 4 decades and after 2018 for the heterogeneous overall schedule, up to 250 t_{rrw}/y over 2 decades and after 2028 for the homogeneous overall schedule. Whatever is the considered case, such RW flows have to be analyzed to set out the RW management strategy for the next decades so that the decommissioning actions of PWR units permanently shut down are not unduly disrupted.

7 CONCLUSION

The management of numerous LWR units permanently shut down in parallel with those which may be still under operation needs to address some key issues. The simultaneously removal of the SNF and management of all the RW generated by the related decommissioning actions can be analyzed by considering the estimates of SNF and RW flows, in particular for the radioactive waste that cannot be disposed of in a near surface repository. These estimates are notably based on a phasing-scenario and a planning template defined for the decommissioning of the LWR units of one NPP and coupled to an overall schedule for phase out all the LWR units of the fleet. They are relative to the next decades and can be compared to the current experience feedback of flows of SNF and RW for units under operation, in order to identify risks when facing decommissioning. The risks highlighting are driven by key parameters (as duration of the main dismantling actions) of the estimates which can be adapted to minimize their impact. On this basis, it is possible to identify the key-factors to dismantle each unit of NPPs and phase out the fleet regarding SNF and RW management. It is noteworthy that this work needs to be done in any case upstream the studies and the implementation of dismantling actions. Nevertheless, the question of “who should do that?” arises, especially in countries having many operators. In addition, it can be underlined that another issue is the human resources (staff, skills and knowledge) necessary to perform all the decommissioning actions, but this aspect is not addressed in the present paper.

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Overview of the Strategic Research Agenda in the field of safety of radioactive waste geological disposal developed by the Expertise Function in the EC-H2020-SITEX-II project.

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Abstract:

The coordination action SITEX-II was initiated in 2015 within the EC programme H2020 to further develop an independent Expertise Function network in the field of safety of radioactive waste management and geological disposal (the development of the network was initiated by the former EURATOM FP7 SITEX project). SITEX-II brings together, as partners, representatives from 18 organisations involving national regulatory authorities (NRAs), technical support organisations (TSOs), research entities (REs), non-governmental organisations (NGOs), specialists in risk governance and an education institute, and involves interactions with a wider group of civil society (CS) participants. The network is expected to ensure a sustainable capability for developing and coordinating, at the international level, joint and harmonized activities supporting the technical review of safety cases for geological disposal facilities. One key task of SITEX-II consisted in developing the Expertise Function's Strategic Research Agenda (SRA) for strengthening the independent scientific and technical capabilities of the expertise function. This SRA was developed and is now used as an input in the EC-H2020-JOPRAD project for preparing a proposal for the setting up of a Joint Programming in the field of radioactive waste management, including geological disposal of waste, at the EU level, involving waste management organizations (WMOs), TSOs, REs and CS organisations. The main outlines of the SITEX-II SRA will be presented.

1 INTRODUCTION

1.1 Context

A key objective of the SITEX-II project ("Sustainable network for Independent Technical EXpertise of radioactive waste disposal - Interactions and Implementation") is to consolidate at the international level the knowledge base and expertise upon which organisations fulfilling an expertise function[1] in the context of the licensing process of underground radioactive waste disposal facilities can rely on, and to stimulate its sharing amongst all stakeholders, including civil society (CS).

In the context of geological disposal, the mission of the expertise function is to support the regulatory function as illustrated in Figure 1 [1] by ensuring that the disposal facility is

[1]The expertise function provides the technical and scientific basis notably for supporting the decisions made by the regulatory function.

developed, constructed, operated and closed in a safe manner, without imposing undue burdens on future generations i.e. people and the environment are protected against the hazards of ionising radiations emitted by the disposed radioactive waste. This mission involves several types of activities, such as participating in the establishment of regulatory requirements, as well as the development of guidance for meeting these requirements at the different stages of the licensing process.

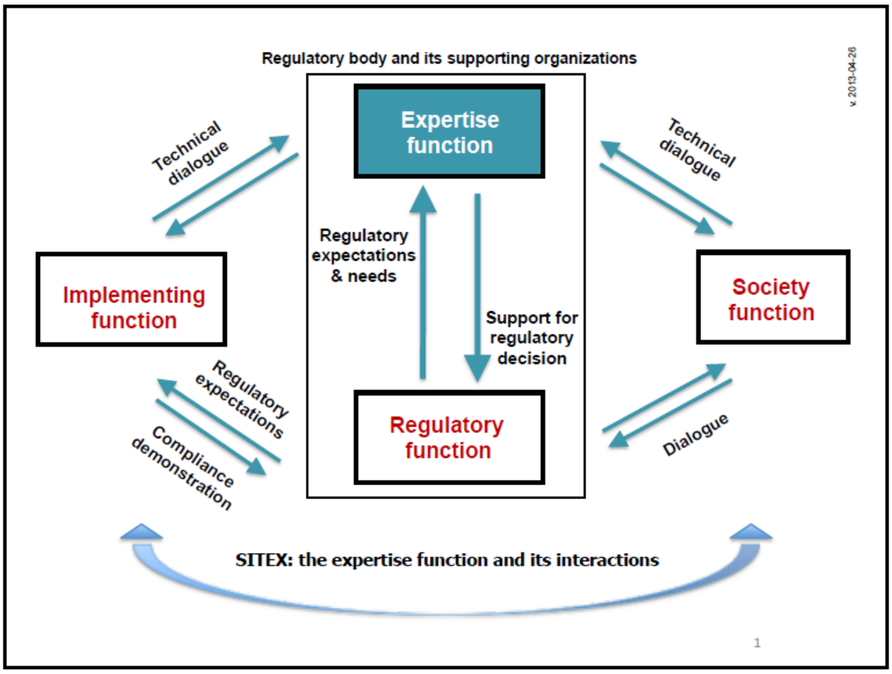

Figure 1: the expertise function and its interactions [1].

As stated by article 6-2 of the EC Directive 2011/70/Euratom of 19 July 2011 [2], the regulatory function has to be independent of the implementing function fulfilled by Waste Management Organizations (WMOs). Accordingly, the independence of the regulatory function calls for the support of an independent expertise function that develops and maintains the necessary know-how and skills in the field of nuclear safety. For complex issues such as those associated with the operational and long-term safety of waste disposal facilities, this can be achieved by performing and/or overseeing R&D in support of safety analyses and horizontal activities such as exchanging on practices, establishing states of the art and transferring knowledge. R&D and horizontal activities performed by the expertise function are also an important contributor to the development of its technical expertise and necessary to build the credibility of its technical competences (e.g. vis-a-vis the civil society), integrity and judgement.

This need for R&D and horizontal activities by the expertise function is clearly identified in international recommendations and requirements. For instance, the 2011/70/EURATOM directive requires the expertise function to carry out its own horizontal and R&D activities:

> *"Article 8 - Expertise and skills - Member States shall ensure that the national framework require <u>all parties</u> to make arrangements for education and training for their staff, as well as research and development activities to cover the needs of the national programme for spent fuel and radioactive waste management in order to obtain, maintain and to further develop necessary expertise and skills."*

It is also stressed in IAEA safety guides that the Regulatory Body (RB), and thus its supporting organisations (see figure 1), may need to conduct or commission R&D in support of regulatory decisions (see IAEA GS-G-1.1 [3] (see §3.33) and IAEA GS-G-1.2 [4] (see §3.68)).

It is important to highlight that the expertise function's R&D objectives may differ from those adopted by the WMOs. For instance, the expertise function's R&D is mostly intended to investigate safety issues with the objective to assess if the concept developed by the WMO fulfils the defined safety requirements. In that way, a special attention is given to the identification of questionable assumptions, knowledge gaps and incompleteness in the safety assessment performed by the WMO. These "challenging" activities are therefore more a "complement to" and "a verification of" than a "duplication of" the R&D activities performed by the WMO. This being said, a type of activity could be challenging at a given time, and could later be integrated e.g. in a WMO's R&D programme or in a European Joint Programming (and thus would not be anymore a "challenging" activity).

1.2 The SITEX-II project

The SITEX-II project gathers National Regulatory Authorities (NRAs), Technical Support Organisations (TSOs) and REs fulfilling an expertise function, as well as CS experts. Its overall objective is the practical implementation of the sets of activities and interaction modes issued by the former EC FP7 SITEX project (2012-2013) [1], with a view to develop at the European and international level an expertise function network. This network is expected to ensure a sustainable capability to develop and coordinate joint and harmonized activities related to the independent technical expertise function in the field of safety of geological disposal of radioactive waste, as a first priority. The followings tasks are carried out within the SITEX-II project:

- The definition of a Strategic Research Agenda (SRA) of the expertise function, taking into consideration the concerns of the CS;
- The production of guidance on the technical review of the safety case;
- The development of a training module for generalist experts involved in the safety case review process;
- The development of interactions between technical experts and CS;
- The preparation of the administrative framework for a sustainable network, by addressing the legal, organisational and management aspects.

The SITEX-II project is also meant to provide an input to the JOPRAD project [5] (Coordination and Support Action "Towards a Joint Programming Project on Radioactive Waste Disposal"). The overall aims of the JOPRAD project are to assess the feasibility and, if appropriate, to generate a proposal for Joint Programming (JP) in the field of Radioactive Waste Management, including geological disposal. This paper presents the SITEX-II SRA, which is an important input to the JOPRAD project as it contributes to identify potential topics for which JP activities could be developed together with WMOs and/or REs.

2 PRESENTATION OF THE SITEX-II SRA

2.1 Objective of the SRA, underlying vision and commitments

The general objective of the SITEX initiative is to meet the vision of fostering at the international level a high quality and independent expertise in the safety of radioactive waste management, including geological disposal as a first priority. The objective of the SRA [6] produced by SITEX-II is to identify and prioritise the needs for competence and skills development of the expertise function, at the international and in particular at the European level. These needs include research activities as well as *horizontal activities* such as exchanging on practices, establishing states of the art and transferring knowledge.

The commitments of the SITEX-II members for the development of the SRA are the following:

- The SRA is developed by applying a transparent methodology;
- The SRA addresses the needs associated with the different states of advancement of geological disposal (GD) programmes;

- The concerns of civil society are taken into consideration.

2.2 Scope of the SRA

The scope of the SRA covers all the topics relevant to the expertise function to assess whether geological disposal facilities are developed and will be constructed, operated and closed in a safe manner. Therefore, topics related to pre and post-closure safety as well as to the technical feasibility of geological disposal are considered. The scope encompasses all topics relevant to any waste type and spent fuel for which geological disposal is envisaged as a solution for its long-term management. Actions dedicated to pre-treatment, treatment, conditioning, as well as transport and storage of radioactive waste having an impact on the safety of geological disposal facilities could also be considered in the SRA. Furthermore, activities related to management options other than geological disposal may be addressed by the future SITEX network if relevant to several national programmes. However, this first version of the SRA is specifically focused on disposal in underground facilities.

In addition to R&D activities, the needs for knowledge transfer (e.g. training or tutoring), for developing state of the art and for exchanging on practices and developing common positions are also identified in the SRA.

The SRA is not an exhaustive list of all the potential topics that could enter into the scope above. It covers topics for which a sufficient level of common interest has been expressed amongst the SITEX-II members.

2.3 SRA Main topics and associated specific issues

Based on the methodology presented in [6], 7 main topics associated to specific issues and activities of common interest for the expertise function were identified and included into the SRA. These main topics are described in the following section. Table 1 summarizes the main topics, issues and activities of common interest.

The applied methodology allowed to consider the concerns of the CS about the R&D needs of the expertise function. For instance, holistic topics were identified in main topic 7, for which both technical and societal aspects need to be investigated in an integrated manner, using specific interdisciplinary methodologies and involving CS participation.

Moreover, it came out essential to embed CS participation also in activities related to other main topics, which are mainly technical. This could be achieved by involving trained representatives from the CS in such activities, to allow them performing knowledge sharing and interpretation.

2.3.1 Main topic 1: Waste inventory and source term

Source terms associated to cemented and vitrified waste, as well as spent fuel will be affected among others by the waste form composition and the conditions in the disposal facility. A reliable prediction of waste form degradation mechanisms, leaching rates of various radionuclides, radionuclide speciations, etc., thus requires systematic broad research. Several EC projects were already dedicated to this field of research (e.g. MICADO [7], FIRST NUCLIDES [8] , NF-PRO [9]). There is nevertheless a common interest for pursuing the R&D efforts in this field. Examples of specific issues of interest are: impact of radiation on cement matrix transport properties, impact of an alkaline environment (cement) on glass leaching, evaluation of long term instant release fraction (IRF) for SF, investigation of unconventional spent fuel dissolution (e.g. MOX fuel and RBMK fuel) and chemistry under disposal conditions, influence of organic matter potentially present in concrete waste forms on radionuclide source term.

Besides the need for R&D activities, there is a common interest in organizing horizontal activities on the methodologies applied to define the radionuclide inventories (e.g. use of radionuclide vectors, uncertainties about databases of radionuclide properties), to characterise the waste forms and to define the waste acceptance criteria (WAC), as well as the verification of the conformity to them. Such horizontal activities should take due account of ongoing international projects such as the IAEA project “Status and trends” [10] or the NEA expert group on inventorying and reporting methodologies (EGIRM) [11]. Exchanges on new treatments and conditioning, such as thermal processes and new mineral matrix other than usual concrete (e.g. geopolymer), are also foreseen. Moreover, the existing knowledge related to release processes and WAC is identified as candidate for transfer of knowledge, notably towards less advanced programmes.

Table 1: SRA Maint Topics and associated issues / activities (continued on next page)

SRA Main Topics and associated issues		Research activities (experiment and/or modelling works)	Horizontal activities		
			Exchange on practices, develop common positions	Develop states of the art	Transfer knowledge (eg. training, tutoring…)
Main Topic 1: Waste inventory and source term					
#1.	Uncertainty about databases and methodologies used for defining waste inventories (including historical waste)		■		
#2.	Evolution of the waste inventory due to possible neutron activation		■		
#3.	Understanding of the release processes and speciation of the radionuclides for different types of wastes	■	■	■	■
#4.	Waste acceptance criteria		■		■
Main Topic 2: Transient THMBC conditions in the near-field					
#1.	Oxidative transient	■			
#2.	Chemical conditions induced by metallic and/or cement materials and components	■			
#3.	Transients associated with gas production and migration		■		
	#3.1 Generation processes and rates of safety-relevant gases other than H2	■			
	#3.2 Influence of gas on geochemistry and microbial activity in HR and EBS	■			
	#3.3 Gas migration through EDZ and EBS	■			
#4.	Co-disposal of waste: interactions between different types of wastes		■		
Main Topic 3: Evolution of EBS material properties					
#1.	Heterogeneous behaviour of bentonite components	■			
#2.	Behaviour of metallic components	■	■		
#3.	Behaviour of cementitious components	■			
Main Topic 4: Radionuclide behaviour in disturbed EBS and HR					
#1.	Competition between sorption of radionuclides and other elements from EBS/waste	■			
#2.	Influence of organic matter on radionuclide migration	■			
#3.	Influence of the thermal transient on RN migration in EBS and HR	■			
#4.	Influence of microbial activity on RN migration	■			
#5.	Transport of volatile radionuclides in the disposal system	■			

SRA Main Topics and associated issues	Research activities (experiment and/or modelling works)	Horizontal activities		
		Exchange on practices, develop common positions	Develop states of the art	Transfer knowledge (eg. training, tutoring…)
Main Topic 5: Safety relevant operational aspects				
#1. Efficiency of the monitoring system over the operational period	■	■		
#2. Assessment of the risk of fire and explosion	■	■		■
#3. Assessment of the risk of flooding		■		
#4. Influence on long term safety of pre-closure disturbances		■		
Main Topic 6: Managing uncertainties and the safety assessment				
#1. Uncertainties associated with site characteristics		■	■	
#2. Management of uncertainties associated with geodynamics and tectonic movements		■	■	
#3. General methodologies for the safety assessment		■		■
#4. Safety assessment models		■		■
Main Topic 7: Lifecycle of a disposal programme and its safety case				
#1. Methods to review the safety case		■		■
#2. Assessment of the technical feasibility of a geological disposal concept		■		
#3. Evolution of the safety case content with the lifecycle of the disposal programme		■		■
#4. Organization of the pre-licensing phase		■		■
#5. Reversibility and Retrievability		■		
Holistic topics for which technical and societal aspects could be investigated:		■		
#6. Application of the optimization principle	■	■		
#7. License of disposal operation	■	■		
#8. Conditions for closure	■	■		
#9. Site selection process	■	■		
#10. Safety culture in the context of geological disposal	■	■		
#11. Intergenerational governance of the operational phase	■	■		

2.3.2 Main topic 2: transient THMBC conditions in the near-field

Chemical transients

The construction and the operation of a disposal facility will give rise to transients in the near-field that could affect the safety functions provided by various components (Engineerd Barrier System - EBS - and/or the host rock). For example, metallic and/or cementitious materials that will be used to condition and to immobilise the waste and to build the geological disposal facilities (gallery lining, groutings, sealing plugs, shaft lining…), coupled with other perturbations (such as the thermal transient), will induce chemical transients in the near-field. An improved understanding of such transients has already been developed in previous EU projects (e.g. former EU projects BENIPA [12], NF-PRO [9]). Nevertheless, there remains a need for further improvement and there is a common interest in pursuing R&D particularly on the spatial extent and evolution as well as the possible impact on safety functions of the following transients: oxidative transient during the construction and operational phase, notably with regard to corrosion of metallic components, and chemical transient induced by metallic and/or cement components on clays.

Transients associated with gas generation and transport

Gas generation and transport in geological disposal facilities have been studied for more than 15 years in a series of successive international projects. These include the PEGASUS [13], EVEGAS [14], PROGRESS [15] and the GASNET [16] projects. While R&D on gas issues continued from the early 2000s within the national programmes, there was a hiatus of several years for comprehensive multinational projects [17]. In 2009, the FORGE project [18], under the auspices of the European Commission, was launched with participants from radioactive waste management organisations, regulators (TSOs included) and academia. The following issues were not addressed or fully resolved during past projects and need to be investigated in the future:

- Generation processes and rates of safety-relevant gases other than H_2 (also investigated in the ongoing CAST project [19] as regards to the release of ^{14}C);
- Influence of gas on geochemistry and microbial activity in host rock (HR) and EBS, and associated impact on radionuclide transport (microbial activity is also investigated in the ongoing MIND project [20]);
- Although considerable amount of work has been carried out on this topic, in particular in the past FORGE EC project [18], uncertainties still exist on processes driving gas migration through Excavation Disturbed Zone (EDZ) and EBS, associated in particular with possible saturation levels and scenarios of bentonite evolution or with other perturbations such as alkaline plume. Therefore, there is still a need to improve the process understanding.

Moreover, a common interest exists for exchanges on the interpretation of the outcomes of the former FORGE project [18].

Transients associated with co-disposal of radioactive waste

The possible interactions between different kinds of waste that would be disposed of in the same facility are of common interest for performing horizontal activities (exchanging on practices and developing common positions), for example how to take into account in the concept of a disposal facility for possible interactions between the different kind of waste.

2.3.3 Main topic 3: Evolution of EBS material properties

The EBS covers a wide range of different components. Each component is a man-made barrier which consists of engineered materials. It is essential to know how these materials behave in different situations which might occur in the development of the site and disposal system. There is a common level of interest for pursuing R&D in the following fields of research:

- Heterogeneous behaviour of bentonite components.
  - Conceptual improvement of existing models is needed to efficiently account for the time-dependence of Hydro-Mechanical (HM) processes.
  - The coupling of these HM processes with Thermal (T) and Chemical (C) processes should also be improved. For instance, the bentonite transformation due to interactions with canister material is of interest at long term. In particular, the consequences on mechanical stability, swelling pressure and related radionuclide migration are not well known for the disturbed bentonite.
  - The influence of these processes on the effective closure of a disposal facility (e.g. performance of seals and plugs on the long term and large scale) should be further investigated.
- Evolution of metallic components. For example, study of metal (e.g. steel, copper) corrosion in repository conditions or of canister design lifetime.

- Evolution of cementitious components. Note that several aspects of cement material evolution are covered by the EC H2020 CEBAMA project [21]. As an example, the impact of radiations on cement material properties important for safety could be investigated.

There is also a common interest for exchanging on container design and manufacturing issues (e.g. modelling codes and standards and QA/QC programs and procedures for container design and manufacturing). If not properly managed these issues could affect the long term behaviour of metallic components.

2.3.4 Main topic 4: Radionuclide behaviour in disturbed EBS and HR

Current performance assessment studies generally include predictions for radionuclide migration using a constant, radionuclide dependent Kd approach, taking into account uncertainty in "all other geochemical processes" by a bandwidth for individual Kd's. The conservatism of such an approach with regards to the impact on radionuclide transport of possible perturbations needs to be investigated. For instance, the following perturbation needs to be considered: degradation product fronts, which could notably include corrosion products, as well as metal fronts (Mn, Cu, Ni, Fe, …) that change RN sorption and sorption competition and bentonite properties in general, organic matter fronts affecting RN sorption in cementitious environment and their interaction with Fe, temperature fronts influencing mineral precipitation/dissolution rates, microbial activity related to these fronts.

Although former EC projects focused on some of these aspects (e.g. EC FP7 SKIN [22] and EC FP7 RECOSY [23] projects), there is still a common interest in starting new R&D activities for these issues. Approaches to explaining and assessing sorption phenomena more sophisticated than the Kd approach are already proposed in the literature (e.g. electrostatic DL, TL or non-electrostatic surface complexation ion-exchange models using sites (multi-site) and sites capacities models). Such approaches could be used and developed further for investigating radionuclide migration in disturbed EBS and host-rock. Furthermore, the transferability of experiment results to in situ conditions is also an important issue that has to be considered when investigating topics related to radionuclide migration.

In the framework of this main topic, the transport of volatile radionuclides in the disposal system needs to be investigated too. Concerning C-14, note that its behaviour and impact strongly depend on its speciation, which is currently investigated in the CAST project [19].

2.3.5 Main topic 5: Safety-relevant operational aspects

While international programs related to the post-closure safety of geological disposal facilities have been carried out for decades, the safety during operation of these facilities came more recently into discussion in international projects (e.g. the IAEA projects GEOSAF [24], the European projects MODERN [25]) as the so-called "more advanced programs" enter their pre-licensing step. Actually, the very specific features of geological disposal facilities currently developed in Europe (underground vaults, tight areas, operation time- and space-scales, co-activity…) question the direct transposition of knowledge developed for the safe operation of already existing (aboveground) nuclear facilities. Furthermore, the state of the facility at its closing stage may depend on the operational phase as events occurring during the operation may impact provisions expected to fulfill post-closure safety functions.

Preventing a massive release of activity due to a fire or an explosion is a major safety issue during the operational phase of a geological disposal facility. The review of fire and explosion hazards assessment in such an environment should account for requirements in both the underground (mining) and nuclear fields. It requires reviewing merged standards and further developing independent modelling tools to simulate the behaviour of a fire and the generated smokes in galleries and disposal vaults, using theoretical laws and parameters values potentially different from those accounted for in the safety case. There is also a need regarding the ventilation of galleries while there may be explosion hazards when hydrogen is released by waste packages. In particular, the main parameters of air fluxes are difficult to anticipate due to the complex network of underground tunnels currently developed in national programs and in some cases, to the piling of waste packages in disposal vaults. Some modelling actions (in situ test being of WMO responsibility) may be needed in the future, especially in view of counter-calculations when reviewing safety cases. Furthermore, the behaviour of the packages from some waste streams (such as bitumen waste) in the case of a fire or of run-away (uncontrolled) chemical reactions, as well as that of its concrete overpacks, concrete liner and even the host-rock (locally), needs to be further studied so as to provide possible levels of containment failure; it thus challenges the provisions made by the implementer to prevent such accidents and to limit their consequences.

Whatever the provisions made, the occurrence of appropriate scenarios of accidents, including major ones such as fire/explosion or flooding, should be accounted for in the safety case – and thus be reviewed - as well as the remediation of the facility, which is an issue to the extent that it may impact the post-accident safety. Besides, events or accidents occurring during the operational phase may impact components with a post-closure safety function or their environment, leading to e.g. a decrease in their performances (see above). Damage to overpacks due to handling, local flooding or heating of host-rock can be mentioned for illustration purpose. On a more general level, the disposal facility shall be operated in such a manner to preserve the safety functions assumed in the safety case that are important to safety after closure (IAEA Specific Safety Guide No SSG-14 [26]). Exchanges in these fields to get hold of outcomes from other international project (e.g. GEOSAF [24] as mentioned above) and further develop common positions would be helpful in terms of challenging the assessment made by the implementer and homogenization of expertise approaches.

At last, monitoring is, in addition to the provisions made to prevent accidents, one of the paramount safety provisions to implement. As stated by the above mentioned IAEA Guide, monitoring provides input to safety assessments, continuing assurance of operational safety of the facility and confirmation that actual conditions are consistent with the assumptions made for safety after closure. The ageing of safety structures and components (SSC) is of particular concern for geological disposal facilities as operations over periods of around one hundred years are foreseen. At present, strategies and tools for such monitoring are still a vast research topic, as shown by the European project Modern2020 [27] launched in 2015. The expertise function must be involved in this research field (beyond Modern2020) to develop its own expertise capability; exchanges in this area would also be needed to share the state of the art and practice in this field.

2.3.6 Main topic 6: Managing uncertainties and the safety assessment

The development and use of appropriate assessment methodologies are essential for building confidence in the results of the safety assessment [28] [29]. Furthermore, as uncertainties are always associated with assessment results, the substantiation that they have been properly identified, characterised and managed is central to the demonstrability.

There is a common interest in exchanging and developing states of the art on the management of uncertainties associated with site characteristics, and more particularly: the present state of the site (e.g. uncertainties associated with the upscaling of lab measurements to site characteristics, the transposition of characteristics from one site/host rock to another, transfer of (sorption) data from diluted systems to compacted systems, ….) and possible geodynamics and tectonic perturbations of the site at the long term.

Furthermore, there is a common interest in transferring knowledge and exchanging about review approaches for issues such as general methodologies for the safety assessment, and safety assessment models (e.g. of specific issues are the limitations, difficulties and uncertainties associated with safety assessment models).

2.3.7 Main topic 7: Lifecycle of a disposal programme and its safety case

Evaluation of experience with different country arrangements would enable the identification of possible gaps or weaknesses in the understanding of expertise function expectations associated with the lifecycle of a disposal programme. This would provide an opportunity to overcome any such gaps or weaknesses and would assist in strengthening a harmonized approach. A common view on areas of significant safety impact could be identified and proposals formulated for an appropriate degree of regulatory control. The following issues are of common interest for horizontal activities: develop guidance for reviewing the safety case (this issue is currently covered by the SITEX-II WP2), assessment of the feasibility of a geological disposal concept (e.g. expectations of the expertise function on the methodology that should be followed to assess the feasibility), evolution of the safety case content with the lifecycle of the disposal facility, organization of the pre-licensing phase and reversibility and retrievability.

Moreover, specific issues for which there is a common interest to address both the technical and the societal aspects, in collaboration with representatives from the CS, were identified:

- Application of the optimization of the radiation protection principle (See [30], [31]) (e.g. how to consider the concerns of the CS in the application of the optimization process);
- License of disposal operation (e.g. develop a structured socio-technical understanding of the possible successive decision-making steps to confirm the design and operation modes of a geological disposal facility in view of a full commissioning license);
- Conditions for closure (e.g. examining the technical and socio-political criteria on which a partial or full closure could be decided);
- Site selection process (e.g. develop a common understanding of the socio-technical expectations about the organization of the process and the criteria for site selection);
- Safety culture in the context of geological disposal: the objective is to investigate the conditions and means for developing interactions between various categories of stakeholders and the public into the context of reviewing the safety of RWM strategies and geological disposal.
- Intergenerational governance of the operational phase (e.g. managing of changes in the socio-political framework, elaborating sustainable societal memory patterns, during the operational and the post-closure phase, …).

3 CONCLUSIONS

The SITEX-II project has developed a Strategic Research Agenda (SRA) identifying 7 main topics associated to specific issues and activities of common interest for the expertise function. Beside potential R&D activities, horizontal activities such as developing states of the art, exchanging on practices and transferring knowledge were identified. The developed SRA is used as an input in the EC-H2020-JOPRAD project for preparing a proposal for the setting up of a Joint Programming on radioactive waste management at the EU level, including geological disposal, and involving waste management organizations (WMOs), TSOs, REs and CS organisations.

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Examples of near-field modelling activities in Bel V for supporting the review of safety assessment for radioactive waste disposal facilities

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Abstract:

For strengthening its expertise in evaluating safety assessments for disposal facilities, Bel V performs modelling activities. A special focus of these modelling activities is oriented towards the study of the flow and the transport of radionuclides into the near-field, since its components (e.g. the waste form or engineered barriers) could have an important impact on long term safety.

This paper first develops the rationale and the benefits for a Technical Safety Organisation for developing models independent from those developed by the applicant. To illustrate these benefits, some lessons learned from near-field models developed by Bel V are presented. The presented models were aimed at (i) verifying the calculations performed by the applicant by reproducing some results of the performance assessment, (ii) developing a better understanding of the possible effects of cracks and voids on transport processes, as well as of the coupling between chemical reactions and those transport processes, (iii) developing a better understanding of the possible evolutions of the flow and the water saturation level and its impact on transport, and (iv) studying the conceptual uncertainty associated to different source term models. The paper concludes with some recommendations and limitations of performing such modelling activities.

1 INTRODUCTION

On 30 January 2013, the Belgian National Agency for Radioactive Waste and Enriched Fissile Material (ONDRAF/NIRAS) applied for a license to build and operate a surface disposal facility for low level radioactive waste (known as “Category A” waste) to be situated in Dessel. Bel V experts collaborated with the Federal Agency of Nuclear Control (FANC) in examining the applicant’s safety case.

At the same time, in Belgium, ONDRAF/NIRAS continues its R&D program on deep geological disposal for Category B&C waste (intermediate & high level radioactive wastes and spent fuel). A clay formation named “Boom Clay” is investigated as potential host rock. An alternative clay formation named “Ypresian clays” is also the subject of an investigation program.

In this context, Bel V performs modelling activities for developing its expertise and supporting its review of safety assessment of disposal facilities. This paper presents the rationale and benefits for a TSO of developing such activities and gives some examples and lessons learned from near-field models developed by Bel V.

2 RATIONALE AND BENEFITS FOR A TECHNICAL SAFETY ORGANISATION OF DEVELOPING INDEPENDENT MODELLING ACTIVITIES

Near-field modelling is aimed at assessing the migration of radionuclides from and in the surroundings of the waste packages. The objective is usually to assess the performance of the waste packages and engineered barriers. For instance, fluxes of radionuclides (Bq/y) at the boundaries of the disposal facility (interface with the aquifer or with the geosphere) may be determined based on near-field models.

Bel V considers this type of modelling especially important for surface disposal facilities which safety functions are mainly fulfilled in the long-term by the waste form and engineered barriers. Moreover, as validation elements, such as natural analogues or experimental results, are difficult to provide considering the long time frames involved, modelling has to be conducted carefully and the results interpreted with caution.

For those reasons, Bel V dedicates significant time and efforts in developing near-field models. These modelling activities may be project-driven, by national projects or by international collaboration projects, such as EC Projects or specific collaborations. Bel V also performs prospective modelling activities in the framework of its R&D program.

As illustrated in the following section, a significant modelling effort was produced to support the review of the license application of ONDRAF/NIRAS for the category A waste disposal facility in Dessel, Belgium (hereinafter referred to as “cAt project”).

For Bel V, the objectives of developing models independent from those developed by the applicants are:

- To verify independently the modelling results obtained by the licensee.
- To better understand the system behaviour (flow and transport of radionuclides) for different scenarios.
- To assess the significance of potential weaknesses identified by reviewers.
- To identify the key parameters, hypotheses and uncertainties (parametric or conceptual uncertainties) in order to focus the review on the most critical elements.
- To verify the adequacy of the parameter values and the hypotheses considered in models:
  - Are the parameters and the hypotheses sufficiently representative of the expected behaviour?
  - What is the level of conservatism of the selected parameters and hypotheses with respect to alternative hypotheses?

Developing such models is also a good way to acquire expertise in flow/transport processes and in performance assessment of radioactive waste disposal. Indeed:

- There is no better learning than when you get your hands dirty;
- By raising new questions, e.g. by questioning the choice of hypotheses, it sometimes drives literature reviews;
- It allows to understand what are the modelling difficulties (e.g. numerical problems) and the current technical limitations of models;
- In the framework of the R&D or through international collaborations, modelling is a way to exchange with other organizations in order to compare and discuss about the modelling and safety assessment practices.

This paper focuses on near-field modelling but Bel V performs as well modelling work in other fields of expertise (e.g. biosphere or hydrogeological models).

3 EXAMPLES

This section gives some examples of near-field modelling performed by Bel V. Some of the presented studies were carried out in the framework of an independent safety review by Bel V of the Safety Case submitted by ONDRAF/NIRAS for the cAt project.

3.1 Model verification

First, the ONDRAF/NIRAS model for the reference scenario of the cAt project was reproduced for a limited set of radionuclides. The radionuclides were selected based on their significance for long-term safety (relative contribution to the total dose), on their representativeness of the different ranges of half-lives and of the different ranges of sorption coefficients in concrete.

At first, the program HYDRUS 2D was used for this modelling but due to mass balance problems and limitations in modelling cracks and voids (lack of time-dependent parameters and no possibility for the user to develop custom modules), the finite element computer program FEFLOW [1] was finally used. This program is different from the one used by the license applicant.

This exercise allowed to verify the modelling results of ONDRAF/NIRAS obtained with a different calculation code. Moreover they also allowed to have a better understanding of the underlying hypotheses of the model of ONDRAF/NIRAS.

3.2 Study of alternative hypotheses

3.2.1 Impact of cracks and voids

Bel V studied the impact of cracks and voids on the release rate calculated with the near-field model by varying the value of main model parameters and developing "*what if*" models.

The reference model considered here is illustrated in Figure 1. The reference model consists of a stack of 5 monoliths containing cemented waste on top of a concrete slab and an embankment. This model represents a slice of a vault. For simplification, the behaviour is assumed to be similar for other parts of the vault (hypothesis of symmetry with limited influence of the boundaries). The earth cover is not modelled here but a time-dependent water flow input on top of the stack is considered. A lateral void and a crack at the bottom of the stack are present.

To illustrate a « what if » model that was considered, Figure 2 compares the release rate of ^{108m}Ag in the reference scenario with a case where the crack and voids are not modelled, for instance assuming that they would become clogged.

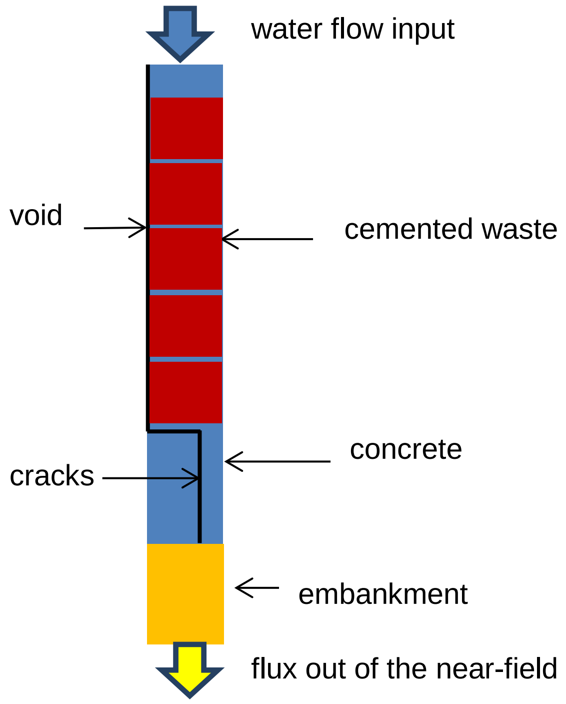

Figure 1: *The considered reference model.*

This comparison suggests that the presence of cracks and voids causes the spreading in time of the radionuclide release and, thus, of the radiological impact of the repository. This results in a lower maximum release rate (factor ~1.7 in Figure 2) compared to the model without cracks and voids. In the case without cracks and voids, the water flow imposed at the top of the module roof crosses the monolith region, causing nearly uniform migration of the radionuclides towards the bottom of the embankment of the repository. This model highly contrasts with the reference model, where the vertical void along the monolith region creates a preferential pathway for the incoming water and therefore allows a significant fraction of the water flow to bypass the monolith region.

The example shows that under certain conditions the presence of cracks and voids could reduce the maximum modelled release rate of the repository over the whole modelling time, which could sound as counter intuitive.

Furthermore, as the impact of cracks and voids depends, among others, on their pattern (number of cracks and their spacing), on the water flow level and on the radionuclide half-life and sorption coefficient, it is a priori not possible without a complete assessment of the model to determine what constitutes a conservative assumption versus an optimistic assumption with regard to maximum release rate indicator. This observation was also exemplified in detail in [2] for the performance of concrete in surface disposal facilities. In this case, performing independent calculations was shown to be a useful tool to support the review of a near-field model.

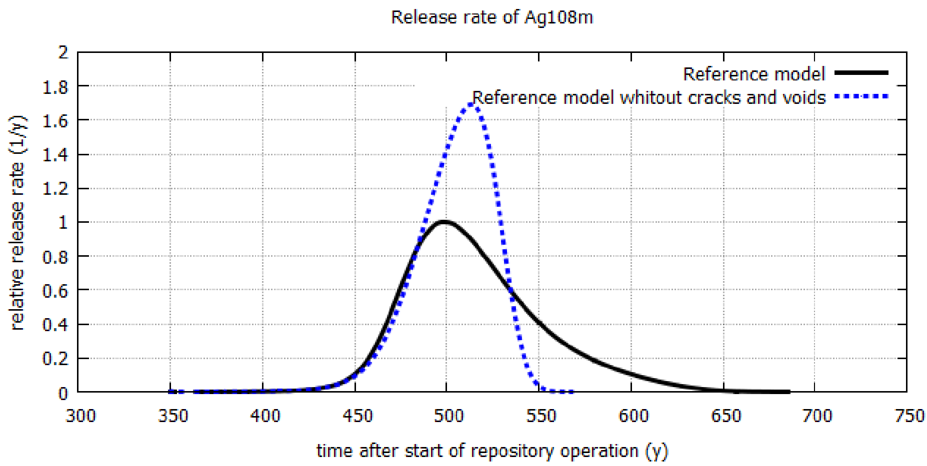

Figure 2: *Relative release rate of Ag108m in two scenarios : one with crack and voids and one without.*

3.2.2 Study of chemical degradations

Chemical degradations of concrete over time can have an impact on the migration of radionuclides in the disposal facility. For example, chemical degradation of concrete can modify its porosity, diffusion properties, permeability, sorption properties and pore water chemistry. In some cases, the volume augmentation caused by a chemical reaction such as rebar corrosion could even cause the formation of cracks in concrete.

Bel V uses the reactive transport code HYTEC [3] to study the coupling between flow, transport and chemical alteration.

The effect of heterogeneous cement leaching was for instance studied by Bel V. Figure 3 shows the heterogeneous alteration of a concrete block subject to leaching by water. A water flow is imposed on top of a concrete block with a void running on the left side of the block. This water flow leads to a leaching of the concrete. Figure 3 shows a map of the pH of the concrete block after 100 years of alteration. We can see that the alteration is heterogeneous because the water flow runs mainly along the void. Along the void, the cement is more degraded (pH around 10) (stadium of CSH phases degradation) than far from the void (intact concrete with pH around 14), leading to different transport properties of the material. We learned from these modelling activities that the cracking pattern is thus very important in assessing chemical degradations and that the heterogeneous alteration could be a phenomenon to consider in the safety assessment.

Performing such modelling was interesting from a point of view of expertise building. However, we experienced that the complexity of reactive transport modelling can cause difficulties in the validation and interpretation of the results. Furthermore, as the coupling with mechanical effects (e.g. development of cracks) is not supported by the calculation code HYTEC for the moment, this type of modelling does not offer an "all in one solution" that accounts for all the phenomena. Therefore, we concluded that such modelling should be dedicated to studying specific questions at limited scale and that the extrapolation of these results should be done with care.

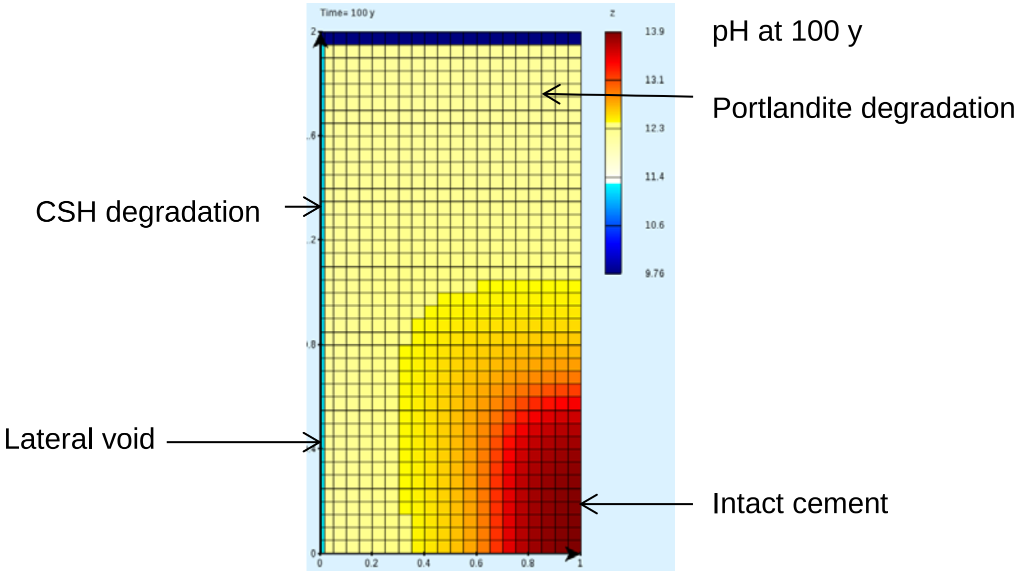

Figure 3: *Map of pH of a concrete block after 100 years of leaching. The block is subject to a water flow on the top boundary and a void is present at the left side of the block.*

3.3 Evolution of flow and water saturation level

Radionuclide transport in a discontinuous porous medium is greatly influenced by its level of saturation. The objective of this modelling was to build an independent understanding of the evolution of the flow and water saturation level in the repository. The repository will be indeed initially unsaturated and its saturation level is expected to evolve after its closure. Moreover, it was desirable to build a model with a larger geometry, to take into account for instance the multilayer cover and the influence of having several monolith stacks.

Therefore, an unsaturated model was first simulated in FEFLOW, see Figure 4. As numerical problems were encountered initially, several adaptations to the model were necessary (simplification of the geometry of some components, improving the mesh and solver options). For instance, as modelling of flow/transport at interfaces between materials was found to be a challenge, no crack was finally considered in this model.

This modelling highlighted the importance of the multilayer cover in the evolution of saturation of the repository, the propensity of the concrete to retain and absorb the water and many computational issues.

In particular, the model results showed that the water retention and absorption properties of the concrete components undoubtedly led, as the simulation progressed, to complete saturation by "accumulating" the water ingress from the top boundary condition. However, the single phase approach used in this model could lead to improper considerations with regard to the flow and a two-phase approach could have led to better representation of the phenomena which occur at the interfaces between concrete and other material (e.g. sand, gravel) and/or where an air layer is expected.

Therefore, it has been decided to implement an alternative model using a different mathematical formulation (consequently with different hypotheses and parameters values) taking a two-phase flow into account. This model uses the two-phase formulation described in [6] and the solver described in [7]0 for the computing tool OpenFOAM [8].

Comparison of the results of those two models has shown that the saturation speed of the concrete is much faster in the FEFLOW model than in the OpenFOAM model. Based on this comparison, it seems that using a single phase or a two-phase model might significantly influence the modelling of water flow through a heterogeneous unsaturated porous media. This difference between the two models is difficult to explain precisely at the moment, and seems to rely on the mathematical formulation used in the models. Due to the time-scales involved it is unfortunately not possible to properly validate these models.

Finally, the assessment of radionuclide transport with such model was judged to be not satisfactory because the lack of a proper source term model led to unrealistic effects during the evolution of saturation.

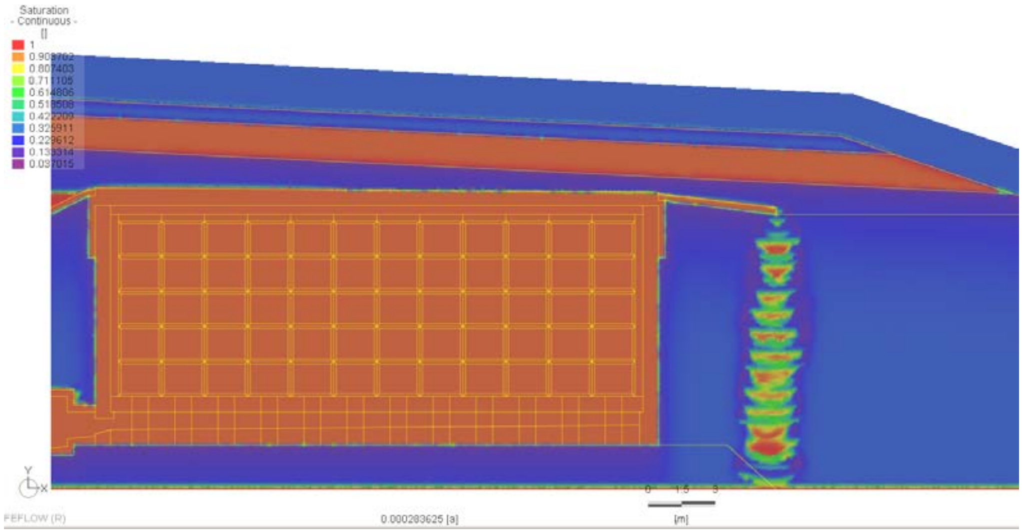

Figure 4: *2D model for studying evolution of water flow and saturation level; This picture shows the drainage of excess of water when the multilayer cover is effective.*

3.4 Source term modelling

As the source term model was identified as one important part of the near-field model, the source term modelling was further studied. First, a literature review was performed. We concluded that 4 types of source term models are usually considered (see e.g. [4], [5]):

- A rinse release model: the release is instantaneous when water comes in contact with the waste.
- A release controlled by a (constant) corrosion rate, e.g. for activated metals.
- An advective model: the waste is embedded in a permeable porous media and is transported by advection only.
- A diffusive model: the contaminant must diffuse over a characteristic length and is then instantaneously transported in fractures by advection. This type of model is generally considered for waste embedded in cement.

For all these models, a sorption coefficient or a solubility limitation can be also considered.

The release rates calculated with these models were then computed using different sets of parameters. For instance the values of the cracking factor (distance between fractures in the diffusive model), water discharge rate (for the advective model) and the sorption coefficients were varied.

Figure 5 illustrates such calculations and allows comparing the release rates calculated with an advective model and a diffusive model. This simple example shows that the parameter “distance between fractures” is very important for the diffusive model; which could then, in this example, becomes more conservative than an advective model (see the results in Figure 5 with a crack spacing of 10 cm).

These rather simple calculations provided a range of release rates that was used to compare the different source term models and their level of conservatism.

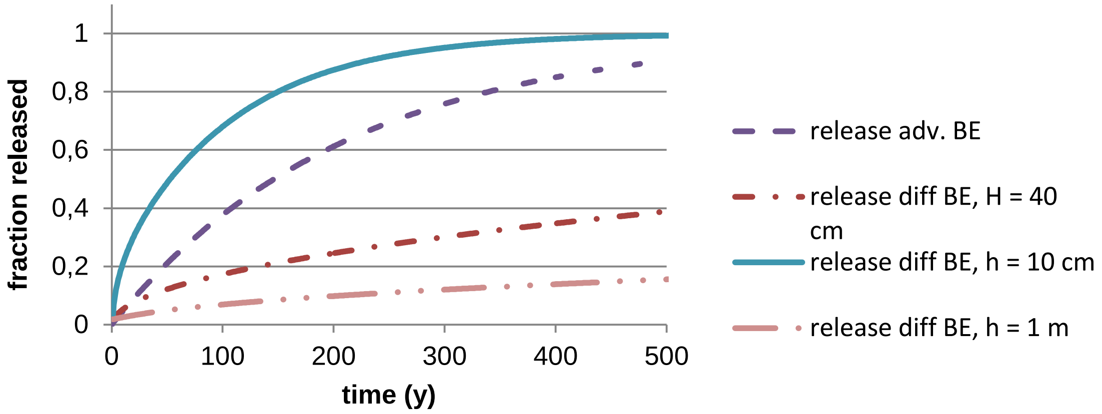

Figure 5: *Illustration of releases calculated with different source term models for I-129 using a best estimate kd. The advective model considers a porosity of 0,25 with discharge rate of 350 mm/a and a length of 11m. The diffusive models consider a spacing between fractures of 10, 40 or 100 cm (H).*

4 CONCLUSION

As near-field modelling constitutes a significant part of the safety demonstration of disposal facilities, Bel V dedicates significant efforts to developing expertise in this field. This modelling work is done in support of national projects, as part of international collaboration projects or as part of Bel V's internal R&D.

The paper gave some examples of the types of models developed by Bel V and some lessons learned. This activity was notably useful for:

- Building expertise.
- Verifying models developed by a license applicant.
- Identifying and analysing in detail potential weaknesses of the models proposed by the license applicant.
- Assessing the level of conservatism of some hypotheses. In particular, as it was shown that the effect of some modelling hypotheses might be counter intuitive (e.g. the effect of cracks and voids, see §3.2.1), independent modelling can be very valuable.

However, the limitations of such exercise are that it may become a time-consuming activity. Furthermore, the obtained results can only be partially validated given the time-scale involved with radioactive waste repositories. Therefore the results must be interpreted with care and it can be sometimes hard to differentiate a modelling artefact from a potential real phenomenon.

The following basic recommendations for performing such modelling work are given:

- The objectives of the modelling activities should be clear from the beginning with well-defined safety indicators.
- The stage of literature review should not be neglected.
- A stepwise approach should be followed, i.e. starting with simple models whenever possible.

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Thermal compatibility of clay with regard to the disposal of highly radioactive waste

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Abstract:

According to the requirements of the German regulating body, the barrier effect of host rocks must not be compromised by a thermal impact resulting from high-level waste and spent fuel (HLW/SF) emplacement. To substantiate and quantify this requirement, a literature survey of research results on thermally-induced changes of clay properties was carried out. As an important outcome, it revealed that temperatures up to 150 °C can be expected not to compromise the integrity of the geotechnical or geological clay barriers, but rather to contribute to their improved performance. This is because of – amongst other positive effects – (i) the increase of the consolidation of the clay and (ii) the sterilization of the repository's near-field and thus inhibition of detrimental microbial activity. The temperatures of >80 °C relevant for other detrimental processes in clays (e.g., illitization of smectites, thermo-chemical sulfate reduction or kerogen transformation) will be in effect no longer than for a few thousand years after the repository closure and will not extend beyond few meters into the host rock. The effects of these detrimental processes during the thermal phase of the repository and afterwards are therefore negligible. It is proposed to set a temperature limit of 150 °C for HLW/SF repositories in clay host rock.

1 INTRODUCTION

The German Parliament, Deutscher Bundestag, established a commission on the storage of the highly radioactive waste in 2014. One of its work groups mandated the GRS in January 2016 to deliver an expert opinion on thermal compatibility of salt, clay, and crystalline host rocks with regard to the disposal of high-level radioactive waste (HLW) and spent nuclear fuel (SF).

The expert opinion of the GRS is based on the results of the applied research in the R&D projects VSG and AnSichT (amongst others on the paper by Jobmann & Meleshyn, 2015, prepared in the frames of the latter project). Project VSG on preliminary safety analysis of a potential repository for HLW and SF on an example of the salt dome Gorleben in accordance with the requirements of the German regulating body was carried out under coordination of the GRS from 2010 to 2013. Project AnSichT on the demonstration of the safety of a repository for HLW and SF in clays according to the requirements of the German regulating body was carried out with involvement of the GRS from 2011 to 2016. This paper concentrates on thermal compatibility of argillaceous host rocks and clay-based geotechnical barriers in clay and crystalline host rocks.

The key topics of the commission's mandate were to give an overview on (i) the temperature limits according to the international and national disposal projects and (ii) the relevant thermally induced processes in the host rocks and geotechnical barriers that necessitate the limitation of the thermal impact of the emplaced waste. Accordingly, this paper starts with an overview of the temperature limits and proceeds with a brief consideration of the most important thermally induced processes identified in the reports by Jobmann & Meleshyn (2015) and Jobmann et al. (2016).

2 RESULTS AND DISCUSSION

2.1 Temperature limits

The overview on the temperature limits according to the disposal projects in some countries as presented in table 1 shows that putative alteration of clay mineral and other detrimental effects at temperatures above 100 °C were the reasons for establishing the corresponding temperature limit in France, Belgium, Sweden, Finland and South Korea. In this relation the statement by Weetjens (2009) is worth of noting that "the criterion limiting the temperature in the buffer to 100 °C … was rather arbitrary" and proposes as an option its reconsideration along with the improvement of its argumentation base.

In Switzerland, on the contrary, no need for such a low temperature limit is seen in the disposal concept, which allows temperatures in excess of 125 °C in the buffer half closest to HLW and SF containers. The temperature in the outer buffer half closest to the host rock, however, should not exceed 125 °C, in order to assure that the temperature at the buffer/host rock interface does not exceed 100 °C. A similar approach is currently under consideration in South Korea. Based on experimental data obtained in the last two decades, in Germany it is proposed in the frame of the R&D project AnSichT to establish the temperature limit of 150 °C for clay host rock and clay-based geotechnical barriers. Insufficient data for thermally-induced processes at temperatures above 150 °C appeared to be the major obstacle for setting even higher temperature limit.

Tab. 1: Temperature limits in the host rock or in the material of the geotechnical barrier (buffer) closest to HLW and SF containers according to some disposal concepts.

Country	Host rock/buffer	Temperature limit	Reason
France (Andra, 2005)	clay/bentonite	100 °C	mineral alteration
Belgium (Ondraf, 2005)	clay/concrete	100 °C	detrimental effects
Switzerland (Nagra, 2002)	clay/bentonite	125 °C[1]	mineral alteration
Sweden (SKB, 2005)	crystalline/bentonite	100 °C	mineral alteration
Finland (Posiva, 2013)	crystalline/bentonite	100 °C	mineral alteration
South Korea (KAERI, 2007)	crystalline/bentonite	100 °C[2]	mineral alteration
Germany (AnSichT, 2015)	clay/crushed clay (+/- bentonite)	150 °C	insufficient data for higher temperatures

[1] temperature limit for the outer half of the buffer

[2] an increase of the temperature limit to 125 °C is under consideration since 2016

2.2 Thermally induced processes

Amongst the thermally induced processes, boiling of water is very often mentioned first when discussing the temperature limit of 100 °C for disposal in clays. However, taking into account the hydrostatic pressure of about 6-7 MPa at the most relevant depth of 600-700 m for a HLW and SF repository, boiling of water does not occur below 260-280 °C (fig. 1). Therefore, the influence of this process in a repository should be considered as negligible.

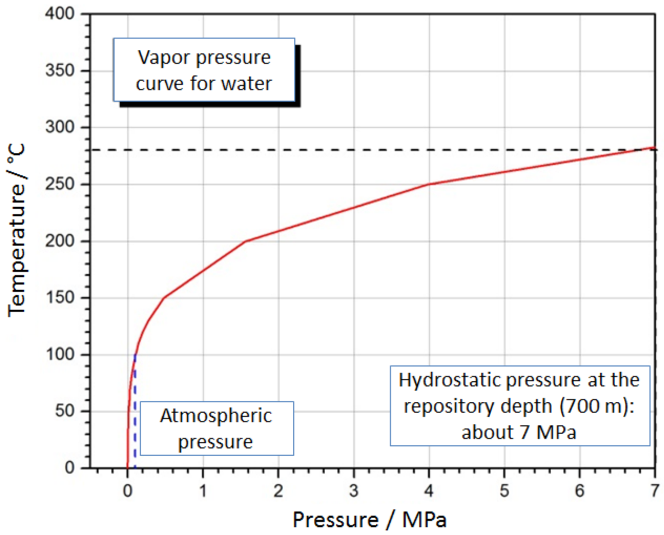

Fig. 1: Vapor pressure curve for water

The next process, illitization of smectites, is considered as the most important one with regard to the potential detrimental effects on isolating properties of clay. In order to prevent it, a number of publications on long-term performance of HLW and SF geological repositories refer to the requirement of limiting the clay temperature to 100 °C. However, the experimental data (Huang et al., 1993) on this process (fig. 2) show that at potassium concentrations of 200 ppm, which is characteristic of clay formations (shales), an illitization of 80 % of smectitic layers of clay particles requires about 100,000 years at 150 °C. At lower temperatures, this process proceeds even slower. The work by Huang et al. (1993) concludes that an observation of a much smaller illitization time of about 3,500 years at 150 °C (illitization of 50 % of smectitic layers of clay particles) reported by Pytte & Reynolds (1989) was due to potassium concentrations of 2.000 ppm, which is characteristic of hydrothermal solutions but not of clays.

A further important thermally induced process is clay expansion and contraction. Experimental studies (Baldi et al., 1991; Sultan et al., 2002; Tang et al., 2008) revealed that heating of clay characterized by low values (≤ 1) of overconsolidation ratio (OCR, a ratio of preconsolidation vertical effective stress to the actual vertical effective stress) leads to its contraction. At high OCR values thermally induced clay expansion converts to contraction at a certain temperature. This characteristic temperature, called expansion-contraction threshold, decreases linearly from 80 °C for OCR of 12 down to 50 °C for OCR of 2 (fig. 3).

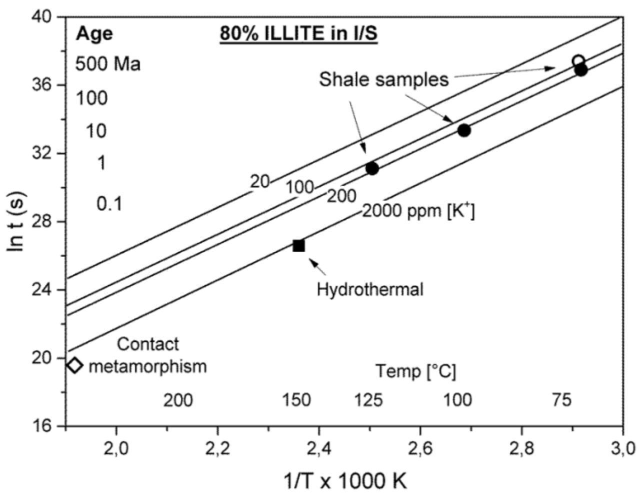

Fig. 2: Kinetics of smectite illitization as a function of temperature (Huang et al., 1993)

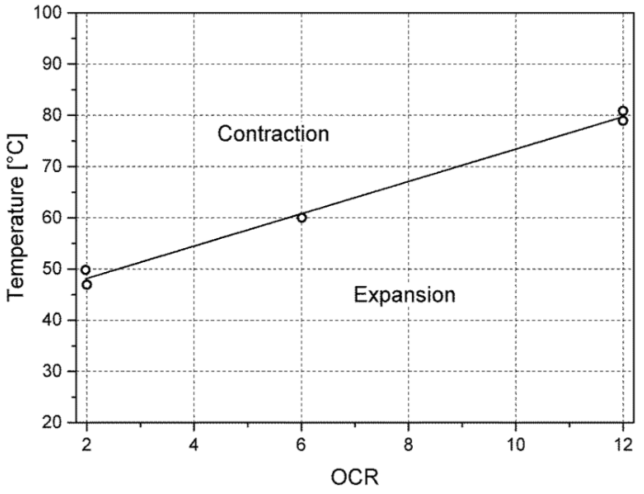

Fig. 3: Temperature of expansion-contraction threshold as a function of OCR (Baldi et al., 1991; Sultan et al., 2002)

Moreover, the clay contraction that exerts positive effects on the mechanical properties of clay because of its consolidation is an irreversible process (Tang et al., 2008), contrary to the clay expansion. Importantly, this contraction leads to a decrease in the water permeability of clay (fig. 4).

Concerning the effect of increased temperatures on the sorption capacity of clay, the available experimental data (fig. 5) is limited to only several chemical elements and to two clay minerals. Different chemical elements can thereby be characterized by opposing trends. However, it can be concluded that the sorption capacity of clays does not show a significant decrease and it can even increase when the temperature exceeds 100 °C.

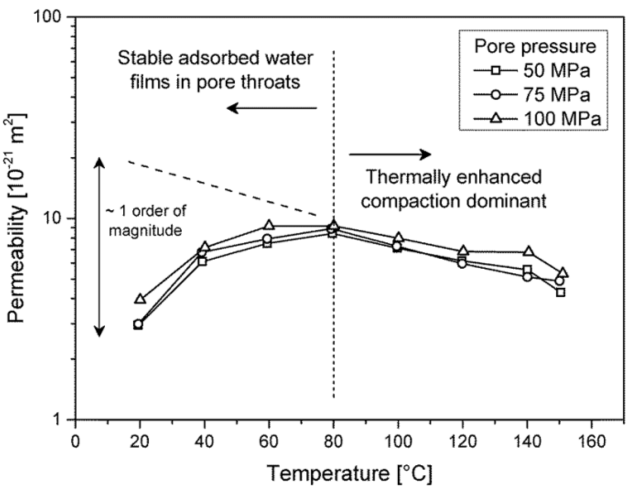

Fig. 4: Water permeability of clay as a function of temperature (the temperature of expansion-contraction threshold is shown as a vertical dashed line) (Faulkner et al., 2003)

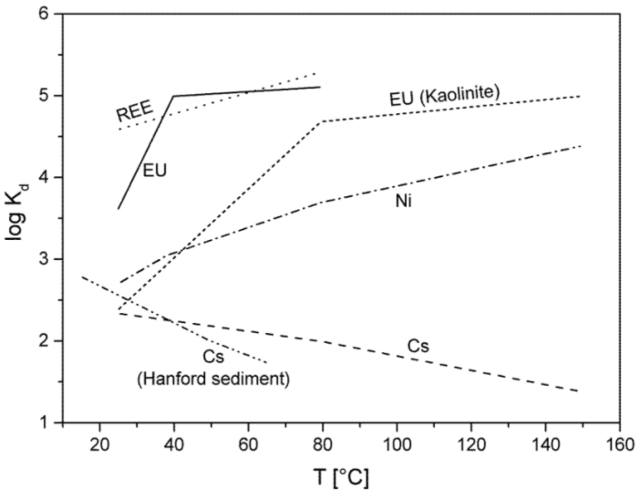

Fig. 5: Distribution ratio (Kd) for Cs, Eu, Ni and rare earth elements (REE) for montmorillonite, kaolinite and Hanford sediment (USA) as a function of temperature (Liu et al., 2003; Tertre et al., 2005, 2006)

Finally, the microbiological activity should be considered as the process with potentially most severe detrimental effects on isolating properties of clay-based geotechnical barriers. To name only one example, hydrogen sulfide is the metabolic product of sulfate-reducing bacteria that is capable to reduce Fe(III) to Fe(II) in the structure of clay minerals and thus to destabilize them and to trigger their accelerated alteration. Fe(III)-reducing and sulfate-reducing bacteria as well as methanogenic microbes can also carry out this reduction reaction directly, without a mediating chemical compound. This reaction can lead to large-scale transformation of swelling clay minerals into non-swelling ones within several hundreds of thousand years (Raiswell & Canfield, 1996). Besides, hydrogen sulfide can strongly accelerate the corrosion of HLW and SF containers.

Fe(III)-reducing, sulfate-reducing bacteria and methanogenic microbes remain active up to, respectively, 95-110 °C, 121 °C and 122 °C. Although higher temperatures at the atmospheric pressure lead to the strong or even complete reduction of their population, many species can form endospores, which increase the temperature limit of their survival by 30-40 °C. Nevertheless, in clays which during the diagenesis experienced paleotemperatures of 140 °C and 145 °C were found, respectively, only trace and zero numbers of microbes (Colwell et al., 1997). Thus, an increase of the temperature limit in a repository to 150 °C can provide a complete sterilization of the HLW and SF containers and the geotechnical barriers closest to them and hence to prevent detrimental microbiological effects for a long period of time.

3 SUMMARY AND OUTLOOK

Taking into account the results presented here as well as other thermally induced effects in clays, which were not considered in this short overview (table 2), a conclusion can be made that it is reasonable to increase the temperature limit for clays in a repository for high-level waste and spent fuel elements to 150 °C.

Tab. 2: Evaluation of the thermally induced effects in clay with respect to the impact on its barrier properties based on available studies for temperatures from 20 °C to 150 °C (Jobmann & Meleshyn, 2015; Jobmann et al., 2016).

Thermally induced effect	Impact evaluation
Expansion/contraction of claystone	$T < T_{ec}$[1]: negative $T > T_{ec}$: positive
Mechanical properties	$T < T_{ec}$: negative $T > T_{ec}$: positive
Hydraulic properties	$T < T_{ec}$: negative $T > T_{ec}$: positive
Thermal properties	$T < T_{ec}$: negative $T > T_{ec}$: positive
Swelling pressure	negligible
Dehydration of swelling clay minerals	negligible
Generation of hydraulic gradients	negligible
Vaporization	negligible
Gas entry pressure	negligible
Illitization of smectites	negligible
Smectitization of illites	positive, but negligible
Cementation by generation of silica and new minerals	negligible
Thermo-chemical sulfate reduction	negligible
Sorption properties	negligible
Kerogen transformation	non existent
Microbiological activity	T < 122 °C: negative T > 122 °C: positive

[1] T_{ec} - expansion-contraction threshold temperature

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Does deep borehole disposal of HLRW have a chance in Germany ?

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Abstract:

Using deep boreholes for disposal of high-level radioactive waste (HLRW) can has its strength in long term safety due to an ample distance between the HLRW and the biosphere (> 1 500 m) and may take advantage of multiple geologic barriers as safety features. The great depth impedes efficiently proliferation. Finally, there may be a benefit in time for technical implementation and costs. Thus, deep boreholes are considered as a viable additional option for disposal of high-level radioactive waste. Open questions are related to technological feasibility, e. g. design of containers, operational safety and compliance with regulatory requirements in Germany such as retrievability and recoverability for 500 years.

A simplified, generic safety concept is presented, which may be the basis for subsequent investigation of technical feasibility. Based on the expected amount and kinds of high-level radioactive waste in Germany minimum requirements for the diameter of boreholes and containers are derived. Furthermore the operational safety of emplacement and sealing of the boreholes is considered. Options for retrieval e. g. using a combination of drill-strings, wireline, and liner extension are presented. Boreholes can be sealed quickly e. g. using the creeping property of salt formations.

This concept is mirrored for its compliance with the safety requirements “Governing the Final Disposal of heat-generating Radioactive Waste” of the German Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety and the site selection criteria defined by the German commission “Storage of high-level radioactive waste”.

It is shown that in principle a disposal in deep boreholes (DBD) can ensure the safe containment of radionuclides using a containment providing rock zone of type Bb. Some developments in technology and concepts are necessary for implementation and assessment of long-term safety of DBD. German safety requirements and the commission “Storage of high-level radioactive waste” recommend recoverability for 500 years after closure. Based on today´s knowledge recovery could be technically feasible in theory, but can not be guaranteed for the whole period.

If disposal of HLRW in deep boreholes shall have chance to be a tested option for geological disposal in Germany further developments and a demonstration of its technical feasibility is required in order to allow for a detailed safety analyses of operational and long-term safety since there is currently no application of deep boreholes with the anticipated large diameters in oil and gas industry. This needs active research, development and demonstration.

1 INTRODUCTION

The disposal of high-level radioactive waste using deep boreholes in geological formations has not been considered in detail in Germany in the past.

Deep borehole disposal (DBD) may offer some advantages such as a better containment due to the greater depth, a faster disposal and less costs. It is discussed currently by the Departement of Energy in the USA /NWTRB 16/, UK /GIB 14/ and in Germany /BRA 15/.

Therefore, the German commission “Storage of high-level radioactive waste”, which was active from 2014 to 2016 /KOM 16/, discussed deep borehole disposal as alternative disposal option and requested an expertise with several questions to be answered /BRA 16/.

In the following an overview on the concept of HLRW disposal in deep boreholes and some results of /BRA 16/ with a conclusion are presented. Furthermore, the DBD concept is discussed with regard to its compatibility with recommendations of the commission and the safety requirements of the federal ministry for environment /BMU 10/, which were designed for a mined deep geological repository.

2 WASTE AMOUNT IN GERMANY

The amount of high-level radioactive waste is limited in Germany due to the phase-out from nuclear energy. The waste forms are mainly spent fuel elements of power reactors (approx. 35 000 pieces), cans with vitrified waste from reprocessing (approx. 8 000 pieces) and some spent fuel elements from research reactors (approx. 2 000 m^3) /PEI 11/.

3 CONCEPT FOR BOREHOLE DISPOSAL

The most important safety requirement is to isolate effectively the high-level radioactive waste from the biosphere in the longterm. The technical feasibility and the operational safety have to be demonstrated as well.

3.1 Safety requirements and disposal concept

The great depth which can be reached by boreholes can contribute to a safe containment if overlying sedimentary rocks provide additional geological barriers (multiple geological barrier concept). The large distance of the disposed waste is also expected to ensure long migration times for radionuclides released from the waste form to human beings and the biosphere. The expected time period for keeping a borehole open for waste emplacement is considerably shorter than for a mined geological repository, thus, proliferation risks can be assumed to be much lower.

The general requirements for the presented basic concept for DBD are:

- The disposal of vitrified waste containers and spent fuel should be technically feasible. Other waste is not considered here.
- The concept should provide a multi-barrier concept.
- The possibility for waste retrieval should comply with the safety requirements of the /BMU 10/.
- The concept should also allow monitoring during the operational and post-closure phase
- The appropriate lithology should be available in Germany and characterisable.

The use of several, independent geological barriers formed by e. g. clay and salt layers together with seals provide the main safety functions of the basic concept. This means that boreholes have to be sealed effectively within these barriers to restore the functionality of the barriers. Furthermore, a slow groundwater movement should be favoured in great depths which ideally restricts radionuclide migration to diffusion only.

The generalized concept envisages the disposal in the geological basement (which is most likely a crystalline basement) which should be overlain by at least two redundant or diverse geological barriers. Ideally a feature is provided by the geology which can act as gas trap below these barriers. The minimum depth for DBD is set to 1 500 m with a maximum depth of 3 500 m. This will allow to find sites with several independent geological barriers and to exclude for sure glacial impacts on barriers and waste.

The maximum depth should be optimized by assessing state-of-the-art drilling, disposal technology and the outcome of safety analyses. A vertical borehole in this basic concept is preferred over inclined boreholes but multiple and deviating boreholes are possible /DEA 15/, /TIS 06/. The concept is depicted schematically in Fig. 3.1.

Possible geological barriers overlying the disposal zone (designated zone) are:

- Clay rock: bedded clay which can ensure retardation and containment. Fig. 3.1 shows an alternating sequence of clay and sandstone.
- Salt rock: bedded salt with high sealing capacity and self-sealing ability based on its visco-plastic characteristics. Fig. 3.1 shows bedded and domal salt.

These barriers shall be combined. At least two independend barriers should be available.

A further possible feature would be porous rock (e.g. sandstone) acting as trap for gases, which could be released from the disposal zone. Fig. 3.1 shows a sandstone formation below the salt layer actinga as a gas trap. Such conditions are found in an undisturbed manner in Germany.

The disposal is planned for a zone, where all geological barriers are working. This zone is called designated zone with safety distance to the geological barriers. The zone where all multiple barriers can function fully and a containment can be provided is called retention zone. The retention zone is characterised by having at least one geological barrier. The transfer zone is located above the topmost geological barrier.

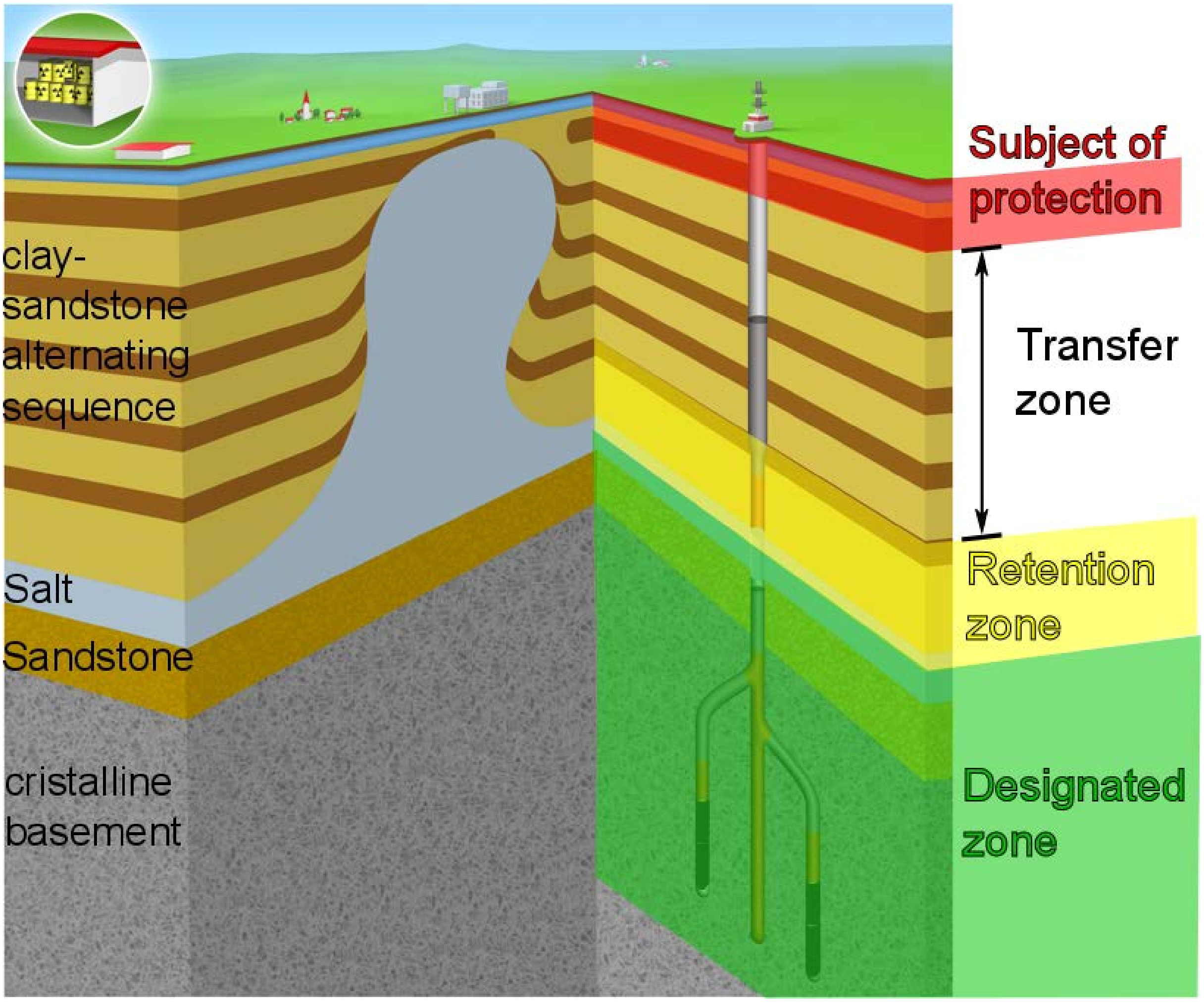

Fig. 3.1: Concept for borehole disposal in deep geological formations (example)

3.2 Operational phase

During disposal operation any container is removed from the biosphere and to pass a transfer zone. Below this zone at least one geological barrier is functioning and can provide complete containment after backfilling and sealing the borehole.

It is obvious that technical measures have to ensure the safety for the biosphere and the transfer zone as it is the case for any other disposal technology.

3.3 Exploration

Geological exploration is necessary to find and characterize the site. This can be performed using a number of standard technologies which include also exploration boreholes with logging and coring. Exploration boreholes may be used at later stages for monitoring or can be developed as boreholes for disposal.

3.4 Container

Containers have to withstand the geomechanical and geochemical conditions during disposal operation. If recovery of containers is required for a certain period of time beyond disposal operation the container should also withstand the conditions at the disposal zone. The latter is a requirement for the design of canisters which may be met by an additional allowance of its wall thickness or selection of material.

A **D**eep **B**orehole **C**ontainer – **R**etrievable (DBC-R) was designed using austenitic steel (Fig. 3.2). The size of the container was derived from the diameter of the canisters with vitrified waste (0.435 m), which are not pressure resistant due to a head space, and the length of the fuel rods from spent fuel elements (approx. 4.5 m) assuming that one container should fit for all waste types. The container can suit three canisters or an assembly of rods from spent fuel elements.

The thickness of the wall of the container depends on the disposal depth and their vertical stacking (Tab. 3.1). The depth of 5 000 m was included by request of the commission /BRA 16/. The diameter of the DBC-R is at minimum about 55 cm for disposal from 3 000 m to 3 600 m depth with allowance for temperature and some corrosion. The length of a container is about 5.6 m. The thickness of the wall (diameter of DBC-R) increases with stacking and depth. Considering a wall thickness with approx. 6 cm the total outer diameter of the casing is 70 cm. Therefore a borehole diameter of 75 cm including some allowance for uncertainties could be sufficient (Fig. 3.3).

Further optimization of canister design and material is expected. By varying the thickness or material of the wall it can be accounted for corrosion or other degrading mechanisms.

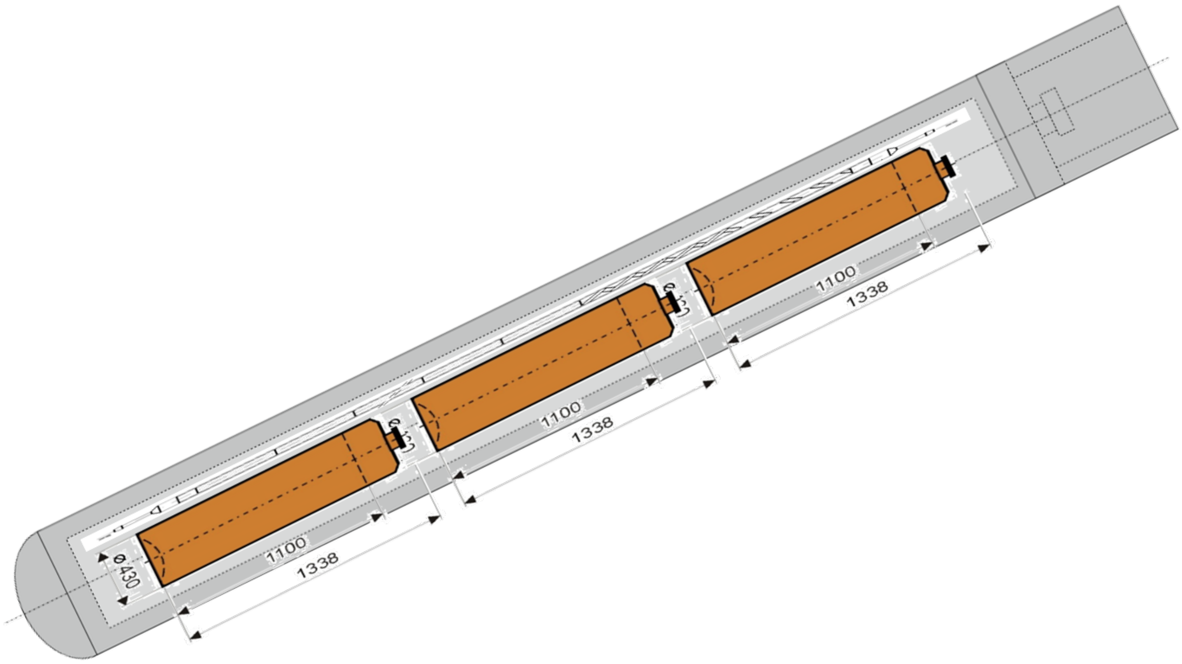

Fig. 3.2: Deep Borehole Container – Retrievable (DBC-R) with cans and rods (rod is not to scale)

Tab. 3.1 Maximum depth, wall thickness of DBC-R, number of boreholes for German HLRW waste (assuming disposal between 3 000 m and maximum depth) and approx. diameter of borehole (assuming a wall thickness of 6 cm for the casing)

Max. depth of borehole	Wall thickness of DBC-R	DBC-R per borehole	Number of boreholes	Approx. diameter of borehole
3 600 m	4.5 cm	103	107	75 cm
4 200 m	6.5 cm	205	55	80 cm
5 000 m	10 cm	363	31	90 cm

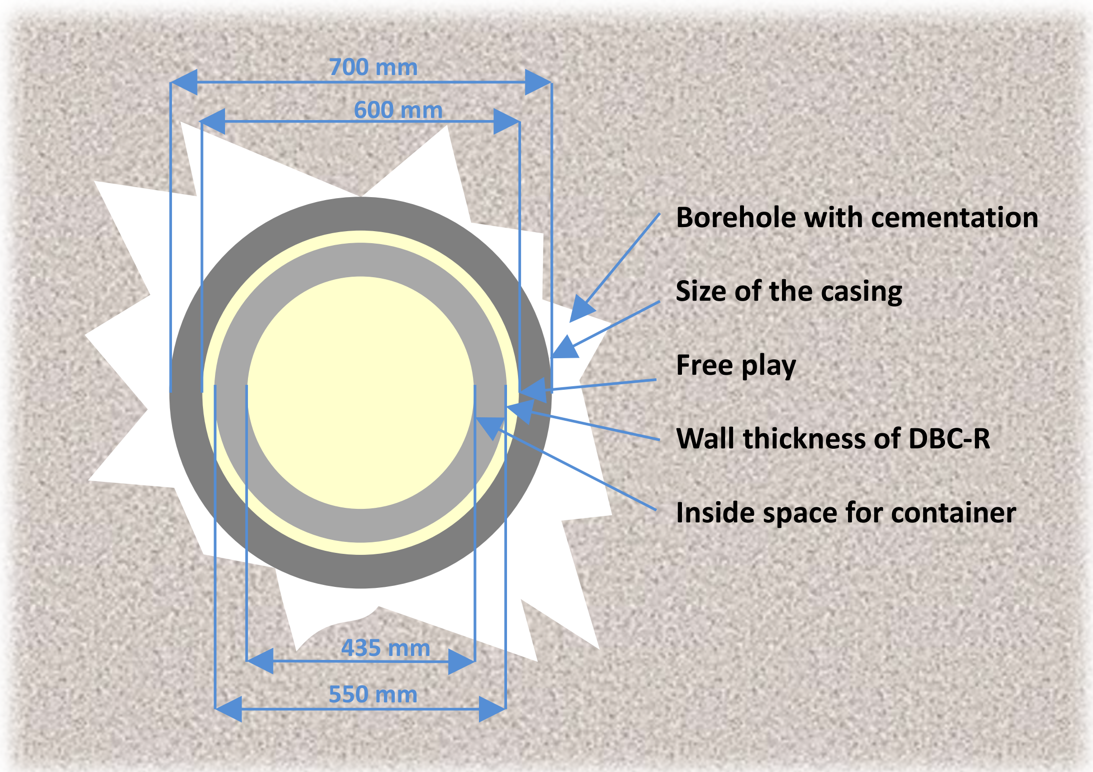

Fig. 3.3: Section through a borehole with casing and DBC-R

4 BOREHOLES AND DRILLING TECHNOLOGY

Technical feasibility and costs of drilling are important factors for disposal in geological formations. Classical drilling technologies are used in conventional and unconventional oil and gas production, in geothermics and mining. Experiences are also available from experimental drilling and research.

The diameter of boreholes ranges from cm to m (shaft sinking) in drilling technology. Since DBD of waste containers with diameter of 55 cm an more is envisaged the borehole diameter must be large (Tab. 3.1). Physical constraints have to be considered, when boreholes become wider and deeper.

The difference between the pressure of overburden of the rock and the pressure inside the boreholes increases with depth. The differential stresses can be so high that boreholes can collapse. This differential stress can be lowered when the boreholes are filled with a fluid. The hydrostatic pressure of the fluid reduces the effective rock pressure on the walls of the borehole or the casing.

Thus the geomechanical stability is a limitating factor for the diameter and the depth of boreholes. This is relevant during site selection. The in-situ stress field has to be observed when setting the orientation of the borehole and avoiding the failure of the wall of the borehole.

Using drilling fluids (drill mud) boreholes can be drilled safely in great depths as provided by the status of technology and experience from more than 100 000 boreholes for oil and gas in different geological formations. Larger diameter than in oil and gas industry were realized in research drillings (see e. g. /ENG 96/). Segmental logging of the boreholes gives information on alteration, lithology, porosity, conductivity, density of the formation in the vicinity of the wall of each borehole.

Most drilling technologies use a drilling fluid which supports drilling and cleaning of the borehole. In addition to the stabilization by the drilling fluid boreholes are completed with several steel casings which prevent wall collaps, inflow and separates different hydrological layers. The composition of this fluid may vary. After well completion, the drilling fluid is usually replaced by a borehole fluid with different characteristics. Cementation of the casing is usually required.

4.1 Borehole design

Boreholes are cased in order to safely reach the desired depth for disposal by drilling. The detailed design of the casing set is based on subsurface data such as formation pressures, rock strengths, wellbore orientation and stress field.

The borehole drilled must be large enough to fit in the casing string and to allow room for cement between the outside of the casing and the hole. A wellhead usually is installed on top of the first casing string after it has been cemented in place.

The inside diameter of the casing must be large enough that the next bit fit into it to continue drilling. Thus, each casing string will have a subsequently smaller diameter.

Casing design for each size is done by calculating the worst conditions that may be faced during drilling and production. Mechanical properties of designed pipes such as collapse resistance, burst pressure, and axial tensile strength must be sufficient for the worst conditions.

Casing strings are supported by casing hangers that are set in the wellhead.

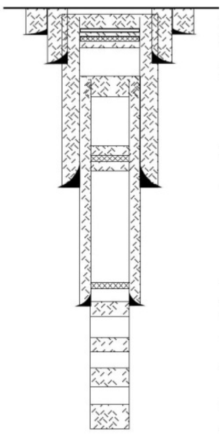

Fig. 4.1: Casing (schematical)

The minimum diameter of a borehole with casing for disposal is approx. 75 cm assuming DBC-R-containers with canisters of vitrified waste. A safety margin against collapsing and other criteria should be provided by the casing and by the container. The experiences on longterm stability and tightness of casings of boreholes cover more than 100 years. More recent experiences are provided by the drillings at the KTB site /ENG 96/, Groß-Schönebeck /KWI 08/, Großbucholz /KWI 08/, Urach /TEN 00/, Soultz-sous-Forets /TIS 06/, Gravberg /JUH 98/ and Kola SG 3 /FUC 12/, which are publicly documented. These wells have reached deeper depths than considered here for final disposal in the concept but exhibit smaller diameters.

Fig. 4.2 shows the casing set of the KTB site, which had been drilled ca. 25 years ago. The outer diameter of the borehole with 44,5 cm in 3 000 m is near to the outer diameter of approx. 75 cm given in Tab. 3.1 for a depth of 3 600 m. The red retangular show the diameter of the DBC-R of 55 cm.

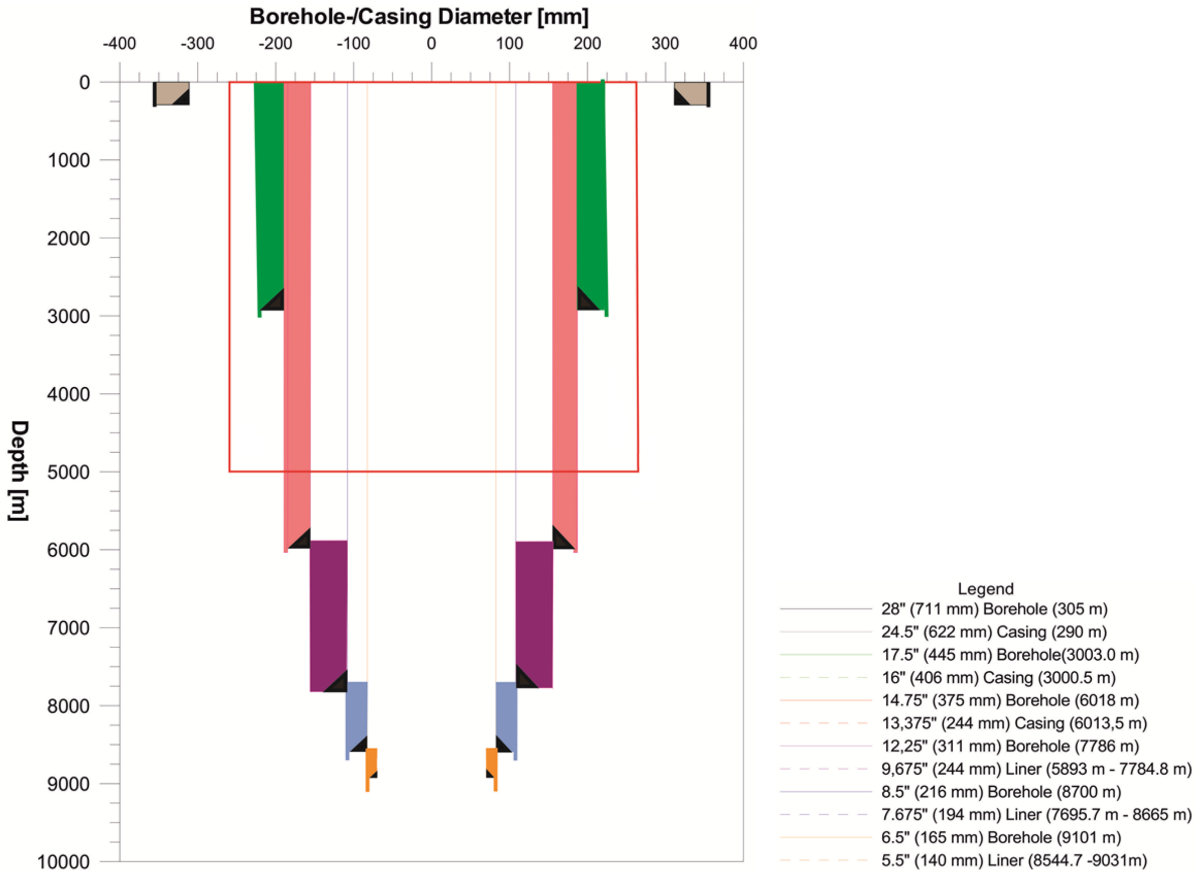

Fig. 4.2: Casing of the KTB (schematical)

4.2 Drilling and Wellbore Logging

The most widely used method is rotary drilling. Oil well drilling utilises tri-cone roller, carbide embedded, fixed-cutter diamond, or diamond-impregnated drill bits to wear away at the cutting face. This is preferred because there is no need to return intact samples to surface for assay as the objective is to reach a formation containing oil or natural gas. Sizable machinery is used, enabling depths of several kilometres to be penetrated. Rotating hollow drill pipes carry down bentonite and barite infused drilling muds to lubricate, cool, and clean the drilling bit, control downhole pressures, stabilize the wall of the borehole and remove drill cuttings. The mud returns back to the surface around the outside of the drill pipe, called the annulus. Examining rock chips extracted from the mud is known as mud logging. Another form of well logging is electronic and is frequently employed to evaluate the existence of possible oil and gas deposits in the borehole. This can take place while the well is being drilled, using “Measurement While Drilling tools”, or after drilling, by lowering measurement tools into the newly drilled hole.

Deviated boreholes currently reach 10 kms distance to the wellhead and are frequently used especially in offshore-drilling.

/ARN 11/ concluded that with todays readily available technology drillings can be done with diameter of 43,2 cm (17‘‘) down to 5 000 m. Research and development is necessary to generate drilling technology for standard larger diameters since current drilling technology aims to be as small as possible in diameter in oil and gas industry due to reduce costs. The expert assessment is that borehole diameters up to 75 cm in 3 500 m are feasible with enhanced technology. Borehole diameters of 100 cm in 5 000 m depth as it was requested by the commission /BRA 16/ were assessed that they could not be safely operated with current technology.

4.3 Boreholes and disposal operation

A drilling fluid and a cemented casing is necessary for drilling and operational safety aspects during disposal. A typical diameter of a borehole including casing was assessed to be 75 cm for a disposal depth from 3 000 to 3 600 m and is increasing with depth and number of stacked containers (Tab. 3.1). Assessments show that a distance of 50 m between disposal boreholes should be sufficient to exclude mutual thermal and other effects.

The complete borehole is cased and cemented. At least two casings are foreseen within the biosphere (subject of protection). The disposal zone is equiped with a cemented casing. An additional casing (liner extension), which is not cemented, is installed for disposal and represents a safety feature within the subject of protection and transfer zone, potentially reaching down to the retention zone, when the disposal operation of containers is performed. This liner extension can be removed completely with a container in case of sticking (Fig. 4.3). A hydraulic gate allows recovery of any contaminated borehole fluid.

After installation of the cemented casing the drilling fluid can be replaced by a borehole fluid specifically designed for disposal operation. Solidfree borehole fluids are used in order to allow recovery during the operational phase. This borehole fluid should be compatible with casing and containers to minimize corrosion, have a sufficient density to ensure the borehole stability, suitable viscosity, and have a low complexing ability for radionuclides. After disposal, each individual canister may be cemented replacing the borehole fluid by cement to minimize fluid volumes and to separate the disposed canister from following disposal operations.

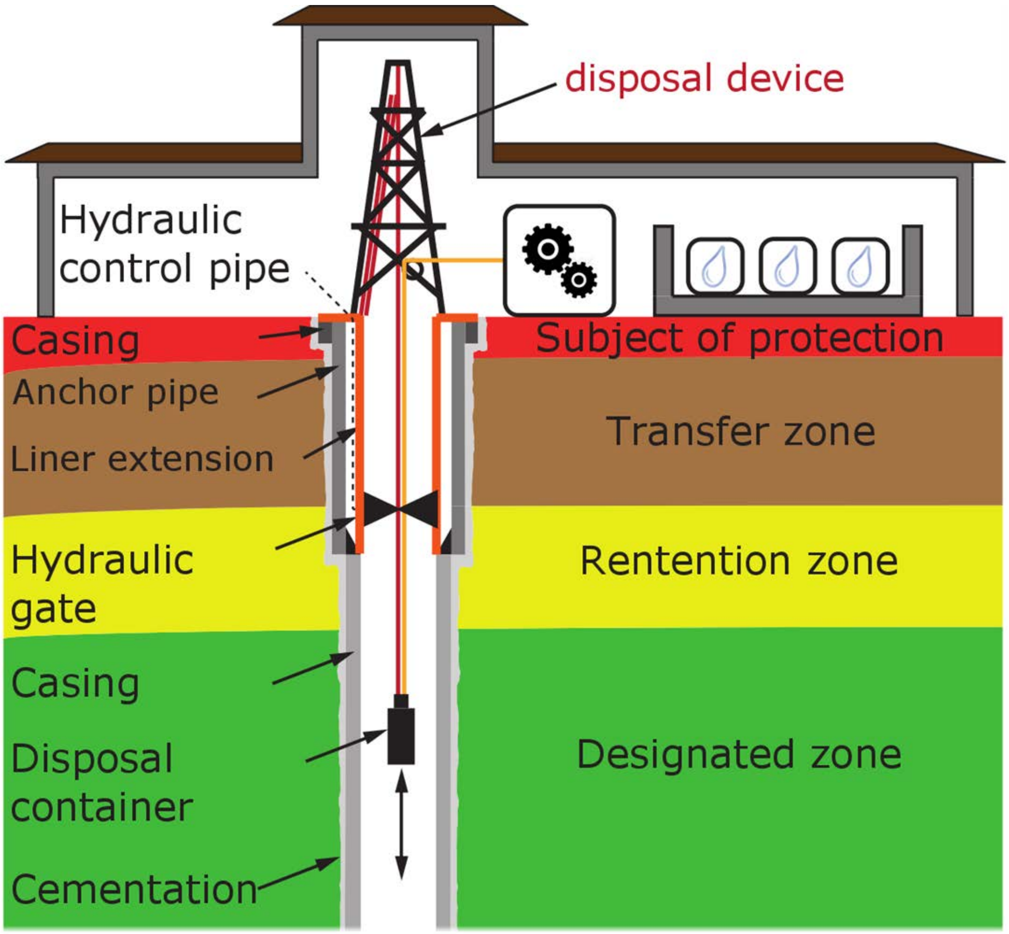

Fig. 4.3: Borehole and disposal operation

4.4 Borehole seals

After completion of disposal of the canisters and cementation of the well within the designated zone, boreholes must be fully sealed and abandoned. Borehole seals shall ensure that there is no intrusion of groundwater to the disposal zone and that no contaminants from the disposal zone are released. The seals shall reestablish the characteristics of the geological barriers drilled through. The sealing of the borehole may be achieved in several ways with materials of proven longterm stability (salt, clay, bitumen) and over the entire length of the borehole. A favourable method for sealing a borehole uses the creeping of salt rock /KRE 09/ which is foreseen in the concept. Fig. 4.4 shows a cased borehole (1.), which is reamed wholly including removal of the casing in the salt formation (2.). Due to the relatively small diameter of the borehole, the high temperature (above 100 °C) and high pressure of the salt rock, the salt is creeping within hours or days (3.) and the borehole is sealed effectively on its reamed length (4.). The sealing process may be enhanced and supported by filling the reamed part with crushed rock salt. Sealing operations can also be performed in clay horizons. Any seal can be reamed by drilling.

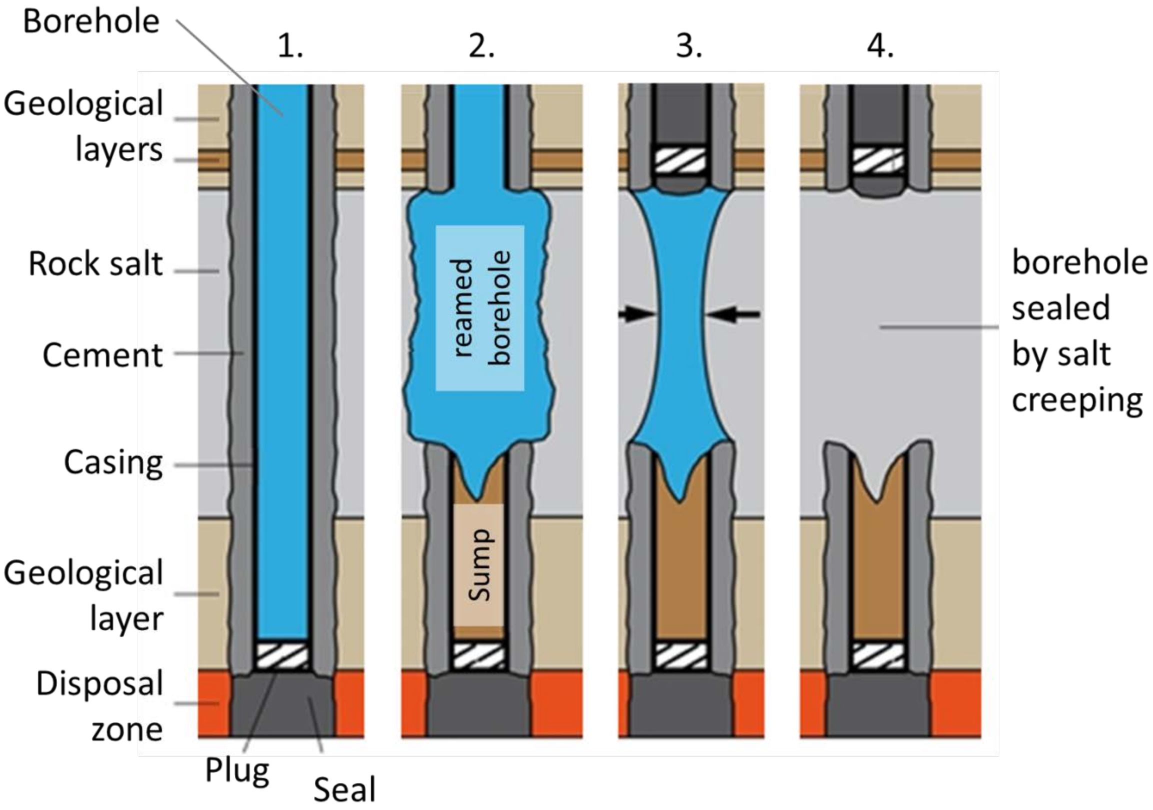

Fig. 4.4: Steps of borehole sealing (geology not to scale, modified after /KRE 09/)

5 SAFETY OF DISPOSAL OPERATION

Available technologies for disposal can include wireline, drill string, coiled tubing, free fall (in the borehole fluid) and liner emplacement modes for instruments of any kind. The main difference to conventional oil and gas industry is the manless operation of the transfer and the huge load in waste disposal compared to probes. A disposal technology for containers in shallower boreholes has already been demonstrated successfully /FIL 10/.

The proposed containers are not self-shielding. Therefore measures for radiation protection have to be foreseen and a detailed design of such a facility for emplacement in deep boreholes has to be developed to comply with radiation proctection during operation. A technical concept for an emplacement facility should include completey encased surface facilities and the borehole with its casings. The container should be delivered in a transfer cask through a lock to the disposal facility and connected to the emplacement string for disposal. An additional wire as backup serves as an additional safety measure. When the container is connected to the string, the preventer can be opened to lower the container into the well. The lowering of each container is slowed down also hydraulically by the borehole fluid. Displaced borehole fluid is collected in a dedicated tub. The fluid is monitored for contaminants and radionuclides. The container is released from the emplacement string, when the final disposal position is reached. The emplacement string is removed and monitored for contamination after the disposal of every container. The facility should run in automatic mode that there are no humans required close to the HLRW.

The safe emplacement has to be demonstrated.

The design of the containers need not to be self-shielding but should be tight for aerosols since it is asked for manageability for a period of 500 years after emplacement /BMU 10/. The present design of the container using steel will inevitably lead to some corrosion due the availability of water and the high temperature in the disposal zone. Therefore, research and development is necessary from the viewpoint of the design of containers for borehole disposal.

Provided that the borehole exists and is ready for disposal, there is no need for dedicated research on installation of rigs or on transportation of loads for disposal since this is standard technology. If deviating boreholes are foreseen a very low inclination should be envisaged to minimize friction of containers during emplacement.

6 SITE SELECTION CRITERIA AND SAFETY ASSESSMENTS

/KOM 16/ favours disposal in geological formations using mining technology with retrievability and recoverability. /KOM 16/ also mentioned disposal using boreholes in geological formations is the only real alternative option over transmutation or long-term interim storage. If disposal in deep boreholes should have a chance in Germany the concept and the selected site has to comply with German safety requirements (currently under revision) /BMU 10/ and has to undergo a site selection procedure with (preliminary) safety assessments /KOM 16/.

The containment providing rock zone (CPRZ) is a key element of the safety requirements /BMU 10/. The concept for borehole disposal takes advantage of multiple geological barriers. These barriers are clay or salt rock layers, which can be used to define several and different possible CPRZ´s of types A (enclosure of waste by CPRZ in the host rock) or Bb (large lateral extension of CPRZ overlying the waste in the host rock) /AKE 02/. If disposal takes place directly in a salt or clay formation the type A is possible. Disposal takes in a crystalline basement should have at least a CPRZ of type Bb if the containment can not be provided by host rock and technical barriers /KOM 16/.

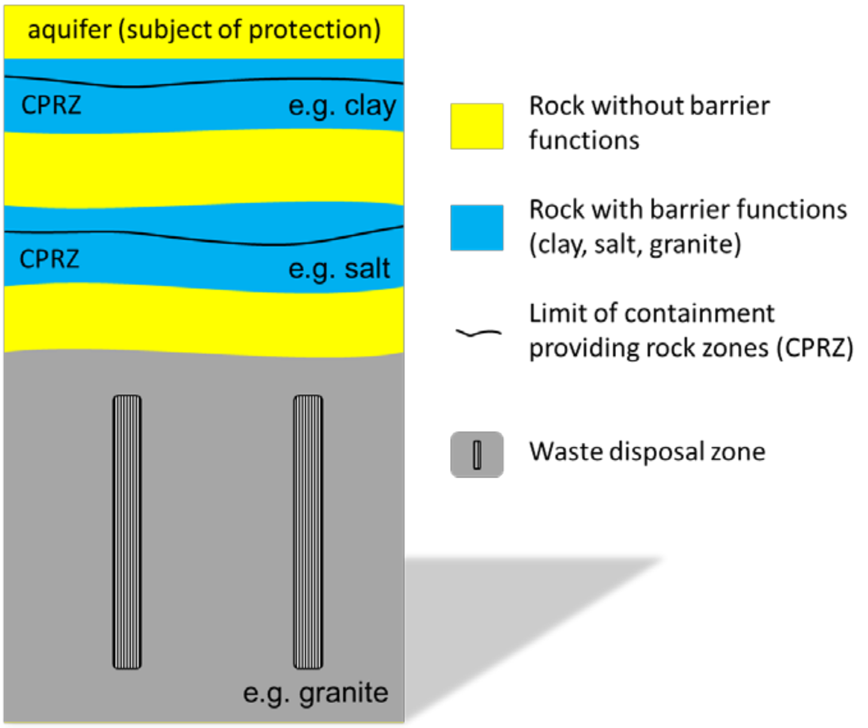

Fig. 6.1: Possible CPRZ for borehole disposal

A generic concept for disposal in deep boreholes has been shown above. The general methodology for safety assessments - which is also discussed in /KOM 16/ - has to be adapted in some technical details (e. g. shaft seals could be seen as equivalent to borehole seals).

Any assessment will be based on the safety requirements /BMU 10/ and will make analogous use of the requirements and criteria for site selection for geological disposal in a mine given by /KOM 16/. The technically important requirements for site selection have exclusion, minimum and weighing geo-scientific criteria.

Although these requirements and criteria of /KOM 16/ are intended to be applied on sites in geological formations using a mined repository they are discussed here if they can be applied to a concept using deep boreholes for disposal or if an adaptation of the requirements and criteria should be considered to be applicable. Specific test criteria for deep borehole disposal are not available presently and planning criteria need not to be discussed here.

6.1 Geo-scientific requirements and criteria for a mined repository applied to deep borehole disposal

The geo-scientific criteria for exclusion (large scale vertical movements, active faults, impact from mining, seismic and vulcanic activity, age of groundwater) can be applied directly on the concept of borehole disposal since it is also a disposal in a geological formation.

The minimum geo-scientifc criteria (permeability of formation, thickness of the CPRZ, depth of CPRZ, disposal area, period for proof) can be applied on the concept of borehole disposal in the same way as it is for a mined repository when considering that the location of the CPRZ does not need to coincide with the location of the disposed waste.

Eleven weighing geo-scientifc requirements with criteria in three groups /KOM 16/ have to be discussed in more detail. The three groups were: quality of containment and reliability of its evidence, validation of containment, additional safety relevant features (Tab. 6.1, Tab. 6.2, Tab. 6.3).

All requirements and criteria are assessed using the generic concept for deep borehole disposal. At a later stage a site specific assessment is required. Due to the disposal in deep boreholes some weighing criteria can be fullfilled favourably when sticking strictly to the definition of the CPRZ and host rock. Since in the basic concept of the DBD the CPRZ is not part of the host rock (disposal zone) but part of the overlying geological barriers, some criteria for safety relevant features should be applied to different rocks as well (No. 4 of Tab. 6.1, Tab. 6.2). A favourable weighing seems to be possible, but is relative to the underlying basic concept of DBD only.

Two requirements (No. 2 and 3 of Tab. 6.3) have been assessed to be not applicable for the concept of borehole disposal.

Due to the disposal depth the temperature will be above 100°C even without disposal of heat generating radioactive waste. The temperature of rock and container will be in any case significantly above 100°C. The heat generating waste will heat further up the rock temperature, but the relatively small size of the container limits the rise in temperature. The compatibility of the rocks to the rise in temperature has to be proven prior disposal. No general temperature limit can be given here as it seems to be specific to the lithology.

The present concept for containers and casing foresees steel which will lead inevitably to some gas generation due to the presence of steel and the expected presence of groundwater. Even if the gas generation rate may be low a future concept should minimize the use of steel to minimize the potential generation of gas.

Two requirements (No. 4 and 5 of Tab. 6.3) are recommended for reassessment to be applicable for DBD. The requirement of having a “high capability of retention of CPRZ for radionuclides” could be extended to other rock formations available below the CPRZ of the basic concept. The requirement of “favourable hydrochemistry“ has to be reassessed for the host rock to be useful for DBD since containment is not provided there.

Tab. 6.1 Weighing group 1: quality of containment and reliability of its proof

No.	Requirement	Criteria	DBD
1.	No or slow transport with groundwater in the CPRZ	• Effective flow velocity less than 1 mm/a • Low available ressources of groundwater • Low rate of diffusion.	Can be fulfilled depending on the definition and size of the CPRZ of the overlying rocks (salt / clay layer).
2.	Favourable configuration of rock body, host rock and CPRZ	• Barrier efficiency (thickness and containment) • robustness and safety margins • Size of CPRZ • Clay rock: connection to water bearing strata and high hydraulic potential.	Can be fulfilled depending on the definition and size of the CPRZ of the overlying rocks (salt / clay layer) and host rock.
3.	Good spatial characterisation	• Ascertainability (low variation and distribution of charateristic features of the CPRZ, low tectonic overprint) • Transferability: large scale uniform or similar formation of rock of the	Can be fulfilled depending on the definition and size of the CPRZ of the overlying rocks (salt / clay layer). Tools to characterize host rock properties are available for greater depths.

No.	Requirement	Criteria	DBD
		CPRZ.	
4.	Good predictability of the longterm stability of favourable conditions	Changes with time in • Thickness of the CPRZ • Size of the CPRZ • Permeability of rock formation.	Can be fulfilled depending on the definition and size of the CPRZ of the overlying rocks (salt / clay layer). The rock formation relevant for assessment of the permeability should be defined.

Tab. 6.2 Weighing group 2: validation of containment

No.	Requirement	Criteria	Borehole disposal
1.	Favourable rock mechanics	• Low tendency of generation of secondary permeability by rock mechanics in host rock and CPRZ.	Can be fulfilled depending on the definition and size of the CPRZ of the overlying rocks (salt / clay layer). The host rock formation relevant for assessment of the rock mechanics is in the disposal zone.
2.	Low tendency of generation of groundwater flows in host rock and CPRZ	• Variability of permeability of rock formation. • Reconstitution of cracks and secondary permeability by self-healing or closure of cracks.	Can be fulfilled depending on the definition and size of the CPRZ of the overlying rocks (salt / clay layer). The host rock formation relevant for assessment of the groundwater flows is in the disposal zone.

Tab. 6.3 Weighing group 3: further safety relevant features

No.	Requirement	Criteria	Borehole disposal
1.	Protective composition of overlying rocks	Protection of the CPRZ by • Coverage of the CPRZ with rock inhibiting groundwater flow • Distribution and thickness of such rocks • Distribution and thickness of erosion resistant rocks • No structural issues in coverage of rocks	A number of geological units with different but favourable characteristics, including additional redundant or diverse barriers, are possible in the basic concept due to depth of CPRZ´s and disposal zone.

No.	Requirement	Criteria	Borehole disposal
2.	Good conditions to avoid or minimize gas generation	• The gas generation should be as low as possible for final disposal.	The present concept for container and casing foresees steel which will inevitably lead to some gas generation. Gas generation may be minimized or slowed down by choice of suitable borehole fluid or cementation of containers. A future concept may also minimize the use of steel or may foresee physical gas traps underneath the CPRZ. Gas generation can not be completely avoided in the basic concept.
3.	Good compatibility to temperature	• A temperature of 100°C is recommended for the surface of the canister.	The temperature in the disposal zone will be already higher than 100°C. It is therefore not a criterion for site selection and not applicable or useful for DBD.
4.	High capability of retention of CPRZ for radionuclides	• High sorption capacity of rocks of CPRZ • High content of minerals with reactive surface in rock of CPRZ • High ion strength of the groundwater within the CPRZ • Size of the pores in the CPRZ within nm-scale.	Can be fulfilled depending on the definition and size of the CPRZ of the overlying rocks (salt / clay layer). Other rock formations could be relevant for assessment and should be defined. These criteria should be reevaluated also for other rock formations and the host rock of the disposal zone.
5.	Favourable hydrochemistry	The groundwater of the host rock / CPRZ shall • be in chemical equilibrium with the rocks • have pH of 7-8 • favourable redox conditions (anoxic-reducing) • have a low content of colloids and complexants • have a low concentration of carbonates	Can be fulfilled depending on the definition and size of the CPRZ of the overlying rocks (salt / clay layer) which is not part of the host rock. Other rock formations could be relevant for assessment and should be defined. These criteria should be reevaluated also for other rock formations and the host rock of the disposal zone.

6.2 Safety analysis and assessment

Safety analyses have to be done to assess operational safety and the long-term safety of borehole disposal. A recent preliminary safety analysis on a generic concept showed a good retention of radionuclides /ARN 13/. However /ARN 13/ did not consider all possibly relevant processes (gas flow, groundwater flow through faults, fractures and the EDZ). Therefore a more detailed operational and longterm safety analysis of the basic concept of DBD still has to be performed /BRA 16/.

6.3 Retrievability / Recoverability

The /KOM 16/ sets the general requirement of a reversibility of the site selection process. On one hand this concerns the site selection process itself, but on the other hand it also has some technical implications which concern retrievability and recoverability of disposal.

Retrievability is the planned technical option for removing emplaced radioactive waste containers from the repository facility during its operational phase. This is also required by /BMU 10/. The containers and the borehole must allow the retrieval until sealing and closure of the boreholes. It has to be decided whether closure means a single borehole, a borehole field or all boreholes. Here is the understanding that the disposal can be performed within a few years and the borehole is sealed and closed. It was assessed based on experiences in conventional drilling that a retrieval of containers should be possible within a time period of at least 5 years after closure.

Recovery is the retrieval of radioactive waste from a final repository as an emergency measure after the operational phase is over. This emergency may happen after the borehole is closed. The /BMU 10/ asks for a time period of 500 years for recovery of waste containers. The understanding is that the container and the casing should be designed to withstand this time period without releasing radioactive contaminants. Experience on recovery of lost objects in conventional drilling cover 100 years but does not include recovery of corroded containers with HLRW.

Whereas from the perspective of experts the retrievability seems to be manageable once the borehole and casing exists, the recoverability from deep boreholes needs research and development to show if it is feasible.

6.4 Hazards

Hazardous incidents during the operational phase may cause relevant releases of radionuclides. Whereas volcanoes, earthquakes and other hazardous geological events should be excluded as far as possible due to the site selection criteria, nevertheless some operational incidents have to be assessed.

An incident would be a crack in the wall or break of the single container to be disposed and subsequent contact of the spent fuel or glas with the borehole fluid. Dissolution of the glas and the instant release fraction of spent fuel would contaminate the borehole fluid. The amount of released radionuclide depend on the contact time of fluid and waste and on the composition of the waste package. Measures for retrieval and repair have to be provided and comply with radiation protection.

A worst case, which should be unlikely, would be the loss of container within the subject of proctection and which can not be retrieved for any reason. A release of radionuclides takes place in the long- or shortterm and has to be assessed considering the geochemical and hydrological conditions.

The long term safety analysis has to consider that the currently proposed containers in deep borehole disposal are not corrosion resistant in the long-term with respect to the groundwater. Even if the corrosion may take place slowly some generation of hydrogen gas and other corrosion products will occur. This may increase the pressure within the sealed borehole and may impact the transportation of released radionuclides. The heat generation and high temperatures at the disposal zone will speed up chemical processes, which can not be modelled in detail due to the lack of thermodynamic data.

The total mass of fissile radionuclides in a deep borehole will be above the critical mass. Therefore a possible critical excursion and safety measures have to be assessed. Although preliminary safety analysis excluded critical excursions due to the low solubility and mobility of U(IV), an assessment and optimisation has to be done.

7 RESEARCH AND DEVELOPMENT

Boreholes with diameters of 0.75 m in 3 500 m depth are still beyond today´s standard technology but are considered feasible. Based on this a concept for disposal of radioactive waste in deep boreholes is drafted. Further development and demonstration of dedicated boreholes technology is necessary to show the feasibility.

The purpose of disposal also require research and development on the longterm behaviour of container and casing.

The operational phase for disposal of radioactive waste in deep borehole require investigations in details for its safety and radiation protection. There is a need for development testing and demonstration. This includes retrievability before closure.

The current proposal of the commission for recoverability for 500 years leads to high needs of research and development on containers and technology, if this proposal becomes a prerequisite.

8 SUMMARY AND CONCLUSION

Using deep boreholes for disposal (DBD) of high-level radioactive waste (HLRW) can have advantages in long term safety due to an ample distance between the HLRW and the biosphere and may take advantage of multiple geologic barriers as safety features. The great depth and short disposal operation impedes efficiently proliferation. Finally, aside from site selection process there may be a benefit in time for technical implementation and costs for implementation.

A basic concept for DBD of HLRW has been developed applying containment providing rock zones (CPRZ). Although further technical developments are required for HLRW disposal in deep boreholes due to larger than usual diameter and depth of boreholes, DBD seems to be feasible as an alternative option for geological disposal of radioactive waste. Further research and development with a feasibility demonstration is necessary. Operational and longterm safety analyses and assessments have to be performed.

A major challenge is the requirement for possible recovery of waste for 500 years after closure. On the other hand if disposal is intended to be a permanent and the most safe solution, a recovery might not be in the main focus of the decision when for best possible safety is strived. If there are clear advantages in long-term safety by DBD this could outweigh the disadvantage in recovery when decision-making.

The commission asked for borehole diameters of 1 m in 5 000 m based on plans for DBD in the USA /NWTRB 16/ setting the initial framework for the here developed, presented and discussed basic concept. It was assessed based on the current state of knowledge that this can not be safely operated. Reducing the depth and therefore necessary borehole and container diameters for disposal lowers the technical challenges without jeopardizing the potential safety benefits of DBD.

DBD was discussed to have advantages in safety, speed, als well as costs and that it might be considered as an alternative option in Germany. To have this option as a proven technology to decide for, it seems necessary to follow up the DBD by installation of a real scale demonstration along with detailed safety analyses. DBD could then provide a technical redundancy - if required - in case the siting or implementation of a mined repository fails or can not be pursued any longer for any reason.

9 ACKNOWLEDGEMENT

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Remediation of sites containing Uranium mining and milling waste: accomplishments and remaining research needs

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Abstract:

Between 1948 and 2001 more than 200 mines were operated in France to produce 76 000 t of uranium. Most of them were very small and produced less than 1 000 t of U. They all were remediated by the operator in accordance with current mining regulation. Tailing were disposed of over 15 specific disposal sites. A large amount of waste rocks was deposited as dump while a part was used to cover the disposal of tailings, built dams and restore the topography. The water in open-pit/underground mine and disposal was stabilized at a defined level to ensure its harnessing. The 238U and 226Ra content of water is controlled and, if necessary, lowered before release into the environment.

Gradually, responsibility of sites is being transferred from the operator to the French authorities following a regulative procedure. In this context, an important work was made some years ago by a pluralist group to assess the difficulties resulting from the historical management of the sites. The main conclusions were that remediation work carried out had contributed to manage certain risks appropriately, but didn't solve all the problems. Moreover, the question arises of the effectiveness of these measures in the medium and long term.

1 FRENCH URANIUM MINES REMEDIATION

From 1948 to 2001, more than 200 sites were operated in France by different companies for Uranium mining, milling and/or advanced prospection. They are allocated over 12 regions and 26 departments (Figure 1). About 52.10^6 tonnes of ore were extracted to produce almost 80.10^3 tons of uranium and $50.\ 10^6$ tons of tailings. In addition, 200.10^6 tons of waste rocks were removed and drop of in the vicinity of mines.

Remediation works were undertaken during the 90s and ended in 2003 [1]. The main objectives of the last operator (COGEMA presently AREVA mines) were:

- Long term stability of the remediated area in terms of safety and public health;
- Reduction of the impact as low as reasonably achievable;
- Prevention of risk resulting from intrusion;
- Reduction of total land consumption and resulting needs for institutional control;
- Favour possible industrial or leisure activities on the land and remaining buildings;

- Landscape integration, in co-operation with local stakeholders.

Figure 1: places of Uranium extraction in France [2]

Uranium ore milling factories were dismantled and decommissioned. Offices and non-radioactive sections of factories may have been used for new purposes but all equipment linked to the industrial process was considered as waste and have been stored with tailings.

Tailings were disposed of over 19 dedicated storages consisting of former open pit or natural talweg closed by a dam. Dams were also erected to enlarge storage capacities of former open pit. Waste rock has been often used as a first cover to reduce external exposure and radon emanation. Grass has been planted to stabilize cover and reduce water infiltration. In two specific cases, a few meters of water have been used instead of grass. Whatever the design of storage, impermeability was not the objective. The concept is based on collection of outflowing water without pumping and control before release into the environment. If necessary, water is chemically treated before the release.

Underground mines were backfilled with the less radioactive part of tailings and with waste rock to avoid any intrusion and reduce the risk of caving. In addition, underground works were flooded to reinforce the stability and groundwater emergences were set up for passive (without technical means) outflowing. Like in the tailing case, collected water is controlled and if necessary treated before to be released into the next waterflow.

Open pits were filled up with waste rock (several thousand or million tons of rock were moved). Others were redeveloped as water impoundment for irrigation, sport fishing, diving or other purposes. The choice between the two options (with or without water) was made on the base of previous commitment, such as ones taken to obtain the administrative authorisation of extraction. Topography, local needs of population, cost and available material, as well as technical aspects were also considered.

Remaining waste rocks piles were reshaped to reduce erosion and infiltration. The landscape Integration was enhanced by vegetation.

Most of the time, sludge produced by water treatment is managed like tailings, i.e. disposed of in specific storages.

2 FORMER AND PRESENT REGULATION

Remediation was undertaken under the control of French authorities. The mining code and corresponding order impose rules relative to radiological impact and surveillance of mines. In accordance with international radiation protection principles, it introduces the concept of added exposure defined as the difference between natural exposure in the area and exposure attributable to mining operations even after remediation.

The following limits were established:

- 5 mSv/year for external exposure
- 170 Bq of long half-life alpha emitters from the 238U decay chain in atmospheric dust and 2 mJ of potential alpha energy (PAE) of inhaled short half-life emitters daughter of 222Rn
- 3 kBq of long half-life emitters of uranate dust with a daily limit of 2,5 mg
- 7 kBq of 226Ra ingested over a year
- 2 g of U ingested over a year with a daily limit of 150mg for hexavalent forms

An indicator (TAETA) was defined as follow:

TAETA = TAET final - TAETinitial

With: $$TAET = \frac{\substack{external\\exposure}}{5mSv} + \frac{\substack{\text{238U}\\\text{in dust}}}{170\ Bq} + \frac{\substack{\text{PAE}\\\text{222Rn}}}{2mJ} + \frac{\substack{\text{PAE}\\\text{220Rn}}}{6mJ} + \frac{\substack{\text{ingested}\\226Ra}}{7\,000Bq} + \frac{\substack{ingested\\U}}{2mg} + \frac{\substack{U\ in\\uranate\ dust}}{3\,000\ Bq}$$

Radiological impact must be lower than 5 mSv corresponding to a TAETA lower than 1.

In addition, specific national rules were introduced in the 80's for tailing management. The main recommendations were:

- In situ confinement of tailings is the best option;
- Radiological impact is of course of great importance, but uranium mines are firstly mines and classical risk(s) like dam stability should not be neglected;
- Change of field or building use must be supervised by administrative registration and control;
- Effectiveness of storage design must be demonstrated over a period of 300 years for predictable normal evolution;

- Long term impact must be assessed through 5 scenarios corresponding to degraded evolution like (loss of cover, major damage on dam, house building on the storage with and without cover, construction work of a road);
- Surveillance must be built to progressively switch from active to passive state;
- Stakeholders must be informed.

At the end of mining operations, the operator declares to the administration the end of works. He describes in a specific document the way of planed remediation. The administration may impose complementary dispositions and delivers a certificate to engage remediation. When completed, remediation is controlled by the administration and a second certificate is delivered if the final situation fits with the dispositions settled in the first one. Then the operator may renounced to the concession. Finally a ministerial order is pronounced to act the transfer to the state which is presently retrieving the responsibility of the sites.

In the process of delivering the second certificate and the ministerial order, the French State has to integrate the modification of international rules relative to radiation protection. Indeed with the 96/26 Euratom directive, the limit of total annual exposure for members of the public have been reduced from 5 mSv to 1 mSv. This value has been confirmed in the revised directive published in 2013 [4]. In addition, the long term evolution of mine and tailing have to be better considered.

3 RESEARCH NEEDS

To prepare the transfer to French authorities, an important work has been engaged. Initially, mines were operated by many small companies. They have been progressively purchased by the main operator (COGEMA, presently AREVA Mines). The first item of research was to bring together all the available information. To get a complete and precise knowledge a program (MIMAUSA) was set up at the beginning of the years 2000. In collaboration with the ministry of the environment and the operator, IRSN started to undertake an inventory of main information for each site: location, type of works, volume of production, volume of waste…The MIMAUSA program is still under progress and reaches the level of radiological description. A data base computing the collected information is presently used to identify the main issues [2].

In addition, a pluralist expertise group (GEP) was established from 2004 to 2010 to produce recommendations for the long term management of uranium mines. Indeed, a long term stewardship is of a great importance and should be considered [5]. The group brought together more than 40 persons from a varying range of disciplines and background. It included representatives of IRSN, operator, administration, academic scientist and associations. Over the 15 recommendations formulated by the GEP some imply the development of research program [6].

Even if the method for radiological impact assessment has changed with the implementation of the 96/29 Euratom directive, we still need to determine the exposure in addition to the natural background. Unfortunately, chemical and radiological characterisation was rare previously to mining operations. No reference state is available. The present methodology of background determination consists in measurements upstream and downstream the mines. The difference is supposed to correspond to the mine impact. This method becomes very uncomfortable when the difference is in the range of natural background variability. Then we need to develop a new methodology to quantify industrial inputs.

Water is one of the most important items of post mining management whatever the ore. As already mentioned, water is used to stabilise underground works and some open pits were flooded for local purposes. Finally, water flowing out the mine is released into the environment. It is an important pathway of pollutants including radionuclides from the 238U family as well as chemicals used for water treatment (most of the time, Ba, Cl, Fe).

- The first question related to mine's water quality is the effectiveness of chemical treatment in terms of radioactive pollution reduction. Of course we also have to consider chemical pollution and waste production from the treatment. The general idea is to progressively switch from active to passive solution [7]. The concentration of radionuclides in mine's water is decreasing over time and the effectiveness of available passive treatment must be improved [11].

- To establish the long term strategy, we need to assess the evolution of mine's water quality. How long the treatment will be necessary? To answer this question, we need a good knowledge of water/rock interactions for each encountered situations (waste rock piles, tailings, underground works, open pit). We also need hydrological and chemical models to predict reactive transport. On this base we will be able to assess the impact of geochemical changes over the long term like climatic change. German studies highlighted the difficulties to predict flooding time [8] and [9]. This illustrates the difficulties to get a robust hydrological model, even if we don't consider chemical reactions. If modelling the water flow is hard, modelling the water quality is very ambitious and requires more studies.

- We also have to integrate that water is a resource. From place to place, mine water is used for agriculture purpose (irrigation of crops, animals drink). Due to the increasing pressure on water resources, multiplication or different uses of mine water have to be considered for environmental and human health impact.

Radionuclides and chemicals released into the environment are transported in the watershed. Several cases of accumulation in river or lake sediments were reported in the literature [10], [11], [12], [13]. These open three main questions:

- Does the chemical speciation of radionuclides in mine's effluent (with or without treatment) favour their retention in the watershed? Chemical characteristics of underground and shallow waters are quite different especially in terms of O2, organic matter contents which may influence the behaviour of U and Ra. In addition, chemical treatments applied aim to increase the proportion of radionuclide associated with particles. If decantation recovery is incomplete, a portion of radionuclides catch by particles may escape the water treatment station and be trapped into river or lake sediments.

- Are radionuclides durably trapped into the sediment? Once radionuclide is in the sediment, early diagenesis conditions favour the reduction of U(VI) species into low solubility U(IV) species [14]. Determining the stability of U(IV) species present in the sediment is of a great importance to predict the behaviour and fate of uranium in lacustrine environment [15]. To study the early diagenesis process we need to assess the sedimentation rate to distinguish historical record from diagenetic redistribution in sediment cores. New methodologies of sedimentation rate estimation have been developed to face the impossibility to use the classical method based on 210Pb in excess [16].

- Should the contaminated sediment be removed and what are the consequences? Locally, French authorities imposed the removal of contaminated sediments and their storage on tailing storage site. These operations imply a reoxydation of sediment which may favour the remobilisation of radionuclides.

Several dams were erected to maintain tailings. The recent accident in Hungary on red sludge storage [17] remind us of the importance to avoid the risk of a dam collapse. During a remediation administrative process, the resilience of dam has to be demonstrated for a period of 300 years. 226Ra is the main long lived radionuclide in tailings. Considering its half-life (1 600 years), the period of 300 years is too short to drastically reduce the radioactive content of tailings. Then we have to ensure the resilience of dam over periods of several thousand years. In addition, a significant attenuation of aqueous 226Ra activity was observed through a dam [18] highlighting the process of Ra accumulation in waste rock which could be considered as a passive treatment on the long term.

Finally, addressing the impact of radon is essential. It is presently the main source of exposure to natural radioactivity due to geologic sources [19]. Harming surrounding rock, underground works may favour the radon emanation and migration toward the surface and contributes to increase the background exposure. In addition, tailing storage represents a major source of radon. Present administrative dispositions aims to avoid the risk of intrusion and building on storage but we have to consider the possibility. To assess the corresponding risk we need to develop models of radon migration in soil and house. This work is already useful in the context of radon from geological origin policy establishment.

To face these questions, a new team has been established in the department dedicated to waste and geosphere interaction studies.

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Safe Management of Emergency-related Radioactive Waste in the Chornobyl Exclusion Zone

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Abstract:

Very large amounts of radioactive waste (RW) are temporarily stored/localised in Ukraine, mainly in places of their generation. In particular, these include emergency RW, originated as a result of the Chornobyl accident. In order to determine the optimized approaches to the management of emergency RW, a methodological guide “Guideline for safety assessment of temporary localization emergency radioactive waste sites in Chornobyl Exclusion Zone” was developed in the framework of INSC (Instrument for Nuclear Safety Cooperation) project U3.01/10 (UK/TS/46), supporting the State Nuclear Regulatory Inspectorate of Ukraine. Via the INSC instrument, the European Union (EU) supports the promotion of a high level of nuclear safety, radiation protection, and the application of efficient and effective safeguards of nuclear material in third countries. The Guideline establishes the recommendations to apply the requirements of the Ukrainian regulatory documents in force, and also provisions of documents of international organizations, taking into account creation of radioactive waste temporary localization sites (RWTLS) under accident conditions and location of RWTLS in the Exclusion Zone.

1 INTRODUCTION

During the acute phase of the accident at Chornobyl NPP (ChNPP) Unit 4, which occurred 30 years ago, emergency radioactive waste (RW) was collected in the area around the NPP for its further localization. The major part of high- and intermediate- level RW is located in buildings with engineered barriers, called radioactive waste disposal sites (RWDS); intermediate- and low- level RW is localized in the trenches and clamps – called radioactive waste temporary localization sites (RWTLS). Around ChNPP, there are 9 RWTLS: "Stanciya Yaniv" (Yaniv Station), "Naftobaza" (Oil Storage), "Pischane Plato" (Sand Plateau), "Rudyy Lis" (Red Forest), "Stara Budbaza" (Old Construction Base), "Nova Budbaza" (New Construction Base), "Prypyat'", "Kopachi", "Chystogalivka" with total area of about 12 km^2, with trenches and clamps where RW are located. Estimated number of trenches and clamps in RWTLS is about 1000 pieces, and total volume of RW in RWTLS is estimated as 10^6 m^3 [5]. To date, for some RWTLS, there remains unknown location of trenches and clamps, as well as characteristics of RW, located in these trenches/clamps, remain unknown. Nowadays, research efforts are carried out and control is ensured over the condition of RTWLS. In addition, limited number of measures are being implemented to support and improve their security.

RTWLS were created without safety analysis, forecast assessment and impact on the environment and population in long-term perspective. Strategy for management of mentioned RTWLS should be based on their long-term safety assessment taking into account their location in the Exclusion Zone where there is no population and gradual zone size reduction. Based on results of these assessments, needs for removal and re-disposal of RW are determined, and the possibility of keeping RWTLS in their current locations are justified with ensuring adequate institutional control, etc.

It is suggested to consider these objects as a practice in the past (situation of the existing exposure) to make justified decisions. Based on the results of the safety re-assessment for each RWTLS, there should be made decisions on terms, sequence of RW removal or inexpediency of RW removal, and considered corrective measures to improve safety (e.g. increase of protective properties of the upper cover, increased scope of control, etc.).

In the framework of INSC project U3.01/10 (UK/TS/46) supporting the Ukrainian regulatory body (SNRIU) in regulation of safe radioactive waste management, financed by the EU, a “Guideline for safety assessment of temporary localization emergency radioactive waste sites in Chornobyl Exclusion Zone” was developed by RISKAUDIT IRSN/GRS International and the State Scientific and Technical Center of Nuclear and Radiation Safety experts. The Guideline defines the methodological recommendations for implementation of iterative safety assessment of RWTLS in the trenches/clamps and determines measures for their rehabilitation. It is necessary to take into account peculiarities associated with the Exclusion Zone as a barrier function aimed at access restriction.

The Exclusion Zone and zone of unconditional (obligatory) resettlement (EZ) is a part of contaminated area that suffered of the most intense radioactive contamination by long-lived radionuclides. Within EZ, as defined in 1991 (about 2600 sq. Km), the calculated effective equivalent exposure dose for population may exceed 5.0 mSv per year. In this regard, the legislation of Ukraine prohibited free public access to the territory of the EZ, and EZ was withdrawn from the fund for land resources of Ukraine, as unsuitable for living and economic activity.

The existence of the EZ including its land and water resources and support measures of administrative control provides the barrier function on the way of distribution radioactive contamination outside. It is supposed that in the future the area of the EZ will gradually decrease according to the decisions to be taken in the future based on the research of radiation situation and forecast assessment dose for the population at the closest and distant future.

The paper presents a recommended algorithm for carrying out the safety assessment of RTWLS in the trenches/clamps. This algorithm includes recommendations for the first step based on the conservative approach: collecting data and carrying out research activities, modeling of RWTLS in the trenches/clamps behavior for different scenarios, comparison of the results with radiological criteria. In case non-compliance of assessment results with radiation criteria is identified, additional research efforts and assessment under reduced conservatism should be performed and measures on rehabilitation and assessment of their sufficiency should be developed based on ALARA principle.

2 OBJECTIVE AND GENERAL CONTENT OF THE RWTLS SAFETY ASSESSMENT

RWTLS safety assessment is carried out to: determine whether, at current conditions of RWTLS, it is possible to achieve adequate level of safety according to existing Ukrainian regulatory documents and IAEA documents related to situations of existing exposure situation (taking into account that RWTLS were created under accident conditions and are located in the Exclusion Zone) and determine, based on ALARA principle, the need for actions on maintenance and increase of RWTLS safety level without RW removal or with their partial removal and redisposal.

Safety assessment does not include systematic detailed assessment of the conditions of each individual trench/clamp, nor issues of safety assurance for routine activities of the operator at the territory of RWTLS and during implementation of measures on maintenance and increase of RWTLS safety level. These issues have to be considered in the framework of routine operation procedures, as well as safety justifications of appropriate designs for activities at specific RWTLS.

Taking into account uncertainties, related to incompleteness of study of RWTLS sites, RW characteristics, as well as insufficient knowledge about possible evolution and extreme change of conditions in the future, iterative approach is used for assessment.

At the first stage of assessment, for prediction and RWTLS behavior and carrying out appropriate calculations, assumptions and values of RWTLS and RW characteristics are used that obviously lead to envelope (penalizing) radiological impacts. If conservative assessments of radiological impacts are unacceptable, on the second stage further reduction of the conservative assessment is made as a result of the gradual increasing of the level of detail of research and realistic simulation. At the third stage, rehabilitation measures to ensure the proper level of RWTLS safety shall be determined.

Sequence of assessments provided in Diagram 1.

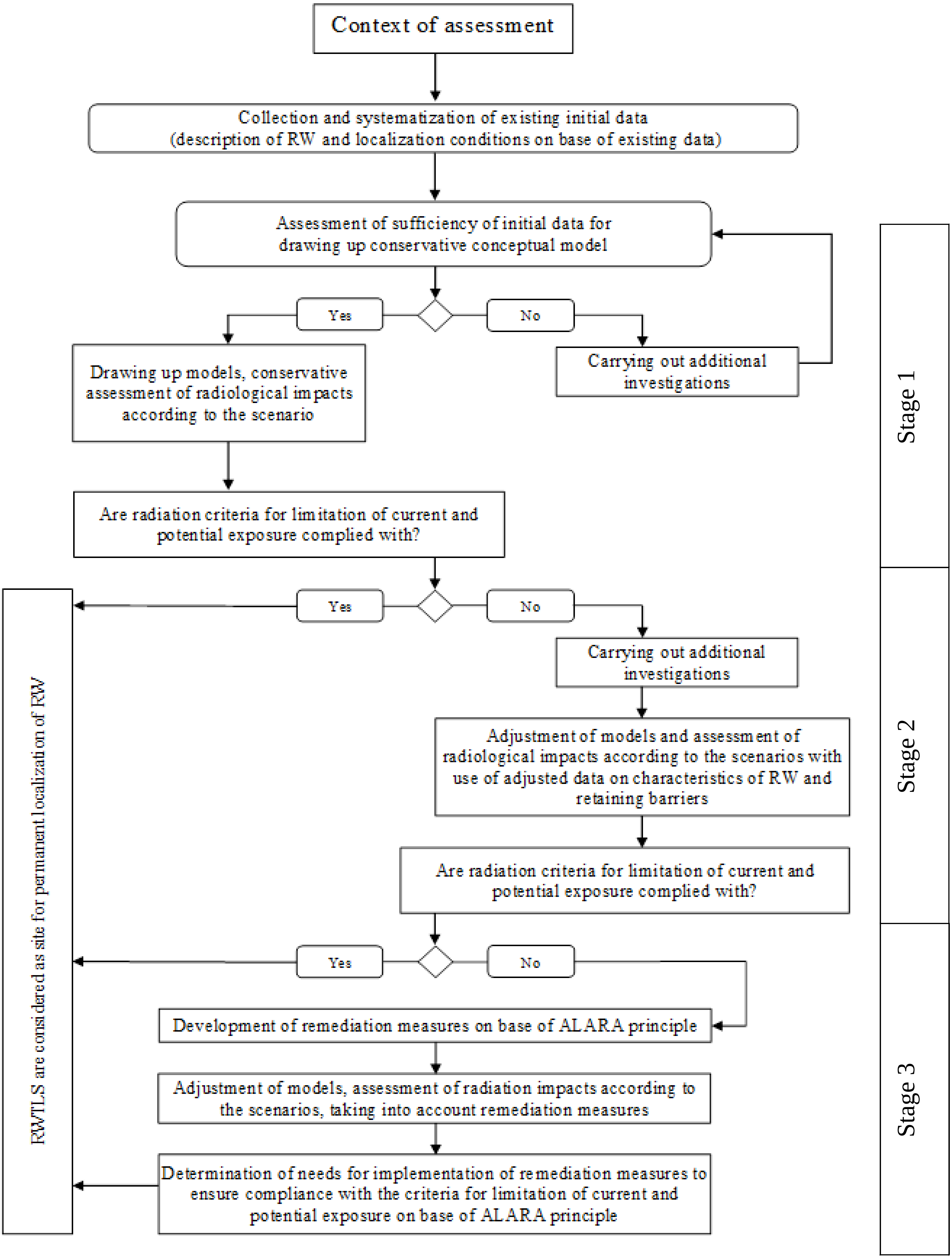

Diagram 1 – Sequence of safety assessment

Assessment of radiological impacts of RWTLS is carried out for the time interval, for which potential hazard of RW, located at RWTLS, is existing. Taking into account the presence of long-living transuranium radionuclides in RW of Chornobyl origin, assessment is carried out for long-term period, for which forecast with acceptable level of confidence may be made (maximum time of 1 million years, if the maximum radiological impacts on critical groups can not be detected).

It is supposed that activity related to RWTLS is coordinated with the activity related to the disposal of RW on the existing sites in EZ (Vector site, Buryakivka disposal facility). Duration of existence of Vector site is described in the Guideline for the Assessment of the Radiological Impact of the "Vector" Site with Multiple Facilities for Radioactive Waste Processing, Storage, and Disposal [3] (based on strategy and program documents for RW management in Ukraine). RW disposal facilities at Vector site must reach the compliance with the conditions for conditional release from regulatory control in up to 500 years (300 years after closure of the last disposal facility). In this period, RW disposal facilities at RWTLS also must reach the compliance with the conditions for conditional release from regulatory control (for those RW that are not removed).

After that, reduced EZ still exists. It performs function of restriction of access, in particular, to Vector site, Buryakivka disposal facility, RTWLS. In the period of reduced EZ existence, no population is residing there, and only restricted activities are carried out. However, temporary unintentional presence of humans in reduced EZ cannot be excluded.

In the remote period of time, reduced EZ may terminate its existence. After that, it is assumed that there are no limitations on residence of population and carrying out activities directly on RWTLS site.

3 SCOPE OF CONSERVATIVE ANALYSIS OF RADIOLOGICAL IMPACTS OF EACH RWTLS

3.1 Conservative assessment at stage 1

Conservative assessment of radiological impacts of RWTLS includes:

- systematization of data on the EZ, required for assessment;
- identification of trenches/clamps at RWTLS sites based on systematization and analysis of existing data, determination of sufficiency of initial data and carrying out investigations;
- development of conservative model, scenarios and carrying out conservative analysis of radiological impacts on critical groups of population and staff of nearby facilities;
- comparison of the results of calculations with criteria of admissible radiological impacts;
- determination of main factors that contribute to radiological impacts and ranking of RWTLS and trenches/clamps by the degree of their potential hazard.

During safety assessment of RWTLS, there are determined characteristics of the EZ. For safety analysis of RWTLS, that may be carried out conservatively, limited initial data about EZ characteristics may be required. In fact, more detailed data are determined for 10-km area around Chornobyl NPP, whereas less detailed – for the rest of 30-km area of the EZ.

At stage 1 of safety analysis of RWTLS a collection, systematization and analysis of available information on RWTLS is carried out, at that based on conservative approach, there are determined:

- number and arrangement of trenches/clamps of each RWTLS;
- geometrical parameters of trenches/clamps;

- RW inventory and characteristics in trenches/clamps;
- presence of water in trenches and its contamination, spreading of contamination from trenches/clamps;
- characteristics and condition of caps of trenches/clamps;
- hydrogeological conditions of RWTLS;
- characteristics of contamination of the area of RWTLS location.

For conservative analysis of the radiological impact of RWTLS, it is allowed to divide of each RWTLS area into sections so that within the limits of specific "uniform" sections there are no significant changes of characteristics as regards the above-mentioned issues. In case of such division of RWTLS into sections, it is allowed to characterize them by generic (with certain conservatism) values of parameters. In particular, it is allowed to use for particular "uniform" section generic (conservative) accumulative values of RW amount, area of trenches/clamps with RW, etc.

At stage 1 it is assessed cumulative radiological impact of each RWTLS on humans; at that, there are defined a number of "uniform" sections of RWTLS with the biggest contribution in total radiological impact (one or several sections per RWTLS). It is not necessary to carry out detail assessments for each specific trenches/clamps.

Safety analysis of RWTLS is carried out by determination of scenarios of evolution of conditions of RW localization, release and spreading of radioactivity from trenches/clamps and outside the borders of RWTLS and development of appropriate conceptual and mathematical models.

Evolution scenarios are developed under the assumption that measures on maintenance and improvement of RWTLS safety level are not implemented, and evolution of RW local conditions takes place in a natural way.

RWTLS safety assessment should be based on prediction calculations of impacts as a results of both gradual leaching of radionuclides and their migration to the location accessible for humans and occurrence of events that violate retention properties of cap of trenches/clamps or enhance release and transport of radionuclides outside the places of their localization (e.g. extreme natural events, potential activities of population at the territory of RWTLS in the distant future, etc.).

When determining routes of radionuclide transfer and locations of critical groups of population, there are taken into account gradual changes of EZ with time as regards reduction of EZ (reduced EZ), as well as easing of restrictions on prevention of access and types of activities at the territory of reduced EZ, including termination of its existence or loss of its memory.

Following potential routes for spreading of contamination are taken into account:

- with groundwater;
- with surface water;
- through atmospheric air during transfer of gases, aerosols, dust, parts of vegetation from the surface of RWTLS;
- propagation of contamination to biomass.

Condition of cap of trenches/clamps will gradually deteriorate (degrade), in particular, due to wind erosion. Therefore, assessments of radiological impacts of RWTLS are carried out taking into account gradual changes of local conditions of RWTLS sites.

Also, the spreading of contamination due to fire on RWTLS sites is considered taking into account contamination of vegetation and wind transport.

Due to the presence of long-living radionuclides in RW, located in RWTLS, the long-term assessment of routes for spreading of radionuclides with groundwater, should take into account:

- every potential groundwater surface outlet identified using the hydrogeological conceptual model;
- presence of two hydraulically connected aquifers (first and second) and possibility of existence of different discharge points;
- distance and time of movement of groundwater through geosphere rocks with different parameters (in particular, dispersion coefficients);
- uncertainties (e.g. presence of perched water) in initial data used for development of local/regional hydrogeological model.

When considering future existence of reduced EZ that includes sufficient administrative measures on prohibition of permanent (long-term) residence/presence of people at the territory of RWTLS but not includes sufficient measures to prevent human intrusion to this territory, it is taken into account that unauthorized stay of people on RWTLS site can take place in future.

In the period when only reduced EZ is the element of passive control, and there is no direct administrative control of RWTLS sites (i.e. after limited release of RWTLS sites from regulatory control in 300 – 500 years), there are considered scenarios of temporary intrusion of people to RWTLS sites. One of these may include presence of tourists on RWTLS site for a certain time (until the tourists are detected by the control bodies) with maximum use of natural resources (bonfires use of water from the site, collection of berries, mushroom, etc.).

For the remote period after termination of existence of reduced EZ or after loss of memory of the site, there are considered permanent residence on site of RWLS and possible types of human activities (road construction, civil construction, water use, etc.).

Staff of nearby facilities: staff that works at the facilities within the borders of reduced EZ that do not belong to RWTLS sites. As regards RWTLS sites, such staff is considered as category B staff [7] that is not directly involved in activities with ionizing radiation sources (in this case, on site of any RWTLS); however, due to location of its workplaces at industrial sites of other facilities in reduced EZ with use of radiation and nuclear technologies, it may be subjected to additional exposure.

3.2 Comparison of the results of calculations with the criteria for admissible radiological impacts

Radiation and hygiene regulations, established in regulatory documents of Ukraine and in IAEA documents [7-8; 11], are applied as criteria of admissible radiological impacts of RWTLS.

Results of calculations of predicted doses of normal exposure of population and staff of nearby facilities according to the NES are compared with radiation and hygiene regulatory values for limitation of normal exposure, stated in Table 1.

RWTLS safety level as regards normal radiological impacts is considered as sufficient if conservative assessments of normal exposure doses do not exceed regulatory values stated in Table 1. For population that may stay at the territory of RWTLS after termination of existence of reduced EZ, the regulatory value of 1 mSv/year must not be exceeded.

If these conditions are not met, the improved safety assessment of RWTLS shall be made (Stage 2).

Table 1. Radiation and hygiene regulations for limitation of normal exposure due to radiation impact from all RWTLS

Period	Radiation and hygiene regulations, individual annual effective exposure dose	
	Critical group of population	Staff of adjacent facilities
Existence of REZ	0,3 mSv – total annual dose from all RWTLS outside REZ (item 2.15, SSR-5 [11])[1)]	2 mSv/year (Table 5.1 of NRBU-97 [7] for category B staff)
After termination of REZ existence or loss of memory of the site	0,3 mSv – total annual dose from all RWTLS outside REZ (item 2.15, SSR-5 [11])[1)] 1 – 20 mSv/year – total annual dose at RWTLS territory from all RWTLS and other sources of exposure (item 2.15, SSR-5 [11])[2)]	

1) It is recommended to use these limits or demonstrate that dose limit of 1 mSv/year (Table 5.1 of NRBU-97) is not exceeded taking into account total radiation impact of all radiation and nuclear objects (RWTLS, Vector site, Buryakivka, etc.) for the members of respective critical group of population.

2) Taking into account of existing exposure situation, acceptable value of limit of potential exposure from the range of 1-20 mSv/year is determined and justified according to the results of safety assessment based on ALARA principle.

Results of calculations of predicted doses of potential exposure of population and staff of nearby facilities according to alternative scenarios are compared with radiation and hygiene regulatory values for limitation of potential exposure, stated in Table 2.

Taking into account that probability of critical event, caused by extreme wind, is higher than 2·10-5/year, estimated doses of potential exposure of population and staff of nearby facilities in this case must not exceed 50 mSv. Potential exposure dose due to critical event in case of F3.0 class tornado (with probability of 10-6/year) may exceed 50 mSv, but lethal exposure doses must not occur (see Table 2).

RWTLS safety level as regards potential radiological impacts is considered as sufficient if conservative assessments of potential exposure doses do not exceed regulatory values stated in Table 2. For population that may stay at the territory of RWTLS after termination of existence of reduced EZ, regulatory value of 1 mSv/year must not be exceeded.

If these conditions are not met, the improved safety assessment of RWTLS shall be made (Stage 2).

Table 2. Radiation and hygiene regulations for limitation of potential exposure due to radiation impact from all RWTLS

Period	Radiation and hygiene regulations (NRBU-97/D-2000 [8]), D – annual effective dose of potential exposure P – probability of critical event
	Critical group of population and staff of adjacent objects

Before conditional release from the regulatory control	$D \leq 50$ mSv, $P \leq 1 \times 10^{-2}$/year $D > 50$ mSv[1)], $P \leq 2 \times 10^{-5}$/year
After conditional release from the regulatory control	$D \leq 1 - 50$ mSv[2)], $P \leq 1 \times 10^{-2}$/year

1) Probability of events, as a result of which, within the short period of time, lethal exposure doses may occur, must not exceed 5×10^{-7}/year.
2) Taking into account situation of existing exposure, acceptable value of limit of potential exposure from the range of 1-50 mSv/year is determined and justified according to the results of safety assessment based on ALARA principle. At that, during reduced EZ existence or during a reasonable time of site memory preservation, dose of potential exposure of population outside reduced EZ must not exceed 1 mSv/year.

3.3 Ranking of RWTLS and sections by the degree of their potential hazard

For each RWTLS site, for the most hazardous scenarios of normal and altered evolution that call forth the main radiological impacts on the critical groups of population and on staff of nearby facilities, the contributions of "uniform" sections to the radiological impacts are determined. The sections that give the highest radiological impacts (or trenches/clamps for unintentional human intrusion scenarios) are selected.

Taking into account the results of ranking of "uniform" sections of RWTLS by the extent of radiological impact, for the most hazardous sections, there are determined routes of spreading of radionuclides that give the main contribution to radiological impacts.

During ranking, there are separately determined those scenarios, "uniform" sections, routes of spreading of radionuclides that lead to exceeding of radiation and hygiene regulatory values for normal and/or potential exposure of population and staff of nearby facilities.

3.4 Enhanced assessment of radiological impacts of RWTLS at stage 2

If the conservative calculations of radiological impacts at stage 1 show that radiological criteria of admissible impacts are exceeded, then calculations based on a more realistic approach are made in stage 2. To do this, for scenarios and "uniform" sections, which lead to exceeding of radiological criteria of admissible impacts and which were determined at stage 1, there are used data and models that, as far as possible, adequately reflect RWTLS and environment, taking into account existing uncertainties.

Enhanced assessment of radiological impacts of RWTLS includes:

– determination of need for additional data and carrying out additional studies of the most hazardous sections of the RWTLS;

– upgrade of models and carrying out more realistic assessments of radiological impacts of RWTLS;

– comparison of results of enhanced calculations with criteria of admissible radiological impacts;

– additional ranking of RWTLS and trenches/clamps by the degree of their radiological hazard.

There are determined needs for additional data and carrying out additional studies for those "uniform" RWTLS sections or individual trenches/clamps that, by conservative assessments at stage 1, lead to the highest radiological impacts according to the determined scenarios and routes of spreading of radionuclides. In this case there should be used fulfilled ranking of the certain “homogenous” sections of RWTLS. As that use made ranking separate “homogenous” sections RWTLS.

Based on upgraded models and/or scenarios, there is carried out adjustment of calculations of radiological impacts of RWTLS. It is allowed to carry out calculations only for individual RWTLS, "uniform" sections of RWTLS and scenarios that, according to the results of conservative assessment at stage 1, lead to the highest radiological impacts.

Comparison of the results of adjusted calculations with the criteria for admissible radiological impacts and assessment of sufficiency of RWTLS safety level are carried out, as on the stage 1.

3.5 Definition of remediation measures for ensuring appropriate safety level of RWTLS at stage 3

RWTLS safety assessment at stage 3 includes:

- development of potentially possible measures for RWTLS remediation;
- adjustment of models, scenarios and carrying out assessments of radiological impacts, taking into account possible RWTLS remediation measures;
- comparison of calculation results with the criteria for admissible radiological impacts and determination of scope of remediation measures based on ALARA principle.

RWTLS remediation measures are developed to reduce radiological impacts of RWTLS.

If it is impossible to reduce doses of normal exposure to 1 mSv/year for population that may stay at the territory of RWTLS after termination of reduced EZ existence, there is defined and justified acceptable value of permissible dose in the range of 1-20 mSv/year based on ALARA principle.

If it is impossible to reduce doses of potential exposure to 1 mSv/year after conditional release of RWTLS from regulatory control, there is defined and justified acceptable value of permissible dose in the range of 1-50 mSv/year (during reduced EZ existence, potential exposure dose for population must not exceed 1 mSv/year beyond reduced EZ) based on ALARA principle.

Remediation measures are determined, first of all, for those "uniform" sections of RWTLS that cause the highest radiological impacts.

Remediation measures are determined also for individual trenches/clamps:

- located near water reservoirs, ravines, etc.;
- those in direct vicinity of which there is planned activities that may have a negative impact on condition of RW localization in trenches/clamps;
- where significant damage of cap integrity was detected;
- on which or near which there were detected accumulation of water during strong rainfalls and thawing;
- which contain explosion hazardous and self-igniting materials.

Following possible remediation measures may be considered:

- full or partial removal of RW from individual trenches/clamps;
- improvement of isolation properties of the cap;
- water drainage from the relief to prevent flooding of trenches/clamps;
- carrying out decontamination near the trenches/clamps, in particular, removal of the most contaminated sections of soils;
- organization of barriers on main routes of spreading of radionuclides to the water objects;

- change of flora/fauna vital activities to reduce release of radionuclides and their accumulation in the vegetation, as well as reduction of risk and consequences of fire;
- other upcoming technologies, if applicable.

Adjusted assessments of radiological impacts are carried out taking into account the final state of the section in case of implementation of respective remediation measures.

Based on the results of assessments, it is determined how much each remediation measure reduces radiological impacts (doses of normal and potential exposure of population and staff of nearby facilities). To make decisions on implementation of specific remediation measure at RWTLS, the ALARA optimization principle is applied.

In case of waste removal the option for removed RW management and provisions for prevention of removed water accumulation in the long time scale should be envisaged. The assessment of feasibility of new facilities for disposing the removed RW should also be addressed.

The radiological hazards associated to the possible removal actions should be defined in view of evaluating dose optimization.

4 CONCLUSIONS

In the framework of INSC project U3.01/10 (UK/TS/46) of the EU, supporting the State Nuclear Regulatory Inspectorate of Ukraine (SNRIU) in regulation on safe radioactive waste management, a “Guideline for safety assessment of temporary localization of emergency radioactive waste sites in Chornobyl exclusion zone” was developed by RISKAUDIT IRSN/GRS International and the State Scientific and Technical Center of Nuclear and Radiation Safety experts.

This Guide presents methodological recommendations for implementation of iterative safety assessment of RWTLS in the trenches/clamps and to determine measures for their rehabilitation. An algorithm for carrying out the safety assessment of RWTLS in the trenches/clamps is recommended taking into account peculiarities associated with the Exclusion Zone as a barrier function aimed at access restriction.

Criteria for admissible radiological impacts are proposed for a different time frame of RWTLS existence taking into account existing exposure situation in the EZ. In case a non-compliance of assessment results with radiation criteria is identified, additional research efforts and assessment under reduced conservatism should be performed and measures on rehabilitation and assessment of their sufficiency should be developed based on ALARA principle.

Acknowledgements

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Methodology for the optimization of waste management resulting from the clean-up of contaminated sites

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Abstract:

The primary objective defined for the management of sites potentially contaminated by radioactive substances is the complete remediation of the site and implies to remove the maximum amount of pollution, so that no complementary remediation will be needed. However, depending on the situation and difficulties that may be encountered by the complete removal of the contamination, several remediation options can be chosen depending in particular on the existence of an exposure or not (caused by existing or planed uses). The choice of the remediation options is based on the assessment of the compatibility of the extent and level of contamination of the site with the existing or planned uses on it.

The decision to remediate a site and the choice of the remediation option primarily depends on the analysis of the results of a diagnostic that must be supported by a clear and precise methodology. The aim is to characterize the source, the extent of pollution or the presence of radiometric anomalies and the way of transfer of pollutants in the light of current or possible uses in the future. The level of detail of the diagnosis must be commensurate with the issues associated with each site. In any case, regardless of the remediation option selected, the assessment of the benefits (in terms of the global impact of the site, particularly on the long term) and disadvantages (in terms of risk of exposure or nuisances induced during remediation works) of each option should be assessed prior to the choice of the option. In addition, the remediation options should be discussed with all the stakeholders as far as necessary for sharing their validity with civil society.

1 INTRODUCTION

France has a very large nuclear industry covering all nuclear fuel cycle activities and generating substantial amounts of radioactive waste. Considerable effort has been expended in remediating contaminated sites and the former mining and processing sites, and work continues on a number of legacy sites contaminated by various industrial activities involving radioactive material. In addition, numbers of nuclear facilities will be dismantled and remediated in the future. These different activities are controlled and regulated by legal instruments in France, and methodological documents have been developed for the remediation operations. France has also a robust framework for the management of radioactive waste, as described in the National Plan for Management of Radioactive Materials and Waste and has a dedicated disposal facility since 2003 for very low level waste (VLLW). Nevertheless, according to the data of the national inventory 2015, the volume of VLLW that might result from the dismantling of all existing nuclear facilities (excluding contaminated soils), would be at least 3 times the capacity of this disposal facility.

In a context where the opening of a new disposal facility could raise major societal concern while paths for the management of VLLW will come to saturation in a few years time, it appears necessary to investigate other methodologies in order to optimize the production of waste from the clean-up operations at these sites.The paper summarizes the main methodological approaches that are applied or may be envisaged to tackle remediation issues.

2 ASSESSMENT CRITERIA FOR REMEDIATION OPTIONS

Several methodological documents have been developed for the management and remediation of contaminated sites, mining sites or nuclear sites. Even if the requirements in terms of remediation objectives may vary to some extent for these activities, the approaches for evaluating the radiological and chemical impact and site monitoring plans are quite similar and refer to the guide for the management of sites and soils potentially contaminated by radioactive substances [1].

Whatever the situation and the activity concerned, the remediation objective is the withdrawal of the totality of the source of pollution, wherever this is achievable, regardless of the existence or not of a plan for the reuse of the site. This remediation option is thus the reference option aimed at avoiding having to proceed to additional depollutions at a later date. Two possible situations are distinguished, according to the existence of an exposure on sites (eg : presence of dwellings, in that case the choice of remediation actions can be limited) or not (any remediation actions can be implemented). In both cases, when a remediation plan is decided, it is defined according to the requirements of the guide for the management of sites and soils potentially contaminated by radioactive substances [1], so as to eliminate or reduce the added exposure. When the total withdrawal of the pollution is not possible, several alternative remediation options are then proposed and evaluated on the basis of a cost benefit analysis. This cost benefit assessment for the alternative options must, in particular take into account the risks (radiological, chemical or conventional) and the techno-economic constraints (such as the saturation of the pathways for waste management, the cost of operations, transport, …). Whatever option is chosen, the residual radiological impact of the site must be assessed in order to verify that the exposures that may arise are sufficiently low to allow the reuse of the site and to decide whether any reuse can be accepted or if restrictions other than memory keeping are to be set.

Regarding the radiological criteria for remediation, these may differ depending on the activities (nuclear or contaminated site for example) and the exposure situation arising from the use of the site (planned or existing). Even if the overall approach and remediation objective is quite similar for both activities (contaminated or nuclear sites), the guide for the remediation of a nuclear site [2] introduces a precision in the radiological objective to be attained. This difference is explained below.

- For nuclear sites the French Nuclear Safety authority has considered to date that the total withdrawal of the pollution was to be implemented. However, a recent evolution of this doctrine has been proposed [2] . Total whithdrawal remains the primary goal but where major difficulties for achieving the total removal of pollution are identified, the principle realising an “in depth remediation” of the sites has been intoduced. This principle consists in achieving sufficient clean up so as to make possible all plausible future uses of the site but the guide [2] doesn’t precise the methodologies and criteria that are necessary to apply in this goal,
- For contaminated sites, when the total withdrawal of pollution is not possible (in particular when the site is already subject to reuse or when remediation induces clearly unafordable costs), a residual impact remaining below 1 mSv is considered to be the goal to achieve. Where the site is free of use, which enlarges the choice of remediation options that may be envisaged, more stringent requirements may be encompassed. A reflexion is ongoing to examine whether the dose constraint of 0,3 mSv widely recommended by international standards may be generally applied.

Discussions are currently under way to merge these two approaches in order to ensure consistency of treatment of different cases and to be sure that the remediation objectives (dose constraints for example or “in depth remediation”) can be matched. These discussions aim at promoting a graded approach to the cleanup of radiological contamination (thorough level of decontamination: total withdrawal of pollution, in depth remediation, control of uses), and taking into account the specificity of cases which may be encountered.

In this goal, it is necessary to supplement the existing methodological approach [1], [2] by more precise methodological elements.

3 PROPOSAL FOR THE IMPLEMENTATION OF THESE NEW METHODOLOGICAL APPROACHES

The definition of new methodologies and criteria are necessary to ensure that the various sites are treated in a consistent manner and to avoid a case-by-case judgment (basis by the operator and the authority). In this perspective, the methodological work mentioned above must be engaged on the following elements :

- The methodology of characterization of the radiological background level (state of reference) of a site in the absence of such study prior to the start of operation of the activity concerned. On this point, the remediation guide for a nuclear site [2] refers to a comparison of the characterization of the site with the characterization of the surrounding soil presenting similar geological and geochemical characteristics. IRSN believes that this approach may also be considered for the treatment of mining sites considering that these sites are located in areas where the natural radiological background level (in terms of uranium concentration of the soil) is usually higher than the average in the national territory, a remediation. IRSN believes that this alternative approach for the remediation sites (whatever the activity concerned) deserves to be studied;
- Characterization methods of pollution to be applied to different categories of sites in order to ensure a reliable estimate of this activity. In order to have the best estimation of the extent of the pollution, it is essential to use the best available techniques in terms of sampling and measurement and to reduce as far as possible the uncertainties associated with the estimation of the volumes of contamination;
- Elements of methodology to make the demonstration, with a substantial margin of safety, that the residual impact of sites after remediation is sufficiently low to be considered compatible, in terms of radiation protection, with all plausible use of the site. It aims more particularly to define the key elements (characterization data, choice of exposure scenarios and associated parameters, choice of assumptions reasonably penalising...) in order to ensure the robustness of the assessment of the residual impact of the remediated sites. This is an essential element to assess whether an additional remediation effort would be justified.

On the basis of experience feedback of remediation of sites, in order to ensure that the various sites are treated in a consistent manner, it would be appropriate to consider the benefits of define a reference of radiological and chemical exposure value, as a minimum remediation target to achieve and from which the available margins (in terms of remediation) would be appreciated. This methodological work could be done in a multidisciplinary working group including all the stakeholders, particularly the civil society, considering the importance of associated societal challenges regarding the problem of radiological waste management.

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Seminar 4

Radiation Protection, Environment and Emergency Preparedness

HERCA-WENRA approach for better cross-border coordination of protective actions during the early phase of a nuclear accident

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Abstract:

The Heads of European Radiological protection Competent Authorities HERCA and the Western European Nuclear Regulators Association WENRA are voluntary association of the Radiation Protection and Nuclear Safety Authorities in Europe. They work together on significant issues of common interest with the aim to propose harmonization and/or practical solutions for improvements.

Concerning emergency preparedness and response (EP&R), HERCA and WENRA have developed the HERCA WENRA Approach (HWA). It consists of a general mechanisms for coordination in Europe of the protective actions that are independent of the accident scenario. It further includes a simplified scheme for coordination in the highly unlikely event of a severe accident in a nuclear power plant, requiring rapid decisions for protective actions while very little is known about the situation. A minimum common level of preparation for protective actions in Europe has also been defined.

The HWA contains overarching principles and provides an incentive for joint actions between neighbouring countries. The regulators have initiated cooperation at national and international level with the competent authorities in charge of civil protection for the implementation of the corresponding measures.

1 INTRODUCTION

Since its creation, HERCA has identified the need for a harmonized approach on Emergency Preparedness and Response (EP&R) in Europe as a top priority. The events at the Fukushima Daiichi NPP in March 2011 dramatically illustrated that similar needs for a common understanding and approach also exist for accidents happening at great distance from Europe.

HERCA set up a Working Group on Emergencies WGE to come up with practical and operational solutions leading to a uniform way of dealing with any serious radiological emergency situation, regardless of national border lines. The aim is to develop a comprehensive approach to harmonization, not only limited to an agreement on individual parameters and concepts, such as reference levels. The final goal is to obtain a uniform cross border application of protective actions.

On top of the many actions undertaken by IAEA, WHO, NEA and the EC, further harmonization efforts are needed with regard to recommendations to citizens who reside in the vicinity of the accident site, to people arriving at (air)ports with luggage, goods, foodstuffs, etc. The evaluation of the radiological impact, the sharing of information and, last but not least, the streamlining of the communication efforts are of upmost importance.

2 THE HERCA WENRA APPROACH HWA

2.1 HWA - Developpment

After the accident in Fukushima Daiichi NPP, HERCA set the priority on the harmonisation of the reactions in European countries to any distant nuclear or radiological emergency. The outcome of the WGE has been approved by HERCA and published in 2013 [1].

In a second phase, the approach for better cross-border coordination of protective actions was focused on accidents happening within Europe.

The third phase was dedicated to the early phase of a nuclear accidents within Europe with no or very little information available. A strong collaboration between HERCA and WENRA has been established and a consensus was reached by a high level working group within six months.

The outcome of these second and third phases was put together as HWA, approved by HERCA and WENRA and made publicly available on the respective websites in November 2014 [2]. The HWA is divided into 3 steps:

1. In the preparedness phase a shared understanding of the existing national emergency arrangements shall be achieved and maintained.
2. During the early phase of an accident, rapid information exchanges shall enable the countries to do, as far as possible, the same as the accident country.
3. In the later phase a common situation report, accepted by all impacted countries, including the accident country, shall further support coordinated protective actions.

2.2 HWA - Content

Each European state defines its own priorities and objectives in planning for nuclear emergencies directly affecting its own territory. Emergency planning has evolved in all states over many years, mostly without giving great priority to cross-border issues. At the same time, the international framework for planning and response has changed, too. This has led to differences, sometimes significant, in Criteria for intervention levels for introducing protective actions, Types of protective actions, Operational intervention levels, Methods for assessing source terms, Methods for radiological impact assessment and dispersion modelling, Definitions of emergency planning zones, etc. In case of a nuclear emergency in Europe, these differences could potentially have a significant effect, especially if the location of the emergency is close to a national border. Figure 1 illustrates schematically how a particular protective action could be implemented when the decision is purely based on national considerations.

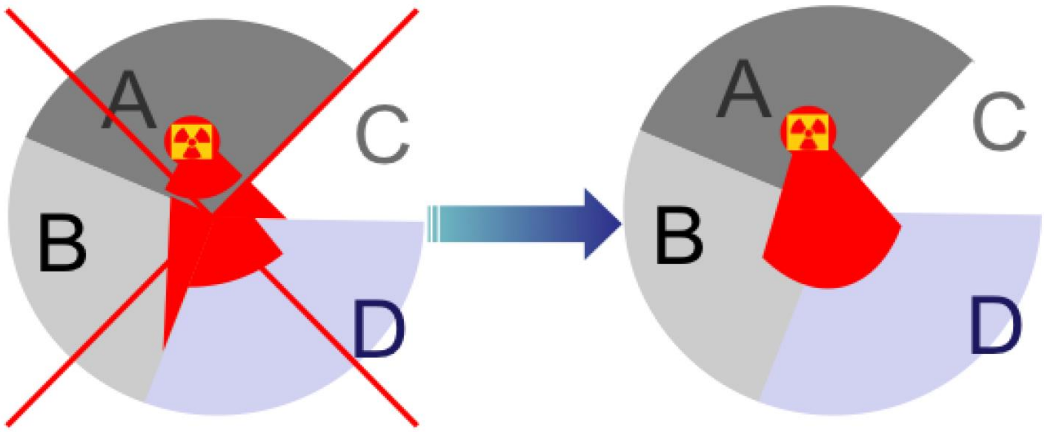


Figure 1: From uncoordinated to aligned protective actions.

Beside the difficulties to achieve harmonization of all emergency arrangements, such harmonization would not give the assurance for a consistent response. During the early phase of a response the appreciation of the uncertainties in assessing source terms and dispersion will remain different in each country. This may significantly influence the decisions.

The HERCA-WENRA-Approach therefor proposes an alternative solution. The aim is to achieve and maintain a shared understanding of the existing national emergency arrangements by developing or improving already existing bilateral and multilateral arrangements, to test these arrangements and implement improvements. In the early phase of an accident, rapid information exchange through existing bilateral and international arrangements should take place. If the response is thought consistent, the neighboring countries will be able to recommend their governments to follow these recommendations, i.e. adopting the principle “We do the same as the accident country” in the first hours of the accident. In the later phase a common situation report, accepted by all impacted countries, including the accident country, will further support coordinated protective actions.

Concerning protective actions, the HWA considers only sheltering, evacuation and iodine thyroid blocking. It is recommended that the protective actions are planned and prepared up to the distances mentionned in table 1.

Protective Action	Distance
Evacuation + ITB	up to 5 km
Sheltering + ITB	5 to 20 km

Table 1: Protective actions and distances to which a detailed preparation is recommended

Unlike stated in a technical document of the IAEA, the HWA clearly recommends that sheltering shall be preferred against evacuation under the plume. The reason for this is that there is, up to now, not enough evidence that the evacuation under the plume causes more good than harm.

As the accident in Fukushima has shown, it may become necessary to extend protective actions over the distances mentioned in table 1. For such a case, a detailed planning is not considered to be commensurate, but general strategies for such an extension should nevertheless be developped.

Protective Action	Distance
Evacuation + ITB	up to 20 km
Sheltering + ITB	up to 100 km

Table 2: Protective actions and distances to which general strategies should be developped.

The Fukushima accident was a reminder that a severe nuclear accident cannot be completely excluded anywhere in the world, including Europe. Considering the safety level of European nuclear power plants and their improvements, the probability of such a severe accident is very low. However, as improbable such an accident might be, emergency preparedness and response (EP&R) arrangements have to take into account such cases too. The HERCA-WENRA approach includes a simplified scheme for situations requiring rapid decisions for protective actions, even if very little is known about the situation and reliable dose calculations are not available yet. The scheme is based on three so called Judgment Evaluation Factors (JEF) dealing with the risk of core melt, the containment integrity and the stability of meteorological conditions.

JEF	Description	Possible values of JEF		
1	Is there a risk of core melt?	Yes	No	Unknown
2	Is the containment integrity maintained?	Yes	No	Unknown
3	Is the wind direction?	Steady	Variable	Unknown

Table 3: Definition of the Judgment Evaluation Factors JEF

The need for rapid decisions using a simplified schemes for protective actions will only be applicable during an initial phase. As soon as the accident country is in a position to present a more elaborate assessment of the plant status and the expected off-site impact, it will take the necessary steps to align its decisions and cross-border coordination mechanisms accordingly.

2.3 HWA - Implementation

The HWA has been adopted by the européen authorities in charge of nuclar and radiation safety. Therefore it is not yet a position shared by States, although EU Member States have engaged in the implementation of some of the recommendations in Council Conclusions [3]. HERCA and WENRA are committed to engage their national Authorities in charge of Civil Protection and to track the implementation of the HWA in the european countries.

2.3.1 Workshop with Civil Protection Authorities

HERCA and WENRA held a first Workshop on the Implementation of the HWA with European Radiation Protection, Nuclear Safety and Civil Protection Competent Authorities in June 2016 in Bled, Slovenia.

Nearly 80 high-level representatives from 23 countries and from international organisations, such as the International Atomic Energy Agency (IAEA), the European Commission (EC) and the Nuclear Energy Agency (OECD/NEA) attended to the workshop.

The aim of the workshop was to discuss with the key actors involved in nuclear EP&R operational and pragmatic means to implement the HWA and therewith contribute to an enhanced protection of the population, particularly in a cross-border context.

Participants agreed to the following main conclusions from the workshop [4]:

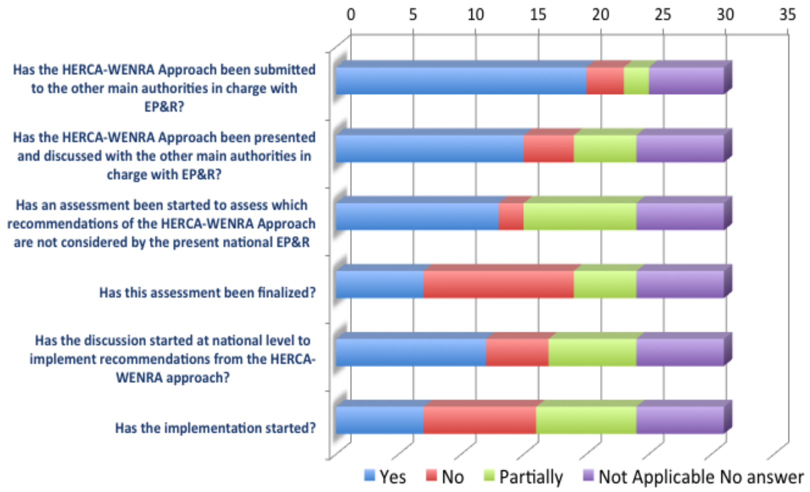
1. During the workshop participants had fruitful exchanges on a better understanding of the HWA, its recommendation and possible ways of implementation.
2. Participants agreed that trust between the relevant competent authorities and other key stakeholders is of fundamental importance. Trust needs to be built at preparation stage and maintained.
3. Participants identified issues for further work on food chain protection, the extension of protective actions at distances beyond the emergency planning zones and the use of non-radiological criteria for deciding on protective actions.
4. Alignment of planning zones and the alignment of protective actions during the response have proven to be difficult, even during exercises, due to political, historical, local and financial issues.
5. Participants identified some areas with NPPs near national borders in Europe where in-depth work for implementing HWA should be prioritized, allowing for experience feedback to be used by other sites.
6. Authorities competent in radiation protection, nuclear safety and civil protection need to continue to work on the implementation of the HWA while taking into account existing international mechanisms, standards etc.
7. Participants underlined the need for setting up an effective and coordinated cooperation among all relevant authorities involved in disaster management, with the support of EC DG ECHO.

2.3.2 ENSREG-Survey

The European Nuclear Safety Regulators Group ENSREG organised early 2016 a quick survey to get a first view of the implementation of the HWA in the member states. The results of the survey was presented at the above mentionned Worshop in Slovenia [5].

25 out of 31 countries answered to the questionnaire. 2 countries stated that they were not concerned since the closest NPP is at distances above 100 km. In one country the competent authority for nuclear safety has no role in EP&R and forwarded the questionnaire to the relevant agency. The later one considers the HWA as not compulsive and did not feel obligated to reply. No answer has been received from 3 countries, one with Nuclear Power Plants.

The answers varied considerably between countries, but as shown in figure 2, they nevertheless indicate that the assessment of the HWA and its implementation have in most cases started and seem to be ongoing, mainly in the framework of the transposition of the Euratom BSS directive which should be done until 6 February 2018. Given this situation, member states will be able to provide more concrete information on the HWA implementation



status once these transposition work is completed.

Figure 2: Responses to the HWA implementation status

It is therefore be important to continue multilateral exchanges and coordination during the implementation processes. The HERCA Working Group on Emergencies (WGE) has been mandated by both HERCA, WENRA and ENSREG to follow-up the HWA implementation in European countries.

2.3.3 Continous tracking

By end of April 2016, the HERCA Board of Heads approved the tracking system proposed by the WGE to follow the implementation of the HWA. The results from the ENSREG survey are integrated in the new tracking system.

The HWA implementation in the participating countries and the way how the discussions at national level with civil protection authorities progress will be intensively discussed twice a year in the meetings of WGE.

Up to the 13th meeting of the WGE in September 2016 in Helsinki, 18 member countries filled up the tracking sheets. As in the ENSREG survey, the responses showed that the implementation of the HWA is well ongoing, but that a lot of work remains to do, mainly related to the transposition of the Euratom BSS.

A detailed analysis showed that most of the national assessments need to be revised in order to become comparable to each other. To reach this, a procedure similar to that used for National Reports to the Convention on Nuclear Safetythe Convention will be used.

At meeting of the 14th WGE in March 2017 in Oxford, revised tracking sheets of a selection of countries (Belgium, France, Germany, Luxemburg, Netherlands and Switzerland) will be analysed and benchmarked.

3 CONCLUSIONS

Efficient EP&R arrangements have been established in Europe since many years and are tested and challenged regularly. They allow authorities to issue recommendations for effective public protective actions. In case of a nuclear emergency in Europe, coordinated protective actions along adjacent national borders are highly desirable.

During the very early phase of a nuclear accident, the status of the reactor and the estimation of the amount of radioactivity released (source term) are unlikely to be precisely assessed. Despite, decision-makers have to take appropriate health protection measures. This inevitably leaves room for flexibility in decisions, even where there is a rigid national framework. The HWA makes use of this freedom for coordination between neighboring countries in order to align early decisions across borders. As a result, the respective national arrangements do not necessarily need to be changed. Instead, the prevailing differences are respected and taken into account, and the response is based on ‘compromise’ solutions, which are understandable and explainable in each given situation.

The HWA relies on the following principles: shared technical understanding, coordination and mutual trust. It does not propose a uniform cross-border framework. The main strategy is to aim at an alignment of the response between neighboring countries or neighboring territories. This is supported by early information exchanges using as far as possible existing bilateral and international arrangements.

The HWA also includes a simplified scheme for the initial stage of a highly improbable accident (i.e. Fukushima like) requiring decisions while very little is known about the situation.

HERCA and WENRA consider that a minimum common level of preparation for protective actions should be achieved in Europe, following a graded approach:

- Evacuation should be prepared up to 5 km around nuclear power plants, sheltering and iodine thyroid blocking (ITB) up to 20 km;
- A general strategy should be defined in order to be able to extend evacuation up to 20 km, sheltering and ITB up to 100 km;
- Nuclear and radiation safety Authorities in Europe should continue attempts to promote compatible response arrangements and protection strategies amongst the European countries.

This position is shared by radiation protection and nuclear safety Authorities. For its implementation, these Authorities have started discussion with their national Authorities in charge of Civil Protection. This is an ongoing and continuous process which be given a high priority. As a final aim, it would be desirable that the principles of the HWA are shared by all States.

4 REFERENCES

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What the “peaks” in local dose rate measured during the Fukushima accident tell about deposition, air concentration and release of radio-activity

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Abstract:

The investigations presented in this paper aim at the reconstruction of radioactive releases from the Fukushima Daiichi nuclear power plant during the first three weeks of the accident based on local dose rate measurements at the site. These measurements are characterised by discontinuous rapid changes (“peaks”) followed each by a continuous decrease phase. By comparing radiological analysis results with those of severe accident analyses for Units 2 and 3 by GRS within the OECD/NEA project "Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF)", a better understanding of the accident progression and an independent evaluation of calculated source terms from severe accident analysis are endeavoured.

Our analysis provides expected and also unexpected results concerning the relationship between local dose rate measured and the respective radioactive releases. A basis nuclide composition for the analysis was reconstructed based on the first available soil sample taken ten days after the accident. Unexpectedly, the local dose rate behaviour during the first days of the accident, especially after the four large peaks between March 14, 2011 evening and March 16, 2011 cannot be explained by ground shine from this nuclide composition while the agreement improves later in March 2011.

Besides the attempt to explain the measured local dose rates by ground shine, several alternatives addressing atmospheric dispersion or release processes have been tested. Those alternative hypotheses are found incapable to explain observations. Only contributions to ground shine by short lived nuclides generated significantly later than reactor shut down can explain measured local dose rates in during the continuous decrease phases in the aforementioned release period. These contributions can be partly attributed to an excess release of short-lived daughter nuclides of fission products with longer half-live time and lower volatility. However, this process is not sufficient enough to produce the amount of short-lived nuclides necessary to explain the observations after the two large peaks between March 14 evening and March 15 noon. In that time window, only additional fission products generated immediately before the releases, possibly due to recriticality events during reflooding of the partly damaged reactor core of Unit 2, can suitably explain the observations. Whether such recriticality events could have occurred in Unit 2 is currently further analysed by different experts’ organisations.

1 SCOPE AND OBJECTIVES

GRS participates on behalf of the German Federal Ministry of Economic Affairs and Energy (BMWi) in the OECD/NEA project: "Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Plant (BSAF)" [1]. Within Phase I of the project launched in 2012 deterministic analyses for the severe accident (SA) progression during the first days for the Units 2 and 3 of Fukushima Daiichi have been provided for which the coupled GRS codes ATHLET-CD/COCOSYS have been used [2]. The second phase of the OECD/NEA BSAF project started in May 2015 with an extended scope. Besides a continuation and extension (comprising the first 3 weeks of the accident) of the deterministic SA analyses one new focal topic consists of the comparison of measured radiological data (dose rate on-site) with calculated releases of radioactive material from the units to the environment (source term). Forward and backward calculations of possible radionuclide releases are foreseen. This should allow for drawing conclusions related to the appropriateness of the results provided by the SA analyses based on an independent approach.

Within this scope our objectives are

- to reconstruct radioactive releases from measured local dose rate on-site Fukushima Daiichi nuclear power plant (NPP) or nearby,
- to identify relevant processes for radioactive releases from the plant and to derive plausible parameters which describe these processes,
- to draw conclusions on processes and uncertainties which sensitively influence source term estimation and
- to compare our results with those of severe accident (SA) analyses performed within OECD/NEA BSAF Project, Phase II.

2 METHODOLOGY

Several studies address the reconstruction of radioactive releases from Units 1-3 of the Fukushima Daiichi nuclear power plant (Fukushima I NPP) based on environmental data. Several attempts employ measured data on a global scale and corresponding global dispersion and deposition simulations while others focus on the use of regional data (e.g. [3]) while others focus on data available form stations throughout Japan [4]. The focus of our study is, in contrast, on the very local scale, i.e. on-site and in the near vicinity of the plant.

2.1 Data

During the first three weeks of the accident, mainly local dose rate measurements are available at several measuring points on the site ([5], cf. Figure 1) and around. However, a few samples of soil activity concentration at the site (e.g. at locations marked by "X_1" and "X_3 in Figure 1) are also available [6]. Near-ground measured weather data are available at Fukushima Daiichi NPP [5].

The attempt to use on-site measurements, such as the measured local dose rates at several measuring points at Fukushima Daiichi NPP (cf. Figure 2 for example measurements), for reconstruction of radioactive releases leads to substantial difficulties, as it is not easy to distinguish between contributions to the measured signals from radioactivity in the air (cloud shine) and from the ground (ground shine). Moreover, samples of air activity which correspond to local dose rate measurements during the first days and weeks of the accident are not available and samples of nuclides deposited on the ground are scarce during that accident phase. Finally the releases need to be attributed to a special Unit having different source terms, nuclide compositions and timings of releases.

Therefore, assumptions on the nuclide composition have to be made in order to draw conclusions from measured local dose rate to air activity concentration and/or surface contamination. These assumptions may lead to considerable uncertainty and errors in the reconstruction process. On the other hand, the local dose rate measurements at the site witness the releases of radioactivity associated with a minimum complexity of atmospheric transport processes that influence the measured signals. This complexity is in turn a large uncertainty factor which affects the reconstruction of radioactive releases from observational data at larger distances

It is thus our intention to draw as much information as possible from measurements at the site or nearby Fukushima I NPP. For this purpose, we combine available data on local dose rates, specific soil activity and weather data to reconstruct quantities which are not covered by measurements in a step-by step approach. Our investigations are based on measured data at Fukushima I NPP and in the vicinity up to a distance of about 20 km during the first three weeks of the accident.

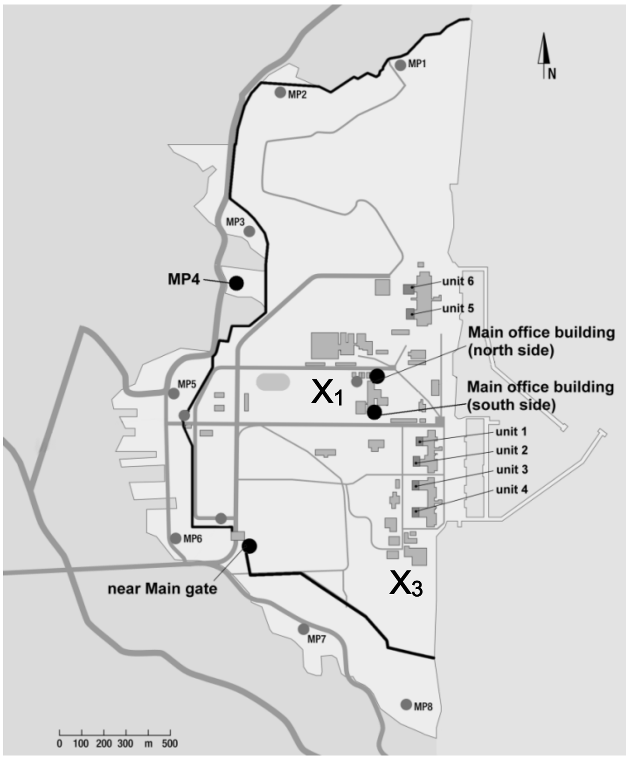

Figure 1: Position of the measuring points (MP) for dose rates and locations of employed soil samples at Fukushima Daiichi NPP

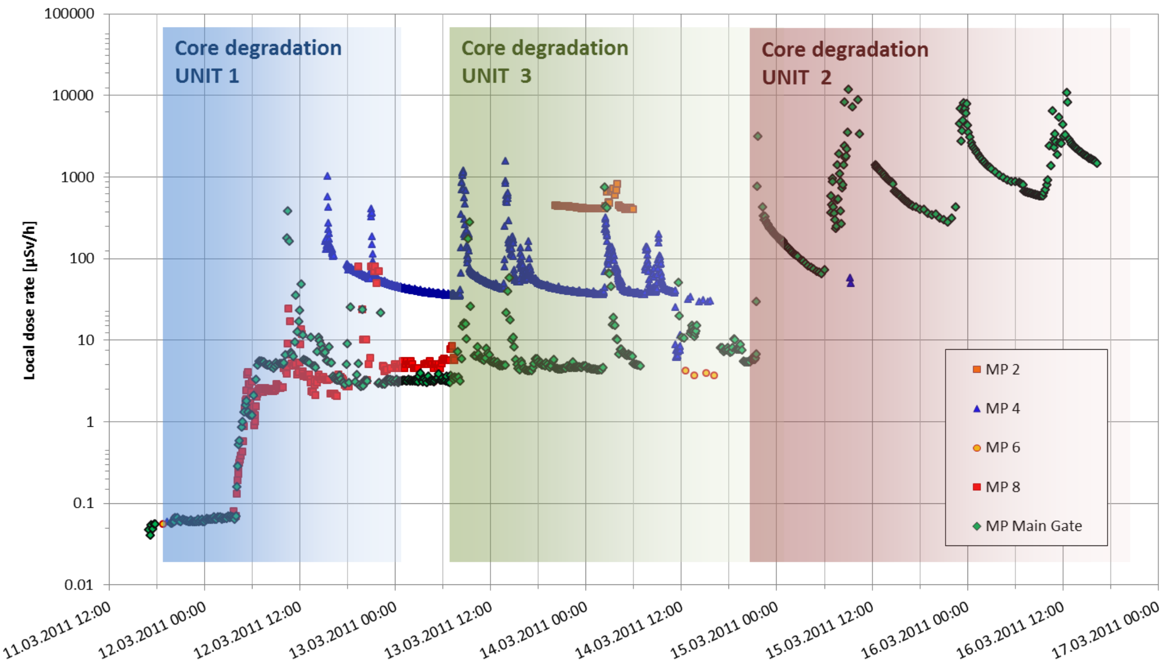

Figure 2: Dose rate measurements performed by TEPCO at selected locations and severe core degradation phases in Units 1 – 3

As shown in Figure 2, several local dose rate peaks have been measured. So far, in spite of considerable efforts, not every local dose rate peak could be clearly linked to indicated events in Units 1 – 3 yet. Nevertheless, it is well established that the main core degradation phase in each of the three units happend at different points in time [7, 8], as indicated in Figure 2. Unit 1 had the earliest core degradation starting already three hours after the tsunami which hit the plant at March 11, 2011 at 15:37 hr. The core degradation in Unit 3 started at the morning of March 13, 2011 around 7:00 hr while Unit 2 experienced the latest core degradation starting at March 14, 2011 around 20:00 hr. All core degradation processes lasted for several hours and releases from the unit to the environment are caused either by containment leakages or by containment venting processes. These processes are still under further investigation within the OECD/NEA BSAF project.

2.2 Reconstruction Scheme

The reconstruction scheme applied for our investigations to reconstruct radionuclide releases from the plant is depicted in Figure 3. It consists basically of the three steps described below.

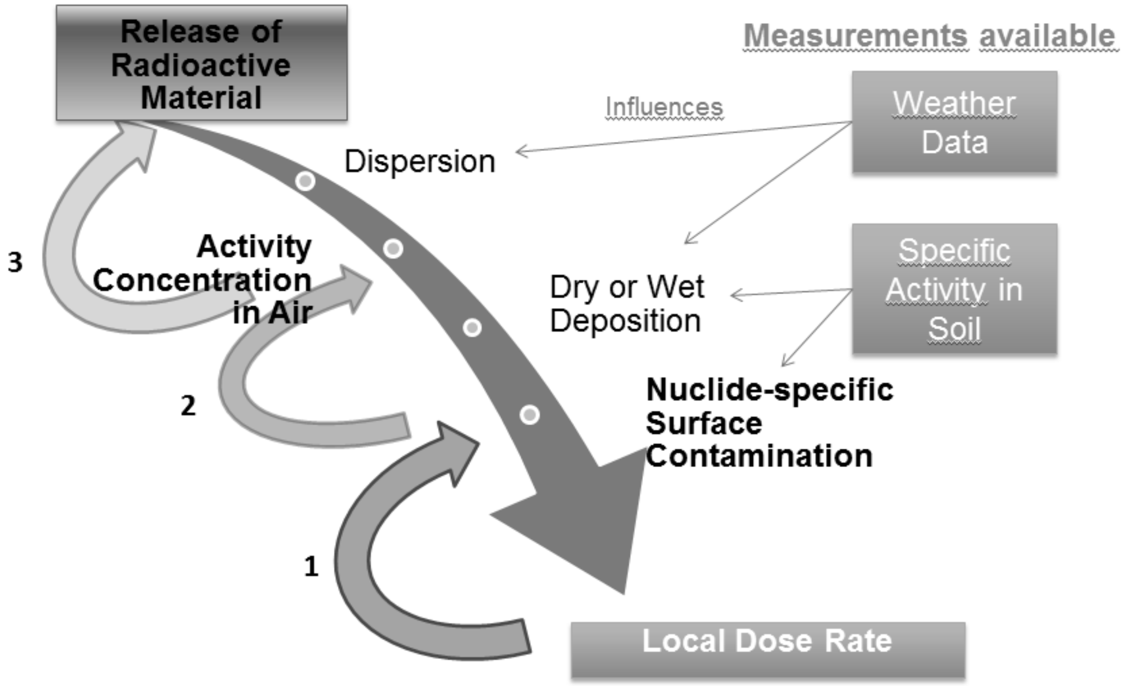

Figure 3: Reconstruction scheme for calculation of surface contamination, air activity concentration and release of radioactivity from measured quantities.

Step 1: Calculation of surface contamination from local dose rate and specific soil activity

For this step, at first measured local rates at a given observation point has to be divided into cloud shine caused by airborne radioactivity and ground shine caused by surface contamination. The accuracy of this separation is crucial for the quality of results from subsequent reconstruction steps. Then, nuclide-specific surface contamination has to be estimated by relating ground shine to the nuclide composition of deposited nuclides which in turn is determined from samples of specific soil activity

Step 2: Calculation of air activity concentration from surface contamination and information on precipitation

Air activity concentration during cloud phases when cloud shine is nonzero and aerosols and elementary iodine are supposed to be deposited are calculated from the difference in surface contamination before and after the respective cloud phase, taking into account the respective deposition mechanism (dry or wet) based on available information on precipitation. Deposition rates are assumed proportional to the strength of the cloud shine signal. This method yields estimates for the temporal development of air concentration of aerosols and elementary iodine, but not possible contributions from noble gases. The latter can, however, be guessed from the difference between measured cloud shine and the contribution to cloud shine calculated from estimated air concentration of aerosols and iodine.

Step 3: Calculation of radioactive releases from local dose rate and air concentration and modelled dispersion

For this step, appropriate dispersion parameters are to be obtained from atmospheric dispersion modelling which is driven by weather information which is available at Fukushima I NPP and, for a shorter period, at very few stations nearby. These calculations are performed with the Lagrangian dispersion model ARTM (Atmospheric Radionuclide Transport Model) [9].

The amount of radionuclides released is then calculated by an appropriate backward calculation method. For this purpose, an optimal solution for radioactive releases to be assumed is sought by minimizing the difference between observed and calculated cloud shine that would result from the release estimate. This minimization problem is solved by the use of the “Non-Negative Least Squares“ (NNLS) algorithm [10].

The reconstruction scheme as a whole has been succesfully tested for local dose rate observations at Fukushima Daiini NPP.

3 ANALYSIS RESULTS FOR MEASUREMENTS AT FUKUSHIMA 1 NPP

For the remainder of this paper, the analysis is concentrated on local dose rate measurements obtained on-site Fukushima 1 NPP and on step 1 and step 2 of our reconstruction method. The focus is on the episode of most intense measured peaks in local dose rates; the peaks measured at the main gate between March 14, 2011 18:00 JST and March 16, 2011 16:00 JST. For further analysis of these peaks and the subsequent phases of decrease, local dose rate $\dot{H}$ as well as its normalized change rate $r := \frac{1}{\dot{H}}\frac{d\dot{H}}{dt}$ is considered.

3.1 Structure of largest peaks in local dose rate

The structure of the four largest peaks measured at a temporal observation point at the main gate is depicted in terms of local dose rate as well as its change rate in Figure 4. All four peaks measured in this time interval exhibit similar structures:

- a phase of strong and discontinuous increase and decrease (**“rapid change phase”**) followed by
- a slow and continuous decrease (**“continuous decrease phase”).**

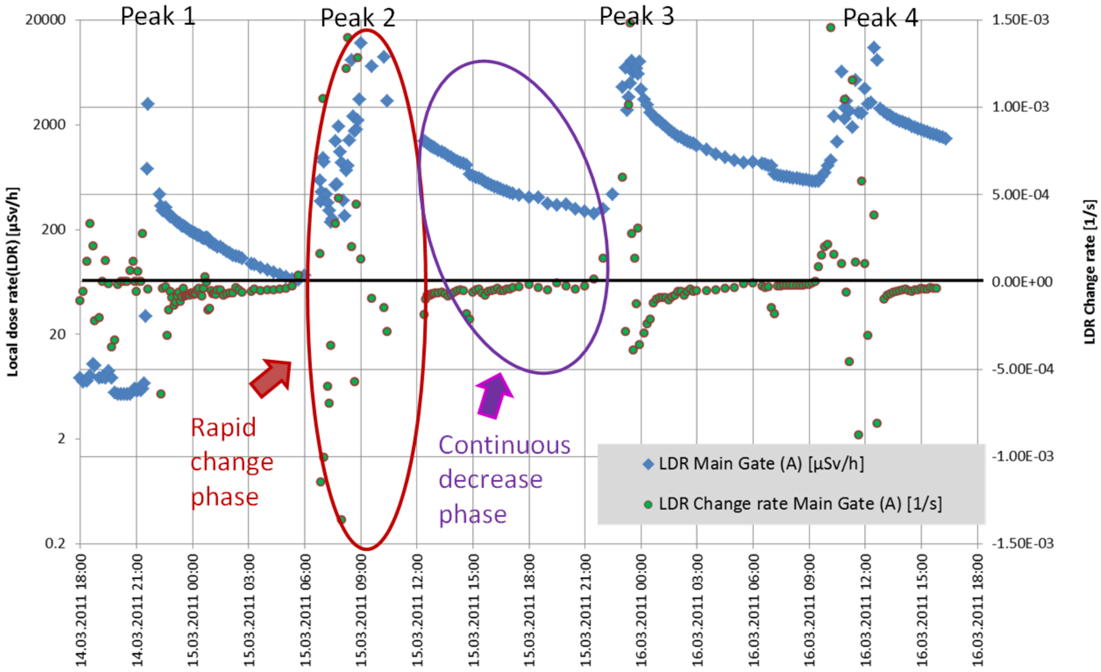

Figure 4: Temporal development of local dose rate and its change rate at MP near main gate between March 14, 2011 18:00 JST and March 16, 2011 16:00 JST.

For a realistic distinction between cloud shine and ground shine, the question has to be answered whether local dose rate in the "continuous decrease phase" is entirely caused by ground shine from surface contamination or not. In order to answer this question, several hypotheses aiming at the explanation of the observed temporal development of local dose and as its change rate in the "continuous decrease phase" are tested.

3.2 Hypotheses for explanation of local dose rate in the continuous decrease phases

The first and fundamental hypothesis to be tested aims at the explanation of local dose rate in the continuous decrease phases by ground shine. The corresponding surface contamination is estimated from soil samples. Thus, the hypothesis can be formulated as follows:

- **Basic hypothesis**: *"The observed local dose rate is dominated by decay of nuclides deposited on the ground. The nuclide composition can be obtained from soil samples."*

In order to test this hypothesis, the corresponding nuclide composition is determined based on an early available soil sample taken at location "X_1" (cf. Figure 1) on March 21, 20112011 is determined. The activity concentration of corresponding hypothetical soil samples within the time range March 14, 2011 18:00 JST to March 16, 2011 16:00 JST can be estimated by decay correction. Contributions from short-lived Iodine isotopes which are no longer evident in the actual sample from March 21, 2011 but could be still present in the investigation period are estimated from the amount of I-131 in the actual sample. The ratio between the respective Iodine isotope in the reactor core at scram and the radioactive decay of this isotope compared the decay of I-131 are used for the estimate and equal release fractions for all Iodine isotopes are assumed.

By this, a nuclide composition consistent with the reference soil sample can be calculated for each time point in question. The derivation and resulting values of this composition (henceforth referenced to as the "basic mixture" of surface contamination) is summarized in Table 1.

Table 1: Nuclides considered in the „basic mixture" derived from soil sample on March 21, 2011

Nuclide	**Dominant Generation Process**	**Half Life Time**	**Typical activity inventory of a BWR with same power as Units 2, 3 at Scram [Bq]** (scaled from [11])	**Activity concentration in soil sample on March 21, 2011 near playground [Bq/m³]**	**"Basic Mixture": Activity concentration in hypoth. soil sample on March 15, 2011 00:00 JST with equal composition [Bq/m³]** (cf. [6])
I-131	Fission	8.02 d	1.9 E+18	5.80 E+06	9.74 E+06
I-132	Fission Decay of Te-132	2.3 h	2.8 E+18	in Equilibrium with Te-132	in Equilibrium with Te-132
I-133	Fission	20.7 h	3.8 E+18	n/a	1.83 E+06
I-134	Fission	52.5 min	4.3 E+18	n/a	<<1
I-135	Fission	6.63 h	3.7 E+17	n/a	5.42 E+03
Ru-106	Fission	1.005 yrs.	1.5 E+18	5.30 E+04	5.36 E+04
Te-129m	Fission	33.6 d	7.0 E+16	2.50 E+05	2.83 E+05
Te-132	Fission	3.18 d	2.7 E+18	6.10 E+05	2.25 E+06
Cs-134	Fission	1.998 yrs.	3.4 E+17	3.40 E+05	3.42 E+05
Cs-136	Fission	13.15 d	1.2 E+17	7.20 E+04	9.88 E+04
Cs-137	Fission	30.108 yrs.	2.4 E+17	3.40 E+05	3.40 E+05
Ba-140	Fission	12.73 d	3.2 E+18	1.30 E+04	1.80 E+04
La-140	Fission	1.67 d	3.2 E+18	3.30 E+04	3.93 E+05

Calculated from I-131 soil activity concentration and inventory ratio at scram

It turns out that, in addition to the nuclides measured in the sample, I-133 is likely to have still contributed significantly to surface contamination around March 15, 2011 while the other short-lived Iodine isotopes generated by fission before scram have already decayed. It should also be noted that I-132 is not only generated by fission, but also produced by decay of Te-132 and thus can be found in equilibrium with Te-132 in the sample.

The temporal change rate of ground shine caused by surface contamination with nuclide composition is calculated and compared to the observed change rate in local dose rate. The result is depicted in Figure 5. It turns out that the absolute values of calculated decrease in ground shine which would result from surface contamination with the “basic mixture” ranges between 2 – 3·10^{-6} s^{-1}. This range is far below the absolute values of observed decrease rate during the “continuous decrease phase” (10^{-4} – 10^{-5} s^{-1}).

The “basic hypothesis” thus cannot explain the behaviour of local dose rate during the continuous decrease phases after the four largest peaks measured near the main gate.

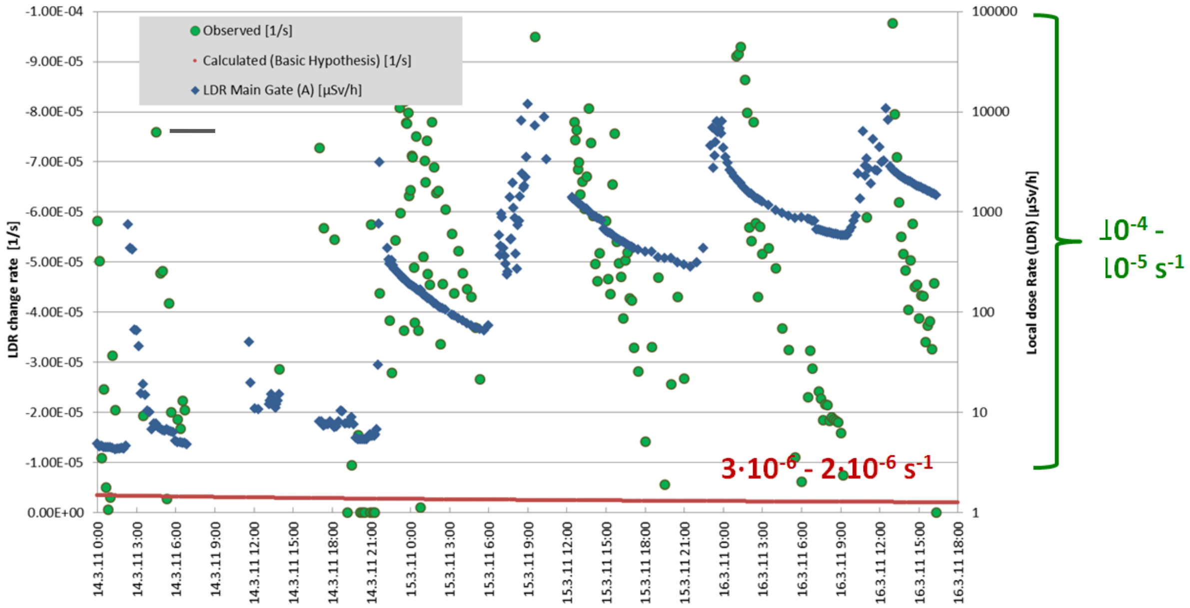

Figure 5: Comparison between calculated change rate in ground shine by „basic mixture“ and observed change rate of local dose rate at measuring point (MP) near main gate between March 14, 2011 18:00 JST and March 16, 2011 16:00 JST.

As the basic hypothesis does not hold in the time interval considered, alternatives for explanation have to be sought. Other processes that might influence the continuous decrease phases might be contributions from airborne radioactivity as well as mechanisms other then radioactive decay which deplete surface contamination. It is also possible that radioactive isotopes with short half times which must have been generated considerably after shutdown of the reactors influence the observations. All these possible explanations are checked by formulating and testing corresponding hypotheses.

Firstly, alternative hypotheses which address possible influences from airborne radioactivity are tested. They are formulated as follows:

- **Alternative hypothesis 1**: *“The observed local dose rate is caused by the* radioactive *cloud, which is slowly drifting away and dispersing.“*
- **Alternative hypothesis 2**: “*The observed local dose rate is caused by reduced and slowly decreasing radioactive releases after a strong release from the* reactor*(s), e.g. caused by an abrupt pressure relief by the opening of a leak followed by a continuous compensation of pressure imbalances via the leak.“*

Alternative hypothesis 1 is tested by calculating lower limits for the change rates in local dose rates which can be caused by very slow drift and weak diffusion of radioactive clouds with our dispersion model ARTM. It turns out that cloud drift is associated with change rates larger than 10^{-3} s^{-1} and cloud diffusion leads to change rates larger than 10^{-4} s^{-1}. These values fit very well with observed change rates in the rapid change phase but not with those of the continuous decrease phase (with 10^{-4} - 10^{-5} s^{-1}). This result fits very well with general results of the time scales of turbulent motions which typically involve periods less than half an hour.

If, according to alternative hypothesis 2, continuous and slowly decreasing releases of radioactivity were the reason for the observed behaviour in local dose rate, the airborne activity concentration would also be modulated by atmospheric transport and dispersion. Their signal at the measuring point would thus exhibit change rates comparable to those caused by a drifting and dispersing cloud as in alternative hypothesis 1. Moreover, they would be subject to changes in mean wind direction caused by the large-scale weather systems, in particular the change from northerly to southerly winds during the daytime hours of March 15, 2011. However, these features are not evident in the dose rate measurements during the continuous decrease phases. Whether release phases have occurred in reality can only be answered by carrying out backward atmospheric dispersion calculations as in step 3 of our reconstruction method (cf. subchapter 2.2).

As a whole, our tests lead to the conclusion that airborne radioactivity does not substantially influence the observed local dose rates during the continuous decrease phases.

The next hypotheses to be tested refer to the possibility that the observed local dose rate during the continuous decrease phases is indeed dominated by surface contamination whose activity decreases more rapidly than expected according to the basic hypothesis.

- **Alternative hypothesis 3**: *"The observed local dose rate is caused by deposited radionuclides which are slowly reduced by wind-driven resuspension and/or runoff by rainfall."*

The test of alternative hypothesis 3 is split into two parts. Firstly, resuspension with the wind would be caused by turbulent motion and thus should lead to fluctuations on timescales shorter than 10^{-4} s^{-1} (cf. test of alternative hypothesis 1). Secondly, precipitation only occurred after March 15, 2011 22:30 JST according to available observations and hence has not influenced the continuous decrease phases after the first two peaks shown in Figure 4. Alternative hypothesis 3 is therefore also not suitable to explain observations, so that another and final alternative hypothesis was tested.

- **Alternative hypothesis 4**: *"The observed local dose rate is caused by deposited radionuclides including short-lived nuclides that must have been produced shortly before a release phase associated with the respective preceding peak. This implies an active generation mechanism for short-lived nuclides considerable time after the shutdown of the reactors."*

The testing of alternative hypothesis 4 is linked to the identification of potential generation mechanisms.

- The *first potential generation mechanism* to be thought of is the production of short-lived daughter nuclides from the decay of fission products. If these daughter nuclides are more volatile and thus released to a larger fraction than their respective mother nuclide, they will be deposited on the ground in excess compared to their mothers. The radioactive decay of the excess activity of these "volatile daughters" will then lead to a stronger decrease in observed ground shine than expected from the decay of the mothers.

  So far, I-132 as daughter of Te-132 has been identified as the only nuclide which could fit such a mechanism. Assuming the same release fraction for I-132 as for I-131, an estimate for excess release of I-132 compared to Te-132 can be guessed from measured ratios of I-131 and Te-132 in the soil sample and those of inventories in the core. This comparison yields an estimated average ratio of about 4:1 between deposited I-132 and Te-132.

- The *second potential generation mechanism* consists of the production of short-lived nuclides by recriticality that might have occurred in the core degradation phase of one of the reactors. Such a recriticality event would generate many kinds of fission products. I-134 and I-132 would however dominate among the more volatile products for short recriticality events up to a few hours.

In order to test possible contributions from both mechanisms, the ground shine resulting from a large variation of hypothetical compositions between the “basic mixture”, I-132 released in excess as volatile daughter of Te-132 and I-132 and I-134 produced by recriticality has been calculated by Monte Carlo simulations and compared to the measured local dose rate curves during the continuous decrease phase by minimization of the relative error. Three sets of simulations have been performed:

- *Simulation A:* Ground shine is calculated by the “basic mixture”. In fact, this set only consist of one member so no optimization is necessary
- *Simulation B:* Ground shine is calculated from different fractions of the “basic mixture” and additional I-132 assumed to stem from release as volatile daughter of Te-132. On average, the efficiency of this process can be assumed to be limited by a ratio between I-132 and Te-132 of 4:1 as described above. In order to allow for temporal variations of this efficiency, a maximum ratio of 8:1 is allowed.
- *Simulation C:* Ground shine is calculated from different fractions of the “basic mixture”, additional I-132 assumed to stem from release as volatile daughter of Te-132 and additional I-134 and I-132 assumed to stem from recriticality events

Within simulation sets B and C, the respective member with the least relative deviation from observed curves (relative error) has been selected as the composition which optimally represents the possible contribution by the aforementioned mechanisms. The result is depicted in Figure 6.

As anticipated above, the agreement between calculated ground shine due to the basic mixture (simulation A, Figure 6 top) and observation is poor for all four continuous decrease phases after the largest peaks. This agreement is largely improved for all four phases by the assumption of I-132 being released in excess as volatile daughter of Te-132 (simulation B, Figure 6 middle). An additional contribution by I-134 and I-132 from assumed recriticality events still leads to a substantial improvement of agreement with observations for the two continuous decrease phases after the first two peaks (simulation C, Figure 6 bottom). In both cases, the relative error is reduced by more than an order of magnitude. For those after the last two peaks, the additional improvement is rather marginal.

The results show that the observed behaviour of local dose rate during the continuous decrease phase after the four strongest peaks can be accurately explained by hypothesis 4. Moreover, the results hint at the possible occurrence of recriticality events in a time window between the evening of March 14 and noon of March 15. This question will be further discussed in chapter 4.

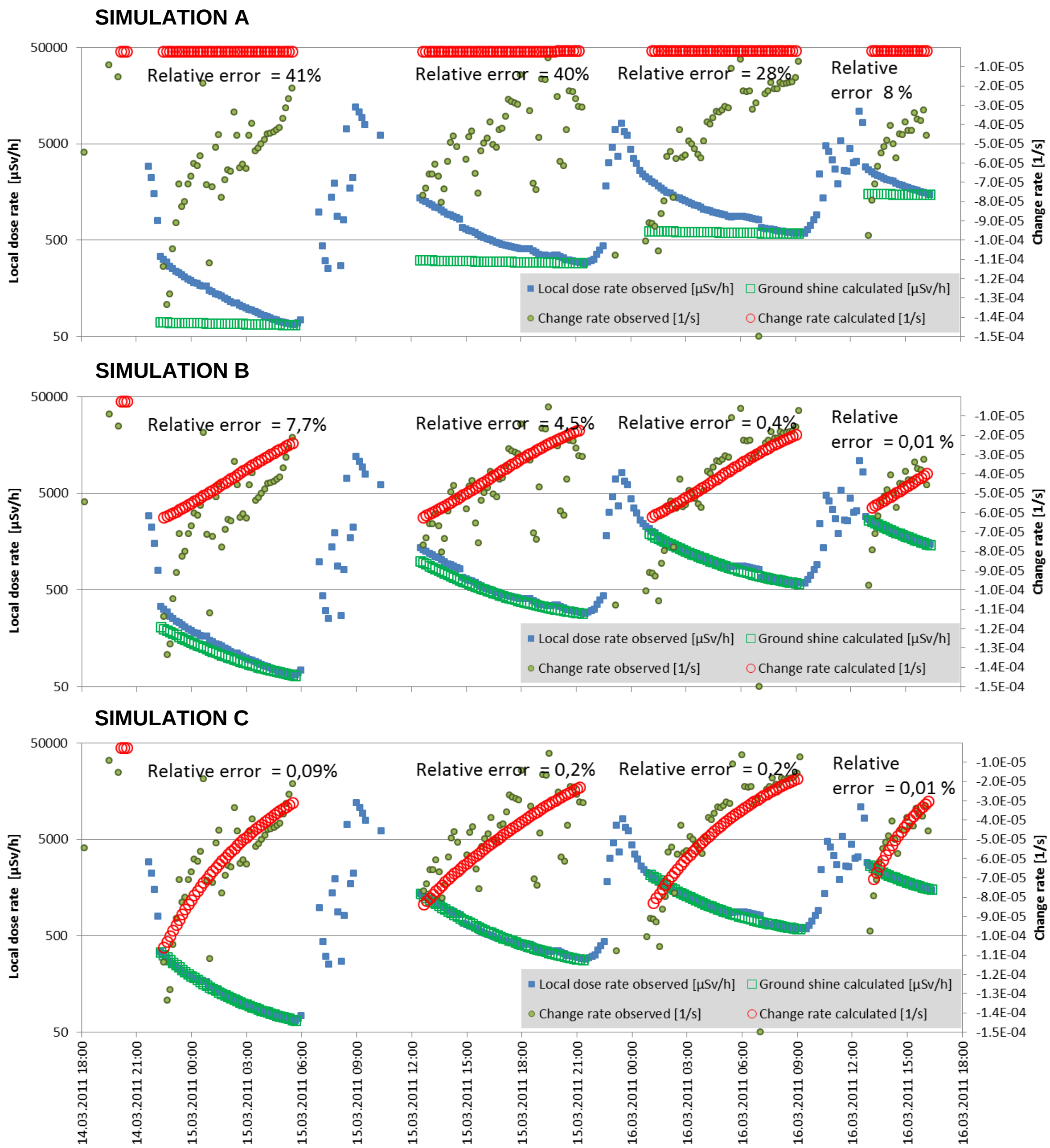

Figure 6: Comparison between calculated local dose rate and its change rate for optimal compositions obtained from simulation sets A, B and C and observations at measuring point (MP) near main gate between March 14, 2011 18:00 JST and March 16, 2011 16:00 JST.

3.3 Results from analysis of other measuring points and times

Our analysis of continous decrease phases after peaks in local dose rate has also been applied to other measuring points and time periods. They can be summarized as follows:

- *Measuring point MP 4, March 12, 2011 afternoon – March 14, 2011 noon:*
  Six peaks and subsequent decrease phases have been identified and analyzed. For all decrease phases, additional contributions to ground shine by excess release of I-132 as volatile daughter of Te-132 are sufficient for a good agreement with observations. Calculated ratios between I-132 and Te-132 range between 1.5 and 3.5. Results thus corroborate the validity of hypothesis 4, but without any additional hints at recriticality events.

- *Measuring Point (MP) main office building, north side, March 17, 2011 morning – March 21, 2011 late afternoon:*
  Observations at this measuring point exhibit many interruptions and large data gaps. Seven fragments of slow decay phases were identified and analyzed. About half of them exhibit additional contributions to ground shine by excess release of I-132 as volatile daughter of Te-132. Calculated ratios between I-132 and Te-132 range between 1.4 and 2. The other fragments are in good agreement with calculated ground shine from the “basic mixture”. In these time intervals, the basic hypothesis seems to be valid.
- *Measuring point (MP) near main gate, March 21, 2011 afternoon – March 26, 2011 late morning:*
  Measurements were restarted at this point after a gap of several days. Five peaks and subsequent decrease phases have been identified and analyzed. The time intervals between subsequent peaks range between 7 and 41 hours. Observations up to the first eight hours after three of the peaks exhibit additional contributions to ground shine by excess release of I-132 as volatile daughter of Te-132. Calculated ratios between I-132 and Te-132 range between 1.9 and 3.7. After about eight hours, local dose rate in the longer decrease phases is in good agreement with calculated ground shine from the “basic mixture”. In these time intervals, the basic hypothesis seems to be valid.

As a whole, the results show (cf. Figure 7), that especially in the first few days of the accident and shortly after strong peaks have occured, contributions to ground shine by excess release of I-132 generated by decay of Te-132 need to be taken into account to explain observed local dose rate. However, in the later phase of the accident, in particular during longer phases between two measured peaks, the agreement between observed local dose rate and ground shine calculated attributable to the “basic mixture” is good.

The need for an additional production mechanism of short-lived nuclides has been uniquely identified for the decrease phases especially after the two strong peaks between the evening of March 14, 2011 and noon of March 15, 2011. Hence, the question whether recriticality events have occured in one of the reactors, especially Unit 2, can be focused on this time window.

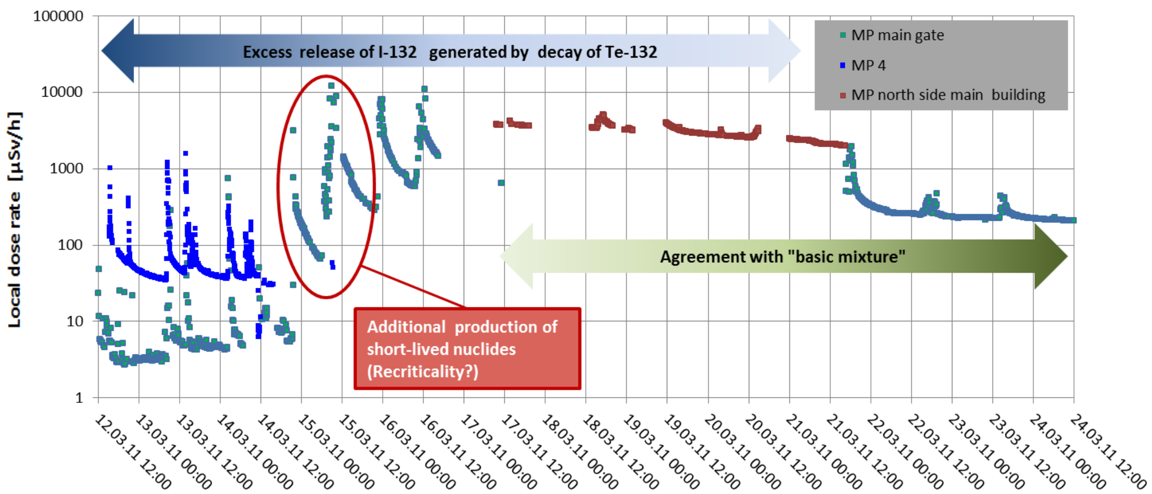

Figure 7: Summary of results from analysis of continuous decrease phases at measuring points MP 4, MP main office building (north side) and MP main gate.

4 DISCUSSION AND OUTLOOK

4.1 Summary of results

Our analysis shows that only ground shine with contributions by short lived nuclides can explain measured local dose rates during the “continuous decrease phases” after the four large peaks between March 14, 2011 evening and March 16, 2011 afternoon. These nuclides must have been produced significantly later than reactor scram occurred and are no longer evident in soil samples, which are available only from March 21, 2011. Alternative hypotheses addressing atmospheric dispersion or other release processes have been found incapable to explain observed results.

These contributions can be partly attributed to excess release of I-132 which is produced in the core by decay of Te-132. This leads to higher I-132 activities compared to Te-132 after end of deposition and a subsequent faster decrease in local dose rate. So far, no other short lived nuclides resulting from a decay process of a mother nuclide could be identified which could relevantly contribute to the effects observed by this mechanism.

For the above mentioned first two peaks, contributions from another production process of nuclides, preferably with even shorter half time is needed to explain observed local dose rate curves. Contributions from I-134 and additional I-132 produced by one or more hypothetical recriticality events between March 14, 2011 evening and March 15, 2011 noon would suitably explain observations. This finding is unique to this time window. Local dose rate observations in other time intervals investigated so far can be explained without the assumption of a recriticality event.

4.2 Clues for recriticality in Unit 2

Our findings show that a recriticality event would suitably explain observations in local dose rate measurements on March 14 / 15, 2011 and so far no valid alternative explanation has been found. This finding should not be mistaken as a positive proof that such an event really occured. Therefore, results from analysis of radiological observations should be compared to plant state information and corresponding findings from severe accident analysis, as such performed within the OECD/NEA BSAF project.

On the basis of calculated accident progession [1], a comparison shows that the releases which caused the two large peaks in the evening of March 14, 2011 and the morning of March 15, 2011 most likely stem from Unit 2, as is illustrated in Figure 8 by comparing local dose rates with the containment (“D/W”) pressure of Unit 2. The pressure shows a temporary halt in increase which corresponds with the first peak and a strong pressure drop which must have occurred during the “rapid change phase” associated with the second peak. Moreover, containment (“drywell)” radiation levels exhibit a continuous increase between the first and the second peak, indicating progressive extension of core damage in this phase. Unfortunately, measuremts of the containment pressure of Unit 2 are not available during the second “rapid change phase” so details of the temporal development in this phase have not been observed.

Measurements of reactor water level in Unit 2 (not shown) show a drop below core bottom level before the first peak occurs. In this situation, based not only on our analyses by ATHLET-CD/COCOSYS [1] it is expected that the core melting has made significant progress and control blades are already destroyed due to their lower melting point compared to the fuel. After this early core degradation an intermittent rise of water level follows as mobile water injecting gets efficient so that the core may be temporary and partially covered by steam and water when the two strong release peaks occur. This partial reflooding of the core with sea water and corresponding moderation of neutron flux could be the reason for temporal recriticality to occur.

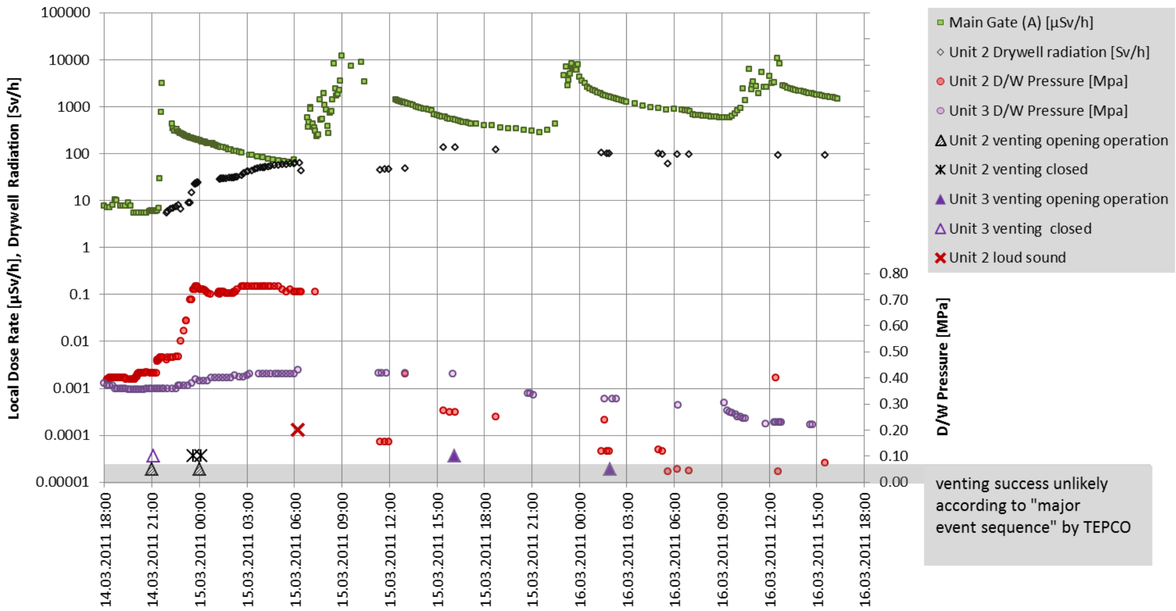

Figure 8: Observed local dose rate at MP main gate compared to secondary containment (D/W) pressure measurements in Units 2 und 3 as well as radiation levels in the secondary containment (drywell) of Unit 2. No such measurements are available for this time interval in Unit 3. The occurrence of venting operations and the „loud sound in Unit 2“ are also indicated.

In comparison to Unit 2, available information from Unit 3 shows no clues for strong releases from the Unit during this time interval. Core degradation of Unit 1 already advanced significantly by March 13, 2011 according to present knowledge of the accident [1, 7, 8]. It is thus very unlikely that the core of Units 1 and 3 still were in a configuration that would enable the occurrence of significant recriticality events on March 14 / 15, 2011 (cf. also Figure 2).

4.3 Open questions and further work

Clarification of occurrence of recriticality in the core during degradation is vital for analysis of fission product release as it alters the composition to be assumed predominantly at on-site measuring points. It is as well essential for explanation of core degradation phase and observed reactor vessel and containment pressure pikes in Unit 2.

For clarification, the question is to be answered whether the inclusion of recriticality events can improve current accident analyses carried out within OECD/NEA BSAF Project, phase 2 or would rather lead to contradictory results. This question could be investigated by introducing hypothetical events into the accident simulations with deterministic models such as ATHLET-CD/COCOSYS used by GRS and test the effect of this additional energy source e.g. on simulated pressure in the reactor vessel and in the containment. Here still none of the analyses presented in the OECD/NEA BSAF Project could explain the observed plant behaviour [1]. Furthermore such analyses can yield an estimate of possible onset times, intensity, and duration of possible recriticality events.

The latter information will be vital to estimate the amounts of additional fission product which have to be included in our reconstruction of air concentration and releases. This reconstructions will be biased towards an over estimation of nuclides such as I-131 or Cs-137 if production of fission products by recriticality is neglected. A solution to this question is thus also for our reconstruction of radiological information which will be carried on.

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TRAGIC CHOICES AT FUKUSHIMA

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Abstract:

The Tchernobyl accident and more recently that of Fukushima have reminded us that workers could face dramatic situations likely to engage their survival. This communication aims to present the mechanisms to assign abhorrent but necessary tasks in emergency situations.

The study is based on an extensive literature review of sources related to the Fukushima Dai-chi accident. A micro-sociological analytical method has been applied.

The results highlight that there is two determining movements in tragic choices. The first line choices define ethical principles that may have guided the choices in the selection of abhorrent but necessary tasks. The second line choices is the definition of the conditions of exposure to realize those abhorrent tasks.

1 INTRODUCTION

On March 11, 2011 at 14:46, an earthquake in eastern Japan caused the reactors in operation at the Fukushima Daiichi nuclear power plant (NPP) to trip. The emergency generators started and suddenly failed following the tsunami which struck the nuclear power plant at 15:27. This event was not only beyond design basis, it was beyond any predictions. The accident management manual no longer could be followed. Faced with the severity of the accident, the operators and managers were confronted with tragic choices at Fukushima Daiichi Nuclear Power Plants. The case of Fukushima Daiichi accident did not involve allocating scarcities, which often make particularly painful choices necessary. It was more about how abhorrent (i.e. could affect the course of a life) but necessary task are assigned (Calabresi and Bobbitt 1978). The decision is perceived as tragic because it determines who is given a better chance to live or to live healthy. The aim of this paper is to identify principles to cope tragic choice.

Although they had followed the procedures before the tsunami and searched the manual for applicable rules afterwards, the operators during the accident of Fukushima Daiichi Power Plant were not automatons following one routine instruction after another and freezing in the absence of guidance. They were attempting to connect the chaos they were experiencing into a comprehensible, describable situation that serves as a springboard into action (Weick, Sutcliffe et al. 2005). What has to be done? What should we decide? In this drastic situation the operators, managers, and politicians found themselves facing difficult, often moral choices. Which operator will take the risk of radiation exposure?

This perspective will enable us to appreciate the tragic struggle in crisis situation, to accept responsibility for tragic choices. Another purpose is to define the principles to allocate abhorrent but necessary tasks in ways that preserve the social structure and the cohesion of the group.

The paper uses data from the Fukushima Dai-ichi nuclear accident in 2011 to examine the underlying norms and principles that played out in practice to make tragic choices. A micro-sociological analytical method has been applied, in order to approximate actors' practices as closely as possible. According to Calabresi and Bobbitt (1978), there are two determining movements in tragic choices. The first-order determination defines the global setting (the ethical principles) and the second-order determination allocates the available resources to achieve that end (the means to implement the ethical principles). Based on the approach taken throughout this work, we will identify the first line choices: ethical principles that may have guided the choices in the selection of abhorrent but necessary tasks. Then we will consider the second line choices: the definition of the conditions of exposure to realize those abhorrent tasks.

2 FIRST LINE DECISION: CHOOSING THE WORKERS

At a very local level, at the control room level, there was a consensus on the ethical principles among the small group of operators, which was never challenged by other actors. According to Izawa's[1] testimony, the shift team of unit 1 and 2, even though the operators had long since realised that venting would be necessary and would require a trip to the highly radioactive reactor building, "*being ordered to 'pick people' was a strange thing*" (Kadota 2014 p. 122).

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[1] The normal shift supervisor for the team on duty in the MCR for reactors 1 and 2 during the disaster, named Hirano, was out for a routine medical examination, and he was replaced by the leader for a different team, Izawa.

The shift team first excluded younger operators from the operation. When nobody immediately volunteered, the shift team said that he would go himself. At that point, some of the more senior operators began to offer to go, while at the same time insisting that the shift team needed to stay in the control room (Kadota 2014 p.87-89). The names of the senior and more experienced operators were written on the white board in order of age. They then began to decide on teams of pairs to do the required tasks.

Inside the control room, they succeeded in thinking about the “common good” rather than that of the individual. Several ethical principles, inspired from Calabresi & Bobbitt’s work were applied to choose the workers:

- The “common good” was a fundamental principle in this situation. Individuals agreed to cooperate even though they were putting their lives at risk, although they tried to limit their exposure to radiation risks.
- Status-related principles: based on biophysical and, more generally, social characteristics, age is a central criterion in this case. The cancers that develop after exposure to low-dose ionising radiation have a latency period that can be as long as several decades. As a result, elderly workers are less likely to develop cancer before their death.
- The need for efficiency: the procedure was the result of a compromise between local efficiency and overall efficiency. Overall efficiency consisted in choosing the workers with the best chance of success in the shortest possible time. Local efficiency consisted in ensuring that the operational teams continued to operate properly by maintaining the integrity of the existing leadership. The chosen compromise therefore consisted in choosing experienced personnel who were not leaders.
- In this case, the “time” principle was particularly complicated to apply in order to define the time of intervention of workers. However, the more time passed, the more the tasks could became dangerous due to the rising radioactivity levels. On the three pairs which had been chosen to manually open the valve, the third had to turn away because of the high level of radiation. As a result the time when the task has to be carry out serves indirectly of criterion (contextual) to define those which are most exposed.

3 SECOND LINE DECISION: THE DEFINITION OF THE CONDITIONS OF EXPOSURE

Two factors – organizational and institutional - contribute to defining in which circumstances the chosen workers must go on the field. The organizational factors plays in the definition of what and when necessary but abhorrent tasks must be done, not in the way that consisted in choosing the workers who would be responsible for performing dangerous. When all possible solutions have been exhausted the need for venting carried out manually is required. The opening of the valve that is normally very simple to achieve by pressing a button in the control room, is extremely difficult to achieve without electrical source. Despite the conditions, the human become a resource for the organizational level to cope to the situation because there is no alternative. At least, the local and organizational levels will both agree to suspend this field mission because of the too high level of exposure.

The institutional factor contribute to formally define the conditions to move on the field. When TEPCO's executive management and the government experts realised that the situation was worsening with the increasing level of radioactivity, they decided to define new exposure limits.

The dose limit for workers applicable to an emergency situation was set at 100 mSv when the Fukushima nuclear disaster began. However, soon afterwards the threshold was exceeded. Following coordination with the Nuclear and Industrial Safety Agency, however, the government raised that limit at a stretch to 250 mSv at 2:03 PM on the 14th of March, three days after the crisis broke out. This decision to define legaly the exposure limits has two issues. First, new guidelines have been decided, although there is also no objective way of predetermining the level of radiation that operators have to face in these situations. The major factor potentially affecting the reliability of the monitoring performed was the use of shared personal dosimeters or the use of damaged dosimeters.

Second, this decision generates some embarrassment inside the plants. For the plant's manager Yoshida, "the government's ceiling made it possible, in institutional terms, for workers nearing their dose limits to stay on the front lines a while longer. But the step did not make human bodies more resistant to radiation" (Kadota 2014).

4 CONCLUSION

The definition of ethical principles, as we have seen, can be both "formal" and "contextual". In concrete situations where abhorrent but necessary tasks must be assigned, the principle of selection of workers is not unique. It is impossible to simply assign a point system or waiting lists (Elster and Herpin). It is not only the principle of seniority or merit that be retained. The ways in which the ethical principles are applied must be pragmatic and democratic. It appears that the definition of what is morally acceptable should not be imposed by institutional rules or highest hierarchy. The highest hierarchy can define some strategic guidelines in the management of the accident leading to the choice of various ethical orientations (including the case of subcontractor’s involvement). However, it seems important in order to maintain the social cohesion of the group that these principles could be discuss and defined locally, before an accident.

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New directions and challenges for research in Radioecology five years after the Fukushima accident

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Abstract:

The Fukushima accident has led to the onset of new research programs in radioecology at national and international levels, in collaboration with Japanese teams. The expected outcomes of ongoing studies are dedicated to the post-accident management of the contaminated areas mainly in the Fukushima prefecture. They are listed hereafter:

1) understand and predict the fate of radionuclides deposited in forest areas, which form the majority of the contaminated territories in Fukushima. The biogeochemical processes governing the evolution of radionuclides in forests imply a high remanence of the radioactive contamination in these ecosystems which represent therefore a long-term source of contamination of surrounding environmental resources (e.g., surface and ground water, food).

2) assess transfers of radionuclides due to the leaching of contaminated lands to streams and rivers, and to the marine environment (sediments and food webs). The ongoing studies at coastal watershed areas of Fukushima revealed that this process annually remobilises less than 1% of the stock of deposited radiocaesiums, mainly during extreme hydrological processes.

3) identify the effects of long-term exposure to ionising radiation on wildlife and understand the underlying mechanisms. Ecological data from fauna and flora observations in the contaminated territories of Fukushima reveal various biological effects on non-human biota, but contradictory conclusions about the main factors governing their occurrence (e.g., absorbed radiological dose by individuals, evacuation of human populations).

1 BETTER MODELLING FOREST CONTAMINATION TO SUPPORT LONG-TERM MANAGEMENT

After a nuclear accident, and once the emergency phase is over, contaminated areas must be duly identified and managed on a long-term basis to optimise the exposure of humans and ecosystems. By predicting spatial and time-based variation in contamination within the biosphere, modelling can provide indispensable scientific input for this management process. After the Fukushima accident, modelling was developed to understand and anticipate changing ambient dose rates, particularly in forests.

When the plume of radioactive particles passes, the cesium is intercepted by the tree canopy, i.e. the tops of the branches and leaves, in proportion to the development of this canopy. As the Fukushima accident took place in March, deciduous trees were little affected. On the other hand, the many coniferous plantations, particularly Japanese cedar trees (*Cryptomeriajaponica*) and Japanese cypress trees (*Chamaecyparis obtusa*), intercepted most of the cesium 137 (^{137}Cs) deposited on the Japanese forest eco-system. 75% of the areas contaminated by radioactive cesium are forest areas. The fastest processes for the transfer of cesium, essentially from trees to the soil, occur over the first 18 months following deposition; they predetermine the forest situation for many years. This involves the dropping of leaves, needles, small and senescent branches, which comprise the forest floor, the “washing” of leaves and needles, and to a far lesser extent, the “washing” of trunks. The term "washing" itself covers various processes: cesium carried along by rain drops running off leaves, placement into suspension by wind, and biological processes such as the desquamation of leaves.

Recent work has particularly allowed the testing of the TREE4 model (Transfer of radionuclides and external exposure in forests), developed over many years at IRSN [1]. This model allowed the satisfactory reproduction of the initial interception and kinematics of the transfer to the soil for both situations, by configuring each of the three processes involved in the natural decontamination of the canopy separately.

Other work has detailed the behaviour of the ^{137}Cs intercepted by the canopy by modelling the ease or difficulty with which it can be leached [2]. The parameters of this model were linked to the area of the canopy (or degree of closure) on the one hand and the intensity of rainfall events on the other hand. The parameters of the model obtained were adjusted using the measurements taken on two cedar tree plots in Japan located at Kawamata (Fukushima prefecture) from July 2011, just after the deposits from the Fukushima accident. The model was then successfully used to describe the leaching of other coniferous forests (cypress, pines, and other plots of cedar trees in Japan). In this way, for conifer trees, 80% of the cesium intercepted and sensitive to leaching remains in the canopy for three months on average. This research led to a better understanding of changes in cesium in the initial months following the accident, a timescale which it was not possible to track after of the Chernobyl accident.

Modelling also led to an understanding of why the decay in dose rates measured by the Japanese authorities between April 2011 and December 2012 was twice as fast as that predicted based on the physical decay of the radionuclides deposited [3].

To understand the apparent inconsistency between the measurements and physical principles, the transfer of radionuclides and changing ambient dose rates were digitally simulated for various typical environments in the Fukushima region (i.e. forests, rice paddies, pastures, inhabited areas, etc.). These simulations demonstrate that in a forest environment, the dose rates at the flight altitudes considered decreased by approximately 40% per year in 2011 and 2012 (consistent with measurements), as this decay was half induced by the progressive decontamination of the canopies and the attenuation of the radiation measured by the airborne sensors enhanced by the plant cover. On the other hand, simulations demonstrate that, in a forest environment, dose rates near the soil increased slightly over the same period, as soil contamination increased due to the decontamination of the canopies.

2 LEACHING OF CONTAMINATED LANDS TO STREAMS AND RIVERS: A SECONDARY SOURCE TERM TO THE MARINE ENVIRONMENT

The Fukushima accident led to levels of radioactive contamination never reached in the marine environment, raising the issue of contaminated fishery products. The evolution of the contamination of the coastal marine environment around the Fukushima plant, after a decrease during the first years after the accident is, five years after the accident, characterized by relatively stable concentrations of radionuclides (no decrease of contamination in sediments, very slow decrease in sea water). Within 30 km around the plant, contamination of sea water is maintained by the effect of three types of contributions it is difficult to quantify (the likely release from damaged reactors site; leaching and drainage of contaminated watersheds (this process is particularly visible during typhoons); resuspension and redissolution from sediment particles). Beyond 200 km from the plant, the concentrations of radiocesium fell back to levels close to those seen before the accident.

As expected, organisms living near coastal areas shows the highest contamination, and the slowest decrease in activity. Outside Fukushima Prefecture, radioactive cesium measured in marine food do not exceed, since late 2014, the marketing limit set by the Japanese health authorities (100 Bq / kg). In Fukushima Prefecture, this limit is still sometimes exceeded.

Modelling is necessary in order to forecast variation in the contamination of seafood products on a medium or long-term basis after a nuclear accident. This prediction would enable their marketing to be managed on the basis of scientific data. The most suitable method was found to be the coupling of an ecosystem model consisting of plankton cycles (nutrients-phytoplankton-zooplankton-detritus model) and regional current circulation, with a radioecological model specifically developed for studying the contamination of pelagic fish [4]. In this approach, each species is described based on its generations, as the number of generations depends on the life cycle, and the frequency of reproduction. Organisms are classified by size, to which are associated a food ingestion rate and diet, parameters that change over time with the size of the individual.

Based on data on the cesium 137 dispersion off the Pacific coast of Japan, the model was first applied to plankton populations, particularly around the nuclear plant (Estimated concentrations of approximately 4 orders of magnitude higher than those observed before the accident): this showed very high levels of contamination. This study demonstrated that zooplankton mainly accumulates cesium from food, and that minor bioamplification exists with increasing zooplankton size classes.

Regarding fish, the model demonstrates that the estimated contamination levels for the different species are well above those observed before the accident, with levels increasing as the size of the individual increases, undoubtedly due to bioamplification via a more ‘carnivorous’ diet. Furthermore, it would appear important to consider the migration movements of some fish species in this type of model. This approach was validated in pre-accident equilibrium conditions, and in post-accident conditions, and the results obtained are satisfactory. Fish living over the sediments which are the most contaminated today now remain to be studied.

3 EFFECTS OF LONG-TERM EXPOSURE TO IONISING RADIATION ON WILDLIFE: THE IMPORTANCE OF PRECISELY ESTIMATING THE ABSORBED RADIATION DOSES

Various studies dedicated to the consequences of nuclear accidents conclude that significant effects exist for the fauna and flora at very low dose rates of ionising radiation, disagreeing with current radiobiological knowledge. However, the effects observed in exposed organisms had been generally related to the ambient dose rate measured at the observation sites. This was considered a weak point by the scientific community of radioecologists.

A precise interpretation of data requires the consideration of the dose actually absorbed by the organisms studied. Calculating this dose will depend on the species, their habits, the radionuclides inhaled or swallowed, and external exposure to contamination at a given site.

In territories contaminated after the Fukushima accident, the dose rates to which the birds are exposed were reconstructed by combining radionuclide measurements (^{134}Cs, ^{137}Cs, ^{131}I) and models [5]. It was found that these rates can be up to 20 times higher than the ambient dose rate. They vary by a factor of 8 among the 57 species examined, and by a factor of 44 within a single species, depending on observations sites. 90% of species are chronically exposed to a dose rate likely to affect their reproduction (With reference to the quality and intensity of the induced effects for these levels of exposure, published by the International Commission on Radiological Protection (ICRP, 2008)).

Variation in the number of birds in a 50 km radius around the Fukushima plant between 2011 and 2014 was analysed using a statistical model representing the entire data set and integrating descriptive variables for the environmental conditions as best as possible (Temperature, time of the observation, cloud cover, type of landscape.). This model demonstrated that the total absorbed dose (reconstructed) has a negative effect on reproduction: for the study period and area, the total number of birds dropped by 22% when the absorbed dose increased from 10 to 100 mGy. By extrapolation, it is estimated that a dose of 550 mGy would decrease the number of birds by 50%.

This work confirms that the accurate assessment of the dose absorbed by the organisms studied helps to guarantee the scientific credibility of the conclusions reached by studies on relationships between levels of exposure to ionising radiation and the biological and ecological response observed. Combining dosimetric reconstruction and advanced statistics to process ecological data sets is a promising approach to improving our understanding of these relationships.

4 CONCLUSION

These research programmes are still underway and help to improve the ability of operational models to predict changes in contamination and the dose rates over different time scales after an accident, on a medium and long-term basis. Currently, Japan implements actively its recovery plan for the territories impacted by the fallouts of the Fukushima accident, mainly through an extensive decontamination programme. The ongoing radioecological research on these territories contributes to the consolidation of elements of post-accident management of aquatic and terrestrial environments

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Radiological Situation at the Chernobyl Site and Main Differences to Fukushima

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Abstract:

Thirty years after the accident the Chernobyl Shelter, i.e. the ruin of the destroyed unit 4 and the erected Sarcophagus, still remains a dangerous nuclear facility. Inside the Shelter remained about 96 % of the irradiated nuclear fuel inventory of the reactor before the accident, i.e. 180 t of Uranium of total radioactivity $7x10^{17}$ Bq. The radioactive releases to the environment were estimated to amount 4 %. Because of the radiation exposure the spent fuel inside the Shelter and the radioactive soil and groundwater contaminations at the site have an essential impact on all human activities which are presently under progress with the erection of the New Safe Confinement (NSC) and later on with the removal of fuel containing materials and other radioactive waste. Essential differences to the situation at the Fukushima site concerning the releases, the environmental contamination and the radiological consequences will be discussed.

1 RADIOACTIVE RELEASES AND SPENT FUEL AT CHERNOBYL

1.1 Radioactive releases after the accident

The radioactive releases after the accident to the industrial site around the Chernobyl NPP were estimated to amount 0,5 - 1,0 % of the spent fuel inventory. Another portion of about 1,5 % was release and settled in the near region of the 30 km zone of the Chernobyl NPP site and almost the same value of 1,5 % was transported by airborne dispersion to far regions in Europa, so that the total release resulted in about 4 %. The release behavior in the first days after the accident is shown in Figure 1.Spent nuclear fuel inside the Shelter

To investigate the potential radiological hazards associated with the radioactive fuel containing materials inside the Shelter and the radioactive contamination at the site in more detail the knowledge of the modifications and local distribution of spent fuel is necessary.

Presently, the spent nuclear fuel exists in four modifications. Radioactive dust can be found in almost all rooms, but predominantly in the central hall with an estimated total mass of about 25 t. Moreover, fuel element fragments are mainly located in the central hall as well as in the southern cooling pool with a total amount of about 120 t. Molten fuel lava of nearly 35 t has moved to the lower rooms and several kg of Uranium and Plutonium solutes are contained in water in the lower levels as shown in Figure 2. The fuel containing materials are often hidden below sand, gravel and concrete thrown e.g. from helicopters after the accident to extinguish the fire and to lower radioactive releases or in non-accessible rooms and thus the numbers have partly big uncertainties.

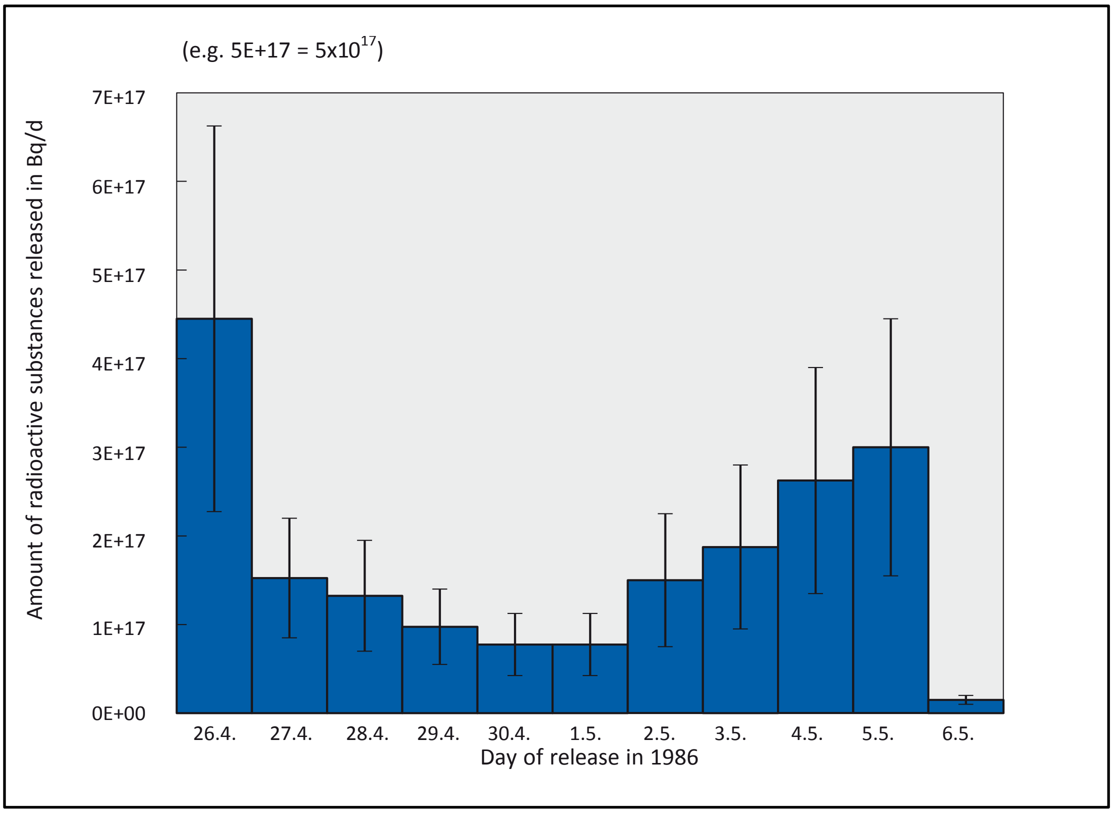

Figure 1: Radioactivity released during the first 10 days (with uncertainty ranges)

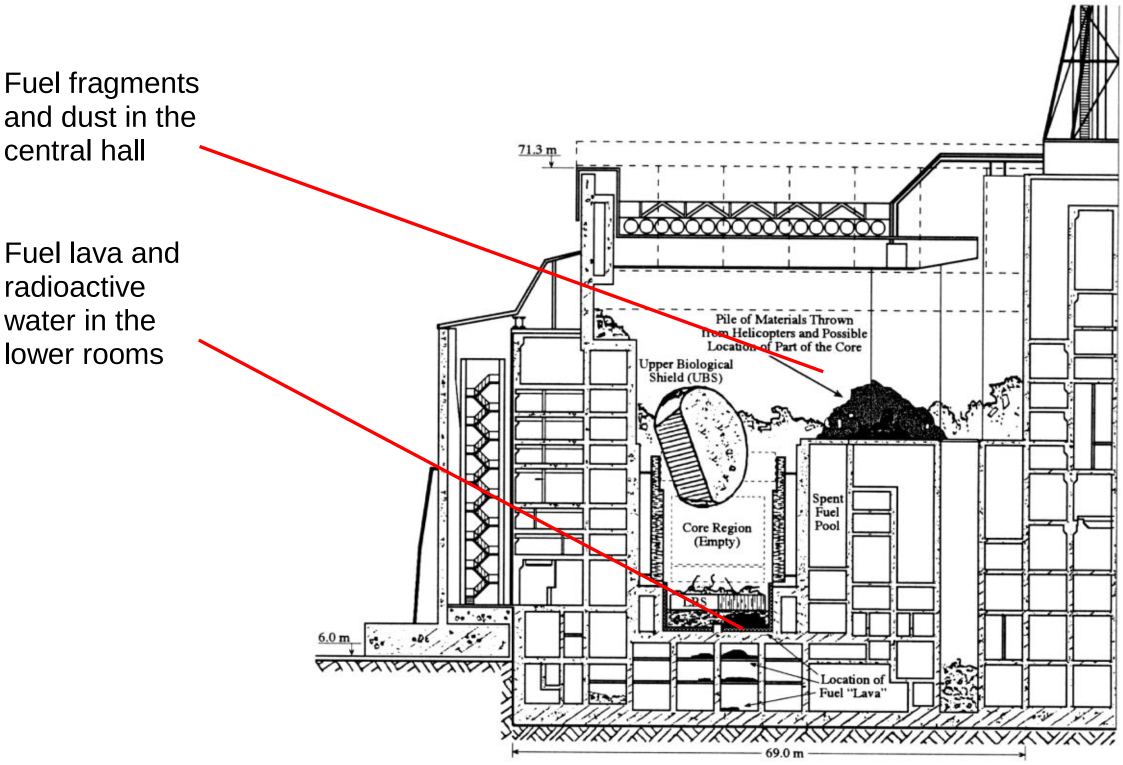

Figure 2: Cross section of the Chernobyl Shelter West - East

1.2 Spent fuel degradation and release of radioactive dust

During the last thirty years the molten fuel due to residual heat decrease, temperature changes and influence of humidity degraded more and more from a glass kind into a brittle and porous matter. Due to water ingress more and more fuel was leached and transported into the lower rooms. Most of the radioactive water moved from the destroyed unit 4 by different pathways into unit 3, e.g. about 200 m^3 in 2015. In the central hall and the upper rooms dry radioactive particles are generated, become airborne and get released through different roof openings and the ventilation stack, deposit outside and may contribute to inhalation and ground shine dose. The radioactive dust release from the Object Shelter into the environment for the last years measured by means of passive aspiration aerosol sampling units as well as ba active filter devices with air pumps is depicted in Figure 3. Most of the dust particles are in the respirable range below 10 µm Aerodynamic Equivalent Diameter, the yearly releases are decreasing e.g. due to the enhanced spray system.

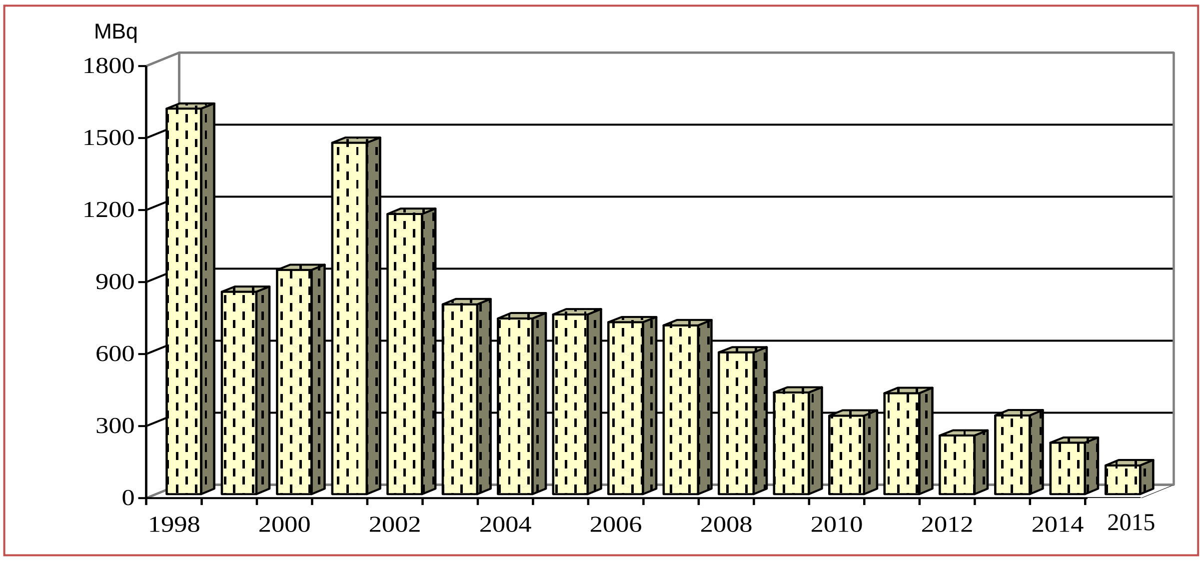

Figure 3: Average yearly releases of radioactive aerosols through roofing leaks and the ventilation stack

2 RADIOACTIVE CONTAMINATION AT THE CHERNOBYL SITE

2.1 Radiation exposure due to ground shine at the Shelter site

Due to construction work for the basement pillars of the rails to shift the NSC from the West over the Object Shelter the radiation situation has significantly changed in the last years. The original contamination layer covered by gravel, sand and concrete in the near vicinity of the Shelter walls was partly released again, new waste was generated and the radiation situation changes with ongoing work. The Dose rate at the Chernobyl Object Shelter site before the start of this work in 2012 is drawn in Figure 4. It can be seen that already several meters far from the Shelter walls and from fuel burial spots the dose rate 1 m above ground reaches values comparable to the natural radiation background.

2.2 Groundwater Contamination at the Chernobyl Shelter Site

The annual groundwater concentrations of Sr-90 and Cs-137 e.g. for well 5-Г decreasing over the last years (see Figure 4) at the flow exit of the Shelter site towards the cooling pond is shown in Figure 5 [1]. The permissible level of drinking water in Ukraine is 96 Bq/l for Cs-137 and 45 Bq/l for Sr-90.

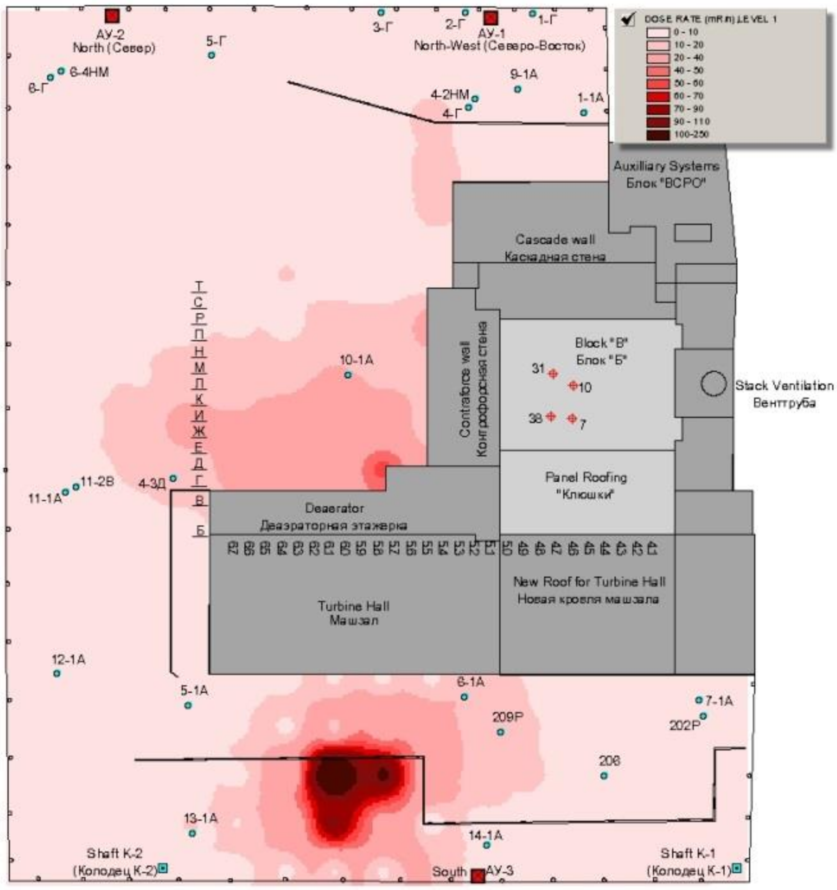

Figure 4: Dose rate 1 m above ground (mR/h) at the Chernobyl Object Shelter site in 2012, Aspiration units AY1 to 3 for aerosol sampling (red), bore holes for ground water inspection (blue), red spots indicate fuel burial.

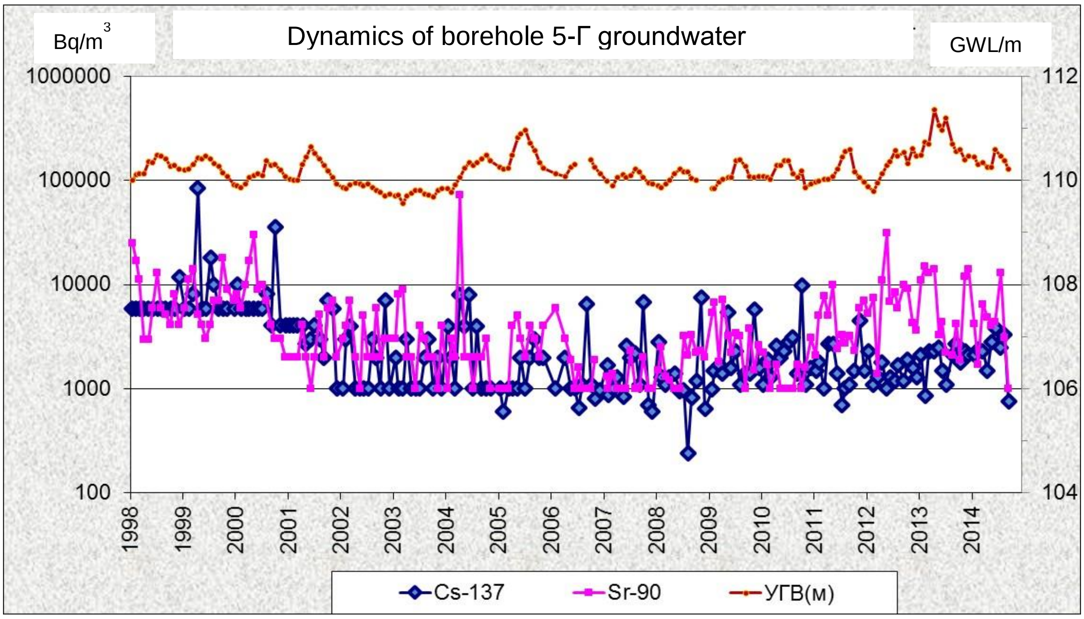

Figure 5: Concentrations Cs-137, Sr-90 (Bq/m3) and Groundwater Level (GWL/УГВ, absolute scale) in borehole 5-Г

3 RADIOACTIVE RELEASES AT FUKUSHIMA AND COMPARISON TO CHERNOBYL

3.1 Radioactive releases due to the accident

In the first days of the accident in Fukushima 11 March 2011, considerable quantities of radioactive substances were released into the environment due to explosions, ventings and other processes. The majority of these came from the reactor units 1 to 3. The radioactive aerosols and gaseous substances liberated into the atmosphere were carried away with the wind and were able to settle in the vicinity, especially in connection with precipitation. In the further course of the accident, radioactive substances were also released into the Pacific Ocean with the emergence of contaminated water [2].

The atmospheric release of I-131 is estimated to be in the range of about 1 to 4 x 10^{17} Bq, that of Cs-134 about 8,3 to 50 x 10^{15} Bq, and that of Cs-137 about 7 to 20 x 10^{15} Bq. Soil samples showed that mainly radioactive I-131, Cs-134 and Cs-137 deposited in the environment.

Water analyzes also showed mainly contamination with Iodine, Caesium, Tritium and Strontium isotopes. The water-borne releases is estimated to correspond to approximately 10% for I-131 and 50% for Cs-137 of the corresponding activity of airborne releases.

3.2 Radioactive contamination in the vicinity of the Fukushima site

In the course of the earthquake and the subsequent tsunami, no automated measurements were available in the first days after the accident at the intended measuring points. As a replacement, the operator used mobile measuring stations (e.g. measuring vehicles), which changed their position several times during the course of the accident.

Figure 6 shows the air dose rate measured at the various measuring points until mid of April 2011. There are two accident phases distinguishable. In the first phase up to approximately the end of March 2011, the visible maxima up to an air dose rate of 12 mSv/h are derived from releases associated with explosions, ventings and other processes. The radioactive substances released into the air partially settled in the environment. The subsequent phase is characterized by a gradual decrease in air dose rate. This trend continued until today, although radioactive substances are still released on a small scale.

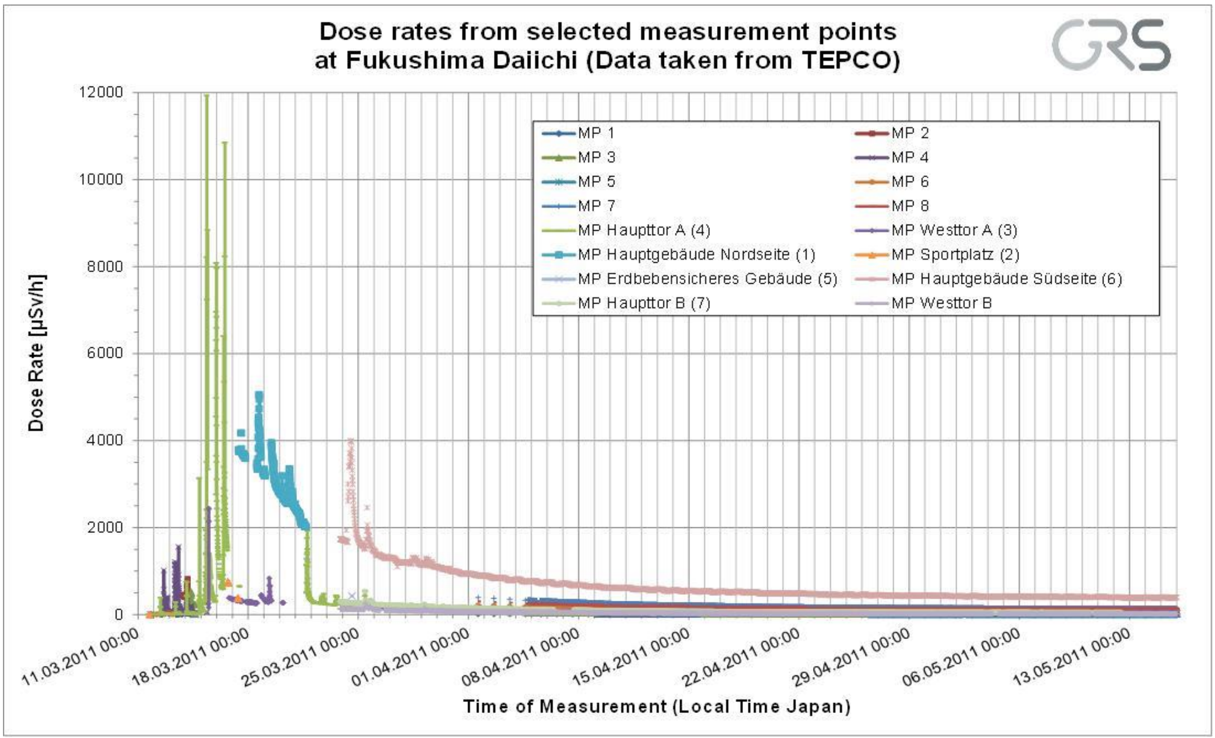

Figure 6: Measured dose rate at different measuring points (MP) of the premise

Aerogamma campaings were started shortly after the accident by the Ministry of Education, Culture, Sports, Science and Technology in Japan (MEXT) and the American Department of Energy (DoE) starting on 17 March 2011 (see Figure 7). In this figure, the area of elevated air dose rate in the north-west direction as well as a gradual decline in air dose rate are clearly visible.

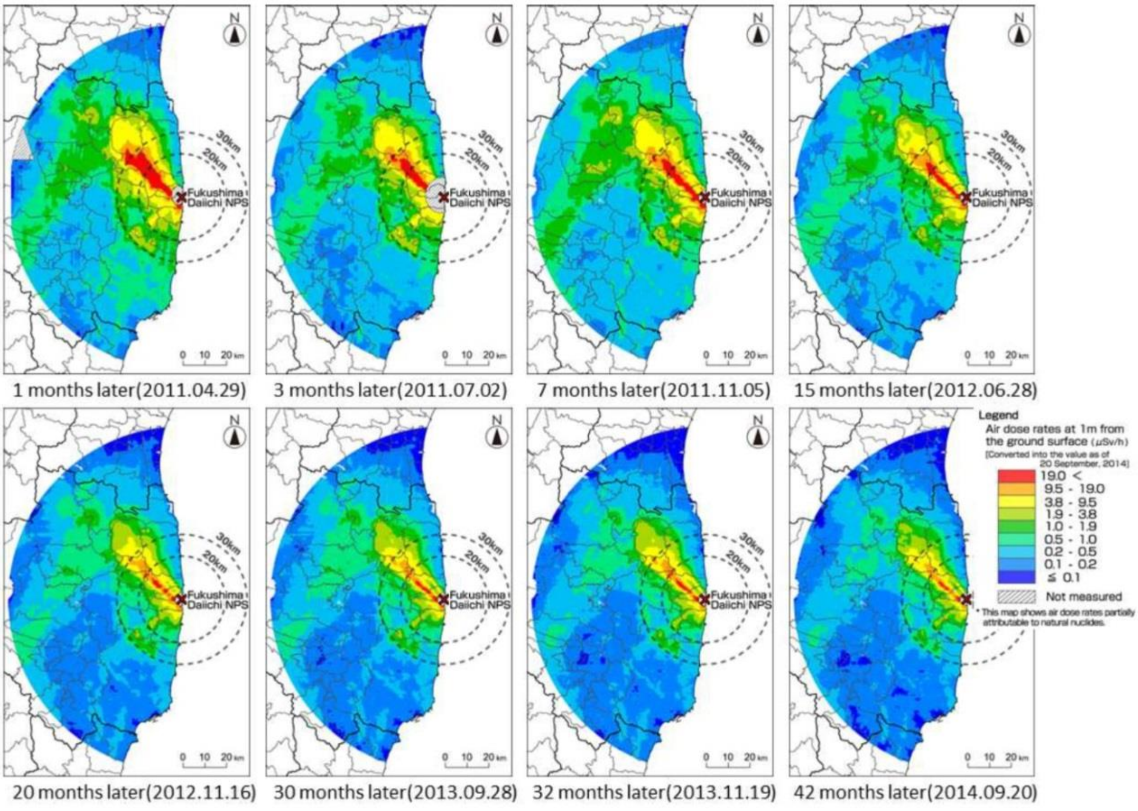

Figure 7: Air dose rate 1 m above ground for different reference dates at Fukushima area

3.3 Main differences to Chernobyl

Compared to the accident in Chernobyl, the total amount of air-borne releases of I-131 and Cs-137 is estimated to be in the order of about 10% of the radioactivity liberated in Chernobyl in the sense of an iodine equivalent. In addition, the composition of the radioactive substances released is also different. In Chernobyl, there was an uncontrolled increase in output in the reactor core, which led to the explosion of the reactor and to a subsequent multiple-day fire of the graphite moderator. As a result, parts of the fuel and thus also larger amounts of hardly volantile radioactive substances, such as plutonium and strontium, were spread into the environment. In addition, the fugitive radioactive substances were transported to high altitudes by a kind of chimney effect and were therefore carried away by the wind over long distances [3].

Different to Chernobyl, most of the releases in Fukushima were blown from the west to the Pacific Ocean due to the predominant airflow. As a consequence, the area around Chernobyl contaminated with radioactive substances is significantly larger and generally shows a higher air dose rate (Figure 7)

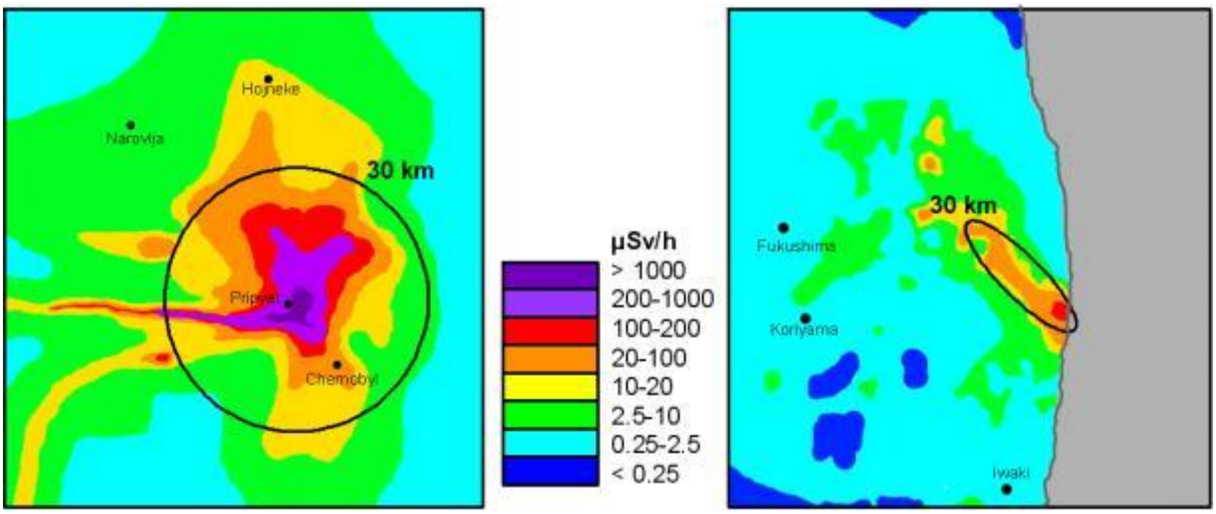

Figure 7. Comparison of the air dose rate 1 m above ground in the vicinity of Chernobyl and Fukushima, referring to about one month after the accidentsl (source: ENSI).

The radiation exposure of the personnel in Fukushima has been lower compared to Chernobyl. In the control of the fire and in the work to cover the open reactor core, the Chernobyl personnel employed were exposed to very high exposures, so that about 300 persons had to be taken to hospitals. Of these, 134 persons exhibited symptoms of acute radiation disease (eg. weakness, vomiting, dizziness). 28 people died of radiation illness despite intensive medical efforts. Up to 1998, further 11 persons died due to received doses between 1,3 Gy and 5,2 Gy.

According to a study of the World Health Organization published in February 2013, a total of seven persons of the Fukushima personnel died in the course of the accident, but the WHO considered that there was no correlation between cause of death and radiation exposure.

4 CONCLUSIONS

Due to the different courses of the accidents especially with the fire at Cherenobyl the source term of the released radioactivity was about ten times higher compared to that of Fukushima and consisted of hardly volantile radioactive substances, such as Plutonium, Uranium and Strontium thrown into hights of several thousend meters. The release lasted for about ten days at Cherenobyl and due to changes of the wind direction the dispersion of radioactive aerosols took place all over Europe whereas the releases at Fukushima lastet for essentially longer time and the most part was tranported with wind from West towards the Pacific Ocean in the first days. Only after changes of the wind direction from southeast the plum and wet precipitation reached a narrow area northwest of the Fukushima NPP. Qualitative countermeasure for the population (resettlement, restriction on residence, limitation of food production and consumption, decontamination of land and living areas etc.) are comparable. At the Cherenobyl territory outside of the direct Shelter site the groundwater meanwhile reached values comparable to the limits for drinking water. At Fukushima this problem is more crucial due to the big amount of sea water pumped into the destroyed units to cool down the molten fuel. At both sides the the upcoming removal and management of fuel containing materials will last for many years.

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Scientific and Technical Support to Technical and Emergency Center of Rostechnadzor

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Abstract

In Russian Federation Unified state system for prevention of and response on emergencies (which in terms of GSR part 7 is all hazard emergency management system) is established and currently functioning. To cope with nuclear and radiological emergencies there is two basic subsystems of Unified system. One of them is headed by Rosatom, which under this competence responsible for response to nuclear and radiological emergencies. Another functional subsystem, namely subsystem for control of radiation hazardous facilities, is headed by Rostechnadzor, who responsible for control of preparedness for response to radiological emergencies. Technical and Emergency Center of Rostechnadzor (Rostechnadzor TEC) is main analytical and information support tool of the Rostechnadzor subsystem, which shall be used as in real emergencies and in exercise mode for dose assessment; assessment and prognosis of integrity of physical barriers and perfomance of safety functions; control of compliance with safety regulations and emergency response plans and instructions; etc.

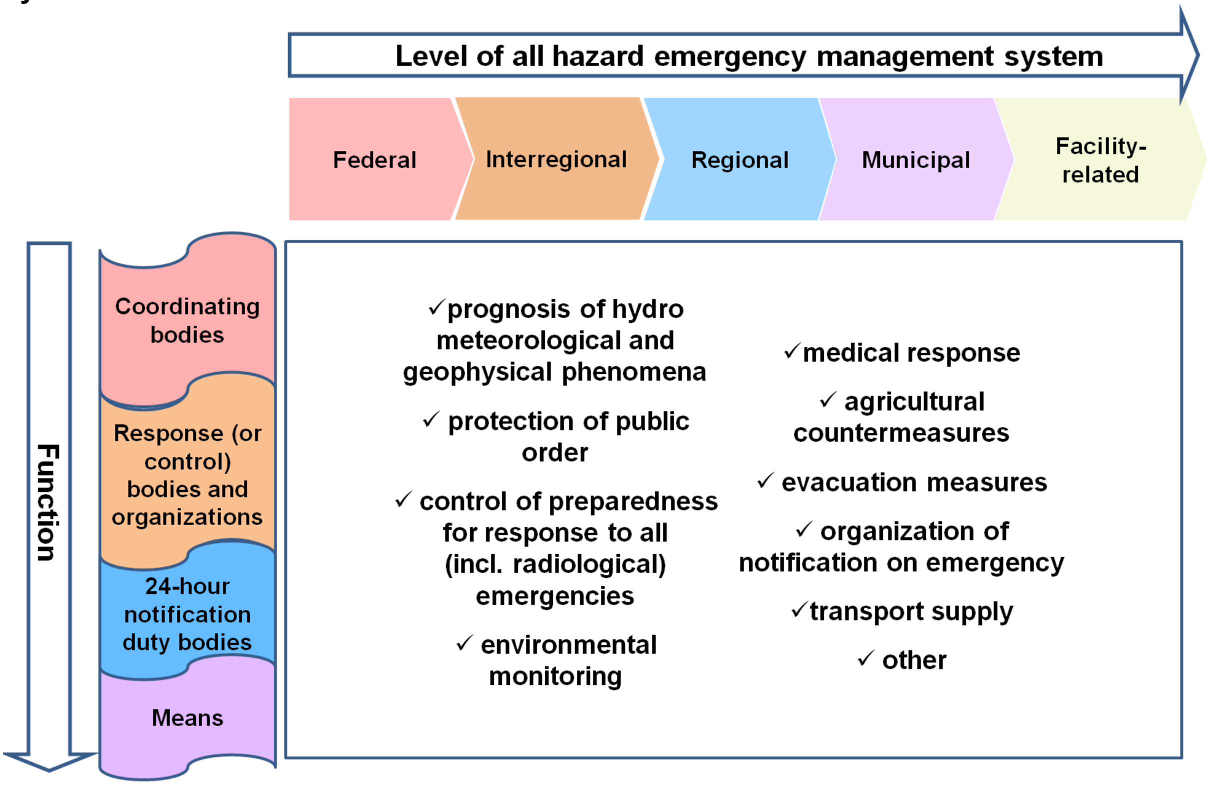
Currently Rostechnadzor TEC have capability to support the most of functions of subsystem for control of radiation hazardous facilities. The report contains the review of results of development of Rostechnadzor TEC, where SEC NRS was involved, and some view on further development of Rostechnadzor TEC.

1ROLE OF TECHNICAL AND EMERGENCY CENTER OF ROSTECHNADZOR IN SUPPORT OF ROSTECHNADZOR FUNCTIONAL SUBSYSTEM

Functional subsystem for control of radiation hazardous facilities [1] is part of Russian all hazard emergency management system [2], under which coordinating bodies, 24-hour notification duty bodies, means (i.e. tools, instruments, supplies, equipment, communication systems, facilities and documentation), related to federal and local authorities, involved as in emergency response and in control on it, are joined. Federal and local authorities under all hazard emergency management system have specific aims for protection of public order, prognosis of hydrometeorological and geophysical phenomena, supply of transport means, medical response, agricultural countermeasures; evacuation measures, etc. Organizational structure of all hazard emergency management system illustrated at figure 1. Russian all hazard emergency management system consists from functional and regional subsystems, which performs its functions on federal, interregional, regional, municipal and facility related levels. Each functional subsystem are chaired by state authority or organization, which esponsible for performance of mentioned aims (figure 1).

It should be noted that the most extensive coordinating power (in terms of inter-agency coordination) during the routine activity assigned to Government commission on prevention of and response on emergencies (hereinafter - the Government Commission) [3], which consists of about 38 representatives of the Government and executive authorities, including - a representative of Rostechnadzor. The Government Commission responsible for inter-agency coordination in case of emergencies as well.

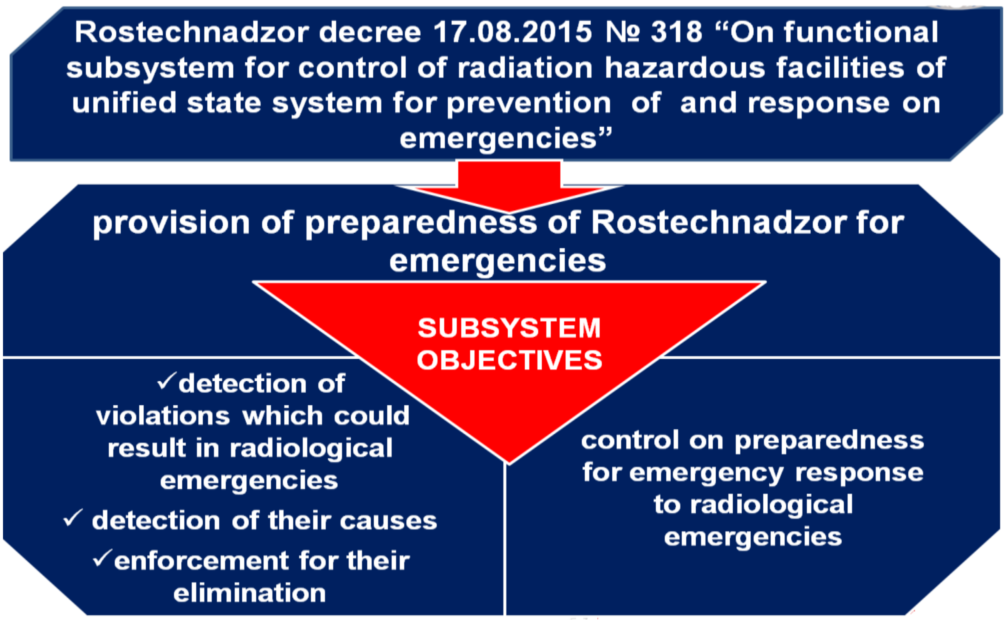
Figure 1: Illustration of structure of Russian all hazard emergency management system



Provision of Rostechnadzor representative in the Government Commission with the necessary information in case of radiological emergencies carried out in the framework of Rostechnadzor functional subsystem.

A crucial step in development of Rostechnadzor functional subsystem is the adoption of the provisions on this sub-system [1]. Figure 2 demonstrates objectives of subsystem, specified in [1].

Figure 2: Objectives of functional subsystem for control of radiation hazardous facilities

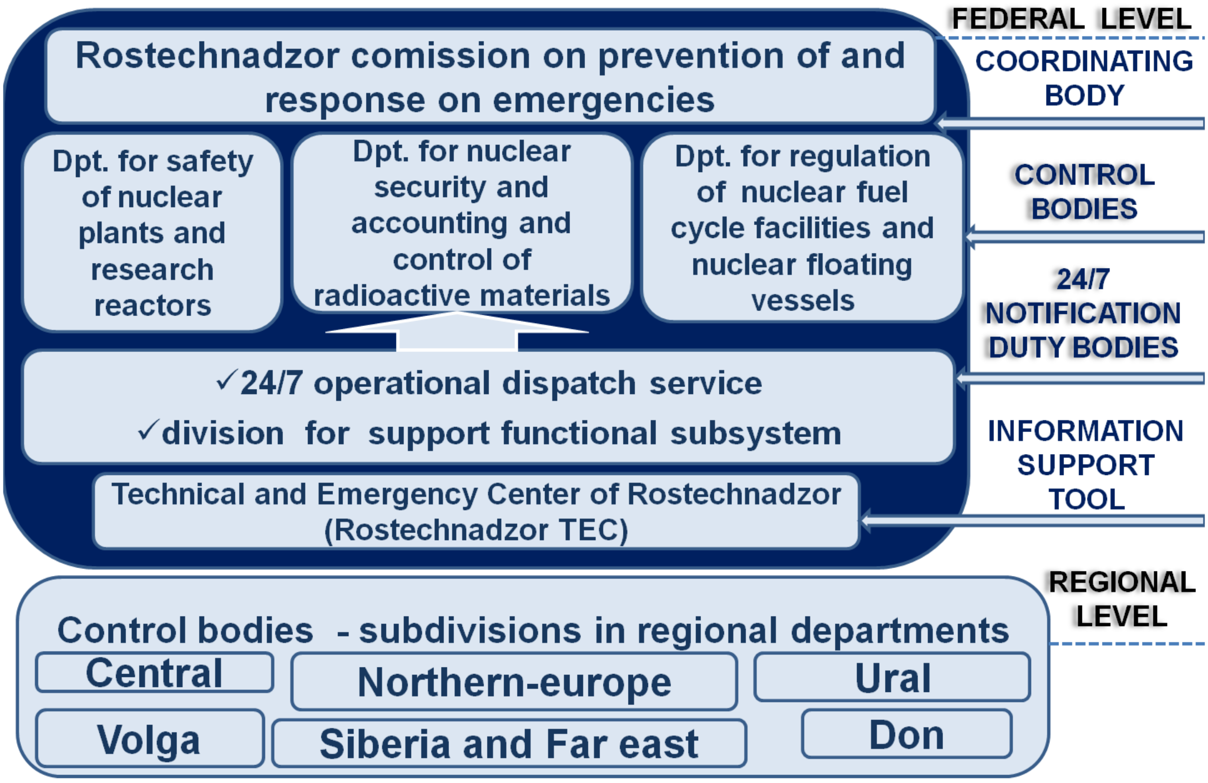


In [1] established two different forms of control of preparedness for response to radiological emergencies by Rostechnadzor, i.e. 1) control by means of inspections and 2) control by means of emergency exercises.

In accordance with [1] in both cases the control of emergency preparedness is done by means of control on compliance with federal norms and rules in the field of use of atomic energy, license conditions, and conditions of permissions to the facilities' workers.

Organizational structure of functional subsystem, which enacted by [1], demonstrated at figure 3.

Figure 3: Organizational structure of functional subsystem for control of radiation hazardous facilities



According to [4] state authorities, that responsible for control on situation on potentially hazardous facilities, have to inform local authorities on potential and current emergencies. In order to comply with that a special coordinating body - Rostechnadzor commission on prevention and response on emergencies is established in the framework of Rostechnadzor functional subsystem. The commission authorized to communicate with other federal authorities and with local authorities as well.

The Technical and Emergency Center of Rostechnadzor is the basic element of Rostechnadzor subsystem, which provides performance of subsystems' functions on communication, notification and analytical and information support. Its current capabilities and some view on future development are described below.

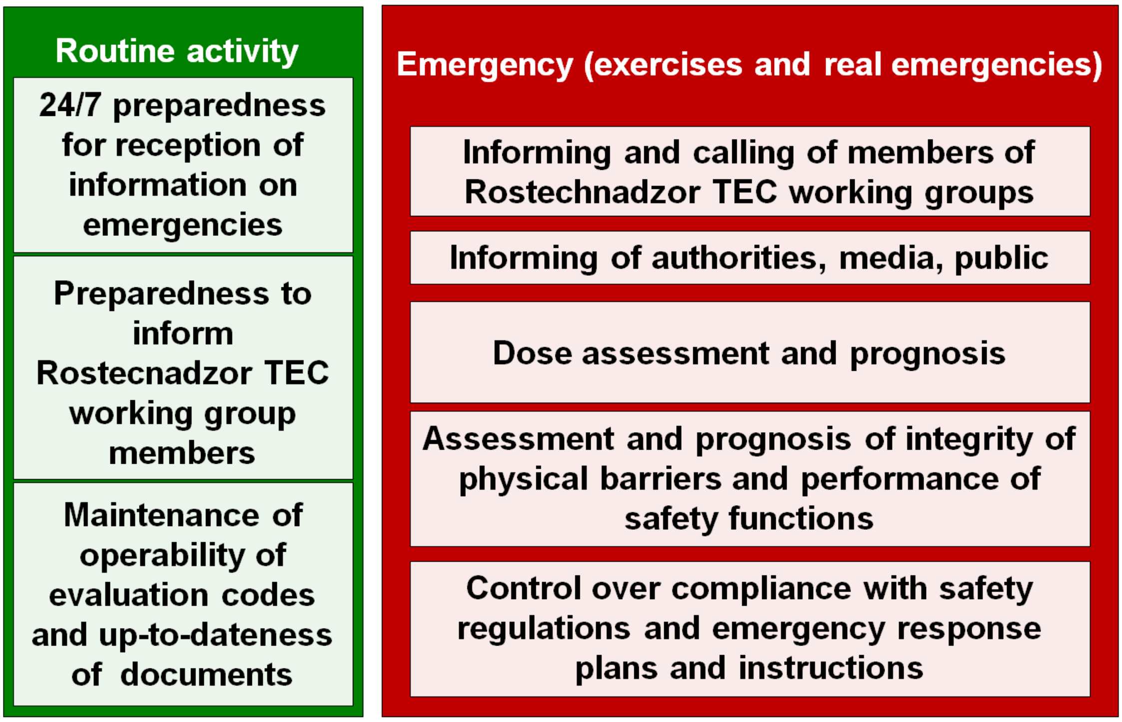
2CURRENT CAPABILITIES OF TECHNICAL AND EMERGENCY CENTER OF ROSTECHNADZOR AND THEIR FUTURE DEVELOPMENT

For purposes of analytical and information support of Rostechnadzor functional subsystem the Rostechnadzor TEC during its routine activity and in case of radiological emergencies (including emergency exercises) performs quite broad range of tasks, which illustrated at figure 4.

Tasks of Rostechnadzor TEC given in figure 4 are defined in Rostechnadzor document, which developed with the participation of SEC NRS experts.

Furthermore, a number of these tasks are performed with direct participation of SEC NRS.

Figure 4: Tasks of Rostechnadzor TEC



Thus, SEC NRS performs maintenance of operability of evaluation codes and models, and up-to-dateness of documents (i.e. emergency plans, guides, instructions etc.), which are used in Rostechnadzor TEC for control over compliance with safety regulations and emergency response plans and instructions. Informing and calling over members of working groups of Rostechnadzor TEC are arranged in case of emergencies and in case of emergency exercises. An organizational structure of Rostechnadzor TEC working groups during an emergency and emergency exercises is shown in Figure 5. SEC NRS are involved in activity of Rostechnadzor TEC working groups on dose assessment and on assessment and prognosis of integrity of physical barriers and performance of safety functions. Control over compliance with safety regulations and emergency response plans and instructions are carried out by both groups.

Management group (where representatives of SEC NRS management are included) is responsible for coordination of Rostechnadzor TEC working groups activities, summation on results of working groups activities, and for informing of Rostechnadzor representative in Government Commission.

Evaluation of compliance with safety regulations (federal norms and rules in the field of use of atomic energy) and with emergency response plans and instructions carried out on following crucial aspects of emergency response:

-activation of emergency response;

-notifying on emergency;

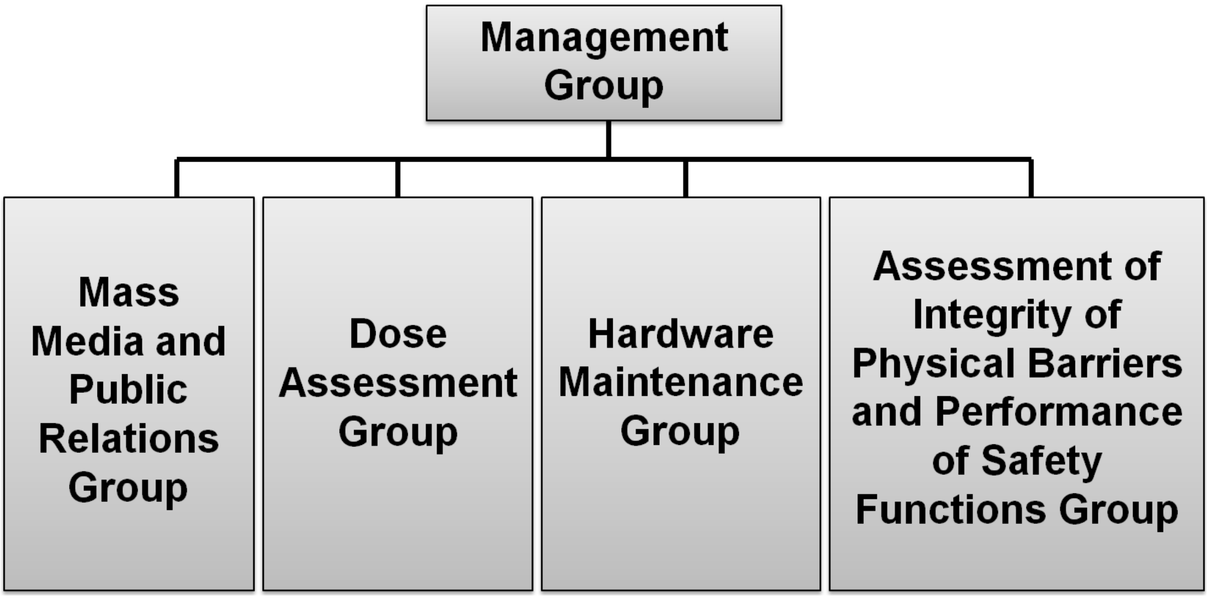
-assistance to operator in emergency response;

-analyzing the radiological emergency and the emergency response, investigation of causes of emergency.

As a methodological framework for the evaluation of exercises working groups are use SEC NRS developed document, which contains the requirements on all aspects mentioned above and presents guidelines for evaluating the exercises on NPPs. Upon completion of emergency exercise the management group, based on proposals of experts of Rostechnadzor TEC working groups and inspection departments, appoint recommendations on improvement of emergency preparedness to NPP operator.

Using of mentioned document in emergency exercises are noted as a good practice by IAEA experts according to the results of IAEA follow-up mission conducted at Rostechnadzor in November 2013 [5].

Figure 5: Diagram of the Rostechnadzor TEC working groups organizational structure



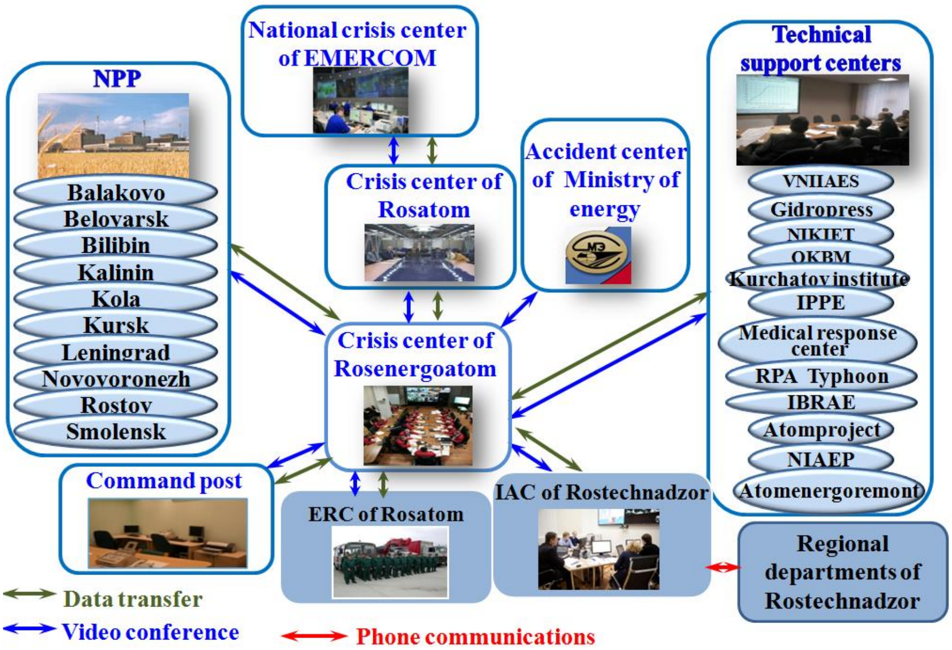
2.1Development of federal norms and rules in the field of use of atomic energy for purposes of improvement of information support of Technical and Emergency Center of Rostechnadzor

The improvement of information support of Rostechnadzor TEC is a crucial task. Significant progress on that have been reached in current year through adoption of federal norms and rules in the field of use of atomic energy “Regulations on activation of emergency response, notification on radiological emergency and on assistance to NPP operator in emergency response” (NP-005-16) [6]. NP-005-16 [6] supersedes a previous document on this matter NP-005-98, due to need for improvement of supply of Rostechnadzor TEC with information and to cope with suggestions of IAEA follow-up mission [5].

Changes on supply of Rostechnadzor TEC with information are introduced in NP-005-16 [6] in order to harmonize regulations on notification on emergency with IAEA safety standard GS-G-2.1 [7], which contain a much more stringent notification time objectives (within 15 minutes after classification of emergency) than that of NP-005-98 (1 hour). In order to account for best practice a requirement to notify authorities and organizations, involved in emergency response, and Rostechnadzor duty officers, responsible for 24/7 reception of emergency notifications, within 15 minutes after classification of emergency incuded to NP-005-16 [6].

There is one more improvement on supply of Rostechnadzor TEC with information, which, however, not connected with suggestions of IAEA follow-up mission [5]. Thus, NP-005-16 [6] contains a requirement according to which Technical and Emergency Center of Rosenergoatom must provide the functioning of the unified information system under which all authorities and organizations, involved in emergency response, are provided with real-time data on state of NPP units, results of process, source and environmental radiation monitoring and of monitoring of meteorological conditions. Scheme of unified information system illustrated at figure 6. This requirement of NP-005-16 [6] cope with requirement 5.17 of GSR part 7 [8]. According to this requirement for facilities in categories I arrangements shall be made to initiate a coordinated and preplanned off-site response, as appropriate, in accordance with the protection strategy. These arrangements shall include suitable means of communication between response organizations [8].

Figure 6: Unified information system



Under this requirement following data are transferred from Technical and Emergency Center of Rosenergoatom to Rostechnadzor TEC:

-dose rates on site and inside of buildings;

-activity concentrations of process streams;

-off site dose rates;

-non radiological process parameters.

Within the framework of development of federal norms and rules in the field of use of atomic energy a regulatory document [9] are developed (by revision of previous one) and adopted. In [9] with account of Fukushima lessons, a requirement that radiation monitoring systems are important for safety, included. As far as equipment for monitoring of off site dose rates wasn't related to important for safety before, the compliance with new requirement will allow authorities and organizations, involved in emergency response and Rostechnadzor TEC to obtain off-site dose rates even in case of external events.

2.2Improvement of assessment tools of Technical and Emergency Center of Rostechnadzor

Currently Rostechnadzor TEC provided with assessment tools shown in table 1. These tools are used by group on dose assessment and group on assessment of integrity of physical barriers and performance of safety functions for carrying out express assessments.

The Nostradamus tool [10] based on lagrangian type atmospheric dispersion model and used in Rostechnadzor TEC for dose assessment and for evaluation of scale of necessary off site protective actions. The Cassandra tool [11] utilize models that account for dispersion as accidental liquid releases to water bodies and depositions from airborne plumes on water body surface (and on its catchment area).

The integrity of physical barriers and performance of safety functions in Rostechnadzor TEC is currently assessed for NPPs with WWER type reactors by modules MVTU, TPP, Rainbow-TPP and Integr of RADUGA-EU [12]. In order to make fast running assessments following features are utilized in RADUGA-EU:

-simulation of basic safety systems and systems of reactor and turbine system important for safety only;

-core thermohydrodynamics are modelled as a few equivalent fuel assemblies ducts and one coolant duct;

-zero dimensional neutronics model.

Table 1: Assessment tools used by working groups of Rostechnadzor TEC

Name of tool	Purpose
Tools used by group on dose assessment	
Nostradamus	Dose assessment due to accidental airborne releases
Cassandra	Dose assessment due to accidental waterborne releases
Tools used by group on assessment of integrity of physical barriers and performance of safety functions	
RADUGA-EU	Assessment of integrity of physical barriers and performance of safety functions before fuel degradation
Auxilliary assessment tools	
SCALE	Core inventory calculations

Currently SEC NRS have developed models of following WWER NPP operated units:

- Balakovo NPP (units 1 - 4);
- Kalinin NPP (units 1 - 4);
- Rostov NPP (units 1 - 3);
- Novovoronezh NPP-1 (units 3 - 5);
- Kola NPP (units 1 - 4).

Also the models of unit 1 (under commissioning) and unit 2 (under construction) of Novovoronezh NPP-2 was developed by SEC NRS.

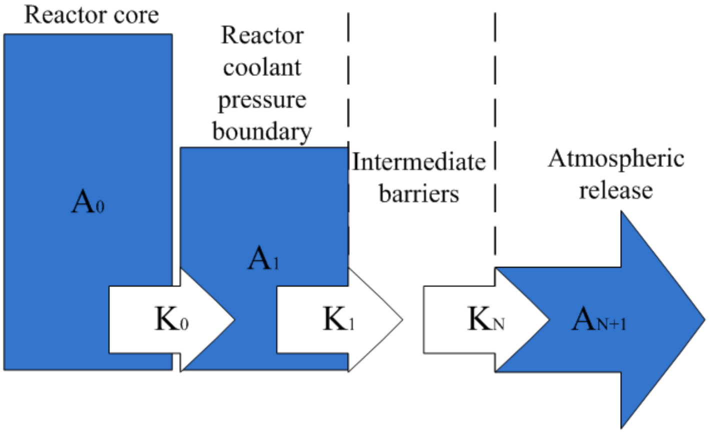
For assessment of airborne radiological releases from NPP accidents the methodology for generic assessment [13], [14], [15] are used. This methodology based on assumption that released activity of specific radionuclide i (A^i) is directly proportional to its activity located within the first physical barrier ($A^0{}_i$). The first physical barrier depends on time phase of accident. Thus, it could be: 1) reactor coolant pressure boundary, when radiological release are primarily defined by coolant activity; 2) fuel cladding, when release are primarily defined by gap activity; 3) fuel matrix. Generally released activity A^i are calculated with following expression:

$$A_i = A_i^0 \cdot k_0^i \cdot k_1^i \cdot k_2^i \cdot \ldots \cdot k_N^i$$

where $k^i_0 \ldots k^i_N$ – factors (specific for each release pathway) that charcterize fraction of activity which not retained within specific physical barrier. Illustration of that approach given on figure 7.

The similar methodology have been developed in SEC NRS for RBMK-1000 reactors. This methodology is currently tested in emergency exercises.

Figure 7: Illustration of the approach for assessment of airborne radiological releases from NPP accidents used by dose assessment group



A computer tools implementing mentioned methodologies are now being developed.

2.3Improvement of Rostechnadzor TEC

In year 2013 in order to improve provision of Rostechnadzor TEC with information and to facilitate of working groups activities Rostechnadzor TEC was provided with equipment for:

-displaying (figure 8);

-selector communication (figure 9);

-wireless control of equipment of Rostechnadzor TEC (figure 9);

-audio gain;

-videoconferencing;

-audio and video record;

-commutation.

The displaying in rooms of management group, group on dose assessment and group on assessment of integrity of physical barriers and performance of safety functions are performed based on two (per each room) LCDs. Each couple of LCDs is supplied with multipoint video conferencing codec and surveillance camera.

Figure 8: Illustration of displaying equipment used in Rostechnadzor TEC

The system of wireless control of equipment of Rostechnadzor TEC is performed based on software Crestron, installed on iPad located in each room.

Figure 9: Illustration of wireless control of equipment of Rostechnadzor TEC

In order to improve the process of notification of Rostechnadzor TEC working group members a system of automated notification (voice message, SMS message and e-mail message) of Rostechnadzor TEC working group members is being selected.

3CONCLUSIONS

Paper contain the summary of activity on scientific and technical support to Technical and Emergency Center of Rostechnadzor.

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Health impact assessment of recovery/disposal options of sewage sludge: methodology and critical parameters

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Abstract:

Health impact of artificial radioactive materials released in sewers is a more and more important topic for the stakeholders of sewers and/or wastewater treatment plants (WWTPs). Many models have been developed to describe the behaviour and to assess the health impact of radionuclides in sewers and in WWTPs. Some models enable also to assess health impact of land application and/or landfill disposal, and sometimes sludge incineration. But the application of these models to the French WWTPs is not straightforward because of the diversity and specificities of the recovery options encountered in France. So the aim of IRSN is to develop a comprehensive methodology to assess the health impact of each type of recovery/disposal option taking into account all potentially exposed persons (workers and public) and all exposure pathways.

1 CONTEXT

The need to apply innovative technologies for maximizing the efficiency of sewage treatment plants and minimizing their carbon footprint has the consequence for sewage sludge management to be now a highly sophisticated research and development sector. Sewage sludge are not to be regarded solely as 'waste'; it is also a renewable resource for energy and material production. In France, the following recovery options for sewage sludge are:

- energy recovery: incineration, gasification, pyrolysis…,
- agriculture uses: farmland application, composting…,
- manufacturing building materials.

In France, waste water treatment plants (WWTPs) often operate several recovery options.

Health impact of artificial radioactive materials released in sewers is a more and more important topic for the stakeholders of sewers and/or WWTPs (owners and operators, national or local authorities), mainly due to high activity released in sewers by hospitals or increasing number refusals of solid wastes in landfill following detection of radioactivity. Several models in different countries have been developed to describe the behaviour and to assess the health impact of radionuclides in sewers and in WWTPs. Some of them enable also to assess health impact of land application and/or landfill disposal, and sometimes sludge incineration. For these three recovery/disposal options however, the above models allow to assess exposure for only some potentially exposed groups.

Since 2000, the Institute for Radiological Protection and Nuclear Safety (IRSN, the French national public expert in nuclear and radiological risks) is more and more involved in studies concerning radionuclides in sewers and WWTPs. The main purpose is the radiological characterisation of wastewater and/or sludge, sometimes associated with workers exposure assessment; and for a few years now, regulators are often interested by the health impact on the population due to sludge recovery.

Lastly, to answer to some stakeholder expectations (especially hospitals, that are responsible for the main releases of radionuclides in sewers), IRSN has developed a generic method to assess radiological exposure of a sewer worker and of a WWTP worker working near sludge storage. This method considers also workers involved in sewage sludge loading and transport operations or in land application. But, it doesn't allow the dose assessment for all sewage sludge recovery options and for all exposed persons, and, as any generic method, hypotheses are very – and even too – conservative.

Finally, to respond to all possible requests of any stakeholder of sewers or WWTPs and to be consistent with its mission, IRSN needs a comprehensive method to assess exposure due to radionuclides in sewage or in sludge. The present paper focuses on the health impact assessment of recovery/disposal options of sewage sludge.

2 OVERVIEW OF WORLDWIDE ASSESSMENT MODELS

Among the models mentioned above and dealing with assessment of worker exposure, some permit also to assess exposure of the population due to some sludge recovery options and are presented below.

2.1 United States of America

A limited survey of radioactivity in sewage sludge was conducted by the Interagency steering committee on radiation standards (ISCORS) across the United States. To assess the levels of the associated doses to people, ISCORS modelled the transport of the relevant radionuclides from sewage sludge into the local environment [1]. Seven general scenarios were considered to represent typical situations in which members of the public or WWTP workers may be exposed to sewage sludge:

- residents of a house built on a former agricultural field where sewage sludge was applied,
- recreational visitors to a park where sewage sludge has been used in land reclamation,
- residents of a town near a sewage sludge land-application site,
- neighbours of a landfill that contains sewage sludge or ash from sludge,
- neighbours of an operating sludge incinerator,
- workers who operate equipment to apply sewage sludge to agricultural lands,
- WWTP workers involved in sewage sludge sampling, transport or loading operations.

2.2 United-Kingdom

The study [2] aims to provide relevant information for the Environment agency to review the acceptability of releases of liquid radioactive wastes to sewer systems. Scenarios are:

- neighbours of a sludge incinerator,
- neighbours of a landfill that contains sewage sludge or ash,
- disposal of sludge off-shore,
- application of sludge to farmland.

2.3 Sweden

The Swedish study [3] assesses the doses to the public and sewage workers in order to provide supporting information to be used for regulation revision. Scenarios are:

- consumers (adults) of food produced in agricultural land where sludge has been used as fertilizer,

- consumers (adults) of food produced in a landfill where sludge has been disposed of and which has been used for agriculture after its closure.

2.4 Conclusions

These models consider all exposure pathways for their different scenarios; however, for residents and consumers, adults are the only considered age group. Furthermore, these models take into account recovery options which are in a strong accordance with their national regulations. Some of these recovery options are not allowed in France and some of the hypothesis which are retained for the recovery options do not comply with the French regulation. This is why IRSN decided to develop its own exhaustive model, taking into account all potentially exposed persons (workers and public) and all exposure pathways, to respond to all public expectations.

3 METHODOLOGY USED BY IRSN

3.1 Methodology

In France, several recovery options for sewage sludge are encountered: energy, agriculture and building materials; when sludge can't be reused, they are disposed in a landfill. Tables 1, 2 and 3 present, according to recovery option, potentially exposed persons and exposure pathways to be considered.

Table 1: Energy recovery - Potentially exposed persons and exposure pathways

Options	Persons	Pathways
Incineration in the WWTP	Patrolman	Irradiation by sludge or ashes Inhalation of resuspended dust Inadvertent ingestion of sludge or ashes
	Residents	Irradiation by plume and deposit Inhalation of the plume and resuspended dust from deposit
	Consumers	Ingestion of vegetables, meat from contaminated surfaces Inadvertent ingestion of soil
Co-incineration in a cement facility	Driver Patrolman	Irradiation by sludge or cement Inhalation of resuspended dust Inadvertent ingestion of sludge or cement
	Residents	Irradiation by plume and deposit Inhalation of the plume and resuspended dust from deposit
	Consumers	Ingestion of vegetables, meat from contaminated surfaces Inadvertent ingestion of soil
Co-incineration in a coal-fired plant or Co-incineration with household refuse	Driver Patrolman Boilermaker Electrician Heap agent	Irradiation by sludge or ashes Inhalation of resuspended dust Inadvertent ingestion of sludge or ashes
	Residents	Irradiation by plume and deposit Inhalation of the plume and resuspended dust from deposit
	Consumers	Ingestion of vegetables, meat from contaminated surfaces Inadvertent ingestion of soil
Wet air oxidation	Driver Patrolman	Irradiation by sludge or sand Inhalation of resuspended dust Inadvertent ingestion of sludge or sand
Gasification	Driver Patrolman	Irradiation by sludge or ashes Inhalation of resuspended dust Inadvertent ingestion of sludge or ashes
	Residents	Irradiation by plume and deposit Inhalation of the plume and resuspended dust from deposit
	Consumers	Ingestion of vegetables, meat from contaminated surfaces Inadvertent ingestion of soil

Table 2: Agricultural recovery - Potentially exposed persons and exposure pathways

Options	Persons	Pathways
Farmland application with or without a storage platform	Driver Platform agent (if storage platform) Farmer	Irradiation by sludge Inhalation of resuspended dust Inadvertent ingestion of sludge
	Residents	Irradiation by sludge Inhalation of resuspended dust
	Consumers	Ingestion of vegetables, meat from contaminated surfaces Inadvertent ingestion of soil
Composting and amendment (agriculture or garden)	Driver Platform agent Farmer (if agricultural amendment)	Irradiation by sludge Inhalation of resuspended dust Inadvertent ingestion of sludge
	Residents	Irradiation by sludge Inhalation of resuspended dust
	Consumers	Ingestion of vegetables, meat from contaminated surfaces Inadvertent ingestion of soil
Mulching	Driver Farmer	Irradiation by sludge Inhalation of resuspended dust Inadvertent ingestion of sludge
	Residents	Irradiation by sludge Inhalation of resuspended dust
	Consumers	Ingestion of vegetables, meat from contaminated surfaces Inadvertent ingestion of soil

Table 3: Building materials recovery - Potentially exposed persons and exposure pathways

Options	Persons	Pathways
Concrete Bricks Ceramics	Drivers Building materials maker	Irradiation by sludge or building materials Inhalation of resuspended dust Inadvertent ingestion
	Building materials user Residents	Irradiation by building materials

Table 4 presents the potentially exposed persons and the exposure pathways for disposal in a landfill.

Table 4: Disposal in a landfill - Potentially exposed persons and exposure pathways

Persons	Pathways
Driver Platform agent	Irradiation by sludge Inhalation of resuspended dust Inadvertent ingestion of sludge
Residents	Irradiation by resuspended dust Inhalation of resuspended dust

Some of these scenarios can be mixed; for instance a patrolman can be a resident and a consumer.

Performing so many assessments requires a lot of information on workers: different operations achieved by each worker, annual duration of exposure for each operation, workers' positions relative to each source, geometry and composition of each source, *etc.*. Information can be obtained from visit of the different places (plant(s), platform(s), landfill(s)), literature and essentially workers interviews.

For population, the main information to acquire are: location of the exposed persons, age groups, food consumption for each age group, weather conditions distribution in case of atmospheric releases.

The model considers the radioactive decay for long-term operations (for instance composting process) or operations taking place a long time after the sludge production (for instance agricultural activities in a field where sewage sludge was applied) , the partition of the radionuclides between ashes and released plume in case of sludge incineration, the dilution of sludge or ashes by other products (for instance in the case of co-incineration in cement facility, concrete production or composting).

In case of lack of specific information (for instance partition factor of radionuclides between ashes and released plume, dimensions of the radioactive sources for some operations, weather conditions, food consumption…), the assessor will retain conservative hypothesis (investigation of literature data, regulatory recommendations in some cases…).

This methodology can appear very - even too - exhaustive but radiological impact, even for small sources and low levels of exposure, can be a huge societal issue; each potentially exposed person wishes to know her own health impact.

3.2 Critical parameters

Many parameters have influence on dose assessment. However the parameters related to the source term (activity of sludge, ash, by-products of flue gas cleaning systems, compost…) are the most critical ones because their influence is linear (partition factor for example) or exponential (duration for storage on a platform for example) on the dose assessed for some or all pathways. These parameters must be considered with great attention, especially:

- the fraction of the radionuclide that is not vented as part of the exhaust gas stream. It depends on the plant design, the element and its chemical form. It can range from 0.0 for noble gases such as radon to greater than 0.99 for metals such as uranium, thorium and plutonium;

- the dilution of sewage sludge by municipal solid waste for the incineration or by green waste for composting;
- the dilution of ashes in concrete or in building materials.

Values of these parameters are in a wide range or are not well-defined in the literature; the choice of relevant values of these parameters must be done with judgement or caution.

The operational times are also important parameters. Most of radionuclides released in sewers are short-lived radionuclides; so the radioactive decay impacts significantly the exposure.

For the exposure assessment due to atmospheric releases, the conditions - such height and surrounding buildings - and the distribution of weather conditions influence significantly the assessed doses.

For external exposure assessment, the knowledge of nature, thickness and density of materials between sludge and the exposed person is essential to assess reasonably the dose.

4 AS APPLICATION: THE MORE EXPOSED PERSONS

In order to show the relevance to look for the more exposed persons among all potentially exposed persons taking into account all exposure pathways, and also to show the importance of one of the critical parameters listed above – radioactive half-life, an arbitrary example is presented below with iodine 131 and caesium 137 for building materials recovery options (concrete, bricks and ceramics).

Iodine 131 is chosen because it is the most frequently measured radionuclide in the sewers, mainly due to nuclear medicine department releases or patients excreta; it is a short-lived radionuclide with a half-life of 8 days. Inversely, caesium 137 is rarely measured in sewers, but can appear in the authorized releases in sewers for some research centres; half-life of caesium 137 is 30 years, much more than iodine 131. Moreover, iodine 131 and caesium 137 are rather similar from the only radiological point of view; the dose coefficient for external exposure of caesium 137 are 50% more than those for iodine 131, and it is the inverse for internal exposure.

To compare the radiological impact of different building materials recovery options for iodine 131 or caesium 137 contaminated sludge, basic hypothesis are the same for the two radionuclides (1 $Bq.g^{-1}$ of iodine or caesium in sewage sludge, sources dimensions except for the room model, transport times, positions in regard to sources, time budget,…). As for iodine 131, 1 $Bq.g^{-1}$ is a relatively high value of mass activity in sludge; the only radiological characterization of sludge known up to now by IRSN for a WWTP downstream nuclear medicine department showed a maximal mass activity around 0.37 $Bq.g^{-1}$.

Table 5 presents the hypothesis for the room model taking into account the type of building materials.

Table 5: Hypothesis for the room model

Parameters	Concrete	Bricks	Ceramics
Dimension of the model room	4 m x 5 m x 2.8 m [4]		
Room structures causing irradiation	Walls Floor Ceiling	Walls	Walls
Thickness	20 cm	7 cm	3 cm
Density	2.35	1.2	0.5

Figure 1, 2 and 3 present a comparison of effective doses, calculated respectively for iodine 131 and for caesium 137, for the main recovery options used in building industries (concrete, bricks and ceramics).

Figure 1: Annual effective doses ($mSv.y^{-1}$) for concrete recovery options

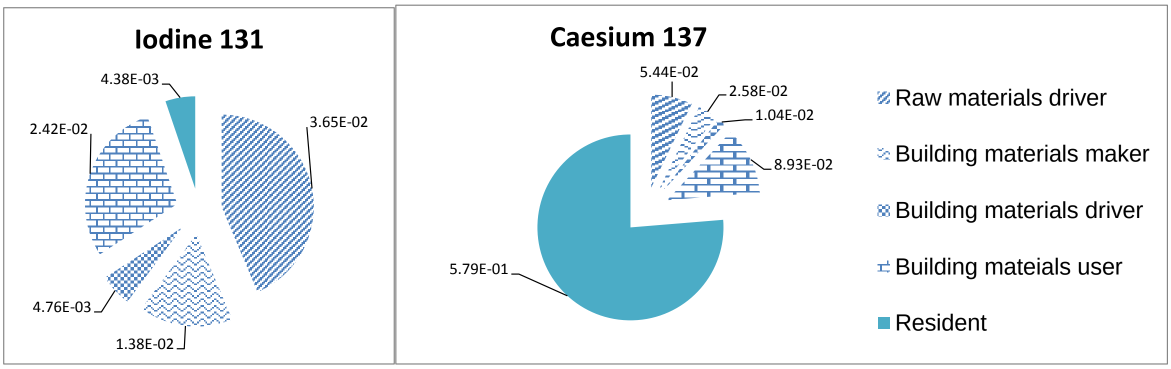


Figure 2: Annual effective doses ($mSv.y^{-1}$) for bricks recovery options

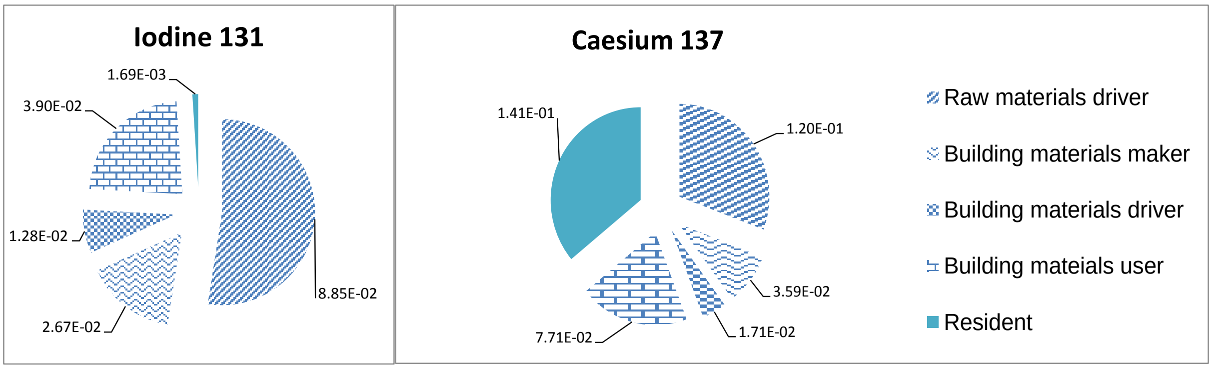
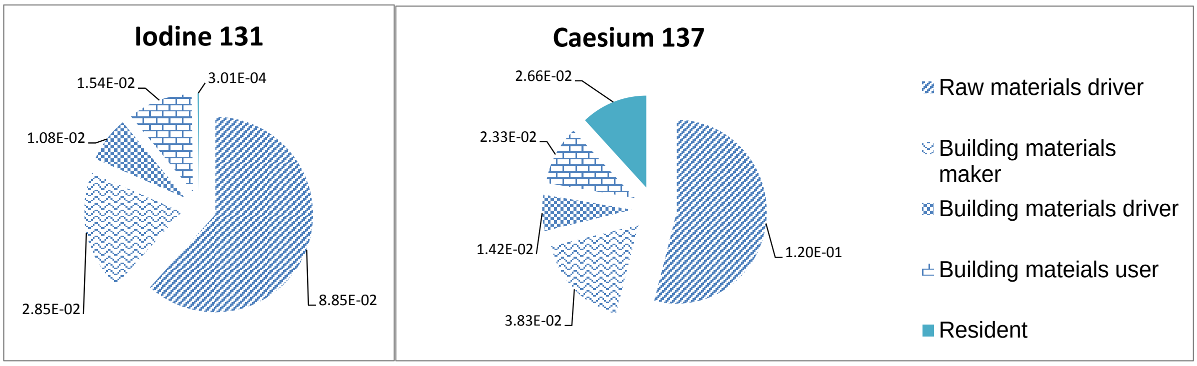


Figure 3: Annual effective doses (mSv.y^{-1}) for ceramics recovery options



Beyond the observation that effective doses for caesium 137 are higher – and sometimes much more than 50% - than those for iodine 131, it is interesting to note that the most exposed persons depend on the radionuclide. In particular, residents are not systematically the most exposed persons. The effect of radioactive decay is obvious: a longer half-life allows pathways concerned by longer duration (because linked to occupation) to become significant; in such cases residents can be the more exposed persons. Inversely, a short half-life doesn't lead these pathways to be significant, and the most exposed person can be a driver or a building materials maker. As the most exposed persons cannot be predicted *a priori*, it is necessary to assess doses for all persons potentially exposed.

5 CONCLUSION

Exposure of persons due to radionuclides in sewage, and subsequently in WWTPs and sludge, is increasingly a subject of interest for many people (sewage network or WWTPs owners or operators, national or local authorities, residents…). IRSN is now often questioned about radiological characterization of sewage or sludge and also about health impact on workers or persons of the general population potentially exposed due to all possible fates of artificial radionuclides in sewage. In order to achieve its mission with very various requests and configurations, and because the more exposed persons cannot be identified *a priori* as demonstrated before, IRSN develops a reliable and comprehensive method to assess doses for all possible persons exposed to radionuclides present in sewage or in sludge. This detailed investigation of actual exposure conditions provides robustness and credibility of IRSN assessments for stakeholders, especially in case of societal issues. Still in development, this method will soon be used in a systematic way for any sewer or WWTP, allowing a global approach for exposure due to artificial radionuclides in sewage.

6 REFERENCES

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Review of Current Q System and the A_1/A_2 Values of the IAEA Transport Regulation

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Abstract:

The current Q system of the Transport Regulations published by the International Atomic Energy Agency (IAEA) was developed in the 1980s for calculation of A_1/A_2 values, i.e. the activity limits for type A packages. Over the years, the need for some additional A_1/A_2 values for nuclides not listed in the IAEA Transport Regulations SSR-6 came up. Therefore, the German Federal Office for Radiation Protection (BfS) and the German Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (BMUB) granted a research project with the objective to analyse the methods used in the current Q system and to establish a program for the calculation of *Q* and A_1/A_2 values.

The calculation tool Ber*QA*Trans enables not only to recalculate already known *A* values for nuclides listed in the IAEA Transport Regulations SSR-6, but also to determine the *Q* and *A* values for new radionuclides. The recalculation results of Ber*QA*Trans are in good agreement with the Transport Regulations SSR-6 for most of the *A* values. Furthermore, it is possible to recalculate *Q* and *A* values not even on the up to now used older data basis of ICRP publication 38 but also by using recent nuclide data presented in ICRP publication 107. Also newer dose rate coefficients published by International Commission on Radiological Protection (ICRP) can be used.

During the development of the calculation program Ber*QA*Trans many lacks and inconsistencies in the documentation and problematic issues of the current Q system were identified and are briefly discussed in this paper.

Other institutions made similar approaches to analyse and/or revise the Q system. In 2013, the work of these groups was also recognized by TRANSSC (Transport Safety Standards Committee) members. To gather their work an international working group was founded in Cologne. This Working Group on review of A_1 and A_2 Values for the IAEA Transport Regulations had several meetings with the aim of a comprehensive review and revision of the current Q system. First results and proposals were presented to TRANSSC in June and September 2015.

1 INTRODUCTION

The current Q system of the IAEA Transport Regulations was developed in the 1980s for calculation of A_1/A_2 values, i.e. the activity limits for type A packages /MAC 86/. The system was integrated into the IAEA Transport Regulations in 1985, superseding the previous A_1/A_2 system of 1973, with a comprehensive revision in the 1990s.

The Q system is based amongst others upon following dose criteria: the effective dose to any person (member of the public or occupationally exposed person) should not exceed 50 mSv and the equivalent dose to skin should not exceed 500 mSv /IAEA 14/. The value of 50 mSv is based on the formerly valid limit of the effective dose for a single year for occupationally exposed persons. In the current Q system described in Appendix I in the Advisory Material

SSG-26 /IAEA 14/ five (main) exposure pathways are taken into account (see Figure 1). For these pathways nuclide specific activity limits (so called *Q* values) are calculated:

- external photon dose (Q_A),
- external beta dose (Q_B),
- inhalation dose (Q_C),
- skin dose and ingestion dose due to contamination transfer (Q_D), and
- submersion dose (Q_E).

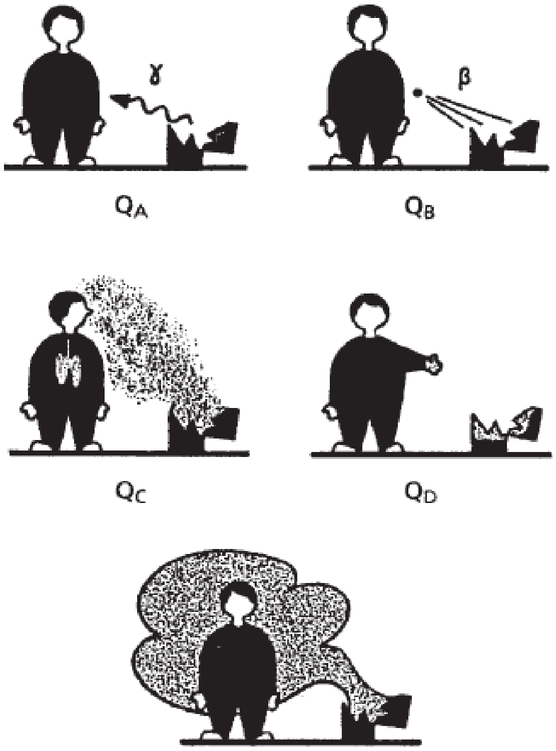

Figure 1: Exposure pathways in the Q system (picture taken from: /IAEA 08/):

In addition, there is a sixth pathway for alpha emitters; this dose value is named Q_F and represents an inhalation dose. All of these *Q* values are considered separately, i.e. it is possible to have in each of these pathways the maximum dose (50 mSv or 500 mSv, respectively). As this is only true for a relatively small number of nuclides, this method is retained /IAEA 14/.

The resulting *A* values for the IAEA Transport Regulations /IAEA 12/ are calculated as follows: The A_1 value (for special form material) is the minimum value of Q_A and Q_B, and (if applicable) Q_F. The A_2 value (for non-special form material) is the minimum value of A_1, and Q_D or Q_E, respectively. There are additional assumptions and calculation rules for deriving *Q* and *A* values according to the Q system, which are not mentioned here in this short description of the system. Even in the Advisory Material /IAEA 14/ not all necessary assumptions are documented /BÜT 14/, /JON 11/.

2 CALCULATION TOOL BERQATRANS

Over the years, in Germany as well as in other countries the need for some additional *A* values for nuclides not listed in the IAEA Transport Regulations came up. Therefore, the German Federal Office for Radiation Protection (BfS) and the German Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety (BMUB) granted a research project with the objective to analyse the methods used in the current Q system, and to establish a PC program for the calculation of *Q* and *A* values.

Therefore, GRS had to analyse the whole Q system in detail. Amongst many other publications (often from the 1970s and 1980s), which are the bases of the current *Q* and *A* values two publications where important: The report of the National Agency for Environmental Protection (ANPA) written in 1994 /BEN 94/, in which new calculation methods for Q_A and Q_B values were laid down; these newly calculated *Q* and *A* values were afterwards introduced into the Transport Regulations in the 1990s. And the report by Health Protection Agency (HPA) of 2011 /JON 11/, which is also a review of the Q system and its calculation methodologies in order to create the calculation tool SEAL /HPA 10/ for calculating *Q* and *A* values, and exemption values according to the current Q system.

The GRS calculation tool Ber*QA*Trans was designed not only to (re-)calculate existing *Q* and *A* values, but also to calculate new values for nuclides not listed in /IAEA 12/ or /IAEA 14/, and to have the ability to introduce new nuclide data /BÜT 14/. The current Q system is based upon rather old data, e.g. ICRP 38 /ICRP 83/, ICRP 51 /ICRP 87/ or ICRP 68 /ICRP 94/. Meanwhile, newer data are available, and therefore partly integrated in Ber*QA*Trans, too.

Ber*QA*Trans is a Microsoft Excel program written in VBA code. It has many options for calculations, e.g. to use newer nuclide data, or to variate dose conversion factors. In /IAEA 12/ values for 373 nuclides are listed, with Ber*QA*Trans it is possible to fully calculate *Q* and *A* values for 768 nuclides, using current calculation methods of the Q system. Results can be shown listed according to the rules mentioned in /IAEA 14/, but it is also possible to show every *Q* value calculable with given data for a nuclide. Also a benefit of Ber*QA*Trans is, that contrary to /IAEA 14/ in the result tables Q_A and Q_F are not shown in a common column, but separately; so for every alpha emitter it is possible to see both simultaneously, the *Q* values for photon dose and for alpha dose. A more detailed explanation of Ber*QA*Trans and the development of its calculation methods can be found in /BÜT 14/.

3 FINDINGS AND RESULTS

3.1 Recalculation of *Q* and *A* values with Ber*QA*Trans

Ber*QA*Trans was used to recalculate the entire *Q* and *A* values as well as the dose rate coefficients $\dot{e}_{\mathrm{pt}}$, $\dot{e}_{\beta}$, and $\dot{h}_{\mathrm{skin}}$ for Q_A, Q_B, and Q_D listed in /IAEA 08/ using current calculation methods and data of the current Q system. Comprehensive tables with results comparing to /IAEA 08/ can be found in the Annex of the GRS report GRS-343 /BÜT 14/. These *Q* and *A* values calculated with Ber*QA*Trans are in good agreement with tabulated values of /IAEA 08/. However, resulting *A* values for eight nuclides (^{26}Al, ^{47}Ca, ^{166}Dy, ^{202}Pb, ^{225}Ra, ^{92}Sr, ^{96m}Tc, and ^{231}Th; see Table 1) showed a higher deviation from values of the current *Q* system by a factor of more than two /BÜT 14/. Deviations of *A* values, e.g. for ^{26}Al, ^{47}Ca, ^{166}Dy, ^{225}Ra, ^{92}Sr, and ^{231}Th are documented in /JON 11/ as well. Reasons for discrepancies are for example different *Q* values restricting the corresponding *A* value due to used calculation methods, or different treatment of progeny.

Table 1: Calculated nuclides with higher deviation from values of the current Q system

Nuclide	Remarks
^{26}Al	Q_B value lesser than in [2]; therefore, Q_A value restricts A_1/A_2 values
^{47}Ca	Q_A and Q_B values lesser than in [2]; now Q_B values restricts A_1 value
^{166}Dy	Q_B value lesser than in [2]; therefore, A_1 value lesser too
^{202}Pb	Q_D value larger than in [2] and “unlimited”; therefore A_1/A_2 values “unlimited” too
^{225}Ra	Q_B value and Q_C value larger than in [2]; therefore, A_1 value and A_2 value higher
^{92}Sr	Q_C value calculated with progeny in /IAEA 08/
^{96m}Tc	Q_C and Q_D values calculated with progeny in /IAEA 08/
^{231}Th	higher deviation of Q_C value, possibly calculated with progeny in /IAEA 08/

Furthermore, dose rate coefficients for pathways Q_A, Q_B, and Q_D were recalculated. For many nuclides there are significant differences between calculated coefficients /BÜT 14/ and listed coefficients /IAEA 08/. All these recalculated dose rate coefficients were used as input parameter for calculating *Q* and *A* values with Ber*QA*Trans giving the very good results discussed above. It seems that dose rate coefficients listed in /IAEA 08/ were calculated backwards from the *Q* values given in the same document. As there is an undocumented limit of 1000 TBq for each pathway, these limits were probably taken into account resulting in too high dose rate coefficients for nuclides with limited *Q* values. For nuclides ^{252}Cf, ^{254}Cf, and ^{248}Cm the tabulated Q_A values are superseded by special *Q* values representing doses from neutron emissions. Even in these cases, it seems that the (neutron dose) *Q* values accidentally were taken to calculate dose rate coefficients for external photon dose. Comprehensive tables with compared dose rate coefficients are given in GRS report GRS-343 /BÜT 14/.

3.2 Current Q system

The description of the Q system and its calculation methods is laid down in /IAEA 12/ and /IAEA 14/. With only these two publications a proper (re-)calculation of *Q* and *A* values is not possible. Even by means of some additional reports, e.g. /BEN 94/, a recalculation of listed values is complicate and in some cases not achievable. It was necessary to partly rely on auxiliary literature from the 1970s and 1980s. Also the HPA report /JON 11/ as well as private communication with colleagues from HPA (now Public Health England, HPE) had a significant impact in investigating the whole Q system.

The calculation methods in the current Q system are not documented sufficiently in order to reproduce all required details. Hence, misinterpretations can occur. The current *Q* and *A* values are based on calculations with rather old data, even if there is newer data available. However, it is not possible to reproduce all current *Q* and *A* values exactly.

Some of the occurred problems in the current Q system are listed below:

- partly, *Q* and *A* values are calculated using outdated input data,
- dose coefficients listed in /IAEA 14/ for Q_C values are partly not in coincidence of dose coefficients of ICRP 68 /ICRP 94/; however, no reference is given for dose coefficients in /IAEA 14/,
- dose rate coefficients listed in /IAEA 14/, seem to be calculated backwards from *Q* values listed in /IAEA 14/, therefore some values (especially for small coefficients) show high differences compared to new calculated ones,
- Q values are limited to 1000 TBq without justification or documentation,
- determination of “unlimited” values for LSA material is not fully documented,
- treatment of progeny is very different between the *Q* value pathways.

More findings as well as detailed explanations of these issues are described for example in /JON 11/ and /BÜT 14/.

As a reaction to these problems, members of TRANSSC asked for an international meeting because other institutions than HPA/PHE and GRS discussed the Q system, too. It was held at GRS premises in Cologne in September 2013. The meeting was joined by participants from Institut de Radioprotection et de Sûreté Nucléaire (IRSN), Japan Nuclear Energy Safety Organisation (JNES; now Nuclear Regulation Authority, NRA), PHE, World Nuclear Transport Institute (WNTI), and GRS. Afterwards, the participants agreed that the current Q system should be reviewed, and an International Working Group on review of A_1 and A_2 Values for the IAEA Transport Regulations was founded.

As part of the work in the International Working Group on review of A_1 and A_2 values for the IAEA Transport Regulations calculations using Ber*QA*Trans were done. While the working group aims for involving Monte Carlo methods for deriving new *Q* and *A* values, it was necessary to use also deterministic methods by SEAL and Ber*QA*Trans. These calculations were done for all nuclides of /IAEA 12/. So it was possible to show the influence of using new data, e.g. ICRP 107 /ICRP 08/, without changing the calculation methods itself. As a second example, *Q* and *A* values for five new nuclides (^{135m}Ba, ^{69}Ge, ^{193m}Ir, ^{57}Ni, and ^{83}Sr) shall be included with the next revision of IAEA Transport Regulations SSR-6 and SSG-26.

Calculations were done by GRS (see Table 2, calculated with data from ICRP 38 /ICRP 83/, ICRP 51 /ICRP 87/ and ICRP 68 /ICRP 94/), HPE, and NRA, using Ber*QA*Trans, SEAL, or BRACSS (calculation tool of NRA /NRA 15/), respectively. Results were given to TRANSSC in February 2016. The new values selected from the results of these organisations shall be added in the new draft revision of the IAEA Transport Regulations.

Table 2: *Q* and *A* values for new nuclides calculated with Ber*QA*Trans

Nuclide	Q_A (TBq)	Q_B (TBq)	Q_C (TBq)	Q_D (TBq)	A_1 (TBq)	A_2 (TBq)
^{135m}Ba	1.6×10^{1}	1.0×10^{3}	3.3×10^{2}	5.9×10^{-1}	2×10^{1}	6×10^{-1}
^{69}Ge	1.3×10^{0}	7.1×10^{0}	1.7×10^{2}	4.5×10^{0}	1×10^{0}	1×10^{0}
^{193m}Ir [a)]	8.3×10^{2}	1.0×10^{3}	4.2×10^{1}	4.2×10^{0}	4×10^{1}	4×10^{0}
^{57}Ni	5.9×10^{-1}	2.0×10^{1}	8.9×10^{1}	3.3×10^{0}	6×10^{-1}	6×10^{-1}
^{83}Sr	1.4×10^{0}	1.4×10^{1}	1.5×10^{2}	8.7×10^{0}	1×10^{0}	1×10^{0}

a) no nuclide data available in ICRP 38 /ICRP 83/, therefore calculated with data from ICRP 107 /ICRP 08/

4 MONTE-CARLO BASED A-VALUE SIMULATOR (MCBAS)

In order to take into account the actual state-of-the-art science and technology, the revision of the Q system will be based on calculations using Monte-Carlo (MC) methods. This will allow to include the actual nuclear physics interaction cross sections of particles also for effects like bremsstrahlung that has not fully been included in the current Q system. Furthermore, all kind of relevant particles can be included based on actual nuclear data published by ICRP. However, the basic principle like the geometry of the current Q system shall remain.

Hence, the international working group already started to perform calculations with different MC tools. A short list of ten nuclides (^{18}F, ^{60}Co, ^{85}Kr, ^{90}Sr, ^{99m}Tc, ^{106}Ru, ^{134}Cs, ^{137}Cs, ^{154}Eu, ^{192}Ir) serves as a basis for comparing the different codes. The main advantage of a short list is the reduction of the intense computing power for all nuclides of interest, since each single nuclide is calculated individually with the MC codes. Recently, GRS also started to develop a code, namely the Monte-Carlo Based A-value Simulator (MCBAS). This code will be able to reduce the computing time significantly. MCBAS is programmed in C/C++ and is not a MC code itself. Therefore, it can be installed by any interested user without the complex MC structure in the background. To achieve this, MCBAS reads the nuclear data inputs as well as conversion coefficients from ICRP publications on the one hand and the flux spectra simulated with MC methods for the required particle and energy combination on the other hand in order to calculate *Q* values. The flux spectra are available in form of a database. Hence, a decoupling of the final calculations and the MC simulations is gained. The time consuming MC simulations are performed independently in advance and only the result, i.e. the database, is used by MCBAS. The energy difference between available particle energies is currently chosen as follows:

- 5 keV for particle energies below 100 keV
- 10 keV for particle energies above 100 keV

In summary, this procedure allows for

- a very fast calculation of each nuclide
- the calculation of new nuclides without further MC simulations
- the calculation of new types of radiation sources, like neutron sources
- an update of input data like new ICRP publications
- a flexible change of input parameters like the calculation of different organ doses (e.g. lens of the eye) or changing the radiation geometry

Hence, MCBAS will be a fast and due to its modularity also a flexible tool to calculate *Q* values on a state-of-the-art basis.

The main disadvantage of the described principle is the fact that the characteristic particle energies of a particular nuclide show a maximal offset to the next available flux spectrum of 2.5 keV or 5 keV for energies below 100 keV or above 100 keV, respectively. However, the energy differences have been chosen in a way that the uncertainties are neglegible in general.

Currently, MCBAS is able to calculate A_1 values only but for all nuclides of interest. It is intended to implement the calculation of A_2 values as well. However, this depends on an ongoing harmonisation process within the international working group. Furthermore, a validation process of the tool is still ongoing.

5 SUMMARY

GRS has analysed the current Q system of the IAEA Transport Regulations /IAEA 12/ and established the calculation tool Ber*QA*Trans for calculating *Q* and *A* values. With Ber*QA*Trans it is not only possible to recalculate known *Q* and *A* values listed in /IAEA 12/ and /IAEA 14/, but also to calculate *Q* and *A* values for new nuclides or to use more up-to-date data published by ICRP. With Ber*QA*Trans according to the current Q system recalculated *A* values are in very good agreement with the tabulated values, e.g. given in /IAEA 08/ and /IAEA 14/, except for eight nuclides. Problems with these nuclides are widely discussed in /BÜT 14/ and /JON 11/, a brief explanation is given in this paper.

While GRS investigated the Q system many inconsistencies in the documentation of the Q system were found. These and similar findings were also made by other organisations. Some of these issues are briefly discussed in this paper. More explanations and extended discussions of this topic can be found in /BÜT 14/ and /JON 11/.

Furthermore, the developments on a new calculation tool, MCBAS, are presented. This tool is based on MC simulations in order to revise the Q system with a state-of-the-art method and actual physical input parameters. Since the MC simulations are decoupled from MCBAS and their results already serve as an input, MCBAS is a very fast tool. Due to its modularity, new input data like from new ICRP publications or data for new nuclides can be used easily.

6 ACKNOWLEDGMENTS

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Activation and dose rate analyses to support NPP dismantling planning

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Abstract:

The planning of the dismantling of a nuclear facility requires knowledge of the installed equipment and the parts of the buildings which are activated during the lifetime of the facility. This activation of the materials can be caused by neutron absorption, especially in facilities with strong neutron sources, e.g. in nuclear power plants. In order to calculate this effect, an activation calculation tool utilising the Monte Carlo transport code MCNP 5 was developed. This tool is described in the paper at hand. At first a simplified benchmark calculation is discussed, which was used to evaluate the basic characteristics and requirements of the activation calculation of a reactor pressure vessel based on the Monte Carlo transport code MCNP 5. Furthermore the current development of the dose rate calculation tool DETECT is described. It is designed to perform automated dose rate calculations of radioactive waste packages. The coupling of both tools is under progress to facilitate a direct dose rate calculation of activated materials.

1 INTRODUCTION

The planning of the dismantling of a nuclear facility requires knowledge of the installed equipment and the parts of the buildings which are activated during the lifetime of the facility. This can be caused by contamination or material activation due to neutron absorption. On one hand this knowledge is needed to estimate the total amount of radioactive waste, on the other hand it is needed to estimate the radiation dose of the working personnel and identify necessary radiation protection measures. The activation of materials becomes particular important in facilities with strong neutron sources, e.g. in nuclear power plants. If the flux and spectrum of the neutron source, the geometrical arrangement of the equipment and the building, and their material compositions are sufficiently known for the whole lifetime of the facility, the activation of the materials inside the neutron radiation field can be calculated.

The calculation sequence of an activation calculation is shown in Fig. 1. At first the neutron flux and spectrum has to be calculated inside the region of interest (ROI) where the material activation should be evaluated. This is typically done using a neutron transport calculation which considers an appropriate geometric model of the neutron source and its surrounding structures. Afterwards the material composition of the ROI and the determined neutron flux and spectrum are needed to compute the change in the material compositions by performing a nuclide inventory calculation.

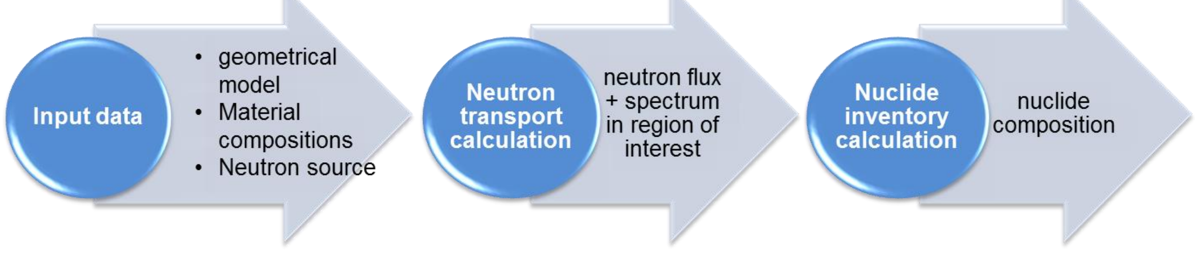


Fig. 1: Schematic sequence of an activation calculation.

With the development of DORTAKTIV [1, 2], GRS started a first attempt to provide an automated tool for activation calculations considering multidimensional geometric models. It was designed to evaluate the activation of a reactor pressure vessel (RPV) due to the neutron radiation field outside the reactor core. DORTAKTIV consists of the two-dimensional deterministic transport code DORT [3] and of the inventory determination code ORIGEN-X for activation calculations [4]. Since the 2D deterministic code DORT shows several limitations of the geometric models, a complementary activation calculation tool utilising the Monte Carlo transport code MCNP 5 [5] was developed. This tool is described in the paper at hand. At first a simplified benchmark calculation is discussed which was used to evaluate the basic characteristics and requirements of the activation calculation of an RPV based on the Monte Carlo transport code MCNP 5. Furthermore the current development of the dose rate calculation tool DETECT is described which is designed to perform automated dose rate calculations of radioactive waste packages. The coupling of both tools is under progress to facilitate a direct dose rate calculation of activated materials.

2 REACTOR PRESSURE VESSEL BENCHMARK

In contrast to deterministic codes, a Monte Carlo transport code provides more complex modelling capabilities. Therefore, detailed realistic geometric models can be created and used in the transport simulations. The drawback of a Monte Carlo transport code is the need of extensive variance reduction measures to ensure statistical meaningful results if highly shielded sources or regions have to be considered. Therefore, a simplified RPV benchmark model (Fig. 2) is used to evaluate the basic characteristics and requirements of neutron transport calculations of RPVs.

The benchmark model includes a homogenized core and control rod drive area, the modelling of only one of the six cooling pipes and only one of the neighbouring rooms, a simplification of the bioshield and the neglect of steam generators and pumps. The goal of the benchmark is the estimation of the total neutron flux at 16 different detector positions (Fig. 2). Therefore, the benchmark model includes also the corresponding tallies to determine the requested fluxes. In order to test an alternative workflow to create the geometric model, it was decided to create first a CAD model of the reactor with the freely available program FreeCAD 0.12 [6]. Afterwards, this model was converted into an MCNP input file with the program MCAM 4.8 [7], which was e.g. also used for the development of neutronic models of the ITER divertors [8]. The geometric model in the FreeCAD representation is shown on the left side in Fig 2.

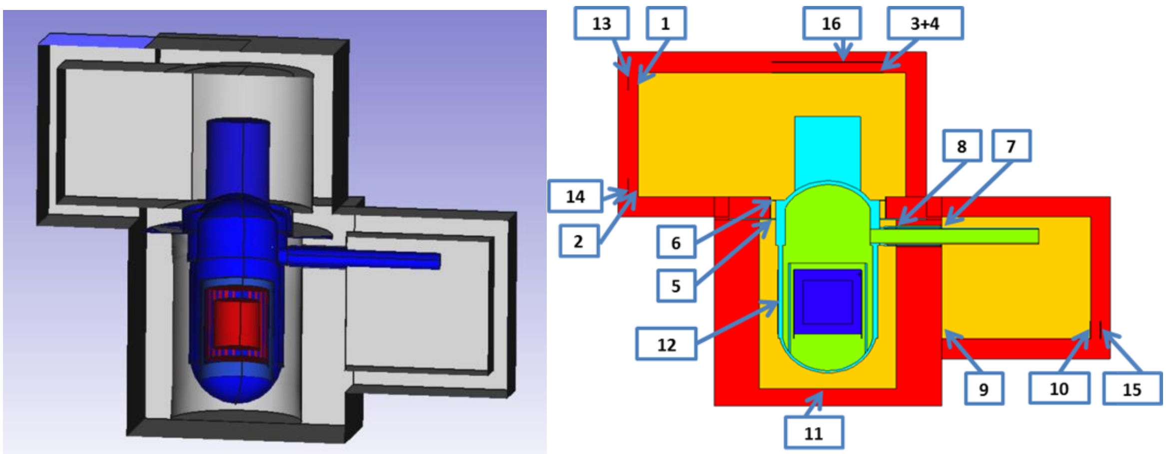


Fig. 2: Geometric model of the simplified reactor implemented in FreeCAD (left) and MCNP with the location of the tallies (right).

Due to the large geometric model which spans several tens of meters, and the relatively small neutron tallies, a variance reduction technique has to be applied. Therefore, the so called weight window method is used in the simulations. This technique is based on a space and energy dependent subdivision and Russian roulette of the simulated particles. It increases the number of particles reaching a desired area of the model, while reducing the number of particles with a low probability to reach this area. Within this process the weights of the particles reaching the tally are reduced, so that the flux stays constant while the number of detected particles is increased. This leads to a significantly reduced uncertainty of the measured flux in the tallies. The weight windows were generated in three energy groups (thermal, epithermal, and fast neutrons) using the MCNP weight window generator card ("WWG") for each tally to reach flux uncertainty of about 1 % in the measuring volumes. Fig. 2 shows the geometric model of MCNP on the right side and the locations of the tallies, which were also calculated by the Swiss "Nationale Genossenschaft für die Lagerung radioaktiver Abfälle" (NAGRA) [9] to allow a comparison of the calculated fluxes for consistency.

The calculated fluxes in the tallies range from 7.08 x 10^{-11} cm^{-2}/source particle for tally 12 close to the pressure vessel down to 6.64 x 10^{-17} cm^{-2}/source particle for tally 9 in the lower left corner of the neighbouring building. This extreme attenuation by the neutron shielding of the moderator, vessel structures and concrete walls shows the enormous importance of the generation of efficient weight windows. Fig. 3 shows the relative deviation of the tally fluxes between GRS and NAGRA calculations and their combined statistical uncertainties.

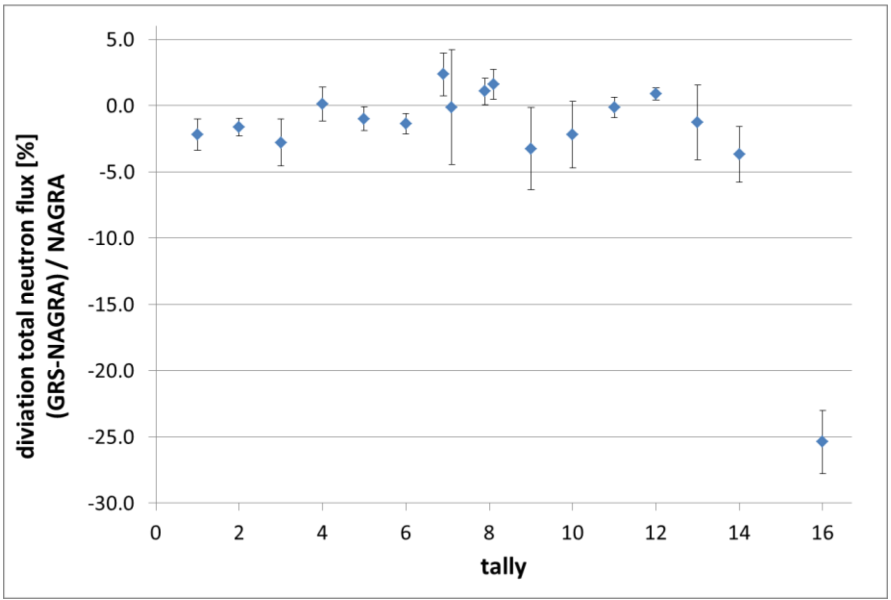
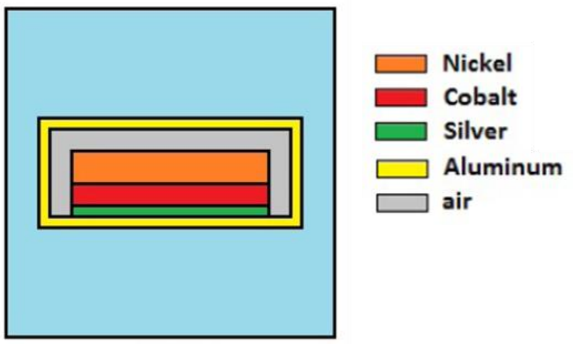


Fig. 3: Relative deviation of the tally fluxes between GRS and NAGRA calculations and their combined relative uncertainties.

For all tallies except for one, the deviation lies within one to two standard deviations. For tally 15, which lies the furthest away and is best shielded no appropriate weight windows could be generated to allow a focussing of the neutrons towards this tally. Therefor no results could be generated. For the large discrepancy between GRS and NAGRA for tally 16 no satisfying reason could be found.

In order to test the activation calculation sequence under realistic conditions, the same benchmark model was used to perform an activation calculation. For this purpose an activation sample (Fig. 4) was included in the benchmark model at the position of tally 11. Three tallies representing the different layers of the sample layers were defined, to determine the total neutron flux and its spectrum in 84 energy groups. The geometry of the sample was taken from [10] and is shown in Fig. 4 on the left side. It consists of thin foils of Ni (0.1 mm), Co (0.025 mm) and Ag (0.01 mm) with a diameter of 2 cm. The three layers of the probe are chosen due to their main nuclear reactions: ${}^{59}_{27}Co\,(n,\gamma)\,{}^{60}_{27}Co$, ${}^{58}_{28}Ni\,(n,p){}^{58}_{27}Co$ and ${}^{109}_{47}Ag\,(n,\gamma){}^{110m}_{47}Ag$. Since these three reactions are sensitive respectively to the thermal, epithermal and fast neutron energy range, they are ideal to measure the entire neutron flux in the probe. An activation cycle of 344 days and a neutron rate of the core of 1.3 x 10^{20} s^{-1} were assumed.

The nuclide inventory calculations were performed using the GRS depletion code GRSAKTIV-II [11]. GRSAKTIV-II is based on ORIGEN-X, but can handle 84 energy group fluxes (see section 3). For comparison also the rates of the mentioned reactions of the specified nuclides were determined by MCNP. These rates can be converted to nuclide masses assuming corrections due to the decay of the created nuclides and due to competing or secondary neutron reactions. The nuclide masses resulting from the two codes are shown in Fig. 4 on the right. The determined nuclide masses are in reasonable agreement for all three analyzed nuclides (^{60}Co, ^{110m}Ag, and ^{58}Co).



	Nuclide mass [g]	
	MCNP reaction rates	GRSAKTIV-II
^{60}Co	3.80e-7	3.73e-7
^{110m}Ag	1.08e-8	1.42e-8
^{58}Co	1.23e-11	9.88e-12

Fig. 4: Geometric model of the activation sample (left, not at scale) and resulting nuclide masses from MCNP and GRSAKTIV-II (right).

3 ACTIVATION CALCULATION TOOL

An activation calculation tool was developed utilising a Monte Carlo transport code to overcome the observed limitations of DORTAKTIV due to the involved two-dimensional deterministic transport code DORT. The developed tool as well as DORTAKTIV is based on the same general calculation sequence, which is shown in Fig. 1. In contrast to DORTAKTIV the new developed tool couples the well-known Monte Carlo transport code MCNP 5 to the depletion code GRSAKTIV-II, which was developed at GRS and is based on the depletion code ORIGEN-X. GRSAKTIV-II was chosen, since it provides the possibility to calculate the nuclide inventory using a neutron spectrum in an 84 energy group structure instead of the three energy groups of ORIGEN-X. In this way arbitrary neutron spectra can be handled. The corresponding calculation sequence is presented in Fig. 5.

The required input data mainly consists of a configuration data set steering the calculation process, the geometrical description including the definition of ROI and the variance reduction data if needed, the material compositions, and the definition of the neutron source. These input data can be organized in several input files to facilitate a clearly arranged input data set and to simplify the reused and exchange of parts of the input data. The required MCNP input data file is created automatically according to the given input data. Afterwards the input file is executed. The generated MCNP output file is analysed and the neutron fluxes and spectra of the ROIs are extracted. Currently the tool is arranged in a way that it favours the use of so-called mesh tallies to determining the neutron fluxes and spectra. These mesh tallies provide some advantages. On one hand they provide an easy way to arbitrarily segment the ROI without any changes to the geometric model. On the other hand the volume of a regular mesh tally cell can be calculated automatically and allows for an appropriate scaling of the results to get absolute values. The use of tallies based on arbitrary geometric cells would require a complex volume calculation which is not yet implemented.

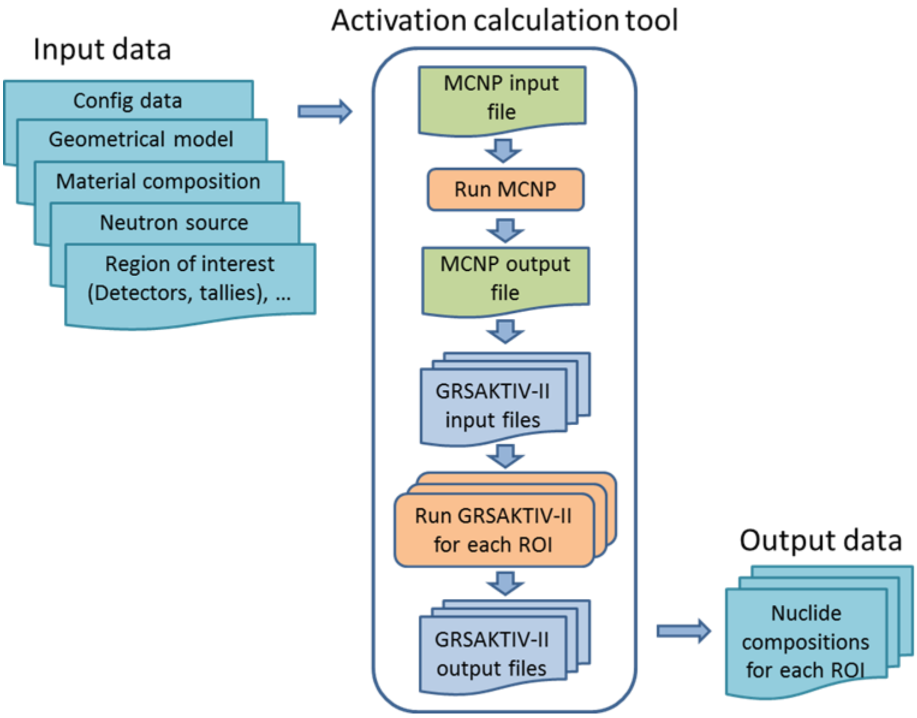


Fig. 5: Calculation sequence of the developed activation calculation tool.

In the next calculation step a corresponding GRSAKTIV-II input file is create automatically for each ROI and each mesh tally cell. These input files are executed by GRSAKTIV-II and the generated output files are analysed. The calculated nuclide inventories are extracted and written to a dedicated output file for further processing of the user.

4 ACTIVATION CALCULATION OF A GENERIC REACTOR PRESSURE VESSEL

The activation calculation tool was applied to a detailed generic RPV model inside a generic reactor building. As for the reactor model in section 2, the geometric model was created with FreeCAD and converted to MCNP using MCAM. Fig. 6 shows the implementation of the RPV in FreeCAD (left and middle) and the resulting MCNP model including the concrete structures around the RPV (right).

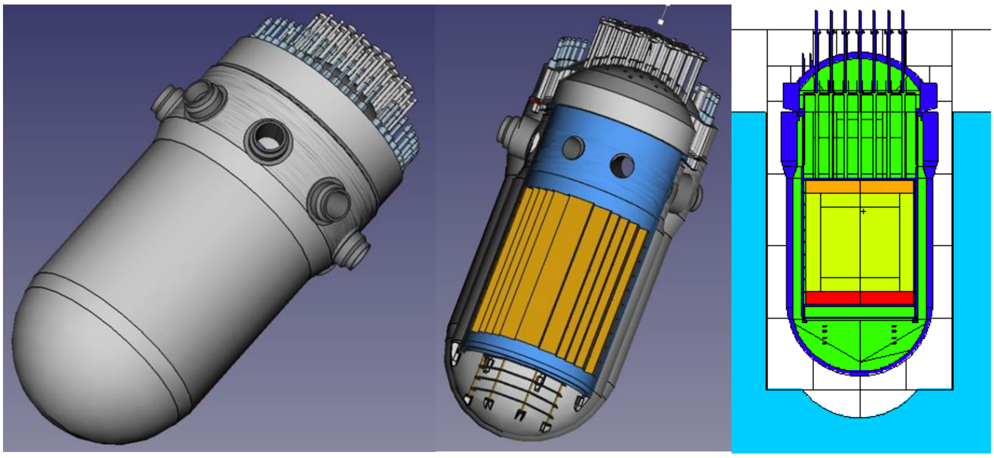


Fig. 6: Detailed model of a generic RPV implemented in FreeCAD (left and middle) and MCNP (right).

For the neutron transport calculation a homogenized neutron source in the outer 30 cm of the reactor core was assumed, since neutrons from further inside have a very low probability to escape the core due to scattering and absorption in the moderator and the fuel. For the activation of the pressure vessel around the core a suitable weight window was created using the MCNP weight windows generator (WWG). Using this input data, the neutron flux around the core calculated by the activation calculation tool is shown in Fig. 7 on the left side. For most of the 84 energy groups of the neutron spectrum at the inner surface of the pressure vessel wall at the height of the core, the relative uncertainty is below 5 %. Only towards the tails of the spectrum, relative uncertainty increases up to 35 %. Towards the outer surface of the wall and towards the upper and lower edges of the core, the uncertainty increases.

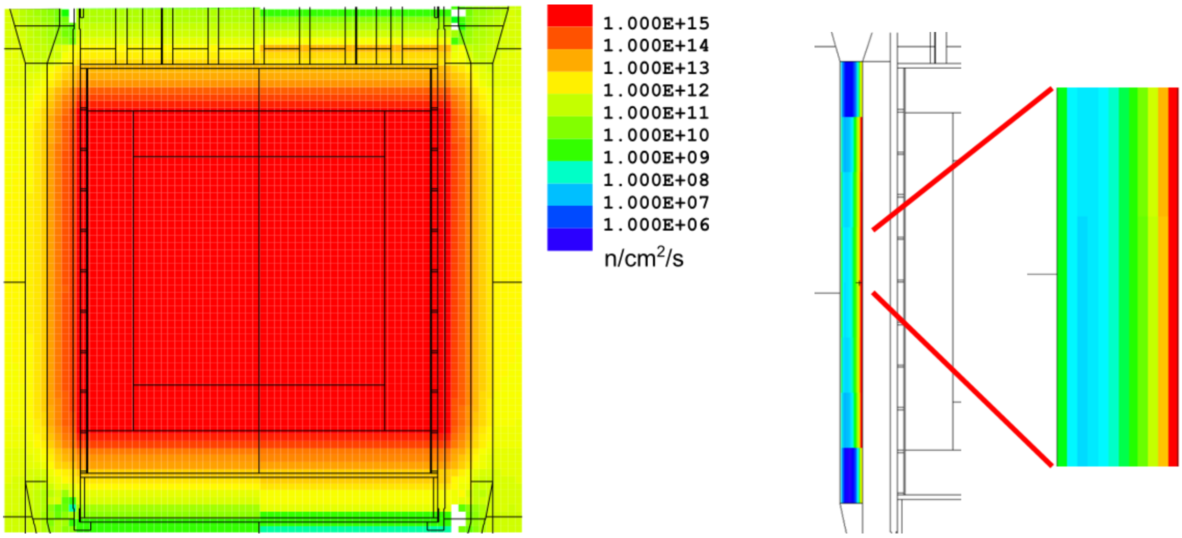


Fig. 7: Total neutron flux (left), and ^{60}Co concentration in the wall of the RPV (right).

The resulting flux was used in the second calculation step of the tool presented in section 3 to determine the activation of the central part of the RPV. A thermal reactor power of 2.2 GW was assumed resulting in a neutron rate of the outer region of the core of about 1 x 10^{20} s^{-1}. A runtime of 25 years split into 10 months of active time and 2 months of down time was used, a cooldown time was neglected. For example, starting with a cobalt content of 0.02 wt.-% the resulting ^{60}Co activity is in the order of 10^8 Bq/kg_{Steal}. Its distribution inside the vessel wall is shown in Fig. 7 in the right-side plot. The nuclide concentration is the largest at the inside of the wall due to the larger neutron flux from the reactor core. Towards the outside of the RPV wall it declines by more than one order of magnitude until it increases again at the very edge. This effect can be attributed to reflected neutrons coming from the concrete wall outside the RPV.

5 DOSE RATE CALCULATION TOOL DETECT

A typical subsequent step after an activation calculation is the evaluation of the dose rate caused by the activated material. In order to provide an efficient way to perform such dose rate calculations, a dedicated tool, called DETECT (dose rate calculation tool), is developed. It is designed to analyse waste packages but it can also analyse radioactive materials in arbitrary 3D geometries.

The main idea of DETECT consists in the definition of a radioactive material (waste) composition by the user and the accompanied selection of a predefined geometric model, e.g. a standardized waste container. The defined material composition is automatically included in the geometric model allowing for an appropriate scaling of the waste masses or volumes. The given input data are automatically translated into a corresponding input file, and executed afterwards. Finally the resulting output file is analysed with respect to the desired dose rate information. The final version of DETECT will also be able to handle large series of calculations considering different combinations of material compositions and geometric models, e.g. to support parameter studies.

A similar older tool called ANITABL [12] already exists at GRS but considers only 1D radiation transport. Therefore, DETECT uses parts of the ANITABL functionalities but couples these functionalities to MCNP 5 to allow 3D radiation transport. Hence, the calculation sequence of DETECT is as follows (Fig. 8): The first step consists of a decay calculation. In this way a storage period of the user-defined radioactive material before the considered time point of the calculation can be considered, i.e. the material composition will be calculated taking into account the corresponding decay time of the involved nuclides. If no decay time should be considered, this calculation step will be skipped. In a next step the radioactive material is analysed regarding emitted gamma rays and neutrons due to decay or spontaneous fission. From their tabulated emission probabilities and energies a corresponding gamma and neutron spectrum is determined. These two steps are performed using ANITABL functionalities. They are realized using ORIGEN-X to calculate the nuclide decay and using a dedicated module, called NGSRC [12], to generate the gamma and neutron spectra. Both spectra are translated into a MCNP input card and integrated into a given MCNP input data set. This translation and integration into the MCNP input is done using a dedicated tool called SRCMCNP. Also SRCMCNP was already developed within the ANITABL environment and can read NGSRC output files. DETECT provides an automated coupling of NGSRC to SRCMCNP and calls SRCMCNP internally to complete the source definition of the MCNP input data. The above mentioned MCNP input data set can be taken from a predefined collection provided by DETECT, e.g. describing different types of standardized waste containers or other typical geometrical configurations. Alternatively it can be defined be the user and has to be included in the DETECT input data. After completion of the MCNP input data, MCNP 5 is executed and the resulting output file is analysed. The dose rates of the ROIs are extracted and written to a dedicated output file.

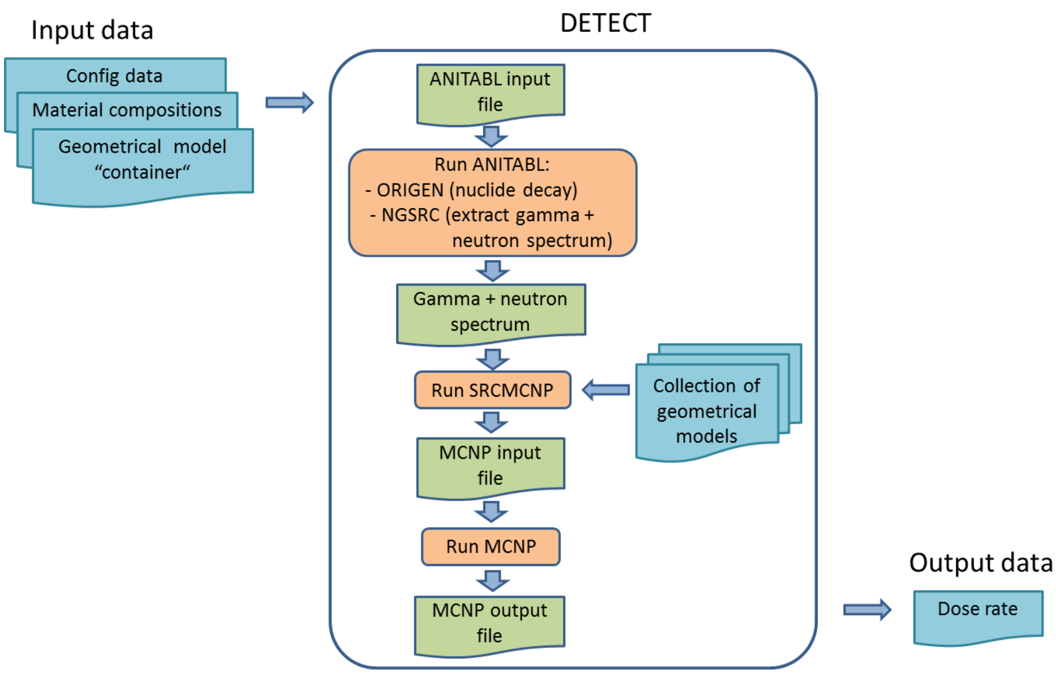


Fig. 8: Calculation sequence of DETECT

A preliminary version is implemented with functionalities still being limited, e.g. up to now only one material composition can be handled at a time. Several enhancements are intended, such as handling an arbitrary number of material compositions, mixing of materials or adjustable geometric models.

In order to minimize user interventions needed to perform a dose rate calculation considering activated materials, also the coupling of both tools, the developed activation calculation tool and DETECT, are currently under development. The goal is the integration of dose rate calculations for the activated materials into the activation calculation tool (Fig. 9). This enhancement is under progress. It should finally allow the user to provide a combined input data set which also includes the DETECT input data. For user-selected ROIs the calculated material compositions after activation will be automatically used as input of DETECT and dose rate calculations will be executed considering a user-specified geometric model. Finally, as described above, the results of the calculations will be extracted from the output files and written to a dedicated output file for further investigation.

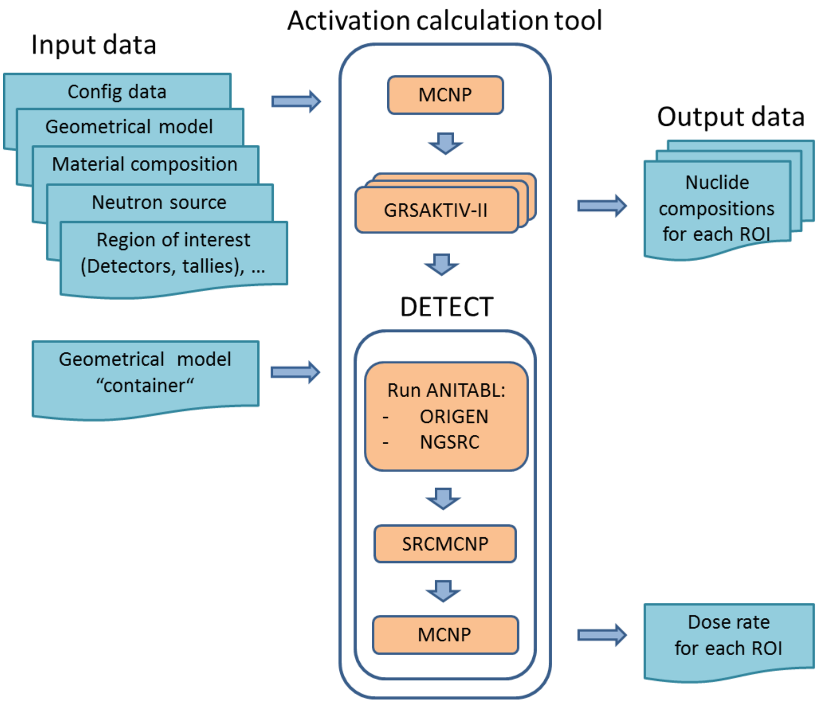


Fig. 9: Calculation sequence of the coupled activation and dose rate calculation.

6 CONCLUSION

The transport and activation analyses shown here demonstrate the general feasibility of and the challenges involved with such calculations. For instance, in the benchmark model total neutron fluxes were determined which are many orders of magnitude smaller than the flux of the source (reactor core). Also the activation of a small sample was calculated. However, the transport calculations of the generic RPV are much more challenging. Reasonable results could be achieved inside the vessel walls at the height of the reactor core, but it is was much more difficult to get a similar number of neutrons above or below the height of the core. Therefore, more elaborated methods or tools to generate appropriate variance reduction parameters will be considered in the future, e.g. ADVANTG [13]. Nevertheless, it is possible to calculate the activation of the generic RPV at the height of the core in detail. The axial and especially the radial distribution of the nuclide concentrations could be determined.

Furthermore, the currently developed tools for activation and dose rate calculations were presented. They enable automated activation and dose rate calculations with detailed 3D geometries. They can support numerical analyses on dose optimization for personnel in the dismantling of NPPs and or waste container loading schemes.

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Seminar 5

Security of Nuclear Installations and Materials

Violations Analysis Methodology in the System of Accounting for and Control of Nuclear Materials in the Russian Federation

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Abstract:

Inspections of control and accounting of nuclear material (MC&A) are integral part of security oversight, performed by Russian regulatory authority, Rostechnadzor. SEC NRS, as technical support organization for Rostechnadzor, performs analysis of violations in MC&A area. Such analysis helps to assess general status of MC&A system at facilities, trends and typical violations, identify gaps in implementation of MC&A system at facilities and oversight activities, identify potential areas for improvement in associated regulations and guides, both for facilities and inspectors.

As a result of analysis proposals are developed for Rostechnadzor on how to improve supervision for MC&A taking into account the risk-based approach.

Violations analysis consists of: analysis of information on MC&A violations; development and application of method for categorization of violations, evaluation of their severity; development of proposals to improve supervision for MC&A.

Method of analysis of MC&A violations is described in the paper. The results of research are also used by Rostechnadzor in revision and update of regulations and guides.

1. INTRODUCTION

Inspections of control and accounting of nuclear material (MC&A) are integral part of security oversight, performed by Russian regulatory authority, Rostechnadzor.

During inspection activity at nuclear facilities Rostechnadzor detects violations of regulatory requirements, such as NP-030-12 “Basic Rules on Accounting and Control of Nuclear Materials” (which corresponds to IAEA Nuclear Security Series No.20 and No.25-G).

Rostechnadzor authorized SEC NRS to perform analysis of MC&A violations. SEC NRS developed methodology to analyze such violations in 2009.

The objective of the analysis methodology is to evaluate MC&A violations and identify weak points in MC&A system, including potential gaps in regulation or inspection procedures. Violations analysis is carried out in order to develop proposals to Rostechnadzor on improvement of supervision of MC&A system taking into account the risk-based approaches.

2. ANALYSIS METHODOLOGY FOR MC&A VIOLATION

Analysis for MC&A violations consists of:

1. Collection of information about MC&A violations (as a result of inspections);
2. Classification of violations;
3. Assessment of the MC&A violations;
4. Development of proposals to Rostechnadzor on improvement of regulatory activities for MC&A.

2.1 Collection of information

Collection of information about MC&A violations is done by receiving of periodic (on a quarterly basis) reports from Rostechnadzor's regional offices and Headquarters. These reports provide information on results of inspection activities: number and type of inspections, details on every inspection: inspected nuclear facility, number of violations detected, description of violations, associated prescriptions for fixing the violations.

2.2 Classification of violations

The objective of classification of violations is comprehensive consideration of the violations influence on MC&A system at nuclear facility in terms of nuclear materials security.

At the moment some classifications of violations are used in violations analysis:
1. Classification of violations per facility type;
2. Classification of violations per MC&A components;
3. Classification of violations per significance;
4. Combinations of the classifications above.

These classifications of violations are considered in more details below.

2.2.1 Classification of violations per facility type

This classification of violations allows detecting the type of nuclear facility where the maximum number of violations was determined in the reported period. In violations analysis all nuclear facilities are classified into eight types:

1. Nuclear power plants;
2. Uranium mining facilities (UMF);
3. Research facilities;
4. Uranium enrichment facilities (UEF);
5. Fuel cycle facilities;
6. Nuclear power installations for vehicles (ships);
7. Higher education institutions;
8. Big chemical combines (complex facilities which combine few fuel cycle stages in one facility).

2.2.2 Classification of violations per MC&A components

To identify the weak points in MC&A system the detected violations are classified per MC&A components:

1. Management of MC&A system at the nuclear facility in general;
2. Determination of nuclear materials balance area (MBA);
3. Technical means and measures applied for access control;
4. Organizing the system of nuclear material measurements;
5. Procedure on nuclear materials transfers;
6. Physical inventory of nuclear materials;
7. Records and reports;
8. Personnel qualification and training;
9. Nuclear materials accounting and exemptions.

2.2.3 Combination of classifications per facility type and MC&A components

After classification of violations per facility type, the types of facilities with the maximum number of violations are detected. Then, for these types of facilities the classification of violations per MC&A components is used.

Such approach allows detecting of weak points of MC&A system for each type of facilities with the maximum number of violations.

2.2.4 Classification of violations per significance

This violations classification allows differentiation of violations per significance in MC&A. There are 3 levels of violations significance in that classification:

1. Low-level (less significant) violation;
2. Medium-level (significant) violation;
3. High-level (gross) violation.

In order to evaluate the violation significance the following parameters are used:

– Assessment of violation nature;
– Assessment of violation scale;
– Detection of the violation causes;
– Assessment of potential consequences of violation.

Specific coefficient is assigned to each parameter by expert decision. The significance index is measured using the numerical coefficients. The significance level depends on the significance index.

After classification of violations per significance, the high-level (gross) violations are identified. The existence of such violations can serve as an indicator that MC&A system does not fully comply with regulatory documents and that may need proper regulatory actions.

2.3 Assessment of the MC&A on the basis of the revealed violation

As a result of such analysis the following is identified:

– facility types with the maximum number of violations;
– MC&A components with the maximum number of violations in the reported period;
– presence of high-level (gross) violations and their percentage in the number of the detected violations;
– Comparison of data for the same period of the previous year.

Typical (most frequent) violations are also identified for each MC&A component.

2.4 Development of proposals to Rostechnadzor

The final stage of the violations analysis of MC&A is the development of proposals for improvement of the regulatory activity.

Based on results of analysis SEC NRS develops recommendations for Rostechnadzor on how to update regulation documents in the MC&A area; develops and updates MC&A guides.

3. CONCLUSION

The methodology for analysis of MC&A violation was developed and is used by SEC NRS. The results of violations analysis are used by SEC NRS for identification of the potential improvements in MC&A regulatory activity of Rostechnadzor.

Master in Nuclear Security (MiNS) at the Brandenburg University of Applied Sciences – A program overview with a focus on curriculum and international partner framework

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Abstract:

Regulatory changes as well as persistent threats are major drivers in the field of nuclear security. To respond to these challenges the International Atomic Energy Agency (IAEA) has called for Universities to set up master programs in nuclear security. Therefore the Institute for Security and Safety at Brandenburg University is currently setting up an innovative Master in nuclear security (MiNS). MiNS will be conducted as a distance learning program and its curriculum will be based on the results of the internal revision process of the IAEAs Nuclear Security Series Nr.12 (NSS 12) on Education in nuclear security. This presentation will shed light on the main elements of MiNS and aims at providing a comprehensive overview over the Master in Nuclear Security.

1 MINS AT A GLANCE

The Master in Nuclear Security (MiNS) is an accredited Master of Science (M.Sc.) provided by Brandenburg University of Applied Sciences, Germany. The Masters' will be provided as an innovative distance learning program. Students need to obtain 90 ECTS points in order to obtain the Masters' degree. MiNS splits into six modules and will be provided as a full-time program (3 terms) or part-time program (5 terms). MiNS adresses a wide varierity of potential students. This includes, for example, bachelor students, diplomatic staff, security professionals, employees of nuclear installations and industry, to give just some examples.

2 PROGRAM OVERVIEW

MiNS is currently implemented by Institute for Security and Safety at the Brandenburg University, on the basis of the already existing Master in Security Management. The Masters' is developed in close cooperation with experts from renowed institutions in the field of nuclear security. For example, one of our project managers (Dr. Johannes Sterba) works at the Atominstitut from TU Vienna and other experts come from Kings College London, Purdue University (US) and Ontario University Institute of Technology (CA).

The content of MiNS is based on the results of the internal revision process of the IAEAs Nuclear Security Series N°12 (NSS12) on Education in Nuclear Security. This was made possible trough the close cooperation of all our experts in the IAEAs International Nuclear Security Education Network (INSEN). Additionally, MiNS builds upon the lessons learned from the previous pilot program "Master's in Nuclear Security" conducted by TU Delft, in which Brandenburg University was a project partner.

MiNS will be conducted as distance learning program and will start on March 1, 2017.

Our Master in Nuclear Security adresses students and professionals alike. The entry requirements are the proove of a relevant Bachelor degree, one year of work experience and a proove of sufficient knowledge in the field of nuclear physics. The latter can be prooven through a degree in physics, chemics or nuclear engineering or by passing an entry examination prior to the regular start of MiNS.

MiNS will allow students to obtain a fully recognized Master of Science (M.Sc.) degree from Brandenburg University. After completion of the M.Sc. in Nuclear Security students will be able to either work in a specialiced work environment focussing on nuclear security or to pursue their academic career. The latter they can do via further in-depth academic work or pursuing doctoral studies in a related field.

2.1 PROGRAM IN DEPTH

2.2 Curriculum

The curriculum design is based on the results of the internal revision process of the IAEA NSS12 and the teaching materials of INSEN. The program is divided into six modules. The first module is Security Management. Apart from security management approaches for the nuclear field, National Security and Counterterrorism will be the second main topic. The module will approach this from the point of view of both the nation state's as well as the operator's. Sub-topics will include: Terrorist threats to nuclear and radiological materials and facilities, national security strategies, different approaches to counter-terrorism, the role of intelligence in countering nuclear and radiological terrorism, as well as different approaches to mitigating nuclear and radiological risks.

The second module International Law and Risk Assessment will address Threat Assessment & Planning and International Cooperation, as well as the Legal Framework and Governance. This will include the following issues with regard to the second topic: Formal international nuclear security instruments, informal multi-lateral nuclear security initiatives, national nuclear security legislation and regulation, security culture at the organizational level, and approaches to security culture assessment.

Fundamentals of Mathematics and Technology will cover Physical Protection, which will deal with physical protection systems design and evaluation, components of physical protection systems (PPS), application of PPS to nuclear material and facilities and other radioactive material and sources, as well as integration with other nuclear security measures. The second topic in this module will be Computer/IT/Cyber Security, which is continuously growing in its importance.

The fourth module Nuclear Security will be split into two parts, too: Nuclear Security in Storage & Detection and Response to Nuclear and Other Radioactive Material out of Regulatory Control (MoRC). In the first one, Security in Transport and Nuclear Material Accountancy and Control will be addressed. The second one will cover detection systems for MoRC, response to incidents involving MoRC, radiological crime scene investigation, as well as nuclear forensics and attribution.

Furthermore, there will be a module "Research and Academic Working", and Master's students will have the opportunity to choose three "Compulsory Facultative Courses" (electives), such as nuclear forensics, nuclear security at major public events, or import/export and transit control mechanism and regime.

2.3 Mode of studies

MiNS will be mainly provided as a distance learning program. Courses will be offered online. Besides paper-based material, provided as so called 'academic letters', three e-learning courses as well as blended learning will be offered. The first courses which will be provided as e-learning courses are Computer Security, Nuclear Security Management and Physical Security. Our objective is to provide all courses as e-learning in the forseeable future. The E-learning lectures will always work together with blended learning. Blended learning stands for the active use of web-based instruments such as social media, e-learning systems for distance learning and video conferencing throughout the entire Masters' program. This will include virtual classrooms and mentoring. It is our outspoken objective to orient those virtual classrooms to different time zones. This will allow students without any geographical restrictions to become part of MiNS and will not require a physical presence at Brandenburg in Germany. The most likely restriction could be a laking availability of the required bandwith for some students. A big share of our blended learning courses will be provided by our international partners.

MiNS students will be aible to take exams either online, in our patner universities or most likely at Goethe Instittutes, as well.

2.4 Summary

The mode of study underlines the Master's uniqueness, as students will be able to participate in the program from anywhere in the world and at certified educational institutions. MiNS prepares participants to use the appropriate analytical tools to make thorough decisions in the various areas connected to nuclear security. students will receive solid knowledge in nuclear security, which enables them to find synergy in thinking between security, safety and business, as well as risk management and corporate governance. MiNS will enable participants to work at a strategic level within the field of nuclear security. Apart from Bachelor students, this applies, for instance, to international diplomatic staff, security professionals, employees of nuclear installations and industry, of research/academic institutions or of regulatory authorities, as well as nuclear security/safety officers in national authorities and federal ministries. In a nutshell, the Master's program is a cost-effective way of educating and rewarding nuclear security managers and strategic talent in various functions.

3 PROGRAM CONTRIBUTERS

Our Team currently consists of four program managers and five program contributers. The project managers are Prof. Dr. Friedrich Holl, Mr. Guido Glusche and Mr. Marco Macori from the Institute for Security and Saftey at Brandenburg University, as well as Dr. Johannes H. Sterba from TU Vienna. Dr. Jason T. Harris (Purdue University, US), Dr. Christopher Hobbs (Kings College London), Dr. Edward J. Waller from University of Ontario University of Technology (Canada), as well as Dmytry Cherkashyn from ISS are further contributing their expertise to MiNS.

4 CERTIFIED EDUCATIONAL PARTNERS

It is our objective to cooperate with educational institutions worldwide for MiNS. Educational institutions that offer a Masters' program or course related to nuclear security based on NSS 12 therefore can become certified educational partners of ISS. This status will allow for different kinds of collaboration with MiNS. For example, certified educational partners could provide an elective course. In fact, we seek for a wide range of different international contributors for our elective courses. Elective courses might be provided by Uiversities and research institutions alike. Another form of cooperation could be that ISS would accredit courses from its partners for MiNS. Partners of course could use courses offered for MiNS to enrich their curriculum, too. To become a certified educational partners ISS and the party interested in cooperation simply would need to sign a memorandum of understanding.

5 PARTNERING AND COOPERATION

There is a wide range of possibilities to partner or cooperate with MiNS. The possibilities we are thinking of include, for example, to provide a joint or double degree by using courses of MiNS (for distance or classroom learning). As mentioned before, Brandenburg University could also accept your course(s) with x ECTS and, thus, your student would reduce his workload in MiNS. Furthermore, students' master thesis could be accepted by both institutions (if it‘s written on nuclear security), or you could include online course(s) from MiNS into your own Master‘s program. You could also provide the blended learning part of MiNS at your university for MiNS students. We are also happy to discuss your ideas and proposals in this regard.

6 NUCLEAR SECURITY CERTIFICATE PROGRAM

Besides the full Master's program in nuclear security, we are able to develop bespoke certificate programs in nuclear security fitting to individual companies needs. Such a certificate program will be an individual composition of courses which strengthen the knowledge base of employees, widen their academic expertise and might fulfill the regulatory requirements in terms of scientific education in nuclear security.

f you would like to offer such a program for employees of your company or organization, we will work together with you to create it. Modules of the Master's program that are relevant in your context can be chosen, and, thus, can build a strong basis for a companies' in-house educational framework.

7 SUPPORTING AND PROMOTING MINS

We address all organizations or individuals that are willing to support our educational activities and the efforts linked to them. Anyone who sees a value in educating students and professionals in terms of nuclear security is welcome to support us, e.g. with promotion, funding or scholarships – in particular with a focus on the Master in Nuclear Security (MiNS) at the Brandenburg University of Applied Sciences. Various ways are possible in this regard.

For example, you could promote MiNS during one of your own events or during a relevant revent that you are participating in. We would also welcome recommodations of MiNS or if you'd provide access to your relevant networks. Also supporting opportunities are manifold. For example, you could become a visible sponsor of MiNS, provide funding and, thus, help to guarantee its sustainability. We are also seeking for scholarships for students. This is crucial since we have a high interest in MiNS from students from developing countries. Or would you prefer to financially support students from your own organization? We would also appreciate if you'd like to become an active part of MiNS as a cost-free expert by providing your valuable expertise on a pro bono basis to the next generation. For instance, as a guest lecturer for an additional elective course. Last but not least, we are seeking for all different kinds of internshipy or work opportunities for our prospective students, too.

8 CONTACT INFORMATION

If you want to support, promote or cooperate with MiNS, or in case you have any question related to MiNS, please contact:

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DOPEX project: Toward Fast-Computing Tools for weapon effects evaluation on nuclear facilities

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Abstract:

This paper presents the DOPEX project supported by IRSN. The aim of this project is to develop fast-computing tools for the specific requirements of nuclear security. It is designed to evaluate weapon effects for an aggression scenario against nuclear facilities and material. These tools are in particular useful for IRSN's engineers with a broad expertise in nuclear facilities design, but who are not necessarily experts in weapon effects. The tools are a great help for a first evaluation in case of technical assessments carried out according to the regulations or in case of a crisis situation.

1 ROLE OF IRSN IN THE FRENCH NUCLEAR SECURITY FIELD

IRSN is a public body with industrial and commercial activities set up in 2002. It provides technical support to all the government authorities in France involved in the security of nuclear material, nuclear facilities and the transportation of nuclear material. In compliance with the agreement between the Ministry for Energy and IRSN, the Institute conducts studies and experiments to support the technical assessments. It is for this reason that IRSN needs and develops different technical tools and softwares for fast or detailed studies especially in the nuclear security field.

2 NUCLEAR SECURITY TOOLS

2.1 Technical needs

For IRSN missions, fast-computing tools developped in the DOPEX project are used by an engineer for a first evaluation of the damages caused further to aggressions against nuclear facilities (for example, fire weapons or rocket launcher…). When necessary the nuclear security computing tools are used for technical assessments to identify the potential vulnerabilities and to estimate a first order of magnitude of damages for aggression means employed. This approach is also to identify and to focus on potential targets.

For crisis situations, the nuclear security fast-computing tools estimates the state of nuclear facilities after an attack or to identify the potential aggravating factors by an aggression scenario in progress. A fast evaluation leads to revision of the projected source term for possible radiological consequences.

The goal of the fast-computing tools developped in the DOPEX project is to give a first order of magnitude of damages in maximum ten minutes time. It's a qualitative approach for a first evaluation, before possible additional detailed studies carried out with Computational Fluid Dynamic (CFD) softwares.

2.2 The principle of the DOPEX project

The DOPEX tools are developped for specfic security requirements for nuclear installations and materials. Each tool, corresponding to a specific aggression mean, is comprised of empirical relations and analytical physical models for weapon effects from the litterature (for example, [1], [2], [3]) or IRSN's research works (with simulation or experimental approaches [4]). Many of these models are well known in the litterature. The tools take into account some particularities of each nuclear plant by a tridimensionnal geographic model in input (for example, topography environment or physical protection equipments).

The users of the DOPEX tools define as input different geographic coordinates (for example, target coordinates) and agression means studied. The different computing steps (loading 3D model, extraction of dimensions and calculations of weapon effects) are automatically performed by the tool employed. In a few minutes, the users, non-expert in evaluation of weapon effects, read and analyze the final results on a map or a meshing of 3D model.

In addition the development goals are to create a catalog of the fast-computing tools for specific aggression means and to constitute a geographic security data base (with Geographic Information System (GIS)). These tools must be used by non-expert in weapon effects evaluation.

3 GEOGRAPHIC INFORMATION SYSTEM WITH NUCLEAR SECURITY DATA BASE

3.1 Geographic security data base

One of the data types used by the fast-computing tools is geographic information. IRSN continuous collection of different geographic information, contributes to create a specific geographic data base for the nuclear security field. This data base includes different georeferenced information in regards with the French nuclear facilities:

- Topography and building (elevation, identification of building's number);
- Physical barrier (localization, type and delimitation of physical protection areas);
- Detection and mitigation equipements for nuclear safety and security;
- Location of response forces inside or outside the plant area.

These geographic data are obtained from different information sources according to the classification levels (public or confidential information), for instance:

- For the localization of public roads or digital elevation model, it is possible to use the data collected by the French Institute of Geographic Information (IGN);
- For the localization and type of different physical protection barriers or equipments, which constitute the specific security information, it is necessary to have an acces of the security plan or study, required by the French security regulations (confidential data).

The plant area data and the topography information constitute many layers of geographic information system which are used as input for the fast-computing tools on the weapon effects evaluation.

3.2 Modeling of nuclear plant's topography

The elevation information collected in the geographic security data base (for buildings and digital land models) are used to create a tridimensionnal meshing with simple geometric shapes (prismatic or cylinder structure). The perimeter of meshing includes the nuclear installations and nearby buildings (one kilometer radius of the site).

A part of the meshing (outside the plant area) is generated on the basis of the data collected by IGN such as the infrastructure, different administration levels ect… called the “BD TOPO” data base or transmetted by others operators. The reading of these data and the elevation grid of meshing is realized with GIS softwares (ARCGIS[5] or GLOBAL MAPPER[6] (figure 1)).

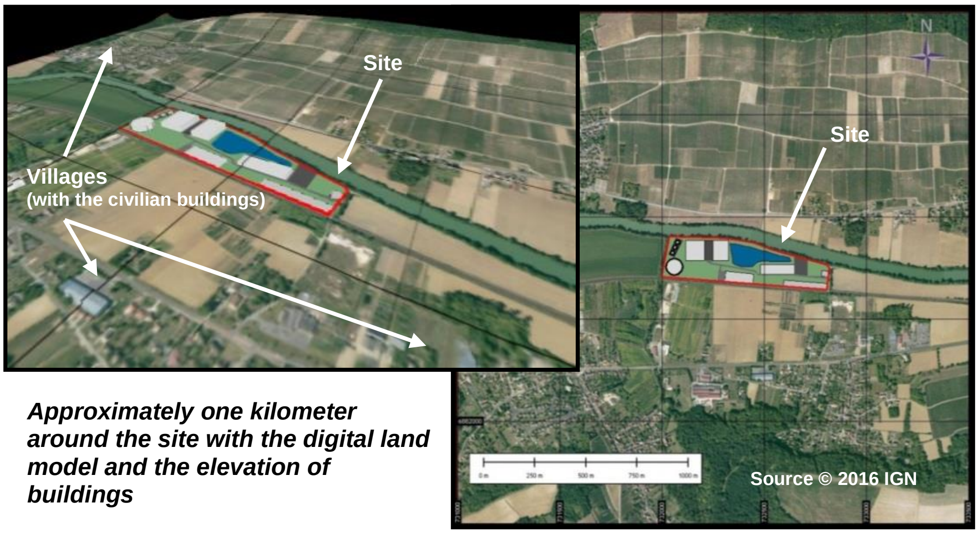
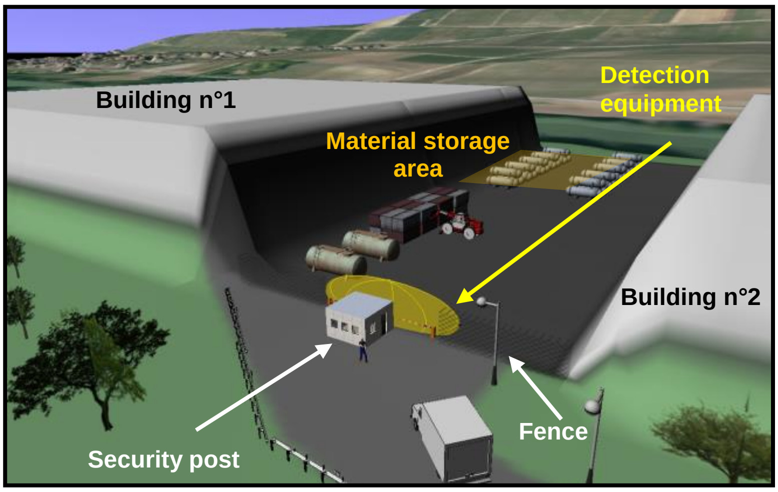


Fig. 1 View of a plant area with a GIS software (fictive factory)

3.3 Geographic input data for Nuclear Security Tools

The tri-dimensionnal meshing of a nuclear site is used by other security softwares than the DOPEX tools. The evaluation of progress outsite facilities is evaluated with a simple tri-dimensionnal model (distance and time). This evaluation is conducted with GIS softwares or the SICAP software developped by IRSN (not detailed in this paper, figure 2). The meshing may also be used by Computational Fluid Dynamic softwares (for example, LS-DYNA).



With a 3D model and the SICAP software, the user can:

- ***View the progress of person on a plant area;***
- ***Identify the physical barriers and the detection equipment to be overcome;***
- ***Estimate the travelled distance;***
- ***Contribute to assess the progress times (on foot or by car).***

Fig.2 View of a progress with SICAP software used a 3D model (fictive factory)

4 FAST-COMPUTING TOOLS EMPLOYED

4.1 Perimeter of the DOPEX tools

In a few minutes, the DOPEX tools show a first order of magnitude of damages caused further to an aggression scenario against a nuclear facility. For detailed evaluation of the damages, or a quantitative approach, the CFD software is necessary. The accuracy is far better, but the computing time is far more important than the DOPEX tools (figure 3).

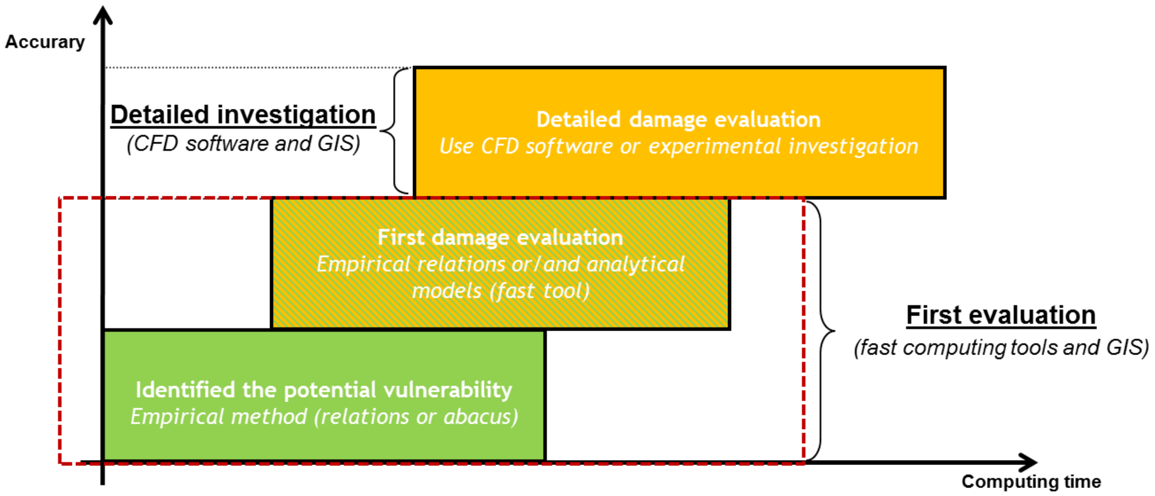


Fig. 3 Perimeter of the DOPEX tools

Main advantages of the DOPEX tools are the accuracy compared to the computing time ratio and the automatic connection between geographic data and weapon effects approaches.

4.2 Operation principle

In case of a technical assessment or a crisis situation, the users choose the most suitable fast-computing tool according to the agression means studied (for example, fire weapons or rocket launcher). In input, the DOPEX tools need geographic coordinates, minimum a target position, and the configuration design (geometric dimensions or type of materials) on a nuclear plant. These data are mainly extracted from a 3D model recorded in the geographic security data base:

- For ballistic impact, the shooting trajectories, and the possible presence of equipement or buildings on these trajectories, must be identified;
- For explosion scenarios, a suitable simple geometric model must be identified in view of the configuration studied on a nuclear plant.

Based on the recorded geometric data input (dimensions), the DOPEX tools will estimate different geometrical parameters needed for the evaluation of weapon effects (for example, angle of incident blast wave on the building or the travelled distance of the rocket). This geometrical estimation is carried out by the first part of the DOPEX tools called “geometric“ modules. Figure 4 present an example of geometric automatic assessing for an aggression scenario with a rocket launcher: geometrical parameters (angle, distance) and identification of possible equipment on the trajectories studied.

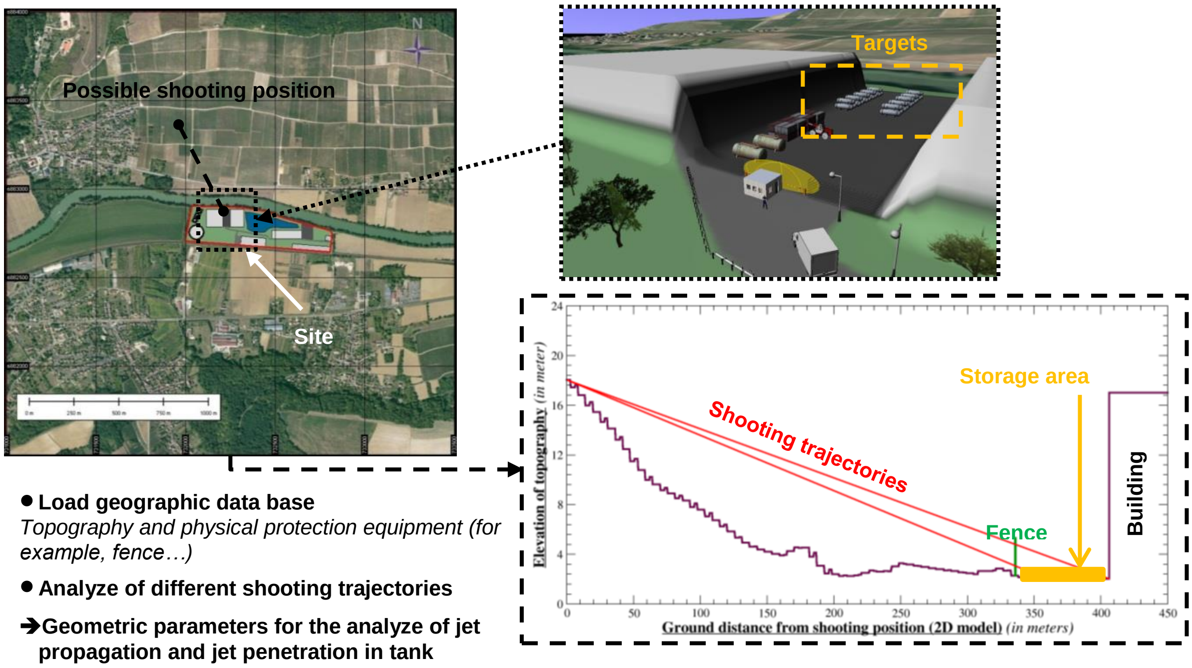


Fig. 4 Example of a geometric automatic processing for an aggression with rocket launcher

The evaluation of weapon effects is carried out with geometrical parameters and physical models from the litterature or IRSN's research works. For each fast-computing tool, the different analytical and empirical relations used are synthesized in a second module. This second part of the DOPEX tools is called "physical" module. Afterwards during the final calculation, these two modules (geometric and physical) share different information, leading to a damage mapping.

4.2.1 Evaluation of blast wave parameters

For an explosion scenario (detonation), the blast wave interacts with surrounding buildings or equipements. During these interactions, the blast wave is disturbed by the presence of different physical phenomena. In the DOPEX tools, the physical modules take into account many of these phenomena, for example (figure 5):

- Decrease by travelled distance (effects of rarefaction waves behind shock) with the empirical relations in free field [7];
- Reflection phenomenon with abacus [8] and analytical relations;
- Rarefaction phenomenon with abacus [3];
- Recombination of blast waves with the method of image [9].

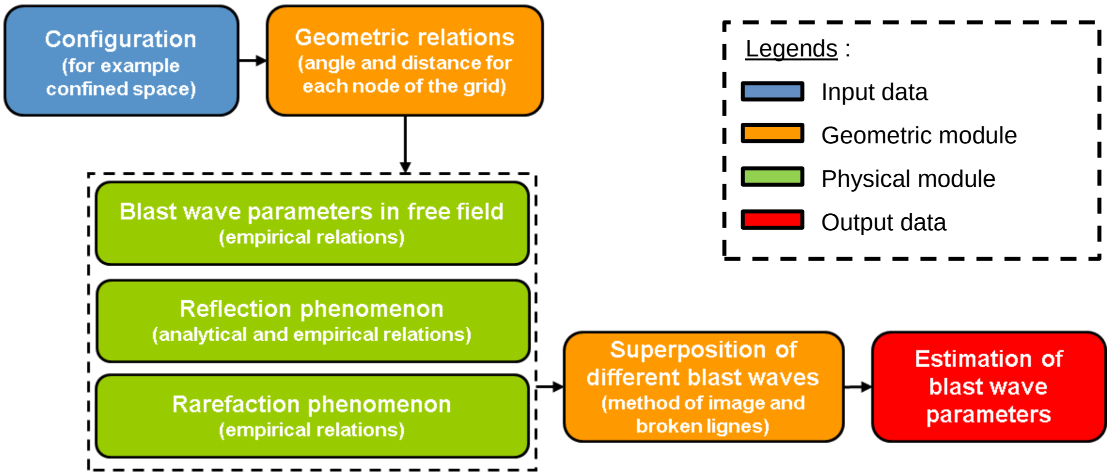


Fig. 5 Example of operation for the evaluation of blast wave parameters

For each point concerning meshing of configuration studied (pool, closed room, space between two buildings…), the pressure signal is estimated by the superposition of different blast waves: incident, recombination and reflected blast waves [3]. The blast wave parameters are evaluated from this pressure signal (maximum overpressure or impulse).

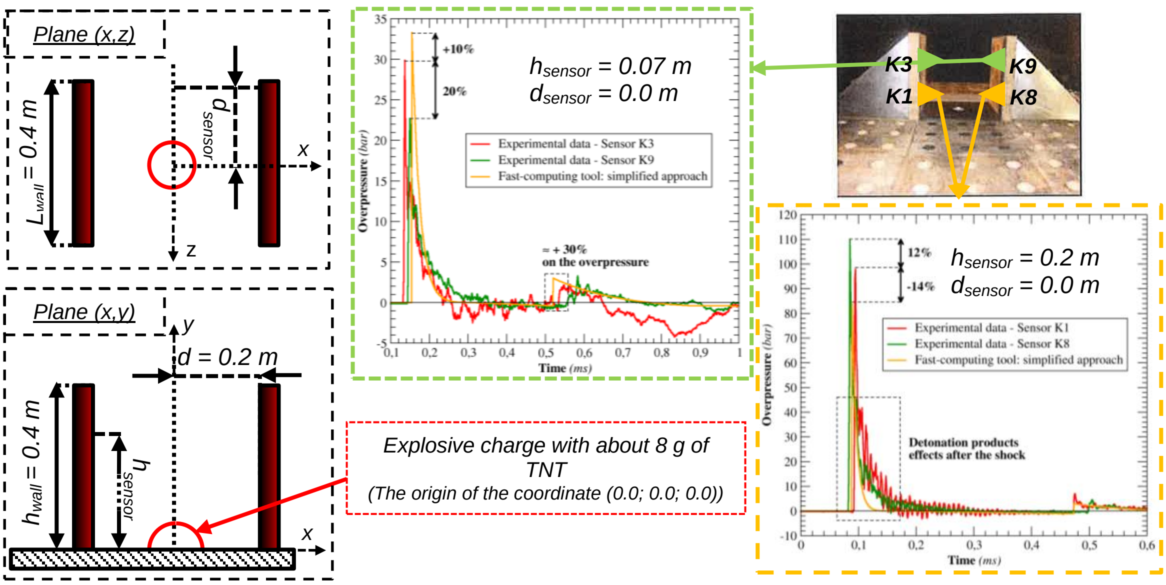


Fig. 6 Example of predictive capacities evaluation of a DOPEX tool

The accuracy of the DOPEX tools is evaluated by a comparison between the pressure profile estimated by a fast-computing tool and experimental data [4] (figure 6). For the tools currently used, the maximum discrepancies is estimated to be around 20-30% error. This is considered acceptable for a prompt first evaluation of the damages.

4.2.2 Evaluation of blast wave effects

The input data for the evaluation of damages (on an equipment and a building) is the map of pressure profiles generated by the physical modules. According to the value of maximum overpressure or positive impulse, the part of post-processing in the DOPEX tools edit a map for the users with different damage areas by standard values from the litterature or the results of IRSN's reserach for example destruction of simple concrete wall, rupture of tanks and pipes…

Figure 7 presents an example of maps edited by the DOPEX tools for an explosion scenario in the storage material area between two buildings on a fictive factory. For this example, the effected area used are defined by the French regulation on the Pyrotechnic Safety [10].

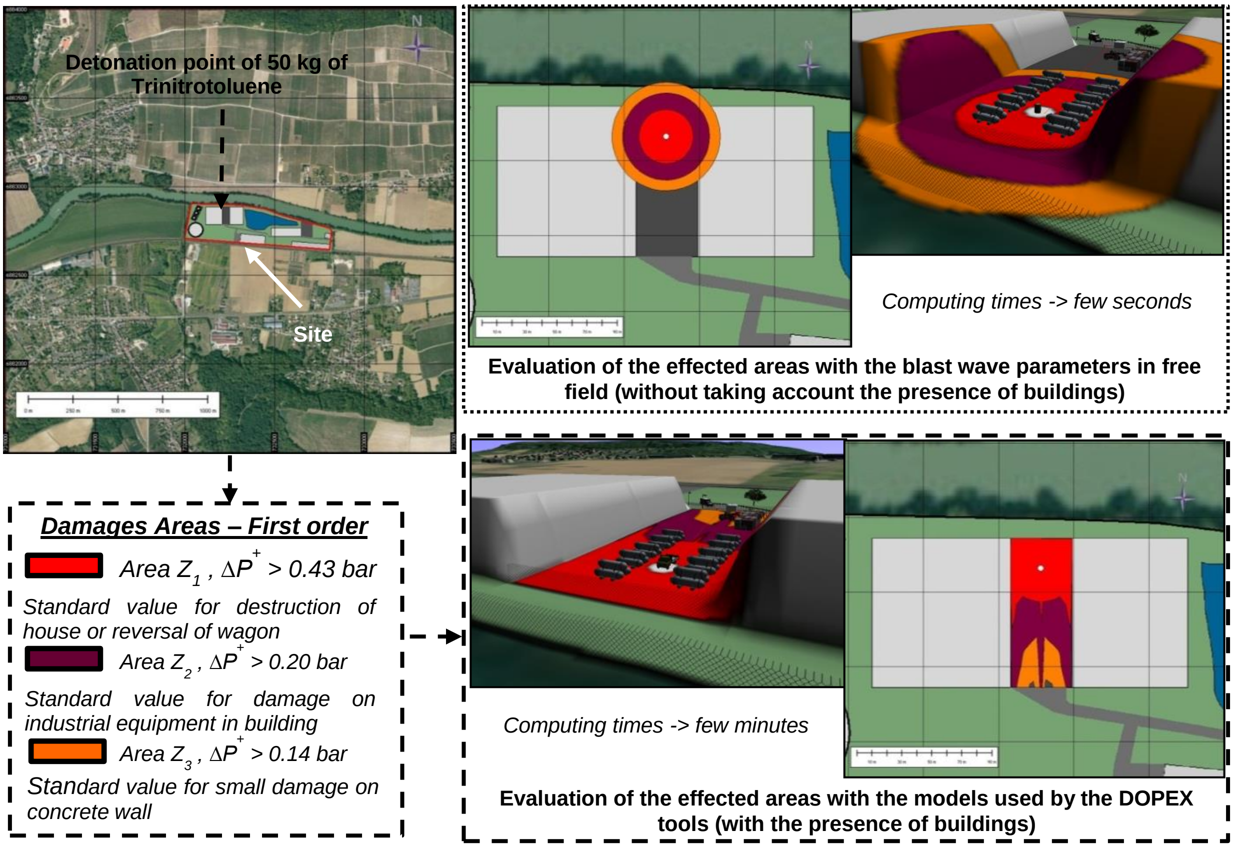


Fig. 7 An example of the map edited by the DOPEX tools for an explosion scenario

In case of a simple concrete wall in front of an explosive device, the yield lines method [8] could be used in the post-processing for a first evaluation: between slight damages and complete destruction.

4.2.3 Evaluation of weapon effects

The potential aggression means include others weapons than the explosive device (for example, fire weapons or rocket launchers). For these others aggression means, there are also specific physical modules (ballistic impact or shaped charge (PER method [11] or [12])). In input data of the DOPEX tools, the users give the geographic data and the tools will return automatically a first evaluation of weapon effects on a map or a meshing.

All fast-computing tools developped in the DOPEX project constitute a first version catalog of tools for the specfic requirements to the nuclear security field.

4.3 Limits of the simple tools

The results obtained with the DOPEX tools are just a first evaluation of magnitude of damages. This qualitative approach uses simple physical models and simplistic hypothesis of the physical phenomena observed:

- Simple geometric conditions and structures, not complex configurations;
- The Physical modules do not consider all physical phenomena observed, just the main phenomena (for example, the presence of detonation products in confined space is not considered).

The DOPEX tools are fast and give a global overview of damages expected or caused. It should be taken into account that the DOPEX tools provide results with a certain degree of uncertainty. The results however are a great indication of how the incident or accident situation could be recovered or/and whether a further in depth study is necessary. In this case, the CFD software will be used, knowing that the computing time is more important than the DOPEX tools.

5 CONCLUSION AND PERSPECTIVE

The aim of the DOPEX project is to develop the fast-computing tool for the specific requirements of the nuclear security according to different aggression means against a nuclear facility. These tools are very easy to use for an engineer with a broad expertise in nuclear facilities design, but who are not necessarily experts in weapon effects. It gives a global overview of the damages expected by an aggression scenario.

The use of the DOPEX tools requires a maintenance and a regular updating of the geographic security database around nuclear sites: elevation model (land and buildings), localization and type of physical protection equipements…

Futhermore, some computer development works are in progress to carry out a fully automatic connection between DOPEX's modules (geometric and physical) in order to improve the man-machine interface and to reduce computing times.

Others tools are still in developpment and the current physical models used in the DOPEX tools are continuously improving to reduce uncertainty, thanks to research such as developped abacus or analytical models. Furthermore, the database of the standard effect values (maximum overpressure or impulse) must be extended, in particular about factors used to evaluate a source term for a first estimation of radiological consequences.

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